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October 4, 2004
L-04-124

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

**Subject: Beaver Valley Power Station, Unit No. 1 and No. 2**
**BV-1 Docket No. 50-334, License No. DPR-66**
**BV-2 Docket No. 50-412, License No. NPF-73**
**License Amendment Request Nos. 318 and 191**

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above licenses in the form of changes to the Technical Specifications. The proposed change requests approval to apply the Westinghouse Best-Estimate Loss-of-Coolant Accident (BELOCA) analysis methodology to Beaver Valley Power Station Unit Nos. 1 and 2, and requests amendment of the respective Technical Specifications. This BELOCA methodology has previously been approved on a generic basis by the NRC as presented in WCAP 12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

This License Amendment Request (LAR) contains one enclosure with four attachments. The proposed Technical Specification changes are provided in Attachments A-1 and A-2 for Unit Nos. 1 and 2, respectively. The proposed changes to the Technical Specification Bases are provided in Attachments B-1 and B-2 for Unit Nos. 1 and 2, respectively.

A BELOCA analysis has been completed for each unit assuming an atmospheric containment design. Therefore, for the corresponding unit, implementation of the BELOCA amendment is contingent upon approval of the containment conversion LAR (317/190 for Unit Nos. 1 and 2) which was submitted by letter L-04-073 dated June 2, 2004. Thus, the implementation dates for the BELOCA amendments are to be consistent with the implementation dates of the corresponding containment conversion amendments.

FENOC requests approval of the proposed BELOCA amendments by October 2005. However, since a number of the Technical Specification changes proposed in the containment conversion LAR (317/190) require a plant outage to implement, FENOC requests the following implementation periods. The Unit No. 1 containment conversion and BELOCA amendments shall be implemented prior to the first entry into Mode 4

A001

during plant startup from the 1R17 refueling outage planned for the spring of 2006. The Unit No. 2 containment conversion and BELOCA amendments shall be implemented prior to the first entry into Mode 4 during plant startup from the 2R12 refueling outage planned for the fall of 2006.

The Beaver Valley Power Station review committees have reviewed this change. The change was determined to be safe and does not involve a significant hazard consideration as defined in 10 CFR 50.92, based on the attached safety analysis and no significant hazard evaluation. No new regulatory commitments are contained in this submittal.

If there are any questions concerning this matter, please contact Mr. Henry L Hegrat, Supervisor, Licensing at 330-315-6944.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 4, 2004.

Sincerely,

L. W. Pearce

L. William Pearce

Enclosure:
FENOC Evaluation of the Proposed Changes

Attachments:
A-1 Proposed Unit No. 1 Technical Specification Changes
A-2 Proposed Unit No. 2 Technical Specification Changes
B-1 Proposed Unit No. 1 Technical Specification Bases Changes
B-2 Proposed Unit No. 2 Technical Specification Bases Changes

c: Mr. T. G. Colburn, NRR Senior Project Manager
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Central File - *Keywords: Best-Estimate Loss of Coolant Accident, BELOCA, Containment Conversion, Extended Power Uprate.*

**ENCLOSURE**

**FENOC Evaluation of the Proposed Changes**

Beaver Valley Power Station
License Amendment Requests
318 (Unit No. 1) and 191 (Unit No. 2)

Subject: Application to Permit Operation with Best-Estimate Large Break LOCA Methodology.

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Attachments


| Number | Title |
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| A-1 | Proposed Unit No. 1 Technical Specification Changes |
| A-2 | Proposed Unit No. 2 Technical Specification Changes |
| B-1 | Proposed Unit No. 1 Technical Specification Bases Changes |
| B-2 | Proposed Unit No. 2 Technical Specification Bases Changes |

## 1.0 DESCRIPTION

This License Amendment Request (LAR) for operating licenses DPR-66 (Beaver Valley Power Station Unit No. 1) and NPF-73 (Beaver Valley Power Station Unit No. 2) requests approval to apply the Westinghouse large break Best-Estimate Loss-of-Coolant-Accident (BELOCA) analysis methodology. It is requested that Technical Specification 6.9.5, "Core Operating Limits Report (COLR)" be amended to allow use of the methodology.

## 2.0 PROPOSED CHANGES

The specific changes to the Technical Specifications (TS) that are proposed are shown on Attachments A-1 and A-2 for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2, respectively. Changes to the respective TS Bases are submitted for information in Attachments B-1 and B-2. The proposed Technical Specification Bases changes do not require NRC approval. The Beaver Valley Power Station Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. The Technical Specification Bases changes are provided for information only.

The proposed changes to the Technical Specifications and Technical Specification Bases have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined. This presentation allows the reviewer to readily identify the information that has been deleted and added. To meet format requirements the Indices, the Technical Specifications, and the Technical Specification Bases pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

Technical Specification 6.9.5.b lists applicable references for the analytical methods used to determine core operating limits identified in TS 6.9.5.a. This list of references includes the Westinghouse topical report that documents the currently approved large break LOCA analysis methodology. It is proposed that this reference would be replaced with the generically approved topical report, WCAP-12945-P-A, for the Westinghouse best-estimate large break LOCA analysis methodology (Reference 1).

The values of major plant parameters used in the large break BELOCA analyses are identified in Tables 1 and 2 for BVPS Unit Nos. 1 and 2, respectively.

Tables 3 and 4 present the 95th percentile peak cladding temperature (PCT), maximum cladding oxidation, maximum hydrogen generation, and cooling results for BVPS Unit Nos. 1 and 2, respectively. The limiting time period discussed in the Note on Tables 3 and 4 pertains to the limiting phase of the following three phases of the analysis: (1) the blowdown phase; (2) the early reflood phase; and, (3) the late reflood phase. Only the limiting phase is used for PCT reporting

purposes (10 CFR 50.46 reporting requirements). The late reflood phase (i.e., Reflood 2) is currently limiting for both BVPS units and the results reported in Tables 3 and 4 apply to that phase.

Tables 5 and 6 represent total minimum injected Safety Injection flow used in the analyses for BVPS Unit Nos. 1 and 2, respectively. The figures and tables in this LAR are based on the BVPS December 2002 analysis of record (AOR).

Figures 1 and 2 illustrate the lower bound containment pressure used for the BELOCA analyses for BVPS Unit Nos. 1 and 2, respectively. Figures 3 and 4 illustrate the operating limits for the Integral of the Power Generated in the Bottom Third of the Core (PBOT) and the Integral of the Power Generated in the Middle Third of the Core (PMID) for BVPS Unit Nos. 1 and 2, respectively.

The 2002 AOR includes the Constant Axial Offset Control (CAOC) methodology for the determination of Axial Flux Difference and Heat Flux Hot Channel Factor $F_Q(Z)$. Tables 3 and 4 list the effects on PCT of the Relaxed Axial Offset Control (RAOC) methodology. The effect on PTC due to RAOC is shown in the Tables 3 and 4 because FENOC plans on submitting LAR 310/182 that will change the BVPS Technical Specifications from the CAOC to the RAOC methodology. The effects on PCT shown in Tables 3 and 4 are being provided for information only and reflect the configuration of the BVPS units when the containment conversion, BELOCA, Extended Power Uprate and RAOC amendments are implemented.

## 3.0 BACKGROUND

Westinghouse has obtained generic NRC approval of its topical report describing large break BELOCA methodology. NRC approval of the methodology is documented in the NRC safety evaluation report appended to the topical report (WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998). Separate plant specific analyses for BVPS Unit Nos. 1 and 2 have been performed using the approved methodology.

These changes are being made to incorporate the best-estimate approach into the licensing basis for BVPS large break LOCA analyses in accordance with the Westinghouse "Code Qualification Document For Best Estimate LOCA Analysis," WCAP-12945-P-A, Volumes 1-5 (Reference 1), 10 CFR 50.46 (Reference 2), and Regulatory Guide 1.157 "Best-Estimate Calculations of Emergency Core Cooling System Performance" (Reference 3). The best-estimate methodology is needed to support a future extended power uprate of the BVPS units and its use is dependent on implementation of LAR 317 (Unit No. 1) and 190 (Unit No. 2), Operation with an Atmospheric Containment Design, submitted separately by FENOC letter L-04-073 dated June 2, 2004. Completed large break

BELOCA analyses have been performed at the planned extended power uprate conditions (2900 MWt) with an atmospheric containment design. The values of major plant parameters used in the large break BELOCA analyses, and other Updated Final Safety Analysis Report (UFSAR) changes resulting from approval of this LAR, will be made in accordance with 50.71(e) (Reference 4).

Both FENOC and its analysis vendor Westinghouse have ongoing processes in place that assure that analysis input values for peak clad temperature-sensitive parameters bound their as-operated plant values.

## 4.0 TECHNICAL ANALYSIS

Separate large break BELOCA analyses have been performed for BVPS Unit Nos. 1 and 2 using the methodology contained in WCAP-12945-P-A (Reference 1). All plant specific parameters used in the analyses are bounded by the models and correlations contained in the generic methodology. Therefore, the BVPS analyses conform to 10 CFR 50.46 (Reference 2) and Section II of Appendix K, and meet the intent of Regulatory Guide 1.157 (Reference 3). The conclusions of the analyses are that there is a high probability that:

1. The calculated maximum fuel element cladding temperature (peak cladding temperature) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (maximum cladding oxidation) will not exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (maximum hydrogen generation) will nowhere exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

5. After successful initial operation of the emergency core cooling system (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Therefore, FENOC has concluded that adopting the large break BELOCA methodology for BVPS Unit Nos. 1 and 2 and making the proposed TS changes would not adversely affect the health and safety of the public.

## 5.0 REGULATORY SAFETY ANALYSIS

### 5.1 No Significant Hazards Consideration

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

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

   No. No physical changes are required as a result of implementing best-estimate large break loss of coolant accident (LOCA) methodology and associated Technical Specification changes. The plant conditions used in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased, since it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph b. No other accident is potentially affected by this change.

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

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

   No. There are no physical changes being made to the Beaver Valley Power Station units. No new modes of plant operation are being introduced. The parameters used in the analysis are within the design limits of the existing plant equipment. All plant systems will perform as designed during the response to a potential accident.

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

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

   No. It has been shown that the methodology used in the analysis would more realistically describe the expected behavior of plant systems during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations and other

variations in properties are analyzed to provide assurance that the most severe postulated LOCAs are addressed. It has been shown by analysis that there is a high probability that all criteria contained in 10 CFR 50.46, Paragraph b are met.

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

Based on the above, FENOC concludes that the proposed amendments present 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 have been evaluated to determine whether applicable regulations and requirements would continue to be met. FENOC has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the UFSAR. Section 4 demonstrates that the proposed change is consistent with 10 CFR 50.46.

## 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.

Based on this evaluation and the fact that neither an environmental impact statement nor an environmental assessment is required, the proposed amendment will not have an adverse effect on the environment and can thus be deemed acceptable.

## 7.0 REFERENCES

1. WCAP 12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.
2. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."
3. Regulatory Guide 1.157 "Best-Estimate Calculations of Emergency Core Cooling System Performance (Draft RS 701-4 published 3/1987)."
4. 10 CFR 50.71(e), "Maintenance of records, making of reports."
5. NUREG-0800, Standard Review Plan, "Emergency Core Cooling."
6. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

Table 1

BVPS Unit No. 1 Major Plant Parameters
Used in the Best-Estimate Large Break LOCA Analysis

| Parameter | Value | Intended Location |
|---|---|---|
| Plant Physical Description | | |
| Steam Generator Tube Plugging | ≤ 22 % | UFSAR Section 14.3 |
| Plant Initial Operating Conditions | | |
| Reactor Power | ≤ 100.6% of 2900 MWt including 0.6% Calorimetric Uncertainty | UFSAR Section 14.3 |
| Peaking Factor | $F_Q$=2.52, $F_{\Delta H}$=1.75 | COLR |
| Fluids Conditions | | |
| RCS Average Temperature ($T_{avg}$) | 566.2 ± 4.1°F ≤ $T_{avg}$ ≤ 580.0 ± 4.1°F | UFSAR Section 14.3 |
| Pressurizer Pressure | 2200-2300 psia | UFSAR Section 14.3 |
| Reactor Coolant Flow | ≥ 87,200 gpm/loop | UFSAR Section 14.3 |
| Accumulator Temperature | 70-105°F | UFSAR Section 14.3 |
| Accumulator Pressure | 575-716 psia | UFSAR Section 14.3 |
| Accumulator Water Volume | 893-1022 $ft^3$ | UFSAR Section 14.3 |
| Accident Boundary Conditions | | |
| Single Failure Assumptions | 1 Train of ECCS Pumps | UFSAR Section 14.3 |
| Safety Injection Flow | Table 5 | UFSAR Section 14.3 |
| Safety Injection Temperature | 45-105°F | UFSAR Section 14.3 |
| Safety Injection Initiation Delay Time | ≤ 17 sec Off-Site Power Available<br>≤ 27 sec Loss of Off-Site Power (LOOP) | UFSAR Section 14.3 |

Table 2

BVPS Unit No. 2 Major Plant Parameters
Used in the Best-Estimate Large Break LOCA Analysis

| Parameter | Value | Intended Location |
|---|---|---|
| Plant Physical Description | | |
| Steam Generator Tube Plugging | ≤ 22 % | UFSAR Section 15.6 |
| Plant Initial Operating Conditions | | |
| Reactor Power | ≤ 100.6% of 2900 MWt including 0.6% Calorimetric Uncertainty | UFSAR Section 15.6 |
| Peaking Factor | $F_Q$=2.52, $F_{\Delta H}$=1.75 | COLR |
| Fluids Conditions | | |
| RCS Average Temperature ($T_{avg}$) | 566.2 ± 4°F ≤ $T_{avg}$ ≤ 580.0 ± 4°F | UFSAR Section 15.6 |
| Pressurizer Pressure | 2200-2300 psia | UFSAR Section 15.6 |
| Reactor Coolant Flow | ≥ 87,200 gpm/loop | UFSAR Section 15.6 |
| Accumulator Temperature | 70-105°F | UFSAR Section 15.6 |
| Accumulator Pressure | 575-716 psia | UFSAR Section 15.6 |
| Accumulator Water Volume | 922-1072 $ft^3$ | UFSAR Section 15.6 |
| Accident Boundary Conditions | | |
| Single Failure Assumptions | 1 Train of ECCS Pumps | UFSAR Section 15.6 |
| Safety Injection Flow | Table 6 | UFSAR Section 15.6 |
| Safety Injection Temperature | 45-105°F | UFSAR Section 15.6 |
| Safety Injection Initiation Delay Time | ≤ 17 sec Off-Site Power Available<br>≤ 27 sec Loss of Off-Site Power (LOOP) | UFSAR Section 15.6 |

Table 3

BVPS Unit No. 1 Best-Estimate
Large Break LOCA Analysis Results

| | Value | Acceptance Criteria |
|---|---|---|
| 95th Percentile PCT (°F)* | 2144 | 2200 |
| | | |
| Maximum Cladding Oxidation (%)* | 8.77 | < 17 |
| | | |
| Maximum Hydrogen Generation (%)* | 0.985 | < 1 |
| | | |
| Coolable Geometry | Core Remains Coolable | Core Remains Coolable |
| | | |
| Long Term Cooling | Core Remains Cool in Long Term | Core Remains Cool in Long Term |

* Calculated using the methodology in the following reference:

WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," March 1998 (Westinghouse Proprietary).

Note: Subsequent to the determination of the 95th percentile PCT, reported above, various evaluations have been performed to estimate the effect of changes in the planned operation of the plant. These evaluations have been introduced since the completion of the original application of the BELOCA Evaluation Model, to the date of this submittal. Future evaluations and assessments against the PCT will be applied under the usual reporting guidelines of 10 CFR 50.46. The summary of effects for the limiting time period includes: +70°F for a containment initial pressure change, -209°F for a peaking factor reduction and +16°F for Relaxed Axial Offset Control implementation.

Table 4

BVPS Unit No. 2 Best-Estimate
Large Break LOCA Analysis Results

| | Value | Acceptance Criteria |
|---|---|---|
| 95th Percentile PCT (°F)* | 1976 | 2200 |
| | | |
| Maximum Cladding Oxidation (%)* | 6.7 | < 17 |
| | | |
| Maximum Hydrogen Generation (%)* | 0.91 | < 1 |
| | | |
| Coolable Geometry | Core Remains Coolable | Core Remains Coolable |
| | | |
| Long Term Cooling | Core Remains Cool in Long Term | Core Remains Cool in Long Term |

* Calculated using the methodology in the following reference:

WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," March 1998 (Westinghouse Proprietary).

Note: Subsequent to the determination of the 95th percentile PCT, reported above, various evaluations have been performed to estimate the effect of changes in the planned operation of the plant. These evaluations have been introduced since the completion of the original application of the BELOCA Evaluation Model, to the date of this submittal. Future evaluations and assessments against the PCT will be applied under the usual reporting guidelines of 10 CFR 50.46. The summary of effects for the limiting time period includes: 0°F for a containment initial pressure change and 0°F for Relaxed Axial Offset Control implementation.

Table 5

BVPS Unit No. 1 Best-Estimate Large Break LOCA Analysis
Total Minimum Injected Safety Injection Flow

(High Head and Low Head Safety Injection from 2 Intact Loops)

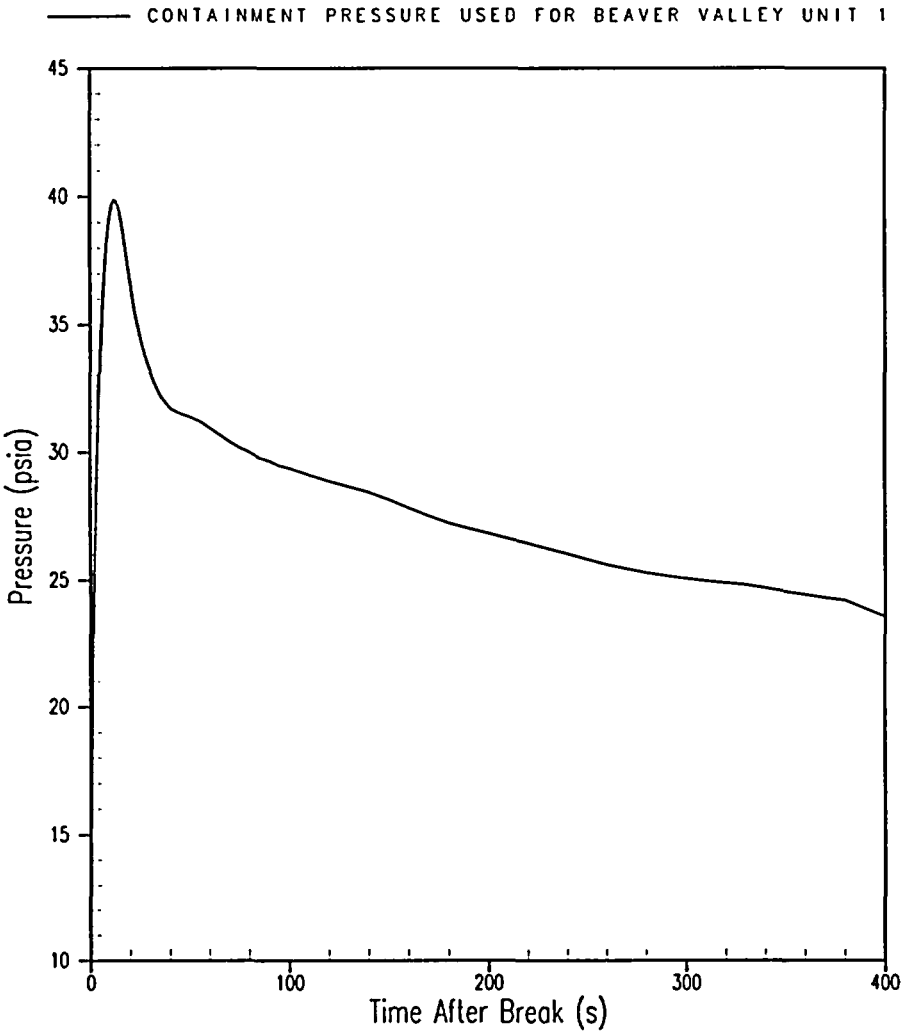
| RCS Pressure (psig) | Flow Rate (gpm) |
|---|---|
| 0 | 2433.0 |
| 10 | 2272.1 |
| 20 | 2106.2 |
| 50 | 1569.1 |
| 100 | 338.1 |
| 105 | 278.4 |
| 150 | 270.4 |
| 200 | 261.4 |
| 400 | 219.2 |
| 600 | 173.4 |

Table 6

BVPS Unit No. 2 Best-Estimate Large Break LOCA Analysis
Total Minimum Injected Safety Injection Flow

(High Head and Low Head Safety Injection from 2 Intact Loops)

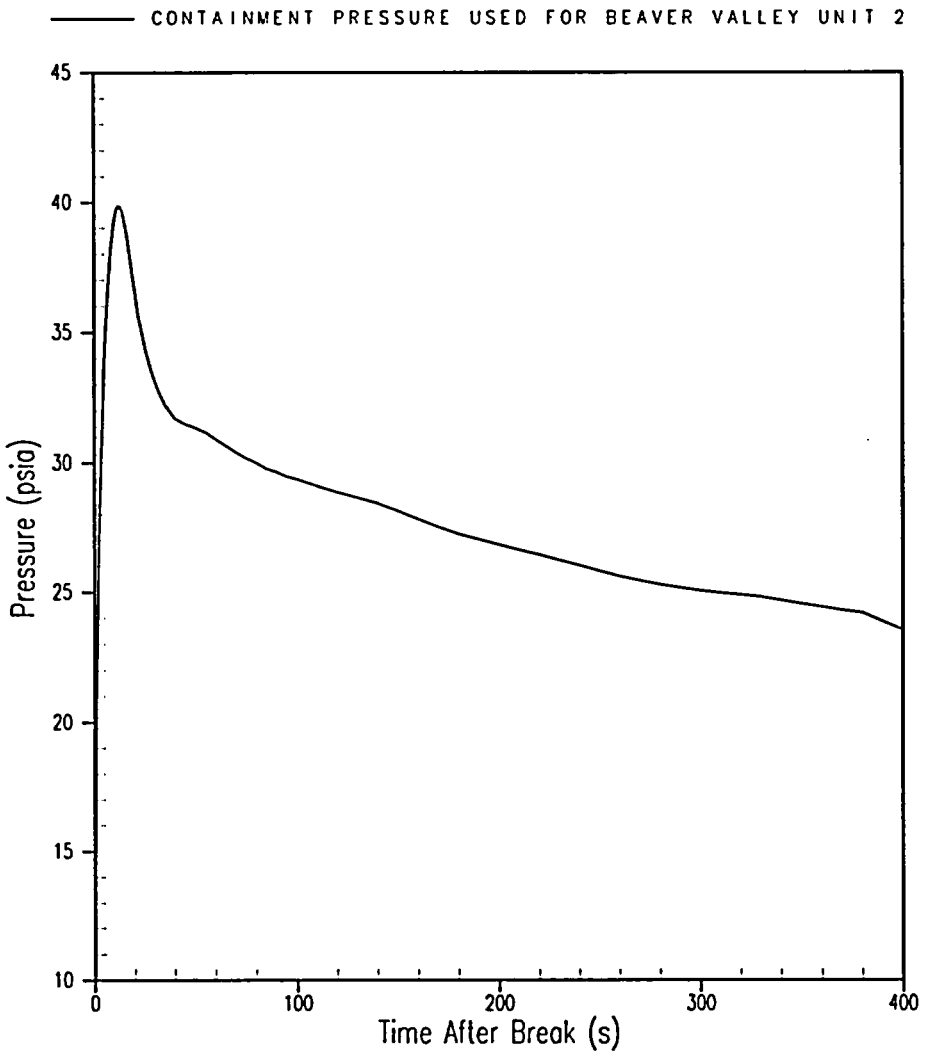
| RCS Pressure (psig) | Flow Rate (gpm) |
|---|---|
| 0 | 2719.5 |
| 10 | 2556.5 |
| 20 | 2385.5 |
| 50 | 1807.6 |
| 90 | 441.3 |
| 100 | 251.5 |
| 150 | 245.2 |
| 200 | 239.1 |
| 400 | 215.0 |
| 600 | 189.1 |




Based on BVPS December 2002 AOR.

**Figure 1**

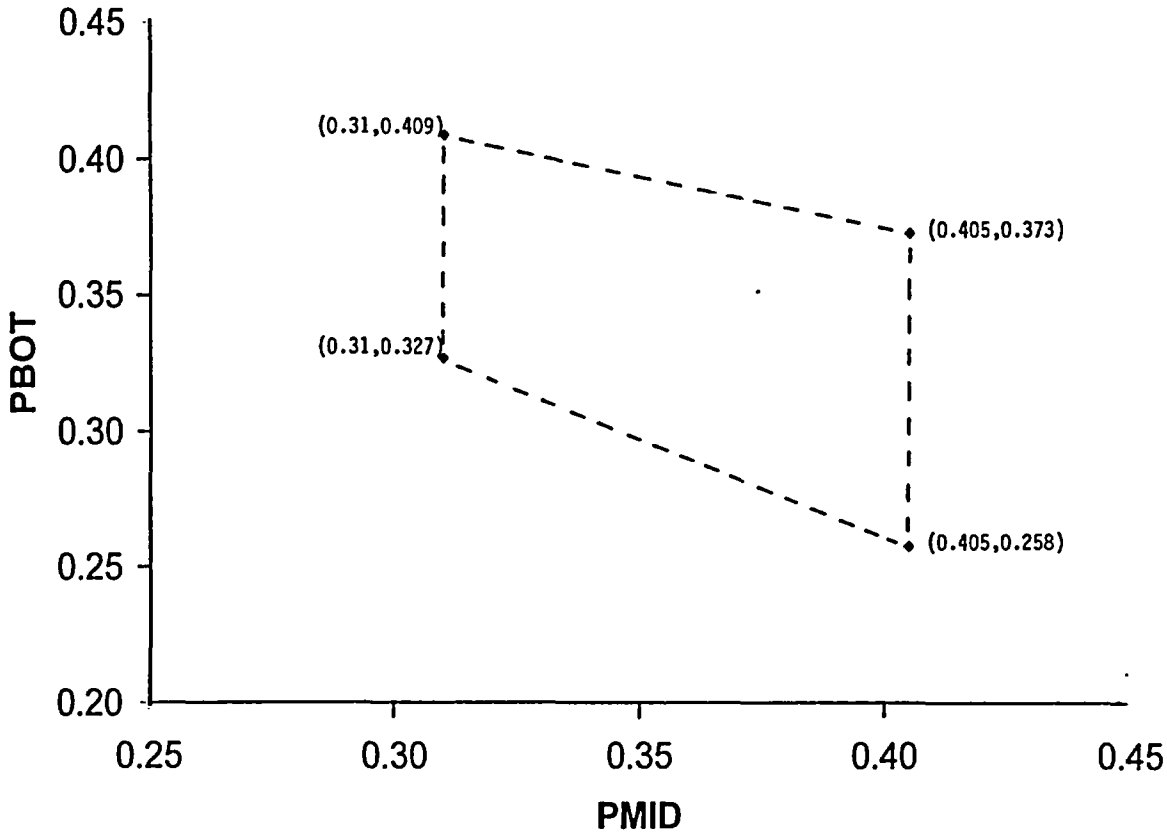
**Lower Bound Containment Pressure Used for BVPS Unit No. 1 Best-Estimate Large Break LOCA Analysis**




Based on BVPS December 2002 AOR.

**Figure 2**

**Lower Bound Containment Pressure Used for BVPS Unit No. 2 Best-Estimate Large Break LOCA Analysis**

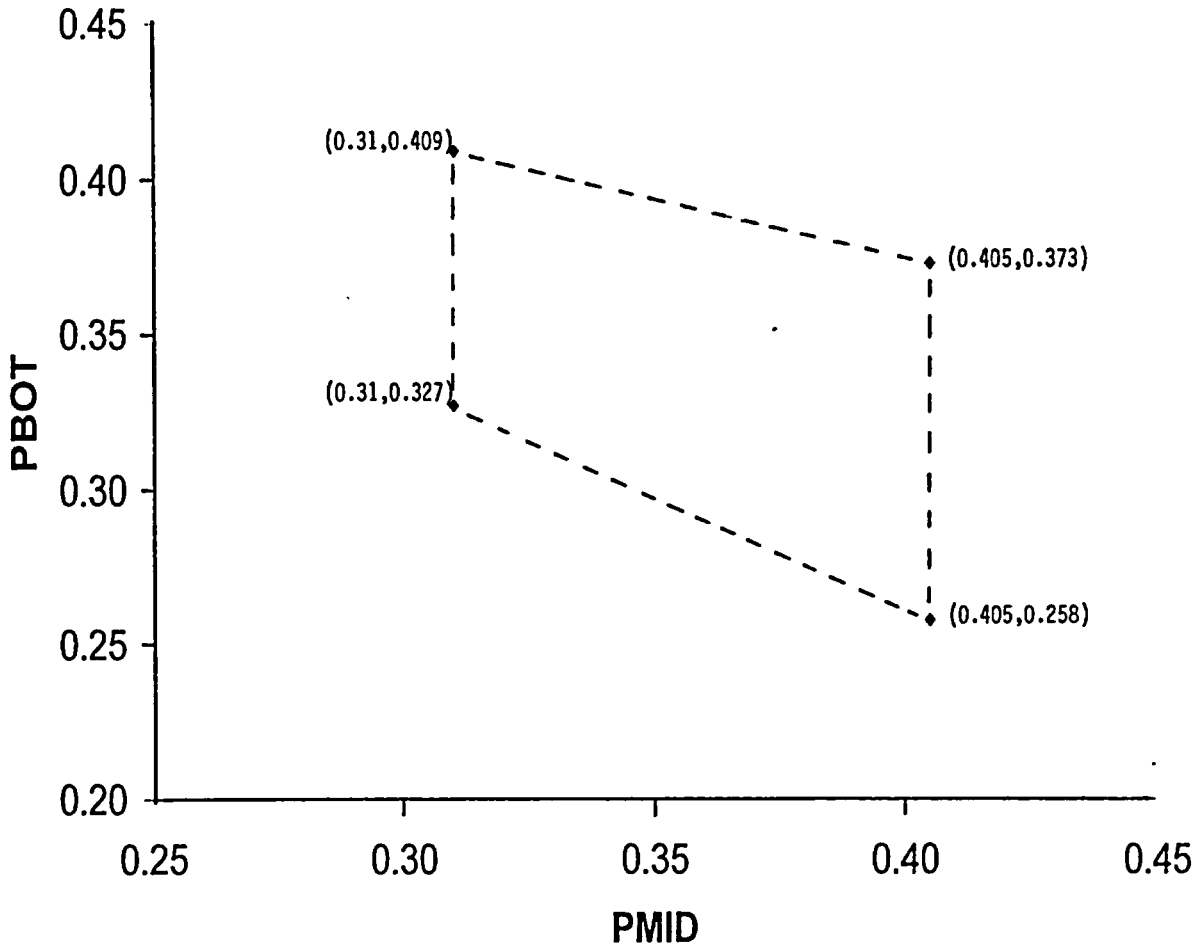



PBOT = Fraction of power in the lower third of the core.

PMID = Fraction of power in the mid-region of the core.

**Figure 3**

**BVPS Unit No. 1 PBOT/PMID Operating Limits**




PBOT = Fraction of power in the lower third of the core.

PMID = Fraction of power in the mid-region of the core.

**Figure 4**

**BVPS Unit No. 2 PBOT/PMID Operating Limits**

**Attachment A-1**

**Beaver Valley Power Station, Unit No. 1**
**Proposed Technical Specification Changes**

License Amendment Request No. 318

The following is a list of the affected pages:

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CORE OPERATING LIMITS REPORT (Continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).

WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

~~WCAP-10266-P-A Rev. 2/WCAP-11524-NP-A Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Kabadi, J. N., March 1987; including Addendum 1-A "Power Shape Sensitivity Studies" 12/87 and Addendum 2-A "BASH Methodology Improvements and Reliability Enhancements" 5/88.~~
WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT." September 1974 (Westinghouse Proprietary).

T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.

**Attachment A-2**

**Beaver Valley Power Station, Unit No. 2**
**Proposed Technical Specification Changes**

License Amendment Request No. 191

The following is a list of the affected pages:

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REPORTING REQUIREMENTS (Continued)

WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions," September 1986.

~~WCAP-10266-P-A Rev. 2/WCAP-11524-NP-A Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Kabadi, J. N., March 1987, including Addendum 1-A "Power Shape Sensitivity Studies" 12/87 and Addendum 2-A "BASH Methodology Improvements and Reliability Enhancements" 5/88.~~
WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT." September 1974 (Westinghouse Proprietary).

T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.

Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.

**Attachment B-1**

**Beaver Valley Power Station, Unit No. 1**

**Proposed Technical Specification Bases Changes**

License Amendment Request No. 318

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BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core ≥ the design DNBR limit during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F as specified in 10 CFR 50.46 is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are

3/4.2.2 AND 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS-$F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the ECCS acceptance criteria limit of 2200°F as specified in 10 CFR 50.46.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and Specification 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ±12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a through d above, are maintained.

When a $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate experimental error allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

The specified limit of $F^N_{\Delta H}$ contains an 8% allowance for uncertainties which means that normal, full power, three loop operation will result in $F^N_{\Delta H} \leq$ the design limit specified in the CORE OPERATING LIMITS REPORT.

3/4.2.4 QUADRANT POWER TILT RATIO (QPTR) (Continued)

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F as specified in ~~accordance with~~ 10 CFR 50.46;

b. During a loss of forced reactor coolant flow accident, there must be at least 95 percent probability at the 95 percent confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm in accordance with the indicated failure threshold from the TREAT results (UFSAR 14.2.6), and

d. The control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin (SDM) with the highest worth control rod stuck fully withdrawn in accordance with 10 CFR 50, Appendix A, GDC 26.

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analysis.

**Attachment B-2**

**Beaver Valley Power Station, Unit No. 2**

**Proposed Technical Specification Bases Changes**

License Amendment Request No. 191

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The following is a list of the affected pages:

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The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core ≥ the design DNBR limit during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F as specified in 10 CFR 50.46 is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time

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AXIAL FLUX DIFFERENCE (AFD) (Continued)

duration limit of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the CORE OPERATING LIMITS REPORT for THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% of RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the ECCS acceptance criteria limit of 2200°F as specified in 10 CFR 50.46.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

## 3/4.2.4 QUADRANT POWER TILT RATIO (QPTR)

### BACKGROUND

The Quadrant Power Tilt Ratio limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation. The QPTR is routinely determined using the power range channel input which is part of the power range nuclear instrumentation (NI). The power range channel provides a protection function and has operability requirements in LCO 3.3.1. While part of the NI channel, the power range channel input to QPTR functions independently of the power range channel in monitoring radial power distribution. For this reason, if the power range channel output is inoperable, the power range channel input to QPTR may be unaffected and capable of monitoring for the QPTR.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.3.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the design criteria and that the power distribution remains within the bounds used in the safety analyses.

### APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F as specified in ~~accordance with~~ 10 CFR 50.46;

b. During a loss of forced reactor coolant flow accident, there must be at least 95 percent probability at the 95 percent confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm in accordance with the indicated failure threshold from the TREAT results (UFSAR 15.4.8), and