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February 11, 2005
L-05-009

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. 310 and 182

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 Beaver Valley Power Station (BVPS) Technical Specifications. This License Amendment Request (LAR) proposes implementation of the Relaxed Axial Offset Control (RAOC) and F_Q surveillance methodologies. These methodologies are used to reduce operator action required to maintain conformance with power distribution control Technical Specifications, and increase the ability to return to power after a plant trip while still maintaining margin to safety limits under all operating conditions.

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. The proposed changes to the Licensing Requirements Manual (LRM) are provided in Attachments C-1 and C-2 for Unit Nos. 1 and 2, respectively. The Technical Specification Bases and LRM changes are provided for information only.

FENOC requests approval of the proposed amendments by January 2006 in order to support the RAOC analysis which assumes Extended Power Uprate (EPU) conditions, including the Best Estimate Loss of Coolant Accident (BELOCA) methodology. For each unit the RAOC amendments will be implemented concurrent with the applicable unit's EPU and BELOCA amendments. Thus, FENOC requests the following implementation periods.

The Unit No. 1 RAOC amendment shall be implemented prior to the first entry into Mode 4 during plant startup from the 1R17 refueling outage planned for the spring of 2006. The Unit No. 2 RAOC amendment 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.

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The Beaver Valley Power Station review committees have reviewed the proposed changes. The changes were determined to be safe and do 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 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 February 11, 2005.

Sincerely,

Richard G. Mende

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-1 Proposed Unit No. 1 Licensing Requirements Manual Changes
C-2 Proposed Unit No. 2 Licensing Requirements Manual Changes

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

ENCLOSURE
FENOC Evaluation of the Proposed Changes

Beaver Valley Power Station
License Amendment Requests
310 (Unit No. 1) and 182 (Unit No. 2)

Subject: Application to Implement the Relaxed Axial Offset Control (RAOC) and Heat Flux Hot Channel Factor (F_Q) Surveillance Methodologies

Table of Contents

Attachments

1.0 DESCRIPTION

This License Amendment Request (LAR) is a request to amend Operating Licenses DPR-66 (Beaver Valley Power Station Unit No. 1) and NPF-73 (Beaver Valley Power Station Unit No. 2). The proposed changes will revise the Operating Licenses to permit implementation of the Relaxed Axial Offset Control (RAOC) and F_Q surveillance methodologies. These methodologies are used to reduce operator action required to maintain conformance with power distribution control Technical Specifications and to increase the ability to return to power after a plant trip while still maintaining margin to safety limits under all operating conditions.

The changes proposed to Technical Specifications (TS) 3.2.1, AXIAL FLUX DIFFERENCE (AFD), and 3.2.2, HEAT FLUX HOT CHANNEL FACTOR – F_Q (Z), are being made to adopt the RAOC calculational procedure of the Standard Technical Specifications (STS), i.e., NUREG-1431, "Standard Westinghouse Technical Specifications Westinghouse Plants" (Reference 1). Changes to the other TS listed in the enclosed table are made to provide consistency with the changes made to TS 3.2.1 and 3.2.2.

The adoption of the RAOC and F_Q surveillance methodologies are supported by WCAP-10216-P-A, Revision 1A (Proprietary), Relaxation of Constant Axial Offset Control F_Q - Surveillance Technical Specification (Reference 2).

2.0 PROPOSED CHANGES

The specific changes to the TS that are proposed are shown in Attachments A-1 and A-2 for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2, respectively. Changes to the TS Bases are shown in Attachments B-1 and B-2, respectively. The changes proposed to the Licensing Requirements Manual (LRM) are provided in Attachments C-1 and C-2 for Unit Nos. 1 and 2, respectively.

The proposed Technical Specification Bases and LRM changes do not require NRC approval. The BVPS Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. The BVPS Licensing Document Control Program controls the review, approval and implementation of LRM changes. Changes to these two documents are controlled by the 10 CFR 50.59

process. The Technical Specification Bases and LRM changes are provided for information only.

The proposed changes to the Technical Specifications, TS Bases and LRM 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 appropriate Indices, the Technical Specifications, the TS Bases and the LRM pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

Changes to the following Technical Specifications are being proposed in this LAR.

Affected Technical Specifications			
No.	**Unit 1**	**Unit 2**	**Title**
1	3.2.1	3.2.1	AXIAL FLUX DIFFERENCE (AFD)
2	3.2.2	3.2.2	HEAT FLUX HOT CHANNEL FACTOR – F_Q (Z)
3	3.2.3	3.2.3	NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$
4	3.2.4	3.2.4	QUADRANT POWER TILT RATIO (QPTR)
5	3.3.1	3.3.1	REACTOR TRIP SYSTEM INSTRUMENTATION (Table 4.3-1, Note 3)
6	6.9.5	6.9.5	CORE OPERATING LIMITS REPORT (COLR)

All of the proposed TS and TS Bases changes are consistent with the STS (Reference 1).

2.1 Proposed TS Changes

Change Number 1

This proposed change is a re-write of Technical Specification 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)". The Limiting Condition for Operation (LCO), Actions and Surveillance Requirements are revised to be consistent with the STS RAOC version of this TS. Surveillance

Requirement 4.2.1.2 is revised to be consistent with the RAOC methodology and is included as a footnote to the LCO statement. Surveillance Requirements 4.2.1.3 and 4.2.1.4 are deleted.

Basis for Change Number 1

The proposed changes are being made to be consistent with the RAOC methodology and the STS (Reference 1).

The current Technical Specification 3.2.1, per the core operating limits report (COLR), specifies a target band of +7%, -7% for normal operation from 0% to 100% Rated Thermal Power (RTP). The RAOC methodology allows an Axial Flux Difference (AFD) operating space relaxation to +10%, -15% ΔI at 100% RTP and linearly increasing to +24%, -32% ΔI at 50% RTP. Penalty minutes are not applicable to the RAOC methodology and therefore reference to them is no longer necessary in the TS or TS Bases. The necessary LRM changes are discussed in Section 2.2.

For the RAOC methodology, the value of the AFD does not affect the limiting accident consequences with thermal power less than 50% RTP. Reducing the power range neutron flux high setpoints is not necessary to provide an adequate level of protection. Reducing the power level to less than or equal to 50% RTP maintains the plant in a benign condition since under RAOC methodology there are no AFD limits below 50% RTP. In addition, a rapid rise in power to greater than 50% RTP, with AFD outside limits, does not immediately create an unacceptable situation. Since the transient analysis setpoint calculations for $f(\Delta I)$, input to the Overtemperature ΔT Trip Function, are based on the same core power distributions used in a reload cycle design, the Overtemperature ΔT Trip Function provides an acceptable level of protection for such an excursion. It is also noted that the event would be successfully terminated by a trip at the previous setpoint level.

Surveillance Requirement 4.2.1.1 has been revised to be consistent with the STS. The first sentence of Surveillance Requirement has been moved to Note 1, which is applicable to the LCO statement. The remaining portion of this surveillance can be deleted because penalty minutes are not applicable to RAOC. The Note on the Applicability has been designated with a number instead of an asterisk for clarity. This is an editorial change that requires no further justification.

Surveillance Requirements 4.2.1.3 and 4.2.1.4 are deleted because they pertain to the target flux difference, which is not applicable to RAOC. With

RAOC there is no target band. In addition, the activity required by Surveillance Requirement 4.2.1.4 is addressed by Notation (3) of Table 4.3-1.

Change Number 2

This proposed change is a re-write of Technical Specification 3.2.2, "HEAT FLUX HOT CHANNEL FACTOR – F_Q (Z)". The LCO is revised to indicate that two limits exist, i.e., $F^C_Q(Z)$ and $F^W_Q(Z)$. The Actions and Surveillance Requirements have also been revised to reflect the $F_Q(Z)$ surveillance methodology. Several Notes have also been added to specify Action and Surveillance Requirement conditions.

Basis for Change Number 2

The proposed changes are being made to be consistent with the RAOC methodology and the STS.

Action "a" of TS 3.2.2 is revised to change the time allowed to reduce the Power Range Neutron Flux-High Trip from 4 hours to 72 hours. As written, the completion time of 4 hours to reduce the Power Range Neutron Flux-High Trip setpoints presents an unjustified burden on the operation of the plant. A completion time of 72 hours will allow time to perform a second flux map to confirm the results, or determine that the condition was temporary, without implementing an unnecessary trip setpoint change, during which there is increased potential for a plant transient and human error. Following a significant power reduction, at least 24 hours are required to re-establish steady state xenon prior to taking a flux map, plus additional time to obtain a flux map and analyze the data. A significant potential for human error can be created through requiring the trip setpoints to be reduced within the same time frame that a unit power reduction is taking place within the current completion time of 4 hours. To account for setpoint adjustments and any necessary initial preparation, a completion time of 72 hours is proposed. The completion time extension is acceptable because Action a.1 requires a reduction in power well before the proposed 72 hours and it provides enough time to safely take the necessary steps to determine if a setpoint change is indeed required.

Also, "after each $F_Q(Z)$ determination" has been added to Action "a" to define the frequency of the action requirement. The proposed change will require the Thermal Power, the Power Range Neutron Flux-High Trip Setpoints, and the Overpower ΔT Trip Setpoints reductions be repeated after each subsequent $F_Q(Z)$ determination if $F_Q(Z)$ is not within limit. This will ensure that Actions are continued until the parameter is within its limit at the current power level.

In Action "a" the requirement to perform the Overtemperature ΔT Trip Setpoint reduction with the reactor subcritical is being deleted. This change is made to provide consistency with Reference 1.

Action "b", which requires the out of limit condition of Action "a" to be identified and corrected prior to increasing THERMAL POWER, is being replaced by an action that requires the verification that $F^C_Q(Z)$ and $F^W_Q(Z)$ are within limits prior to increasing THERMAL POWER above the limits of Action "a". These two actions ensure that core conditions during operation at higher power levels and future operations are consistent with safety analyses assumptions.

Actions a.4 and b.4 have been added to TS 3.2.2. These actions are applicable to the measurement of the peak fuel pellet power within the reactor core at the steady state power, which is known as $F^C_Q(Z)$, and the measurement of the peak fuel pellet within the reactor core that is adjusted for power distribution transients encountered during normal operation, which is known as $F^W_Q(Z)$.

Actions a.5 and b.5 have been added to TS 3.2.2 to define the alternative for not meeting Actions "a" or "b". These Actions are more restrictive than the Actions provided in the current BVPS TS. These Actions require that the unit be placed in Mode 2 within 6 hours if THERMAL POWER is not reduced to comply with the Action.

Footnotes (1) and (2) have been added to Actions a and b respectively, to require that verification of $F^C_Q(Z)$ and $F^W_Q(Z)$ shall be completed whenever Actions a or b are entered.

Surveillance Requirement 4.2.2.2 and 4.2.2.3 have been revised to incorporate the $F_Q(Z)$ surveillance strategy to determine $F_Q(Z)$ is within its limits. These revised Surveillance Requirements are also required for $F^C_Q(Z)$ and $F^W_Q(Z)$ any time that they exceed their limits prior to increasing power as described in their respective Actions.

Footnote (3), added to each of these Surveillance Requirements, allows power escalation and delays obtaining a power distribution map at the beginning of a cycle until an equilibrium power level is reached.

Footnote (4) is added to Surveillance Requirement 4.2.2.3 to indicate that additional actions are required when the maximum over z of $[F^C_Q(Z)/K(Z)]$

increases over the previous evaluation of $F^C_Q(Z)$. This footnote ensures that future operations are consistent with safety analyses assumptions.

The Action and Surveillance Requirement Notes have been designated with numbers instead of asterisks for clarity. Replacing the asterisks with numbers are editorial changes that require no further justification.

Change Number 3

This proposed change consists of two modifications of Technical Specification 3.2.3, NUCLEAR ENTHALPY HOT CHANNEL FACTOR. The first is to change the nomenclature used for CFDH and PFDH. The second is to insert "RISE" between "ENTHALPY" and "HOT" in the TS title.

Basis for Change Number 3

The proposed changes are being made to make the TS nomenclature consistent with the COLR nomenclature for CFDH and PFDH. The COLR nomenclature is $CF_{\Delta H}$ for CFDH and $PF_{\Delta H}$ for PFDH. The TS title change, which is carried through the TS Index and Bases, results in consistency with the STS and COLR. These changes are editorial and require no further justification.

Change Number 4

This proposed change is a modification of Technical Specification 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)", to reflect the $F^C_Q(Z)$ and $F^W_Q(Z)$ surveillance requirements.

Basis for Change Number 4

The proposed change is being made to be consistent with the RAOC methodology and the STS.

Change Number 5

This proposed change is a modification of the Notation to Table 4.3-1 of Technical Specification 3.3.1, "REACTOR TRIP SYSTEM INSTRUMENTATION". Notation 3 of Table 4.3-1 is modified by changing "15%" to "50%" of RATED THERMAL POWER.

Basis for Change Number 5

The proposed change is being made to be consistent with the change to Surveillance Requirement 4.2.1.1.

Change Number 6

This proposed change is a modification of Technical Specification 6.9.5, "CORE OPERATING LIMITS REPORT (COLR)".

The change consists of revising TS 6.9.5.a by replacing "Constant" with "Relaxed" for consistency with the F_Q surveillance methodology.

In addition TS 6.9.5.b is revised:

(a) by adding WCAP-10216-P-A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification (Reference 2) to the list of references, and

(b) by deleting WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", the T. M. Anderson to K. Kniel letter, and NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, from the list of references.

Basis for Change Number 6

The proposed change is being made to be consistent with the RAOC methodology and the STS. A reference to WCAP-10216-P-A is being added because it is applicable to the RAOC methodology. The references to WCAP-8385, the T. M. Anderson to K. Kniel letter, and NUREG-0800 are being deleted because they are applicable to the Constant Axial Offset Control (CAOC) methodology, not the RAOC methodology.

2.2 Proposed TS Bases and LRM Changes

Since TS 3.2.1 and 3.2.2 are being revised to reflect the STS, it is prudent to also expand the TS Bases to what is contained in the STS Bases. The existing BVPS TS Bases have a single discussion for TS 3.2.2 and TS 3.2.3. By adopting the expanded STS Bases, it became necessary to provide a separate TS Bases for TS 3.2.2 and TS 3.2.3. The expanded TS Bases are provided for information only in Attachments B-1 and B-2.

The LRM contains the COLR, which will be modified to reflect the changes proposed in this LAR. These changes consist of revising COLR Specifications 3.2.1 and 3.2.2 to reflect the RAOC methodology. The other changes include revising COLR Figure 4.1-2 and deleting COLR Figure 4.1-4 to be consistent with the RAOC and F_Q surveillance methodologies, and the addition of tables for the $F_Q(Z)$ penalty factor and the W(Z) values.

The COLR also contains various values and figures that are cycle specific. These values and figures are appropriately identified and labeled. The specific operating values for AFD and $F_Q(Z)$ provided in the COLR will be evaluated as part of the reload process using the WCAP-9272-P-A (Reference 3) methodology and RAOC and $F_Q(Z)$ methodology analyses to verify these parameters. The COLR will be revised during the cycle-specific reload core design analysis.

Licensing Requirement 3.4, Axial Flux Difference (AFD) Monitor Alarm, and the Bases for Licensing Requirement 3.8, Leading Edge Flow Meter, are being modified in each unit's LRM. The change is made to maintain consistency with the change to TS 3.2.1. The power level referenced in Licensing Requirement Surveillance 3.4.1 and Licensing Requirement Bases B.3.8 is changed from 15% RTP to 50% RTP to be consistent with the Applicability of TS 3.2.1 and the Action. The statements pertaining to penalty minutes and target band are deleted from Licensing Requirement 3.4 and its Bases, because penalty minutes and target band are not applicable to RAOC. The requirement to monitor a restored channel for 24 hours is deleted because it relates to penalty minutes. All of these changes are acceptable because they are consistent with the proposed changes to TS 3.2.1.

As previously mentioned, the TS Bases and LRM are provided for information only and can be changed by the 10 CFR 50.59 process. As such no further discussion of the proposed TS Bases and LRM changes is necessary.

3.0 BACKGROUND

Axial power distribution control at BVPS is currently achieved by the Constant Axial Offset Control (CAOC) methodology. This methodology was developed and described in WCAP-8385 and WCAP-8403 (Reference 4). This strategy assures peaking factors and departure from nucleate boiling (DNB) remain below the accident analysis limits. The CAOC methodology developed in Reference 4 does this by maintaining the axial power distribution within a band of +7%, -7% ΔI, for the BVPS units around a measured target value during normal plant operation, including power changes. By controlling the axial power distribution, the possible skewing of the axial xenon distribution is limited, thus minimizing xenon oscillations and their effects on the power distribution.

Axial Flux Difference (AFD) is a measure of axial power distribution skewing to the top or bottom half of the core. It is very sensitive to core-related parameters such as control bank position, core power level, axial burnup, and axial xenon distribution. The limits on AFD assure that the Heat Flux Hot Channel Factor $F_Q(Z)$ is not exceeded during either normal operation, or in the event of xenon redistribution following power changes. The AFD limits are used in the nuclear design process and assumed in the safety analyses as a boundary of possible initial condition axial power shapes. Operation outside these AFD limits during Condition I operation influences the possible power shapes and could result in violations of the kw/ft limit during Condition II transients. Condition II transients, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT Trip Setpoints.

The CAOC methodology is presently incorporated into Technical Specification 3.2.1, Axial Flux Difference. The F_{xy} methodology is presently incorporated into Technical Specification 3.2.2 Heat Flux Hot Channel Factor $F_Q(Z)$. Technical Specification 3.2.4, Quadrant Power Tilt Ratio (QPTR) refers to the $F_Q(Z)$ surveillance requirement. Application of the RAOC and F_Q surveillance methodologies requires the alteration of these Technical Specifications. A change to Technical Specification 6.9.5, CORE OPERATING LIMITS REPORT (COLR), is also required to provide the methodology change. In order to provide consistency and to avoid duplicate requirements between the power distribution limits Technical Specifications and the reactor trip system instrumentation Technical Specification, Note 3 of Table 4.3-1 of the TS 3.3.1 also requires modification.

4.0 TECHNICAL ANALYSIS

The implementation of RAOC and F_Q surveillance methodologies have been previously developed and approved by the NRC in WCAP-10216-P-A Rev. 1A (Reference 2). The RAOC strategy was developed to provide wider control bandwidths and more operator freedom than with CAOC. The RAOC methodology provides wider control bands particularly at reduced power by utilizing core margin more effectively. This change provides more operational flexibility in terms of axial power distributions, particularly during power transients such as a return to full power following a power reduction or reactor trip. The wider operating band increases plant availability by permitting increased maneuvering flexibility without a reactor trip or reportable occurrences. The F_Q surveillance allows for a more direct surveillance of the elevation-dependent heat flux hot channel factor and provides margin compared to F_{xy} surveillance.

The overall objective of power distribution limits is to provide assurance of fuel integrity during Condition I (Normal Operation) and Condition II (Incidents of Moderate Frequency) events by:

(a) maintaining the minimum departure from nucleate boiling ratio (DNBR) in the core greater than or equal to 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 loss of coolant accident (LOCA) analyses are met and that the emergency core cooling system (ECCS) acceptance criteria limit of 2200°F is not exceeded.

The limits on Axial Flux Difference in a RAOC strategy 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.

The limits on heat flux hot channel factor ensure that:

(a) the design limits on peak local power density and minimum DNBR are not exceeded and

(b) in the event of a LOCA the peak fuel cladding temperature will not exceed the ECCS acceptance criteria limit of 2200°F.

The heat flux hot channel factor is measurable but will normally only be determined periodically as specified in the Technical Specifications 3.2.2 and 3.2.3. This periodic surveillance is sufficient to ensure 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 Technical Specification 3.1.3.6, "CONTROL ROD INSERTION LIMITS".

(c) The rod insertion limits of Specifications 3.1.3.5, "SHUTDOWN ROD INSERTION LIMIT" and 3.1.3.6 are maintained.

(d) The axial power distribution, expressed in terms of Axial Flux Difference, is maintained within the limits.

When an F_Q measurement is taken, both measurement uncertainty and manufacturing tolerance must be considered. Five percent is the appropriate measurement uncertainty allowance for a full core map taken with the incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

With F_Q surveillance, the heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$, is met. The power factor, W(Z), accounts for the effects of normal operation transients within the AFD band and is determined from expected power control maneuvers over several ranges of burnup conditions in the core.

An evaluation of the potential impact of the RAOC and $F_Q(Z)$ surveillance methodology changes on safety analyses was performed which included:

- Non-LOCA Events
- LOCA and LOCA-Related Events
- Core Design

4.1 Non-LOCA Related Evaluation

The effect on the non-LOCA events for a change from CAOC to RAOC is to increase the number of power shapes that must be considered when developing the Overtemperature ΔT and Overpower ΔT setpoint equations. The Overtemperature ΔT setpoint is designed to ensure plant operation within the DNB design basis and hot-leg boiling limit. The Overtemperature ΔT $f(\Delta I)$ function is designed to ensure DNB protection from adverse axial power shapes. The Overpower ΔT trip function is designed to ensure plant operation within the fuel temperature design basis and its required setpoint reduction to maintain $F_Q(Z)$ within limits is not impacted by the change from CAOC to RAOC.

The $f(\Delta I)$ function is generated based on the expected axial power shapes from the various Condition I and II events. Because RAOC allows for more severe power shapes to be generated, it was necessary to move the negative wing of the Overtemperature ΔT $f(\Delta I)$ penalty to eliminate shapes that may violate the DNB criteria. This will have no effect on the Updated Final Safety Analysis Report (UFSAR) transient safety analyses because they do not model the $f(\Delta I)$ term in the Overtemperature ΔT setpoint equation. The $f(\Delta I)$ term accounts for the axial power shape effects on the DNB criteria and independently lowers the Overtemperature ΔT setpoint to ensure a conservative reactor trip. It is concluded that the implementation of RAOC does not adversely affect the results of the non-LOCA analyses and the conclusions made in the UFSAR remain valid.

4.2 LOCA and LOCA-Related Evaluations

The change from CAOC and F_{xy} surveillance to the RAOC and $F_Q(Z)$ surveillance methodologies has been evaluated for impact upon the existing LOCA safety analyses and in support of the planned extended power uprate (EPU) program. The LOCA and LOCA-related accident analysis remain valid for the methodology implementation given the above parameter changes and their effect on the safety analysis limits.

The RAOC and $F_Q(Z)$ surveillance methodologies do not affect the normal plant operating parameters, the safeguards systems actuation, the accident mitigation capabilities important to a LOCA, the assumptions used in the LOCA-related accidents, or create conditions more limiting than those assumed in these analyses.

The main impact of RAOC implementation on the EPU LOCA analyses is the increased range of permissible axial power distributions prior to an event. The impacts have been evaluated, and a peak cladding temperature (PCT) penalty of 16 °F has been established for the Unit No. 1 large break LOCA limiting time period. The limiting PCT time period for the Unit No. 2 large break LOCA was determined to not require a PCT penalty. Margin to the 2200°F PCT limit remains for both units.

No core design inputs to the EPU small break LOCA analyses changed due to RAOC operation. Thus, these analyses remain unaffected by RAOC. The small break LOCA analysis is not dependent on the specific axial power distributions associated with the change to RAOC.

4.3 Core Design Evaluation

The change from CAOC and F_{xy} surveillance to the RAOC and $F_Q(Z)$ surveillance methodologies have been evaluated for impact upon the BVPS core design. Consistent with the approved RAOC methodology, the

Condition I axial power shapes were analyzed to demonstrate compliance with the LOCA F_Q limit. The normal operation axial power shapes were also evaluated relative to the assumed limiting normal operation axial power shape in the analysis of the DNB-limited events which are not terminated by the Overtemperature ΔT Reactor Trip, e.g. the loss of reactor coolant system flow accident. The Condition II RAOC shapes were analyzed to demonstrate that the fuel melting design criterion was met. In addition, the Condition II axial power distributions were evaluated relative to the axial power distribution assumptions used to generate the DNB core limits. Changes to the axial offset limits and core limits from the extended power uprate (EPU) analyses were made based on these evaluations. The negative wing of the Overtemperature ΔT Trip Setpoints f(ΔI) function from the EPU analyses was revised based on the limiting Condition II axial power distributions such that the DNB design criterion is met for accidents which are terminated by Overtemperature ΔT Reactor Trips.

The axial power shapes generated by RAOC were also evaluated in terms of their impact on fuel rod performance. The transient local power increases experienced by the fuel operating within the RAOC ΔI bands were considered in evaluating the rod internal pressure of the fuel rods and the cladding transient stress and transient strain. Westinghouse demonstrated that all fuel performance limits are capable of being met under RAOC operation. Compliance with the safety analysis assumptions will be performed on a cycle-specific basis during core design analysis.

The use of RAOC and F_Q surveillance therefore successfully provides additional operational flexibility to BVPS while still meeting all corresponding core design bases and limits.

4.4 Other Areas

A review of the areas listed below has been performed for this evaluation, and it has been determined that they are unaffected by the RAOC and $F_Q(Z)$ surveillance methodologies changes.

(a) Emergency Operating Procedures

(b) Instrumentation and Control Systems

(c) Radiological Analyses

(d) Mechanical and Fluid Systems

(e) Plant Operability with respect to operating margin

4.5 Conclusion

The technical analysis demonstrates that the implementation of the RAOC and $F_Q(Z)$ surveillance methodologies does not affect the normal plant operating parameters, protection system actuation, the safeguard system actuation, or any other plant capability important to the mitigation of a Non-LOCA or LOCA accident.

5.0 REGULATORY SAFETY ANALYSIS

This License Amendment Request (LAR) requests approval to implement the Relaxed Axial Offset Control (RAOC) and F_Q surveillance methodologies for the two Beaver Valley Power Station (BVPS) units. These methodologies are used to reduce operator action required to maintain conformance with power distribution control Technical Specifications and to increase the ability to return to power after a plant trip while still maintaining margin to safety limits under all operating conditions.

The Constant Axial Offset Control (CAOC) methodology is presently incorporated into BVPS Technical Specification 3.2.1, Axial Flux Difference. The F_{xy} methodology is presently incorporated into Technical Specification 3.2.2 Heat Flux Hot Channel Factor $F_Q(Z)$. Technical Specification 3.2.4, Quadrant Power Tilt Ratio (QPTR) refers to the $F_Q(Z)$ surveillance requirement. Application of the Relaxed Axial Offset Control (RAOC) and F_Q surveillance methodologies requires the alteration of these Technical Specifications. Changes to Technical Specification 6.9.5, CORE OPERATING LIMITS REPORT (COLR), is also required to provide the methodology change. In order to provide consistency and to avoid duplicate requirements between the power distribution limits Technical Specifications and the reactor trip system instrumentation Technical Specification, Table 4.3-1 of TS 3.3.1 also requires modification. The proposed Technical Specification changes are consistent with NUREG-1431, "Standard Westinghouse Technical Specifications Westinghouse Plants," Revision 3.

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?

Response: No. The proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not initiate an accident. Evaluations and analyses of accidents, which are potentially affected by the parameters and assumptions, associated with the RAOC and $F_Q(Z)$ methodologies have shown that all design standards and applicable safety criteria will continue to be met. The consideration of these changes does not result in a situation where the design, material, or construction standards that were applicable prior to the change are altered. Therefore, the proposed changes will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident.

The proposed changes associated with the RAOC and $F_Q(Z)$ methodologies do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The actual plant configuration, performance of systems, or initiating event mechanisms are not being changed as a result of the proposed changes. The design standards and applicable safety criteria limits will continue to be met, therefore, fission barrier integrity is not challenged. The proposed changes associated with the RAOC and $F_Q(Z)$ methodologies have been shown not to adversely affect the plant response to postulated accident scenarios. The proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

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 will not create the possibility of a new or different kind of accident from any accident previously evaluated.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed changes do not challenge the performance or integrity of any safety-related system. The possibility for a new or different type of accident from any accident previously evaluated is not created since the proposed changes do not result in a change to the design basis of any plant structure, system or component. Evaluation of the effects of the proposed changes has shown that all design standards and applicable safety criteria continue to be met.

Equipment important to safety will continue to operate as designed and component integrity will not be challenged. The proposed changes do not result in any event previously deemed incredible being made credible. The proposed changes will not result in conditions that are more adverse and will not result in any increase in the challenges to safety systems.

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

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

Response: No. The proposed changes will not involve a significant reduction in a margin of safety.

The proposed changes will assure continued compliance within the acceptance limits previously reviewed and approved by the NRC for RAOC and $F_Q(Z)$ methodologies. All of the appropriate acceptance criteria for the various analyses and evaluations will continue to be met.

The impact associated with the implementation of RAOC on peak cladding temperature (PCT) has been evaluated for the planned extended power uprate. This evaluation has determined that implementation of RAOC at the extended power uprate power level will not result in a significant reduction in a margin of safety for either unit.

Therefore, the proposed changes do 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

A review of 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants (Reference 5) was conducted to assess the potential impact associated with the proposed changes. Although some UFSAR description of conformance may require a modification, in no case is an exception to any General Design Criterion (GDC) required.

5.2.1 Discussion of Impact

The following provides a brief description of GDC 10 and a discussion of the impact on the applicable UFSAR discussion.

<u>GDC 10 Reactor Design</u>

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

<u>UFSAR Discussion/Impact</u>

The UFSAR contains analyses of accidents for the Axial Flux Difference parameter. The most important (limiting) Condition 2 events are the uncontrolled bank withdrawal, cooldown and boration/dilution accidents. The most important (limiting) Condition 3 and 4 events are the loss of flow accident and LOCA, respectively. This is the case for both units. Calculation of extreme power shapes that affect fuel design limits is performed with approved methods and verified frequently with measurements from the reactors. The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state. To ensure that the axial profile meets the linear heat rate limit and the departure from nucleate boiling (DNB) limit, excore detector signals are used to provide a top to bottom flux difference which is input, through the f(ΔI), into the Overtemperature ΔT Trip Setpoint. Nuclear uncertainty margin is applied to calculated peak local power. Such margin is provided for the analysis of normal operating states and for anticipated transients.

This compliance with GDC 10 is not adversely impacted by the proposed changes.

5.2.2 Conclusions

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

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. NUREG-1431, "Standard Westinghouse Technical Specifications Westinghouse Plants", Revision 3, June 2004.
2. WCAP-10216-P-A, Revision 1A (Proprietary), Relaxation of Constant Axial Offset Control F_Q - Surveillance Technical Specification, February 1994.
3. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).
4. WCAP-8385 (Proprietary) and WCAP-8403 (Non-proprietary), "Topical Report Power Distribution Control And Load Follow Procedures", September 1974.
5. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants".

Attachment A-1

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

License Amendment Request No. 310

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* No change made to this page. Included for information and readability only.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The ~~indicated AXIAL FLUX DIFFERENCE (AFD)~~ in % flux difference units shall be maintained within the ~~target band~~limits[1] specified in the ~~CORE OPERATING LIMITS REPORT (COLR)~~.

APPLICABILITY: MODE 1 ~~ABOVE~~ with THERMAL POWER ≥ 50% ~~RATED THERMAL POWER~~RTP~~*~~[2].

ACTION:

~~a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band and with THERMAL POWER:~~

~~1. Above 90% of RATED THERMAL POWER, within 15 minutes:~~

~~a) Either restore the indicated AFD to within the target band limits, or~~

~~b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.~~

~~2. Between 50% and 90% of RATED THERMAL POWER:~~

~~a) POWER OPERATION may continue provided:~~

~~1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and~~

~~2) The indicated AFD is within the acceptable operation limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.~~

~~b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.~~

With AFD not within limits, reduce THERMAL POWER to < 50% of RTP within 30 minutes.

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(1) The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

~~*~~(2) See Special Test Exception 3.10.2.

~~LIMITING CONDITION FOR OPERATION (Continued)~~

~~b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION a.2.a) 1), above has been satisfied.~~

~~c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.~~

SURVEILLANCE REQUIREMENTS

4.2.1.1 Verify AFD ~~The indicated AXIAL FLUX DIFFERENCE shall be determined to be~~ within ~~its~~ limits ~~during POWER OPERATION above 15% of RATED THERMAL POWER~~ for each OPERABLE excore channel at least once per 7 days.

~~4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. POWER OPERATION outside of the target band shall be accumulated on a time basis of:~~

~~a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and~~

~~b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.~~

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~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.~~

~~4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and the design end of cycle value. The provisions of Specification 4.0.4 are not applicable.~~

~~(Next page is 3/4 2-5)~~

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$~~,~~ as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$~~,~~ shall be ~~limited by the following relationships:~~ within the limits specified in the COLR.

$$\cancel{F_Q(Z) \le \frac{[CFQ]}{P}[K(Z)] \text{ for } P > 0.5}$$

$$\cancel{F_Q(Z) \le \frac{[CFQ]}{0.5}[K(Z)] \text{ for } P \le 0.5}$$

~~where: CFQ = The FQ limit at RATED THERMAL POWER provided in the CORE OPERATING LIMITS REPORT,~~

~~K(Z) = The normalized FQ(Z) as a function of core height provided in the CORE OPERATING LIMITS REPORT, and~~

$$\cancel{P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}}$$

APPLICABILITY: Mode 1.

ACTION:

~~With $F_Q(Z)$ exceeding its limit:~~

~~a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor subcritical.~~

~~b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.~~

a. With $F_Q^C(Z)$ not within limit[1]:

1. Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds the limit within 15 minutes after each $F_Q^C(Z)$ determination; and

2. Reduce the Power Range Neutron Flux-High Trip Setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds the limit within 72 hours after each $F_Q^C(Z)$ determination; and

3. Reduce the Overpower ΔT Trip Setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds the limit within 72 hours after each $F_Q^C(Z)$ determination; and

4. Perform Surveillance Requirements 4.2.2.2 and 4.2.2.3 prior to increasing THERMAL POWER above the limit of Action a.1.

5. Otherwise, be in MODE 2 within the following 6 hours.

b. With $F_Q^W(Z)$ not within limits(2):

1. Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit within 4 hours; and

2. Reduce the Power Range Neutron Flux-High Trip Setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced within 72 hours; and

3. Reduce the Overpower ΔT Trip Setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced within 72 hours; and

4. Perform Surveillance Requirements 4.2.2.2 and 4.2.2.3 prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits.

5. Otherwise, be in MODE 2 within the following 6 hours.

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(1) Action a.4 shall be completed whenever Action a is entered.

(2) Action b.4 shall be completed whenever Action b is entered.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

~~4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:~~

~~a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~

~~b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.~~

~~c. Comparing the F_{xy} computed F_{xy}^C obtained in b, above to:~~

~~1. The F_{xy} limits for RATED THERMAL POWER F_{xy}^{RTP} for the appropriate measured core planes given in e and f below, and~~

~~2. The relationship:~~

$$\cancel{F_{xy}^L = F_{xy}^{RTP} [1+PFXY(1-P)]}$$

~~where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PFXY is the Power Factor multiplier for F_{xy} provided in the CORE OPERATING LIMITS REPORT, and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.~~

~~d. Remeasuring F_{xy} according to the following schedule:~~

~~1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:~~

~~a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or~~

~~b) At least once per 31 EFPD, whichever occurs first.~~

SURVEILLANCE REQUIREMENTS (continued)

~~2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.~~

~~e. The F_{xy} limit for Rated Thermal Power F_{xy}^{RTP} shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the CORE OPERATING LIMITS REPORT.~~

~~f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured from the bottom of the fuel:~~

~~1. Lower core region from 0 to 15%, inclusive.~~

~~2. Upper core region from 85 to 100%, inclusive.~~

~~3. Grid plane regions ±2% of core height (±2.88 inches) measured from grid centerline.~~

~~4. Core plane regions within ±2% of core height (±2.88 inches) about the bank demand position of the bank "D" control rods.~~

~~g. With F_{xy}^{C} exceeding F_{xy}^{L} the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.~~

~~4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.~~

4.2.2.2 $F_Q^C(Z)$ shall be verified to be within the limit according to the following schedule(3):

a. Once after each refueling prior to THERMAL POWER exceeding 75% RTP; and

b. Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified; and

c. At least once per 31 Effective Full Power Days thereafter.

4.2.2.3 $F_Q^W(Z)$ shall be verified to be within the limit[4] according to the following schedule[3]:

a. Once after each refueling prior to THERMAL POWER exceeding 75% RTP; and

b. Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified; and

c. At least once per 31 Effective Full Power Days, thereafter.

---

(3) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

(4) If measurements indicate that the maximum over z of $[F_Q^C(Z)/K(Z)]$ has increased since the previous evaluation $F_Q^C(Z)$:

a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits, or

b. Repeat Surveillance Requirement 4.2.2.3 once per 7 Effective Full Power Days until Note (4)a above is met or two successive flux maps indicate that the maximum over z of $[F_Q^C(Z)/K(Z)]$ has not increased.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq CF_{\Delta DH} [1 + PF_{\Delta DH} (1-P)]$$

where: $CF_{\Delta DH}$ = $F^N_{\Delta H}$ limit at RATED THERMAL POWER provided in the CORE OPERATING LIMITS REPORT,

$PF_{\Delta DH}$ = The Power Factor multiplier for $F^N_{\Delta H}$ provided in the CORE OPERATING LIMITS REPORT, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limits:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER, subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL power, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

No change proposed. Included for information only.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using moveable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO (QPTR)

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall be less than or equal to 1.02.

APPLICABILITY: MODE 1 greater than 50 percent of RATED THERMAL POWER.[1]

ACTION: With the QPTR not within the limit:

a. Within 2 hours, reduce THERMAL POWER greater than or equal to 3 percent from RATED THERMAL POWER (RTP) for each 1 percent of QPTR greater than 1.00, and

b. Within 12 hours and once per 12 hours thereafter, perform Surveillance Requirement 4.2.4 and reduce THERMAL POWER greater than or equal to 3 percent from RTP for each 1 percent of QPTR greater than 1.00, and

c. Within 24 hours and once per 7 days thereafter, perform Surveillance Requirements 4.2.2.2, 4.2.2.3, and 4.2.3.1, and

d. Prior to increasing THERMAL POWER above the limit of ACTION a or b above, re-evaluate the safety analyses and confirm the results remain valid for the duration of operation under this condition, and

e. After ACTION d above is completed and prior to increasing THERMAL POWER above the limit of ACTION a or b above, normalize the excore detectors to show a QPTR less than or equal to 1.02, and

f. After ACTION e above is completed and within 24 hours after reaching RTP or within 48 hours after increasing THERMAL POWER above the limit of ACTION a or b above, perform Surveillance Requirements 4.2.2.2, 4.2.2.3, and 4.2.3.1.

g. Otherwise, reduce THERMAL POWER to less than or equal to 50 percent RTP within 4 hours.

---

(1) See Special Test Exception 3.10.2.

No change proposed. Included for information only.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
1.	Manual Reactor Trip	N.A.	N.A.	S/U[1], R[10]	N.A.
2.	Power Range, Neutron Flux				
	a. High Setpoint	S	D[2], M[3] and Q[6]	Q	1, 2
	b. Low Setpoint	S	R[6]	S/U[1]	2
3.	Power Range, Neutron Flux, High Positive Rate	N.A.	R[6]	Q	1, 2
4.	Power Range, Neutron Flux, High Negative Rate	N.A.	R[6]	Q	1, 2
5.	Intermediate Range, Neutron Flux	S	R[6]	S/U[1]	1, 2, 3[14], 4[14], 5[14]
6.	Source Range[15], Neutron Flux				
	a. With Rod Withdrawal Capability	S	R[6]	Q[8]	2, 3[14], 4[14] and 5[14]
	b. With All Rods Fully Inserted and Without Rod Withdrawal Capability	S	R[6]	Q[8]	3, 4 and 5
7.	Overtemperature ΔT	S	R[6]	Q	1, 2
8.	Overpower ΔT	S	R	Q	1, 2
9.	Pressurizer Pressure-Low	S	R	Q	1, 2
10.	Pressurizer Pressure-High	S	R	Q	1, 2
11.	Pressurizer Water Level-High	S	R	Q	1, 2

No change proposed. Included for information only.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
12.	Loss of Flow - Single Loop	S	R	Q	1
13.	Loss of Flow - Two Loops	S	R	Q	1
14.	Steam/Generator Water Level-Low-Low	S	R	Q	1, 2
15.	DELETED				
16.	Undervoltage-Reactor Coolant Pumps	N.A.	R	M	1
17.	Underfrequency-Reactor Coolant Pumps	N.A.	R	M	1
18.	Turbine Trip				
	a. Auto Stop Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
	b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19.	Safety Injection Input from ESF	N.A.	N.A.	R	1, 2
20.	Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21.	Reactor Trip Breaker	N.A.	N.A.	M(5,11) and S/U(1)	1, 2, 3(14), 4(14), 5(14)

No change proposed. Included for information only.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
22.	Automatic Trip Logic	N.A.	N.A.	$M^{(5)}$	1, 2, $3^{(14)}$, $4^{(14)}$, $5^{(14)}$
23.	Reactor Trip System Interlocks				
	A. P-6	N.A.	$R^{(6)}$	R	1, 2
	B. P-8	N.A.	$R^{(6)}$	R	1
	C. P-9	N.A.	$R^{(6)}$	R	1
	D. P-10	N.A.	$R^{(6)}$	R	1
	E. P-13	N.A.	R	R	1
24.	Reactor Trip Bypass Breakers	N.A.	N.A.	$M^{(12)}$, $R^{(13)}$, $S/U^{(11)}$	1, 2, $3^{(14)}$, $4^{(14)}$, $5^{(14)}$

TABLE 4.3-1 (Continued)

NOTATION

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15 percent of RATED THERMAL POWER.

(3) - At least once every 31 Effective Full Power Days (EFPD) compare incore to excore axial imbalance above ~~15~~50 percent of RATED THERMAL POWER. Recalibrate if absolute difference greater than or equal to 3 percent.

(4) - (Not Used)

(5) - Each train tested every other month.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-10.

(8) - Below P-6, not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 12 hours after entry into MODE 3.

(9) - (Not Used)

(10) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) - Local manual shunt trip prior to placing breaker in service.

(13) - Automatic undervoltage trip.

(14) - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(15) - Surveillance Requirements need not be performed on alternate detectors until connected and required for OPERABILITY.

6.9.3 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

- - - - - - - - - - - - - - - - - NOTE - - - - - - - - - - - - - - - - - -

A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

- -

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I Section IV.B.1.

6.9.4 MONTHLY OPERATING REPORT

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

6.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1 Reactor Core Safety Limits

3.1.3.5 Shutdown Rod Insertion Limits

3.1.3.6 Control Rod Insertion Limits

3.2.1 Axial Flux Difference-~~Constant~~Relaxed Axial Offset Control

3.2.2 Heat Flux Hot Channel Factor-$F_Q(Z)$

3.2.3 Nuclear Enthalpy Rise Hot Channel Factor-$F^N_{\Delta H}$

3.2.5 DNB Parameters

3.3.1.1 Reactor Trip System Instrumentation - Overtemperature and Overpower ΔT Setpoint Parameter Values

Change proposed in LAR 318 provided for information.

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-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February 1994.

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. 182

The following is a list of the affected pages:

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* No change made to this page. Included for information and readability only.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The ~~indicated AXIAL FLUX DIFFERENCE (AFD)~~ in % flux difference units shall be maintained within the ~~target band~~limits[(1)] specified in the ~~CORE OPERATING LIMITS REPORT (COLR)~~.

APPLICABILITY: MODE 1 ~~above~~ with THERMAL POWER ≥ 50% ~~Percent RATED THERMAL POWER~~RTP~~*~~[(2)].

ACTION:

~~a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band and with THERMAL POWER:~~

~~1. Above 90 percent of RATED THERMAL POWER, within 15 minutes:~~

~~a) Either restore the indicated AFD to within the target band limits, or~~

~~b) Reduce THERMAL POWER to less than 90 percent of RATED THERMAL POWER.~~

~~2. Between 50 percent and 90 percent of RATED THERMAL POWER:~~

~~a) POWER OPERATION may continue provided:~~

~~1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and~~

~~2) The indicated AFD is within the acceptable operation limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55 percent of RATED THERMAL POWER within the next 4 hours.~~

~~b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.~~

With AFD not within limits, reduce THERMAL POWER to < 50% of RTP within 30 minutes.

~~*~~(1) The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

~~*~~(2) See Special Test Exception 3.10.2.

~~LIMITING CONDITION FOR OPERATION (Continued)~~

~~ACTION: (Continued)~~

~~b. THERMAL POWER shall not be increased above 90 percent of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION a.2.a)1), above has been satisfied.~~

~~c. THERMAL POWER shall not be increased above 50 percent of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.~~

SURVEILLANCE REQUIREMENTS

4.2.1.1 Verify AFD ~~The indicated AXIAL FLUX DIFFERENCE shall be determined to be~~ within ~~its~~ limits ~~during POWER OPERATION above 15 percent of RATED THERMAL POWER~~ for each OPERABLE excore channel at least once per 7 days.

~~4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. POWER OPERATION outside of the target band shall be accumulated on a time basis of:~~

~~a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50 percent of RATED THERMAL POWER, and~~

~~b. One half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.~~

~~4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.~~

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be ~~limited by the following relationships:~~ within the limits specified in the COLR.

$$\cancel{F_Q(Z) \le \boxed{\cancel{\frac{[CFQ]}{P}}}[K(Z)] \text{ for } P > 0.5}$$

$$\cancel{F_Q(Z) \le \boxed{\cancel{\frac{[CFQ]}{0.5}}}[K(Z)] \text{ for } P \le 0.5}$$

~~where: CFQ = The FQ limit at RATED THERMAL POWER provided in the CORE OPERATING LIMITS REPORT,~~

~~K(Z) = The normalized $F_Q(Z)$ as a function of core height provided in the CORE OPERATING LIMITS REPORT, and~~

$$\cancel{P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}}$$

APPLICABILITY: MODE 1

ACTION:

~~With $F_Q(Z)$ exceeding its limit:~~

~~a. Reduce THERMAL POWER at least 1 percent for each 1 percent $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1 percent for each 1 percent $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor subcritical.~~

~~b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER. THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.~~

a. With $F_Q^C(Z)$ not within limit(1):

1. Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds the limit within 15 minutes after each $F_Q^C(Z)$ determination; and

2. Reduce the Power Range Neutron Flux-High Trip Setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds the limit within 72 hours after each $F_Q^C(Z)$ determination; and

3. Reduce the Overpower ΔT Trip Setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds the limit within 72 hours after each $F_Q^C(Z)$ determination; and

4. Perform Surveillance Requirements 4.2.2.2 and 4.2.2.3 prior to increasing THERMAL POWER above the limit of Action a.1.

5. Otherwise, be in MODE 2 within the following 6 hours.

b. With $F_Q^W(Z)$ not within limits(2):

1. Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit within 4 hours; and

2. Reduce the Power Range Neutron Flux-High Trip Setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced within 72 hours; and

3. Reduce the Overpower ΔT Trip Setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced within 72 hours; and

4. Perform Surveillance Requirements 4.2.2.2 and 4.2.2.3 prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits.

5. Otherwise, be in MODE 2 within the following 6 hours.

---

(1) Action a.4 shall be completed whenever Action a is entered.

(2) Action b.4 shall be completed whenever Action b is entered.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

~~4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:~~

~~a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5 percent of RATED THERMAL POWER.~~

~~b. Increasing the measured F_{xy} component of the power distribution map by 3 percent to account for manufacturing tolerances and further increasing the value by 5 percent to account for measurement uncertainties.~~

~~c. Comparing the F_{xy} computed (F_{xy}^{C}) obtained in b, above to:~~

~~1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and~~

~~2. The relationship:~~

~~$F_{xy}^{L} = F_{xy}^{RTP}$ [1+PFXY(1-P)]~~

~~where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PFXY is the power factor multiplier for F_{xy} provided in the CORE OPERATING LIMITS REPORT, and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.~~

~~d. Remeasuring F_{xy} according to the following schedule:~~

~~1. When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L}~~

~~a) Either within 24 hours after exceeding by 20 percent of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or~~

~~b) At least once per 31 EFPD, whichever occurs first.~~

~~2. When the $F_{xy}{}^{C}$ is less than or equal to the $F_{xy}{}^{RTP}$ limit for the appropriate measured core plane, additional power distribution maps shall be taken and $F_{xy}{}^{C}$ compared to $F_{xy}{}^{RTP}$ and $F_{xy}{}^{L}$ at least once per 31 EFPD.~~

~~e. The F_{xy} limit for Rated Thermal Power ($F_{xy}{}^{RTP}$) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the CORE OPERATING LIMITS REPORT.~~

~~f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:~~

~~1. Lower core region from 0 to 15 percent, inclusive.~~

~~2. Upper core region from 85 to 100 percent inclusive.~~

~~3. Grid plane regions of core height (± 2.88 inches) measured from grid centerline.~~

~~4. Core plane regions within ± 2 percent of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.~~

~~g. With $F_{xy}{}^{C}$ exceeding $F_{xy}{}^{L}$, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.~~

~~4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3 percent to account for manufacturing tolerances and further increased by 5 percent to account for measurement uncertainty.~~

4.2.2.2 $F_Q^C(Z)$ shall be verified to be within the limit according to the following schedule[(3)]:

a. Once after each refueling prior to THERMAL POWER exceeding 75% RTP; and

b. Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified; and

c. At least once per 31 Effective Full Power Days thereafter.

4.2.2.3 $F_Q^W(Z)$ shall be verified to be within the limit[(4)] according to the following schedule[(3)]:

a. Once after each refueling prior to THERMAL POWER exceeding 75% RTP; and

b. Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified; and

c. At least once per 31 Effective Full Power Days, thereafter.

---

(3) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

(4) If measurements indicate that the maximum over z of $[F_Q^C(Z)/K(Z)]$ has increased since the previous evaluation $F_Q^C(Z)$:

a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits, or

b. Repeat Surveillance Requirement 4.2.2.3 once per 7 Effective Full Power Days until Note (4)a above is met or two successive flux maps indicate that the maximum over z of $[F_Q^C(Z)/K(Z)]$ has not increased.

NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq CF_{\Delta DH}\ [1 + PF_{\Delta DH}\ (1-P)]$$

where: $CF_{\Delta DH}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER provided in the CORE OPERATING LIMITS REPORT,

$PF_{\Delta DH}$ = The Power Factor multiplier for $F^N_{\Delta H}$ provided in the CORE OPERATING LIMITS REPORT, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER, subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50 percent of RATED THERMAL POWER prior to exceeding this THERMAL power, at a nominal 75 percent of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95 percent or greater RATED THERMAL POWER.

No change proposed. Included for information only.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75 percent of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

QUADRANT POWER TILT RATIO (QPTR)

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall be less than or equal to 1.02.

APPLICABILITY: MODE 1 greater than 50 percent of RATED THERMAL POWER.(1)

ACTION: With the QPTR not within the limit:

a. Within 2 hours, reduce THERMAL POWER greater than or equal to 3 percent from RATED THERMAL POWER (RTP) for each 1 percent of QPTR greater than 1.00, and

b. Within 12 hours and once per 12 hours thereafter, perform Surveillance Requirement 4.2.4 and reduce THERMAL POWER greater than or equal to 3 percent from RTP for each 1 percent of QPTR greater than 1.00, and

c. Within 24 hours and once per 7 days thereafter, perform Surveillance Requirements 4.2.2.2, 4.2.2.3, and 4.2.3.1, and

d. Prior to increasing THERMAL POWER above the limit of ACTION a or b above, re-evaluate the safety analyses and confirm the results remain valid for the duration of operation under this condition, and

e. After ACTION d above is completed and prior to increasing THERMAL POWER above the limit of ACTION a or b above, normalize the excore detectors to show a QPTR less than or equal to 1.02, and

f. After ACTION e above is completed and within 24 hours after reaching RTP or within 48 hours after increasing THERMAL POWER above the limit of ACTION a or b above, perform Surveillance Requirements 4.2.2.2, 4.2.2.3, and 4.2.3.1.

g. Otherwise, reduce THERMAL POWER to less than or equal to 50 percent RTP within 4 hours.

---

(1) See Special Test Exception 3.10.2.

No change proposed. Included for information only.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
1. Manual Reactor Trip	N.A.	N.A.	S/U[1], R[10]	1, 2, 3[14], 4[14], 5[14]
2. Power Range, Neutron Flux				
a. High Setpoint	S	D[2], M[3] and Q[6]	Q	1, 2
b. Low Setpoint	S	R[6]	S/U[1]	1[7], 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R[6]	Q	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R[6]	Q	1, 2
5. Intermediate Range, Neutron Flux	S	R[6]	S/U[1]	1, 2, 3[14], 4[14], 5[14]
6. Source Range[15], Neutron Flux				
a. With Rod Withdrawal Capability	S	R[6]	Q[8]	2, 3[14], 4[14] and 5[14]
b. With All Rods Inserted and Without Rod Withdrawal Capability	S	R[6]	Q[8]	3, 4 and 5
7. Overtemperature ΔT	S	R[6]	Q	1, 2
8. Overpower ΔT	S	R	Q	1, 2
9. Pressurizer Pressure-Low (Above P-7)	S	R	Q	1, 2
10. Pressurizer Pressure-High	S	R	Q	1, 2
11. Pressurizer Water Level-High (Above P-7)	S	R	Q	1, 2

No change proposed. Included for information only.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
12. Loss of Flow - Single Loop (Above P-8)	S	R	Q	1
13. Loss of Flow - Two Loop (Above P-7 and Below P-8)	S	R	Q	1
14. Steam/Generator Water Level-Low-Low	S	R	Q	1, 2
15. DELETED.				
16. Undervoltage-Reactor Coolant Pumps (Above P-7)	N.A.	R	M	1
17. Underfrequency-Reactor Coolant Pumps (Above P-7)	N.A.	R	M	1
18. Turbine Trip (Above P-9)				
A. Emergency Trip Header Low Pressure	N.A.	R	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	R	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	R	1, 2
20. Reactor Coolant Pump Breaker Position Trip (Above P-7)	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker	N.A.	N.A.	M(5, 11) and S/U(1)	1, 2, 3(14), 4(14), 5(14)

No change proposed. Included for information only.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2, 3(14), 4(14), 5(14)
23. Reactor Trip System Interlocks				
A. Intermediate Range Neutron Flux, P-6	N.A.	R(6)	R	1, 2
B. Power Range Neutron Flux, P-8	N.A.	R(6)	R	1
C. Power Range Neutron Flux, P-9	N.A.	R(6)	R	1
D. Power Range Neutron Flux, P-10	N.A.	R(6)	R	1, 2
E. Turbine First Stage Pressure, P-13	N.A.	R	R	1
24. Reactor Trip Bypass Breakers	N.A.	N.A.	M(12), R(13)	1, 2, 3(14), 4(14), 5(14)

TABLE 4.3-1 (Continued)

TABLE NOTATION

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15 percent of RATED THERMAL POWER.

(3) - At least once every 31 Effective Full Power Days (EFPD) compare incore to excore axial imbalance above ~~15~~50 percent of RATED THERMAL POWER. Recalibrate if absolute difference greater than or equal to 3 percent.

(4) - (Not Used).

(5) - Each train tested every other month on a STAGGERED TEST BASIS.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-10.

(8) - Below P-6, not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 12 hours after entry into MODE 3.

(9) - (Not Used)

(10) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) - Local manual shunt trip prior to placing breaker in service.

(13) - Automatic undervoltage trip.

(14) - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(15) - Surveillance Requirements need not be performed on alternate detectors until connected and required for OPERABILITY.

REPORTING REQUIREMENTS (Continued)

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I Section IV.B.1.

6.9.4 MONTHLY OPERATING REPORT

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

6.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1	Reactor Core Safety Limits
3.1.3.5	Shutdown Rod Insertion Limits
3.1.3.6	Control Rod Insertion Limits
3.2.1	Axial Flux Difference-~~Constant~~Relaxed Axial Offset Control
3.2.2	Heat Flux Hot Channel Factor-$F_Q(Z)$
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor-$F^N_{\Delta H}$
3.2.5	DNB Parameter
3.3.1.1	Reactor Trip System Instrumentation - Overtemperature and Overpower ΔT setpoint parameter values

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).

Change proposed in LAR 191 provided for information.

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-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February 1994.

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. 310

The following is a list of the affected pages:

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TECHNICAL SPECIFICATION BASES INDEX

BASES

SECTION | PAGE

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TECHNICAL SPECIFICATION BASES FIGURE INDEX

3/4.2 POWER DISTRIBUTION LIMITS

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 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)

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

Relaxed Axial Offset Control (RAOC) is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

~~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~~

~~3/4.2.1 AXIAL FLUX DIFFERENCE (AFD) (Continued)~~

~~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 duration 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% 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.~~

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits. If the AFD monitor is out of service, indicated AFD for each OPERABLE excore channel is manually monitored in accordance with the requirements specified in the Licensing Requirements Manual.

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3/4.2.1 AXIAL FLUX DIFFERENCE (AFD) (Continued)

BACKGROUND (Continued)

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (See WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The RAOC methodology (See WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February,1994) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD) (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

The limits on the AFD ensure that the limits on the Heat Flux Hot Channel Factor, $F_Q(Z)$, are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most limiting Condition 4 event with respect to the AFD limits is the LOCA. The most limiting Condition 3 event with respect to the AFD limits is the loss of flow accident. The most limiting Condition 2 events with respect to the AFD limits include the uncontrolled RCCA bank withdrawal at power, dropped RCCAs, and boron dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (UFSAR, Chapter 7). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR. Figure B 3/4 2-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD) (Continued)

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

ACTION

As an alternative to restoring the AFD to within its specified limits, the ACTION requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A completion time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS (SR)

SR 4.2.1.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. The surveillance interval of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed or the indicated AFD is manually monitored as required by the Licensing Requirements Manual.

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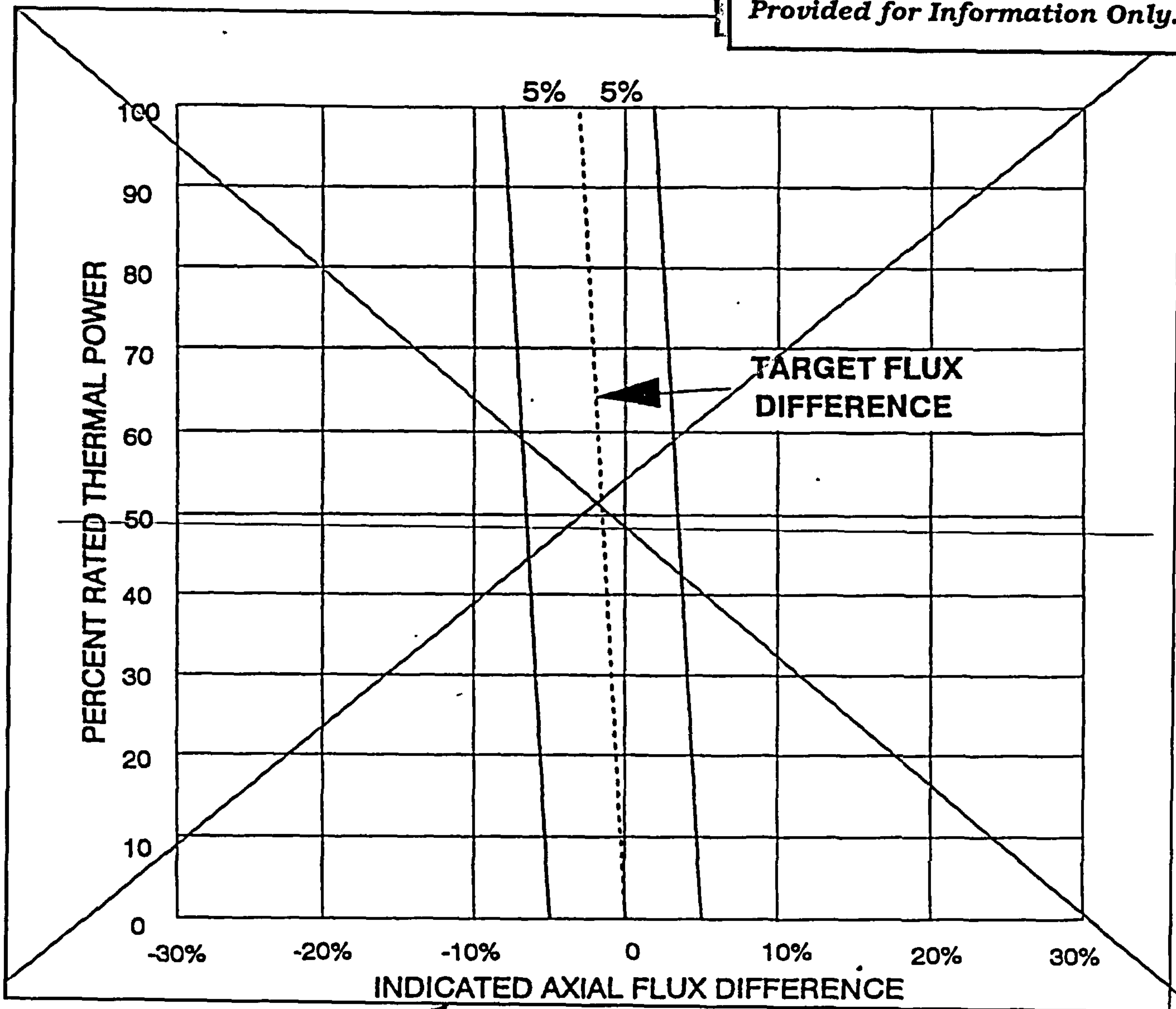

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FIGURE B 3/4 2-1

~~TYPICAL INDICATED AXIAL FLUX DIFFERENCE~~
~~VERSUS THERMAL POWER AT BOL~~
Typical Axial Flux Differences Limits as a Function of %
RATED THERMAL POWER for RAOC

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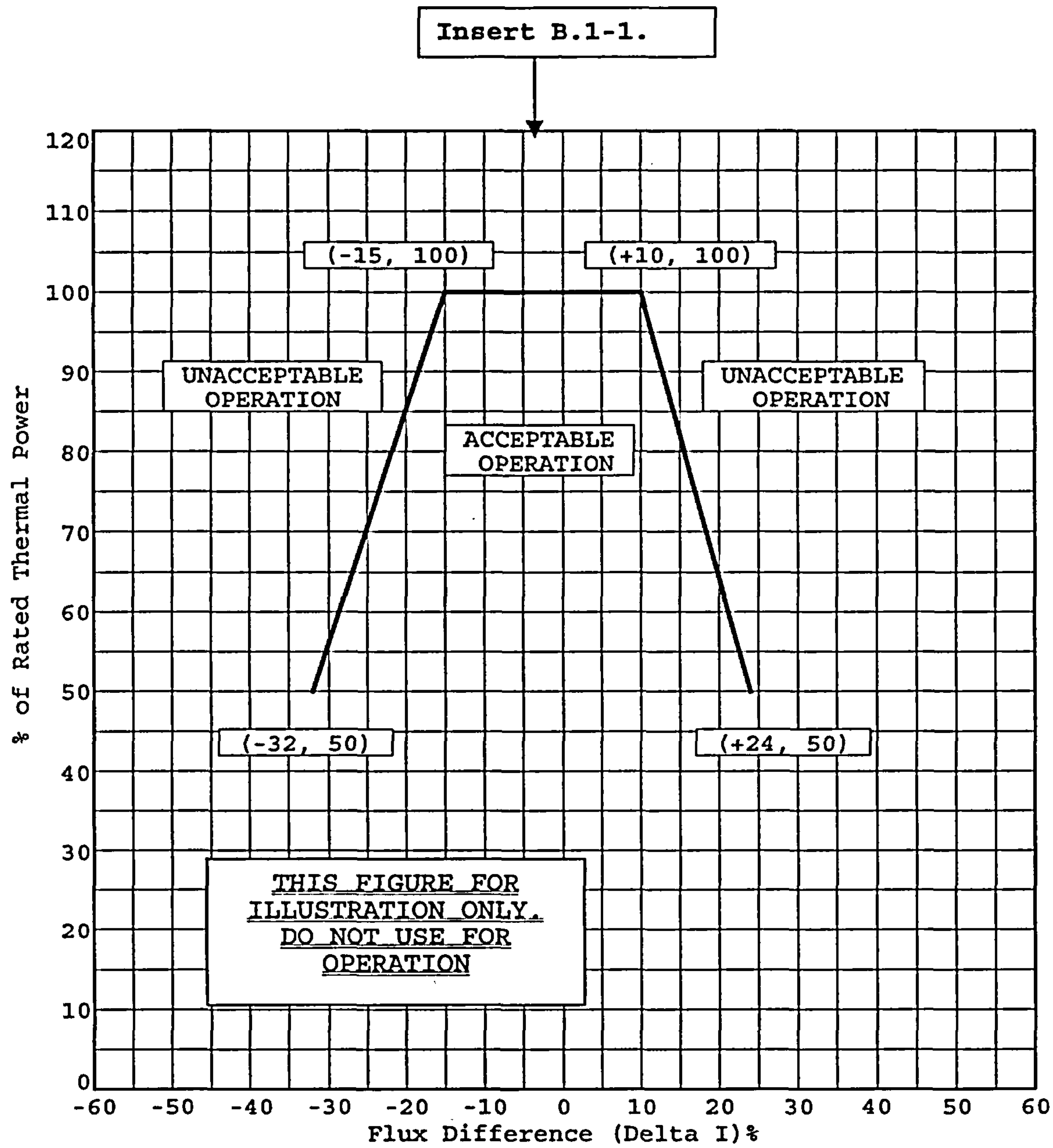

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BASES

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

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

~~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.~~

~~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.~~

~~BASES~~

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

~~Fuel rod bowing reduces the value of the DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses (1.33) and the design limit (1.21) to offset the rod bow penalty and other penalties which may apply.~~

~~The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor, $F_Q(Z)$, remains within its limits. The F_{xy} limit for RATED THERMAL POWER $F_{xy}{}^{RTP}$ provided in the CORE OPERATING LIMITS REPORT was determined from expected power control maneuvers over the full range of burnup conditions in the core.~~

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.3.6, "Control Rod Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ which are present during nonequilibrium situations such as load following or power ascension.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

BACKGROUND (Continued)

To account for these possible variations, the equilibrium value of $F_Q(Z)$ is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

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

a. During a large or small break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F, as specified in 10 CFR 50.46, 1974,

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

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm as specified in Regulatory Guide 1.77, Rev. 0, May 1974, and

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

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

LCO

The Heat Flux Hot Channel Factor $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \le [\ CFQ\ /\ P\ ] * K(Z) \quad \text{for } P > 0.5$$

$$F_Q(Z) \le [\ CFQ\ /\ 0.5\ ] * K(Z) \quad \text{for } P \le 0.5$$

where: CFQ is the F_Q limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

P = THERMAL POWER / RTP

The actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of 2.40, and K(Z) is a function that looks like the one provided in Figure B 3/4 2-2. Figure B 3/4 2-2 is for illustration purposes only. The COLR actual unit specific figures are contained in the COLR.

For Relaxed Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value, $F_Q^M(Z)$, of $F_Q(Z)$. Then,

$$F_Q^C(Z) = F_Q^M(Z) * 1.0815$$

Where: 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty as specified in WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

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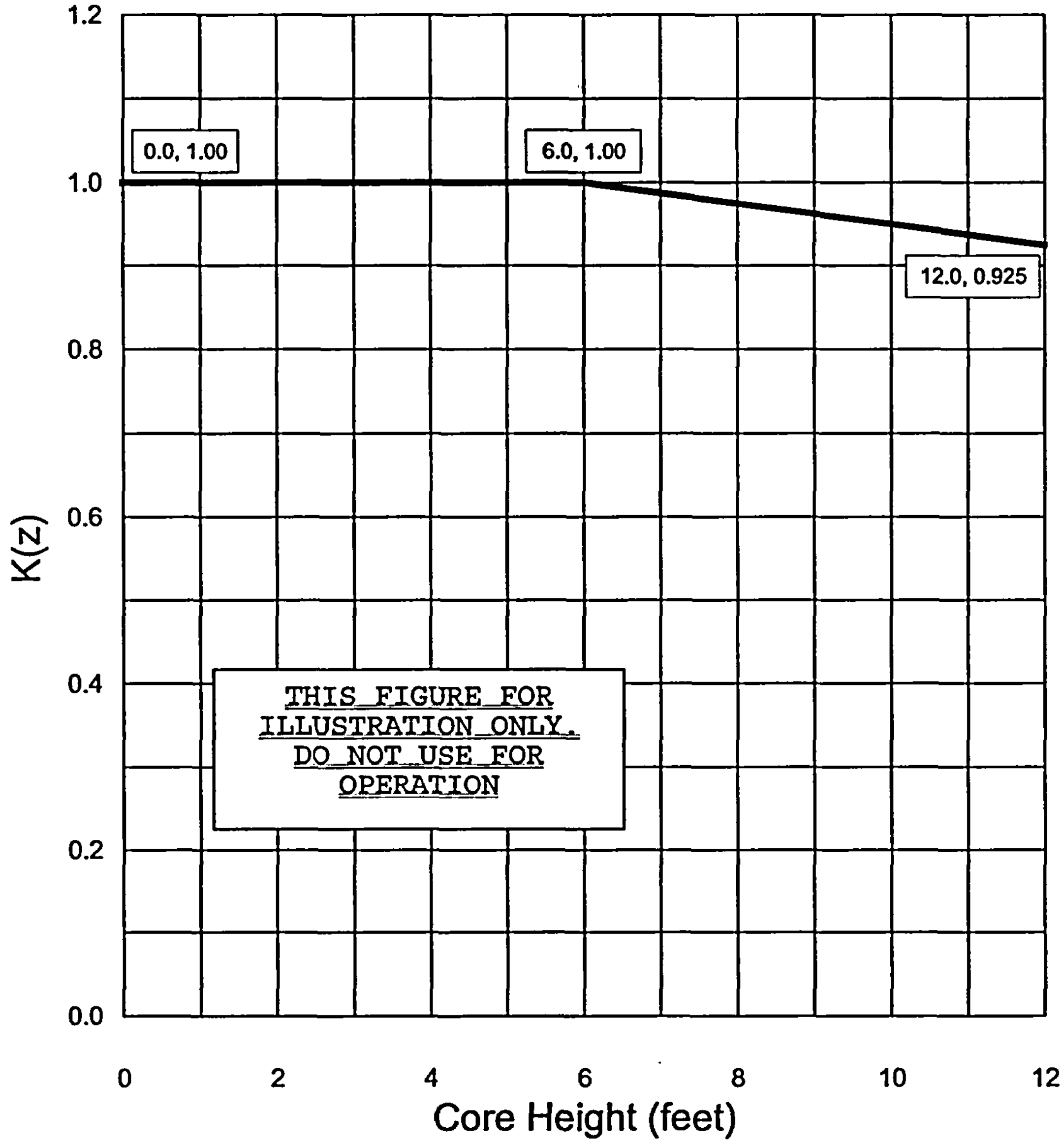

Figure B 3/4 2-2

Typical $F_Q T$ Normalized Operating Envelope, K(Z)

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

LCO (Continued)

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

where: W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required and if $F_Q^W(Z)$ cannot be maintained within the LCO limits, reduction of the AFD limits is required. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

ACTIONS

a.1 Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The completion time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by ACTION a.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

a.2 A reduction of the Power Range Neutron Flux - High Trip Setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The completion time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with ACTION a.1. The maximum allowable Power Range Neutron Flux - High Trip Setpoints initially determined by ACTION a.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux - High Trip Setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High Trip Setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux - High Trip Setpoints.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

ACTIONS (Continued)

a.3 Reduction in the Overpower ΔT Trip Setpoints (value of K4) by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The completion time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with ACTION a.1. The maximum allowable Overpower ΔT Trip Setpoints initially determined by ACTION a.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT Trip Setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT Trip Setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Overpower ΔT Trip Setpoints.

a.4 Verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ have been restored to within its limit, by performing SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the limit imposed by ACTION a.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Action a is modified by Note 1 that requires ACTION a.4 to be performed whenever ACTION a is entered. This ensures that SR 4.2.2.2 and SR 4.2.2.3 will be performed prior to increasing THERMAL POWER above the limit of ACTION a.1, even when ACTION a is exited prior to performing ACTION a.4. Performance of SR 4.2.2.2 and SR 4.2.2.3 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

a.5 If ACTIONS a.1 through a.4 are not met within their associated completion times, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This completion time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

ACTIONS (Continued)

b.1 If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD limits by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed completion time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded.

The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F_Q^C(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for ACTIONS b.2, b.3 and b.4.

b.2 A reduction of the Power Range Neutron Flux-High Trip Setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The completion time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with ACTION b.1.

b.3 Reduction in the Overpower ΔT Trip Setpoints (value of K4) by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The completion time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with ACTION b.1.

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

ACTIONS (Continued)

b.4 Verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ have been restored to within its limit, by performing SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by ACTION b.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Action b is modified by Note 2 that requires ACTION b.4 to be performed whenever ACTION b is entered. This ensures that SR 4.2.2.2 and SR 4.2.2.3 will be performed prior to increasing THERMAL POWER above the limit of ACTION b.1, even when ACTION b is exited prior to performing ACTION b.4. Performance of SR 4.2.2.2 and SR 4.2.2.3 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

b.5 If ACTIONS b.1 through b.4 are not met within their associated completion times, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This completion time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 4.2.2.1

The provisions of Specification 4.0.4 are not applicable because all the following surveillances must be performed in MODE 1.

SR 4.2.2.2 and SR 4.2.2.3 are modified by Note 3. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the surveillance interval conditions, i.e., 4.2.2.2.b and 4.2.2.3.b, that require verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, there is another surveillance interval condition, i.e., 4.2.2.2.a and 4.2.2.3.a, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this surveillance interval condition, together with the surveillance interval condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these surveillance interval conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The surveillance interval condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

SR 4.2.2.2

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)$ 1.0815. $F_Q^C(Z)$ is then compared to its specified limits.

The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

SURVEILLANCE REQUIREMENTS (Continued)

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The surveillance interval of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 4.2.2.3

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive and

b. Upper core region, from 85 to 100% inclusive.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

SURVEILLANCE REQUIREMENTS (Continued)

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This surveillance has been modified by Note (4) that specifies in part "If measurements indicate that the maximum over z of $[F_Q^C(Z)/K(Z)]$ has increased ...". This statement refers to the fact that both $F_Q^C(Z)$ and K are functions of the axial height. At each applicable core elevation the ratio of $F_Q^C(Z)/K(Z)$ is calculated to determine the maximum ratio (maximum over z). If this maximum ratio has increased since the last set of evaluations, then Note 4.b may require that more frequent surveillances be performed. If $F_Q^W(Z)$is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression maximum over z of $[F_Q^C(Z) / K(Z)]$, it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR (See WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q-Surveillance Technical Specification," February, 1994) or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

SURVEILLANCE REQUIREMENTS (Continued)

The surveillance interval of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The surveillance interval of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)" and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)" which address directly and continuously measured process variables.

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3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

BACKGROUND (Continued)

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis ensures the probability that DNB will not occur on the most limiting fuel rod is at least 95% at a 95% confidence level. This is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.22 for typical and thimble cells using the WRB-2M Critical Heat Flux (CHF) correlation, and 1.23 for the typical cell and 1.22 for the thimble cell using the WRB-1 CHF correlation. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition,

b. During a large or small break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F, as specified in 10 CFR 50.46, 1974,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm as specified in Regulatory Guide 1.77, Rev. 0, May 1974, and

d. Fuel design limits required by 10 CFR 50, Appendix A, GDC 26 for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis ensures the probability that DNB will not occur on the most limiting fuel rod is at least 95% at a 95% confidence level. This is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.22 for typical and thimble cells using the WRB-2M CHF correlation, and 1.23 for the typical cell and 1.22 for the thimble cell using the WRB-1 CHF correlation. These values provide a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, DNB events in which the core limits are modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor, $F_Q(Z)$, and the axial peaking factors are also indirectly modeled in the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature.

The fuel is protected in part by Technical Specifications which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.1.3.6, "Control Rod Insertion Limits" LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)" LCO 3.2.2, "Heat Flux Hot Channel Factor $F_Q(Z)$" LCO 3.2.3, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)" and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

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3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is a design radial peaking factor (nuclear enthalpy rise hot channel factor) used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by the value for $PF_{\Delta H}$ specified in the COLR for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other MODES (MODES 2 through 5) have significant margin to the DNBR limit and therefore, there is no need to restrict $F^N_{\Delta H}$ in these MODES.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

ACTIONS

a. With $F^N_{\Delta H}$ exceeding its limit, reduce THERMAL POWER to ≤ 50% RTP and reduce the Power Range Neutron Flux - High Trips Setpoints to ≤ 55% RTP in accordance with ACTION a. Reducing RTP to ≤ 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed completion time of 2 hours to reduce THERMAL POWER provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The allowed completion time of 4 hours to reset the trip setpoints recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

b. Once the power level has been reduced to ≤ 50% RTP per ACTION a, an incore flux map (SR 4.2.3.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 22 additional hours to perform this task over and above the 2 hours allowed by ACTION a. The completion time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this completion time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$. Should a satisfactory incore map not be completed within the required Completion Time, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by reducing RTP to less than 5%, i.e., placing the plant in at least MODE 2, within 2 hours. The allowed Completion Time of 2 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

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3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

ACTIONS (Continued)

c. Identification and correction of the cause of an out of limit condition and verification that $F^N_{\Delta H}$ is within its specified limits prior to increasing THERMAL POWER after an out of limit occurrence, ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

SURVEILLANCE REQUIREMENTS

SR 4.2.3.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD surveillance interval is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this surveillance interval is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

SR 4.2.3.2

The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

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

~~Fuel rod bowing reduces the value of the DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses (1.33) and the design limit (1.21) to offset the rod bow penalty and other penalties which may apply.~~

~~The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor, $F_Q(Z)$, remains within its limits. The F_{xy} limit for RATED THERMAL POWER F_{xy}^{RTP} provided in the CORE OPERATING LIMITS REPORT was determined from expected power control maneuvers over the full range of burnup conditions in the core.~~

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, "QUADRANT POWER TILT RATIO (QPTR)," 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.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Table 3.3-1 Action 2 has been modified by two notes. Note (4) allows placing the inoperable channel in the bypass condition for up to 4 hours while performing: a) routine surveillance testing of other channels, and b) setpoint adjustments of other channels when required to reduce the setpoint in accordance with other technical specifications. The 4 hour time limit is justified in accordance with WCAP-10271-P-A, Supplement 2, Revision 1, June 1990. Note (5) only requires SR 4.2.4 to be performed if a Power Range High Neutron Flux channel input to QPTR becomes inoperable. Failure of a component in the Power Range High Neutron Flux channel which renders the High Neutron Flux trip function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.
The following discussion pertains to Table 3.3-3, Functional Units 6.b and 6.c and the associated ACTION 34. The degraded voltage protection instrumentation system will automatically initiate the separation of the offsite power sources from the emergency buses. This action results in an automatic diesel generator start signal being generated as a direct result of the supply breakers opening between the normal and emergency buses. The failure of the degraded voltage protection system results in a loss of one of the automatic start signals for the diesel generator. Therefore, the ACTION statement requires the affected diesel generator to be declared inoperable if the required actions cannot be met within the specified time period.

The instrumentation functions that receive input from neutron detectors are modified by a note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above ~~15~~50% RATED THERMAL POWER. The power range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 or 1 on unit startup because the unit must be in at least MODE 1 to perform the test. The neutron detector CHANNEL CALIBRATION for the source range and intermediate range detectors consists of obtaining detector characteristics and performing an engineering evaluation of those characteristics. The intermediate range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 on unit startup because the unit must be in at least MODE 2 to perform the test. The source range neutron detector CHANNEL CALIBRATION is performed

Attachment B-2

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

License Amendment Request No. 182

The following is a list of the affected pages:

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TECHNICAL SPECIFICATION BASES INDEX

BASES

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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 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)

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

Relaxed Axial Offset Control (RAOC) is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

~~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~~ $\cancel{F_Q(Z)}$ ~~and~~ $\cancel{F^N_{\Delta H}}$

~~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.~~

~~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.~~

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD) (Continued)

BACKGROUND (Continued)

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits. If the AFD monitor is out of service, indicated AFD for each OPERABLE excore channel is manually monitored in accordance with the requirements specified in the Licensing Requirements Manual.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (See WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The RAOC methodology (See WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February,1994) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD) (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

The limits on the AFD ensure that the limits on the Heat Flux Hot Channel Factor, $F_Q(Z)$, are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most limiting Condition 4 event with respect to the AFD limits is the LOCA. The most limiting Condition 3 event with respect to the AFD limits is the loss of flow accident. The most limiting Condition 2 events with respect to the AFD limits include the uncontrolled RCCA bank withdrawal at power, dropped RCCAs, and boron dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (UFSAR, Chapter 7). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR. Figure B 3/4 2-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

BASES

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD) (Continued)

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

ACTION

As an alternative to restoring the AFD to within its specified limits, the ACTION requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A completion time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS (SR)

SR 4.2.1.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. The surveillance interval of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed or the indicated AFD is manually monitored as required by the Licensing Requirements Manual.

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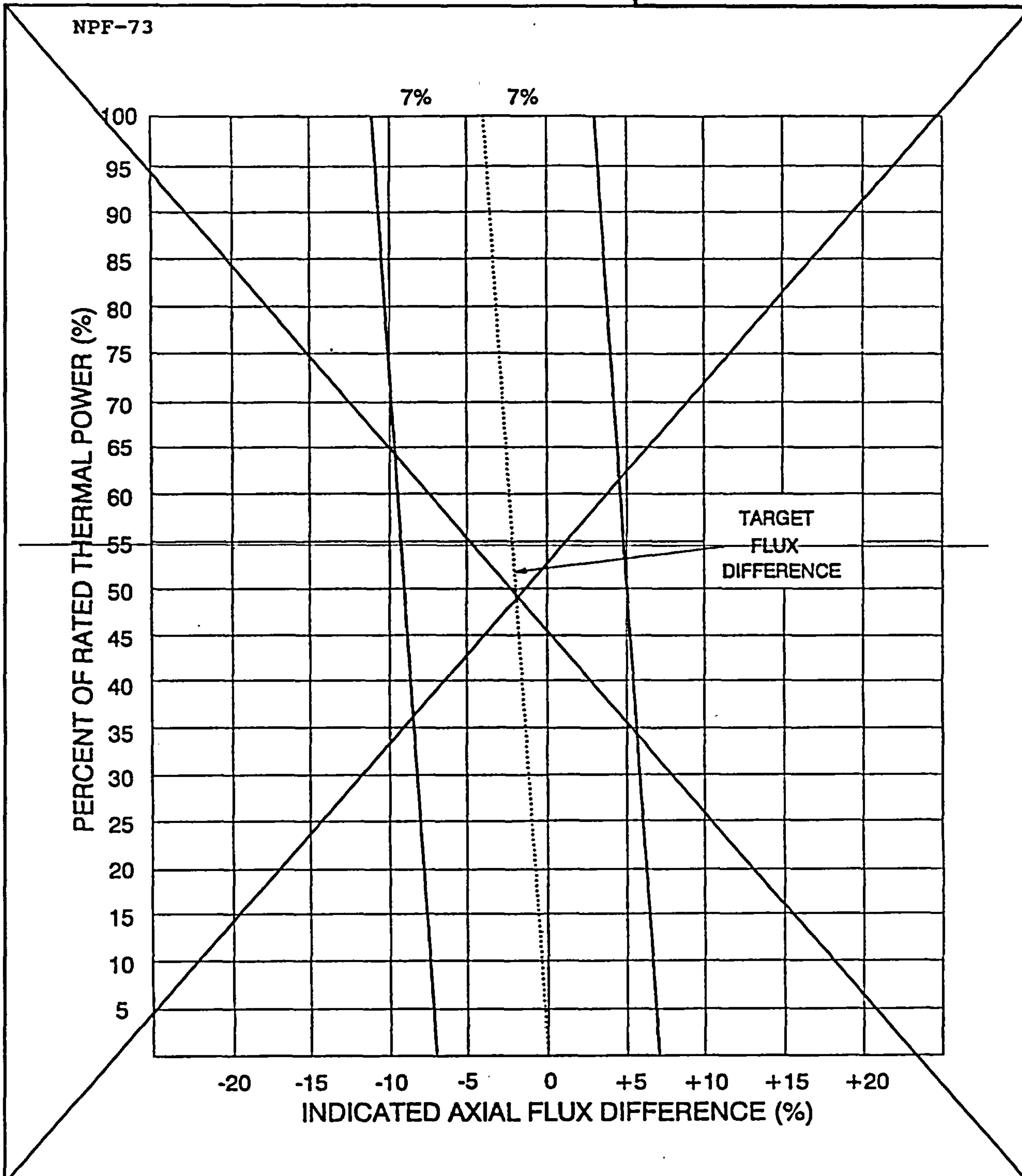
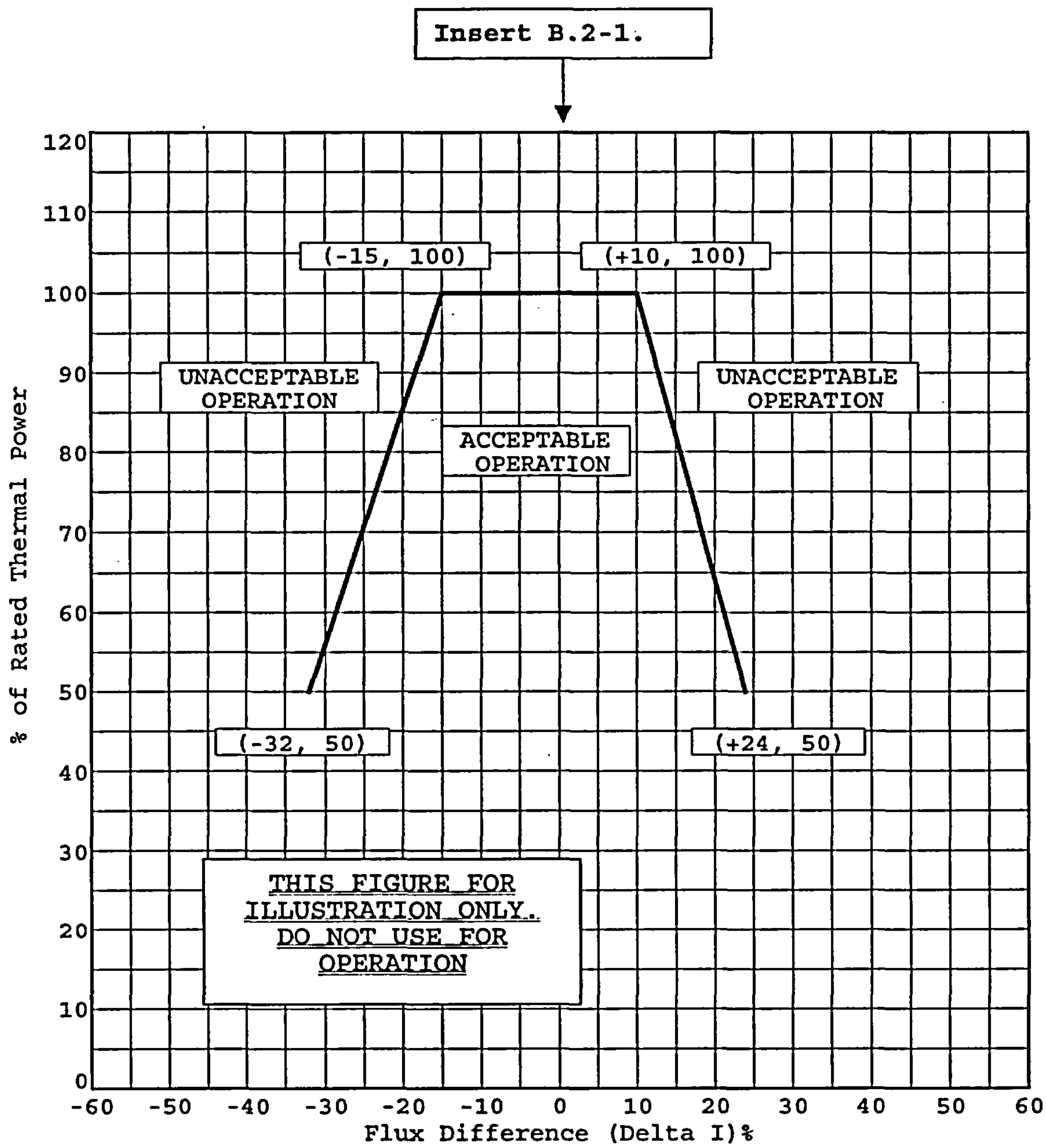


Replace with Insert B.2-1.

~~FIGURE B 3/4 2-1~~
~~TYPICAL INDICATED AXIAL FLUX DIFFERENCE (AFD)~~
~~VERSUS THERMAL POWER AT BOL~~
Axial Flux Differences Limits as a Function of % RATED THERMAL POWER for RAOC


Insert B.2-1.
(-15, 100)
(+10, 100)
UNACCEPTABLE OPERATION
UNACCEPTABLE OPERATION
ACCEPTABLE OPERATION
(-32, 50)
(+24, 50)
THIS FIGURE FOR ILLUSTRATION ONLY. DO NOT USE FOR OPERATION
% of Rated Thermal Power
120
110
100
90
80
70
60
50
40
30
20
10
0
-60
-50
-40
-30
-20
-10
0
10
20
30
40
50
60
Flux Difference (Delta I)%

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

~~c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.~~

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

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

~~When an~~ $\cancel{F_Q}$ ~~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~~ $\cancel{F^N_{\Delta H}}$ ~~contains an 8% allowance for uncertainties which means that normal, full power, three loop operation will result in~~ $\cancel{F^N_{\Delta H}}$ ~~less than or equal to the design limit specified in the CORE OPERATING LIMITS REPORT.~~

~~Fuel rod bowing reduces the value of DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses and the design limit to offset the rod bow penalty and other penalties which may apply.~~

~~The radial peaking reactor~~ $\cancel{F_{XY}(Z)}$ ~~is measured periodically to provide assurance that the hot channel factor,~~ $\cancel{F_Q(Z)}$~~, remains within its limit. The~~ $\cancel{F_{xy}}$ ~~limit for Rated Thermal Power (~~$\cancel{F^{RTP}{}_{xy}}$~~) provided in the CORE OPERATING LIMITS REPORT was determined from expected power control maneuvers over the full range of burnup conditions in the core.~~

Provided for Information Only.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.3.6, "Control Rod Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ which are present during nonequilibrium situations such as load following or power ascension.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

BACKGROUND (Continued)

To account for these possible variations, the equilibrium value of $F_Q(Z)$ is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

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

a. During a large or small break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F, as specified in 10 CFR 50.46, 1974,

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

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm as specified in Regulatory Guide 1.77, Rev. 0, May 1974, and

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

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

LCO

The Heat Flux Hot Channel Factor $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq [\ CFQ\ /\ P] * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq [\ CFQ\ /\ 0.5] * K(Z) \qquad \text{for } P \leq 0.5$$

where: CFQ is the F_Q limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

P = THERMAL POWER / RTP

The actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of 2.40, and K(Z) is a function that looks like the one provided in Figure B 3/4 2-2. Figure B 3/4 2-2 is for illustration purposes only. The COLR actual unit specific figures are contained in the COLR.

For Relaxed Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value, $F_Q^M(Z)$, of $F_Q(Z)$. Then,

$$F_Q^C(Z) = F_Q^M(Z) * 1.0815$$

Where: 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty as specified in WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

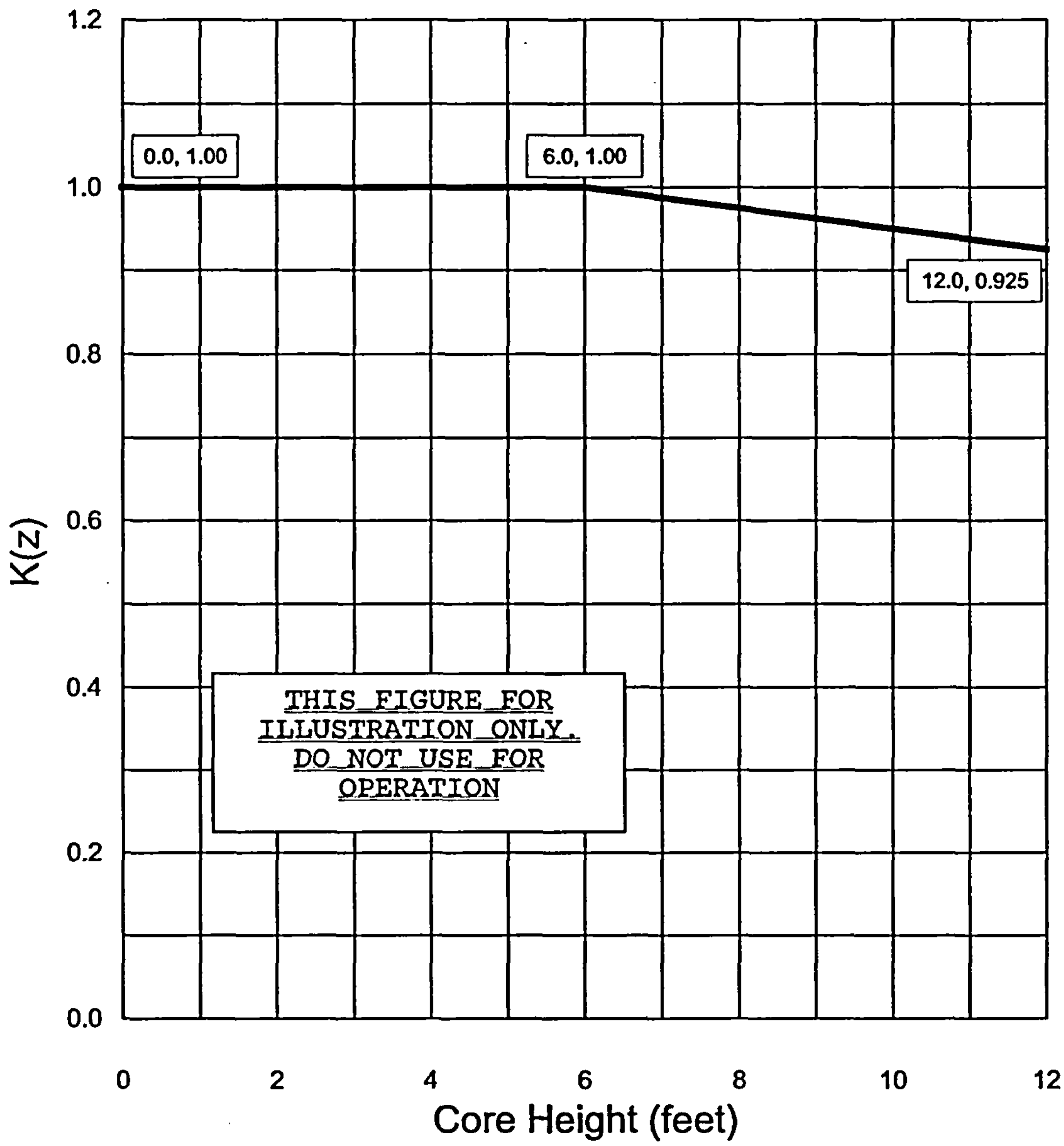


Figure B 3/4 2-2

Typical $F_Q T$ Normalized Operating Envelope, K(Z)

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

LCO (Continued)

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

where: W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required and if $F_Q^W(Z)$ cannot be maintained within the LCO limits, reduction of the AFD limits is required. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

Provided for Information Only.

BASES

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

ACTIONS

a.1 Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of FQ(Z). The completion time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by ACTION a.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

a.2 A reduction of the Power Range Neutron Flux - High Trip Setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The completion time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with ACTION a.1. The maximum allowable Power Range Neutron Flux - High Trip Setpoints initially determined by ACTION a.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux - High Trip Setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High Trip Setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux - High Trip Setpoints.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

ACTIONS (Continued)

a.3 Reduction in the Overpower ΔT Trip Setpoints (value of K4) by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The completion time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with ACTION a.1. The maximum allowable Overpower ΔT Trip Setpoints initially determined by ACTION a.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT Trip Setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT Trip Setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Overpower ΔT Trip Setpoints.

a.4 Verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ have been restored to within its limit, by performing SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the limit imposed by ACTION a.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Action a is modified by Note 1 that requires ACTION a.4 to be performed whenever ACTION a is entered. This ensures that SR 4.2.2.2 and SR 4.2.2.3 will be performed prior to increasing THERMAL POWER above the limit of ACTION a.1, even when ACTION a is exited prior to performing ACTION a.4. Performance of SR 4.2.2.2 and SR 4.2.2.3 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

a.5 If ACTIONS a.1 through a.4 are not met within their associated completion times, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This completion time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

ACTIONS (Continued)

b.1 If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD limits by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed completion time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded.

The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F_Q^C(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for ACTIONS b.2, b.3 and b.4.

b.2 A reduction of the Power Range Neutron Flux-High Trip Setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The completion time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with ACTION b.1.

b.3 Reduction in the Overpower ΔT Trip Setpoints (value of K4) by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The completion time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER as a result of reducing AFD limits in accordance with ACTION b.1.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

ACTIONS (Continued)

b.4 Verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ have been restored to within its limit, by performing SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by ACTION b.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Action b is modified by Note 2 that requires ACTION b.4 to be performed whenever ACTION b is entered. This ensures that SR 4.2.2.2 and SR 4.2.2.3 will be performed prior to increasing THERMAL POWER above the limit of ACTION b.1, even when ACTION b is exited prior to performing ACTION b.4. Performance of SR 4.2.2.2 and SR 4.2.2.3 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

b.5 If ACTIONS b.1 through b.4 are not met within their associated completion times, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This completion time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 4.2.2.1

The provisions of Specification 4.0.4 are not applicable because all the following surveillances must be performed in MODE 1.

SR 4.2.2.2 and SR 4.2.2.3 are modified by Note 3. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the surveillance interval conditions, i.e., 4.2.2.2.b and 4.2.2.3.b, that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$ (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, there is another surveillance interval condition, i.e., 4.2.2.2.a and 4.2.2.3.a, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this surveillance interval condition, together with the surveillance interval condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these surveillance interval conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The surveillance interval condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

SR 4.2.2.2

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)$ 1.0815. $F_Q^C(Z)$ is then compared to its specified limits.

The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

SURVEILLANCE REQUIREMENTS (Continued)

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The surveillance interval of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 4.2.2.3

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive and

b. Upper core region, from 85 to 100% inclusive.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

SURVEILLANCE REQUIREMENTS (Continued)

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This surveillance has been modified by Note (4) that specifies in part "If measurements indicate that the maximum over z of $[F_Q^C(Z)/K(Z)]$ has increased ...". This statement refers to the fact that both $F_Q^C(Z)$ and K are functions of the axial height. At each applicable core elevation the ratio of $F_Q^C(Z)/K(Z)$ is calculated to determine the maximum ratio (maximum over z). If this maximum ratio has increased since the last set of evaluations, then Note 4.b may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression maximum over z of $[\ F_Q^C(Z)\ /\ K(Z)\ ]$, it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR (See WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q-Surveillance Technical Specification," February, 1994) or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

SURVEILLANCE REQUIREMENTS (Continued)

The surveillance interval of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The surveillance interval of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)" and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)" which address directly and continuously measured process variables.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

BACKGROUND (Continued)

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis ensures the probability that DNB will not occur on the most limiting fuel rod is at least 95% at a 95% confidence level. This is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.22 for typical and thimble cells using the WRB-2M Critical Heat Flux (CHF) correlation, and 1.23 for the typical cell and 1.22 for the thimble cell using the WRB-1 CHF correlation. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition,

b. During a large or small break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F, as specified in 10 CFR 50.46, 1974,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm as specified in Regulatory Guide 1.77, Rev. 0, May 1974, and

d. Fuel design limits required by 10 CFR 50, Appendix A, GDC 26 for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis ensures the probability that DNB will not occur on the most limiting fuel rod is at least 95% at a 95% confidence level. This is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.22 for typical and thimble cells using the WRB-2M CHF correlation, and 1.23 for the typical cell and 1.22 for the thimble cell using the WRB-1 CHF correlation. These values provide a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, DNB events in which the core limits are modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor, $F_Q(Z)$, and the axial peaking factors are also indirectly modeled in the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature.

The fuel is protected in part by Technical Specifications which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.1.3.6, "Control Rod Insertion Limits" LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)" LCO 3.2.2, "Heat Flux Hot Channel Factor $F_Q(Z)$" LCO 3.2.3, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)" and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is a design radial peaking factor (nuclear enthalpy rise hot channel factor) used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by the value for $PF_{\Delta H}$ specified in the COLR for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other MODES (MODES 2 through 5) have significant margin to the DNBR limit and therefore, there is no need to restrict $F^N_{\Delta H}$ in these MODES.

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BASES

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

ACTIONS

a. With $F^N_{\Delta H}$ exceeding its limit, reduce THERMAL POWER to ≤ 50% RTP and reduce the Power Range Neutron Flux - High Trips Setpoints to ≤ 55% RTP in accordance with ACTION a. Reducing RTP to ≤ 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed completion time of 2 hours to reduce THERMAL POWER provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The allowed completion time of 4 hours to reset the trip setpoints recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

b. Once the power level has been reduced to ≤ 50% RTP per ACTION a, an incore flux map (SR 4.2.3.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 22 additional hours to perform this task over and above the 2 hours allowed by ACTION a. The completion time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this completion time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$. Should a satisfactory incore map not be completed within the required Completion Time, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by reducing RTP to less than 5%, i.e., placing the plant in at least MODE 2, within 2 hours. The allowed Completion Time of 2 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ (Continued)

ACTIONS (Continued)

c. Identification and correction of the cause of an out of limit condition and verification that $F^N_{\Delta H}$ is within its specified limits prior to increasing THERMAL POWER after an out of limit occurrence, ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

SURVEILLANCE REQUIREMENTS

SR 4.2.3.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD surveillance interval is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this surveillance interval is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

SR 4.2.3.2

The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

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, "QUADRANT POWER TILT RATIO (QPTR)," 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 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

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Table 3.3-1 Action 2 has been modified by two notes. Note (4) allows placing the inoperable channel in the bypass condition for up to 4 hours while performing: a) routine surveillance testing of other channels, and b) setpoint adjustments of other channels when required to reduce the setpoint in accordance with other technical specifications. The 4 hour time limit is justified in accordance with WCAP-10271-P-A, Supplement 2, Revision 1, June 1990. Note (5) only requires SR 4.2.4 to be performed if a Power Range High Neutron Flux channel input to QPTR becomes inoperable. Failure of a component in the Power Range High Neutron Flux channel which renders the High Neutron Flux trip function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.

The following discussion pertains to Table 3.3-3, Functional Units 6.b and 6.c and the associated ACTION 34. The degraded voltage protection instrumentation system will automatically initiate the separation of the offsite power sources from the emergency buses. This action results in an automatic diesel generator start signal being generated as a direct result of the supply breakers opening between the normal and emergency buses. The failure of the degraded voltage protection system results in a loss of one of the automatic start signals for the diesel generator. Therefore, the ACTION statement requires the affected diesel generator to be declared inoperable if the required actions cannot be met within the specified time period.

The instrumentation functions that receive input from neutron detectors are modified by a note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above ~~15~~50% RATED THERMAL POWER. The power range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 or 1 on unit startup because the unit must be in at least MODE 1 to perform the test. The neutron detector CHANNEL CALIBRATION for the source range and intermediate range detectors consists of obtaining detector characteristics and performing an engineering evaluation of those characteristics. The intermediate range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 on unit startup because the unit must be in at least MODE 2 to perform the test. The source range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 or 3 on unit

Attachment C-1

Beaver Valley Power Station, Unit No. 1
Proposed Licensing Requirements Manual Changes

License Amendment Request No. 310

The following is a list of the affected pages:

Page
iv
v
3.4-1
4.1-1
4.1-2
4.1-4
4.1-6
4.1-7
4.1-9
B.3-1
B.3-4

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LICENSING REQUIREMENTS MANUAL (LRM)
INDEX

LICENSING REQUIREMENTS MANUAL (LRM)
INDEX

3.4 Axial Flux Difference (AFD) Monitor Alarm

LICENSING REQUIREMENT SURVEILLANCES

LRS 3.4.1 This surveillance is only required to be performed when the AFD monitor alarm is inoperable and power is above ~~15~~50% RATED THERMAL POWER. ~~Assume logged values of the AFD exist during the preceding time interval.~~ Monitor and log the indicated AFD for each OPERABLE channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter. ~~Monitor the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours after restoring the AFD monitor alarm to OPERABLE status.~~ LRS 1.2.3 is not applicable.

4.1 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report provides the cycle specific parameter limits developed in accordance with the NRC approved methodologies specified in Technical Specification Administrative Control 6.9.5.

Specification 3.1.3.5 Shutdown Rod Insertion Limits

The shutdown rods shall be withdrawn to at least 225 steps.*

Specification 3.1.3.6 Control Rod Insertion Limits

Control Banks A and B shall be withdrawn to at least 225 steps.*

Control Banks C and D shall be limited in physical insertion as shown in Figure 4.1-1.*

Specification 3.2.1 Axial Flux Difference

The Axial Flux Difference (AFD) acceptable operation limits are provided in Figure 4.1-2.
~~NOTE: The target band is ±7% about the target flux from 0% to 100% RATED THERMAL POWER.~~

~~The indicated Axial Flux Difference:~~

~~a. Above 90% RATED THERMAL POWER shall be maintained within the ±7% target band about the target flux difference.~~

~~b. Between 50% and 90% RATED THERMAL POWER is within the limits shown on Figure 4.1-2.~~

~~c. Below 50% RATED THERMAL POWER may deviate outside the target band.~~

Specification 3.2.2 Heat Flux Hot Channel Factor - $F_Q(Z)$ and ~~F_{xy}~~ Limits

The Heat Flux Hot Channel Factor - $F_Q(Z)$ variable is defined by:

$$\cancel{F_Q(Z) \le \left[\frac{CF_Q}{P}\right] * K(Z) \qquad \text{for } P > 0.5}$$

$$F_Q(Z) \le \left[\frac{CFQ}{P}\right] * K(Z) \qquad \text{for } P > 0.5$$

$$\cancel{F_Q(Z) \le \left[\frac{CF_Q}{0.5}\right] * K(Z) \qquad \text{for } P \le 0.5}$$

$$F_Q(Z) \le \left[\frac{CFQ}{0.5}\right] * K(Z) \qquad \text{for } P \le 0.5$$

Where: ~~CF_Q~~ CFQ = ~~2.2~~2.4

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

$K(Z)$ = the function obtained from Figure 4.1-3.

*As indicated by the group demand counter

~~The F_{xy} limits [$F_{xy}(L)$] for RATED THERMAL POWER within specific core planes shall be:~~

~~$F_{xy}(L) = F_{xy}(RTP)(1 + PF_{xy} * (1-P))$~~

~~Where: For all core planes containing D-Bank:~~

~~$F_{xy}(RTP) \leq 1.71$~~

~~For unrodded core planes:~~

~~$F_{xy}(RTP) \leq 1.68$ from 1.8 ft. elevation to 2.3 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.73$ from 2.3 ft. elevation to 3.7 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.79$ from 3.7 ft. elevation to 5.8 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.81$ from 5.8 ft. elevation to 7.4 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.74$ from 7.4 ft. elevation to 8.9 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.60$ from 8.9 ft. elevation to 10.2 ft. elevation~~

~~$PF_{xy} = 0.2$~~

~~P = THERMAL POWER / RATED THERMAL POWER~~

~~Figure 4.1-4 provides the maximum total peaking factor times relative power ($F_Q^T * P_{rel}$) as a function of axial core height during normal core operation.~~

$$F_Q^C(Z) = F_Q^M(Z) * 1.0815$$

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

The Heat Flux Hot Channel Factor - $F_Q(Z)$ limit is defined by:

$$F_Q(Z) \leq \left[\frac{CFQ}{P * W(Z)}\right] * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \left[\frac{CFQ}{0.5 * W(Z)}\right] * K(Z) \qquad \text{for } P \leq 0.5$$

W(Z) values are provided in Table 4.1-1.

The $F_Q(Z)$ penalty function, applied when the analytic $F_Q(Z)$ function changes by more than 2% in a month, is provided in Table 4.1-2.

<u>Specification 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$</u>

$F^N_{\Delta H} \leq CF_{\Delta H} * (1 + PF_{\Delta H} (1\text{-}P))$

Where: $CF_{\Delta H} = 1.62$

$PF_{\Delta H} = 0.3$

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

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Provided for Information Only.

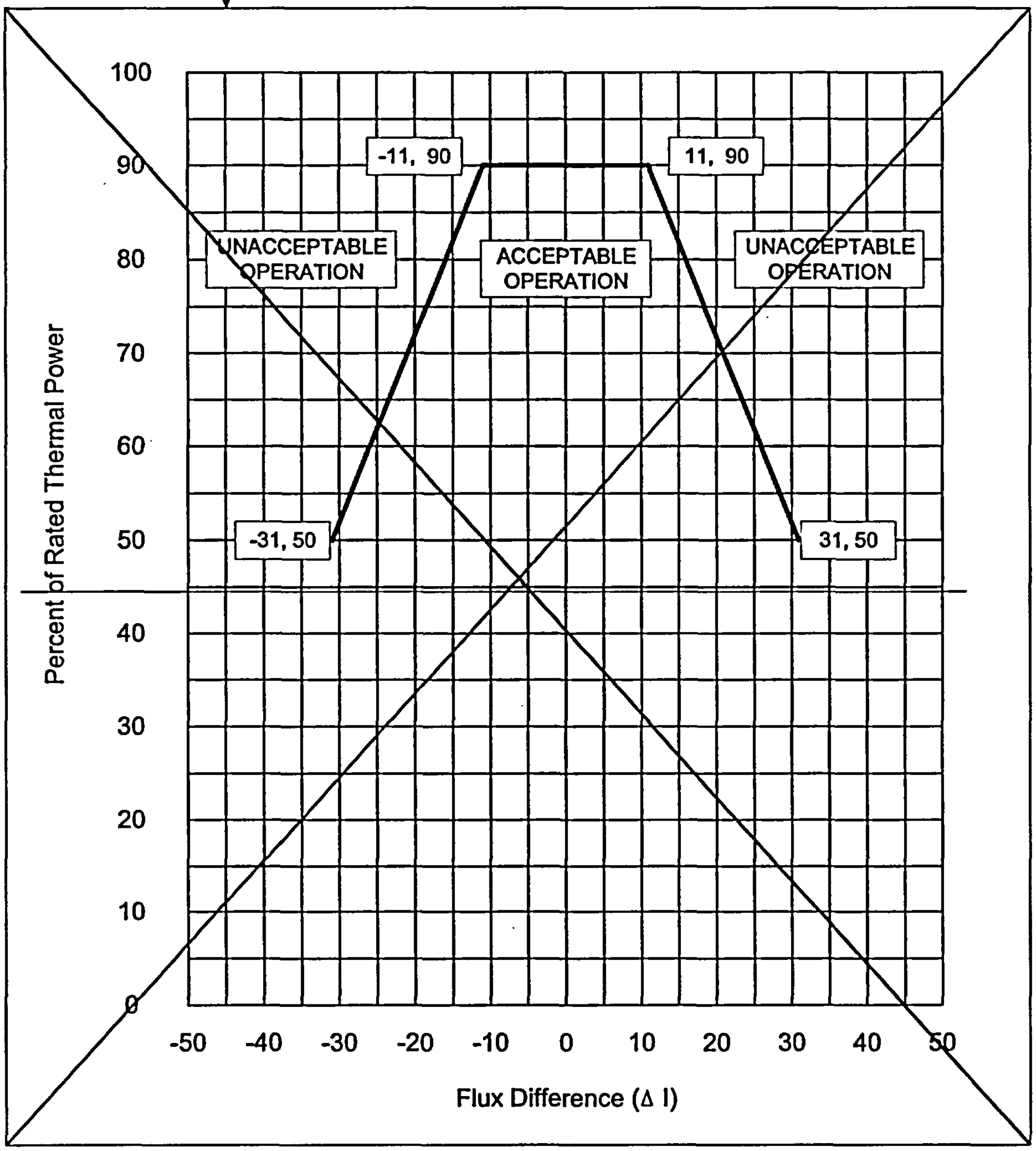

FIGURE 4.1-2
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF PERCENT OF RATED THERMAL POWER FOR RAOC

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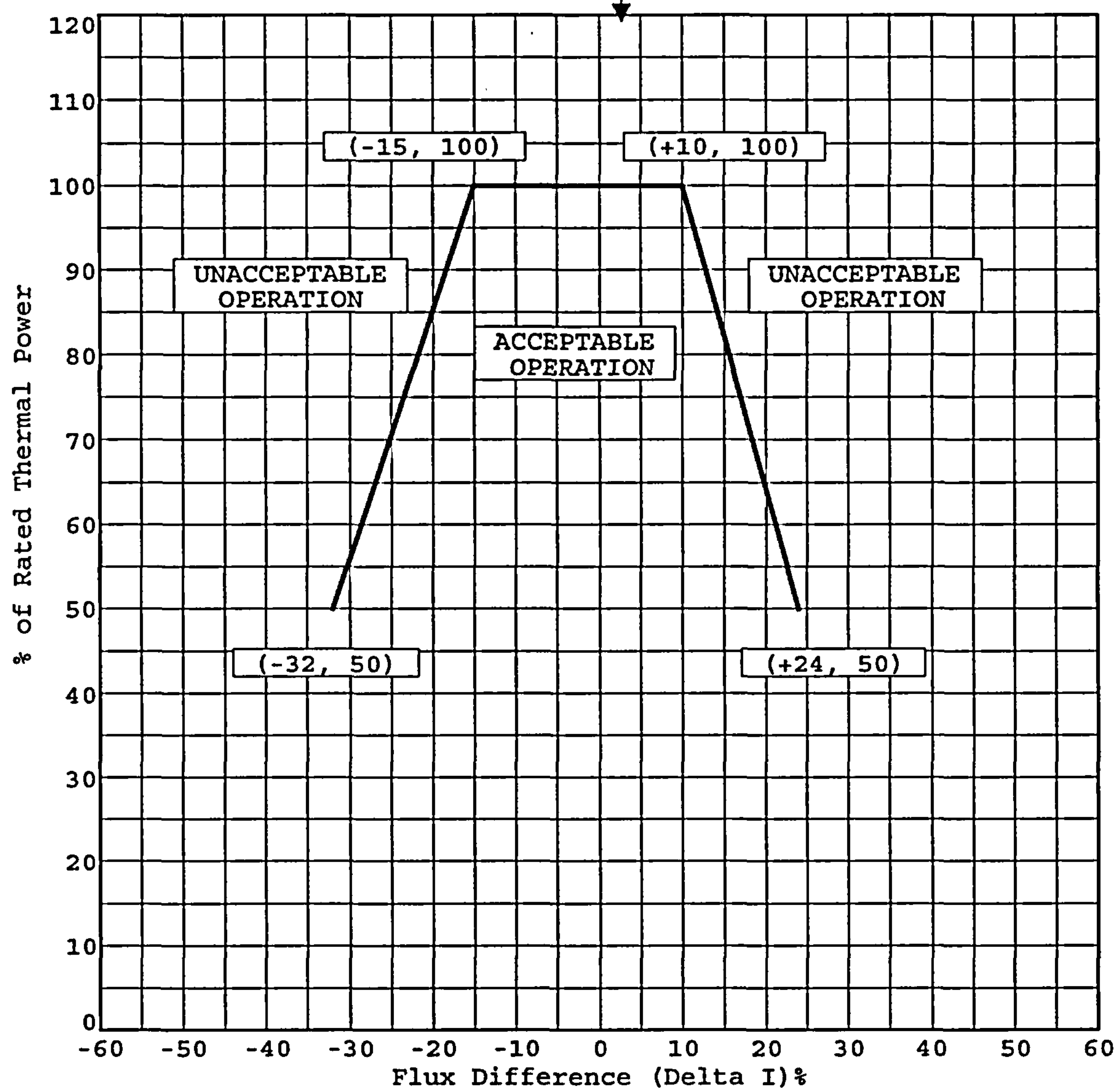

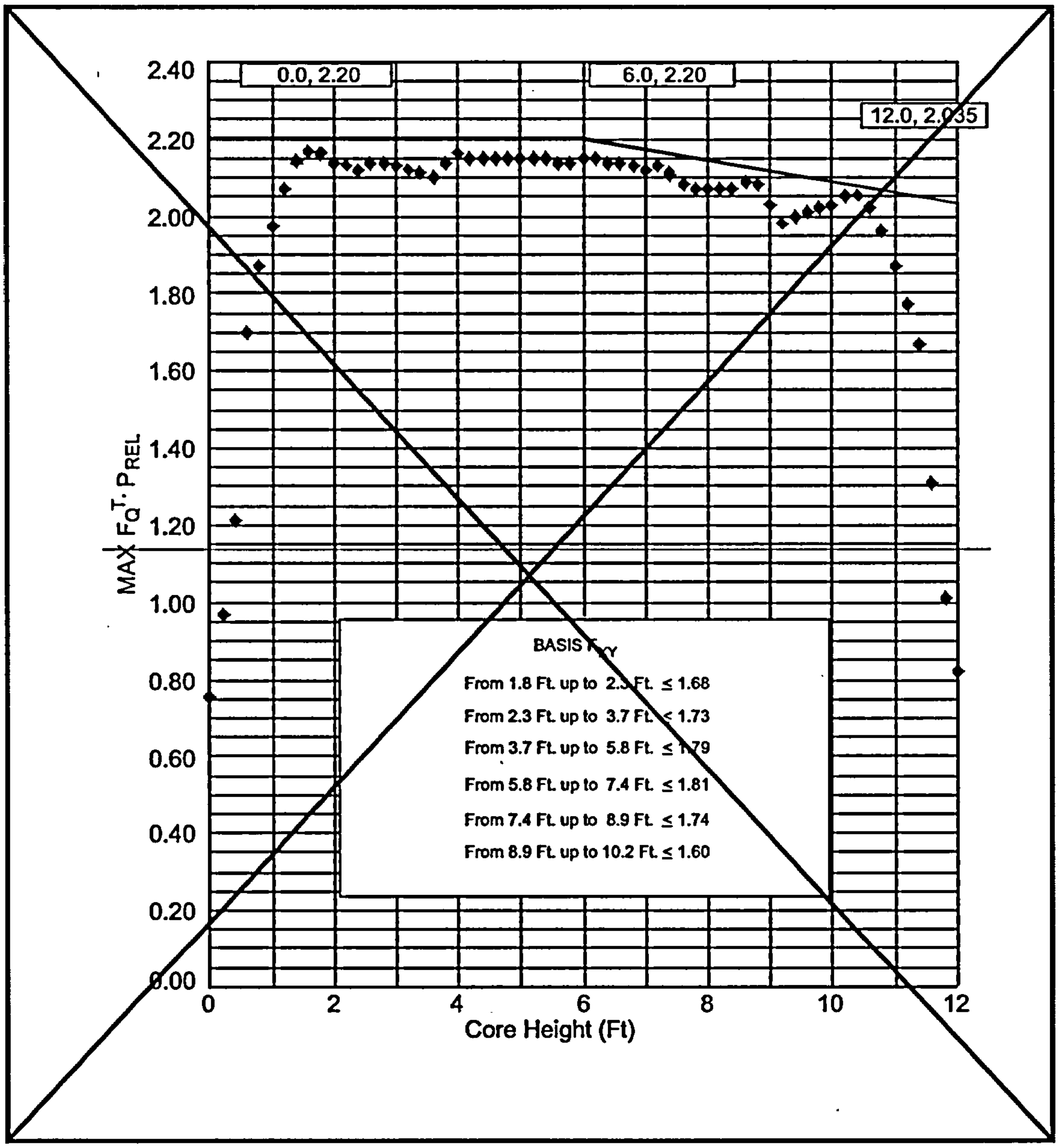

~~FIGURE 4.1-4~~
~~MAXIMUM (F_QT*PREL) VS AXIAL CORE HEIGHT~~
~~DURING NORMAL OPERATION~~

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Specification 3.3.1.1 Reactor Trip System Instrumentation Setpoints, Table 3.3-1 Table Notations A and B

Overtemperature ΔT Setpoint Parameter Values:

Parameter	Value
Overtemperature ΔT reactor trip setpoint	K1 ≤ ~~1.259~~1.242
Overtemperature ΔT reactor trip setpoint Tavg coefficient	K2 ≥ ~~0.01655~~0.0183/°F
Overtemperature ΔT reactor trip setpoint pressure coefficient	K3 ≥ ~~0.000801~~0.001/psia
Tavg at RATED THERMAL POWER	T' ≤ ~~576.2~~580.0°F
Nominal Pressurizer Pressure	P' ≥ 2250 psia
Measured reactor vessel average temperature lead/lag time constants	$\tau_1 \geq 30$ secs $\tau_2 \leq 4$ secs
Measured reactor vessel ΔT lag time constant	$\tau_4 \leq 6$ secs
Measured reactor vessel average temperature lag time constant	$\tau_5 \leq 2$ secs

f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that: 15

(i) for $q_t - q_b$ between ~~-36~~48 percent and ~~+15~~10 percent, f (ΔI) = 0 (where q_t and q_b are percent
37 RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER). 37

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds ~~-36~~ 48 percent, the ΔT trip setpoint shall be automatically reduced by ~~2.08~~4.67 percent of its value at RATED THERMAL POWER. 15

2.52 (iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds ~~+15~~ 10 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.59~~1.47 percent of its value at RATED THERMAL POWER.

Overpower ΔT Setpoint Parameter Values:

Parameter	Value
Overpower ΔT reactor trip setpoint	K4 ≤ ~~1.0916~~1.085
Overpower ΔT reactor trip setpoint Tavg rate/lag coefficient	K5 ≥ 0.02/°F for increasing average temperature K5 = 0/°F for decreasing average temperature

Replace with Insert C1-2.

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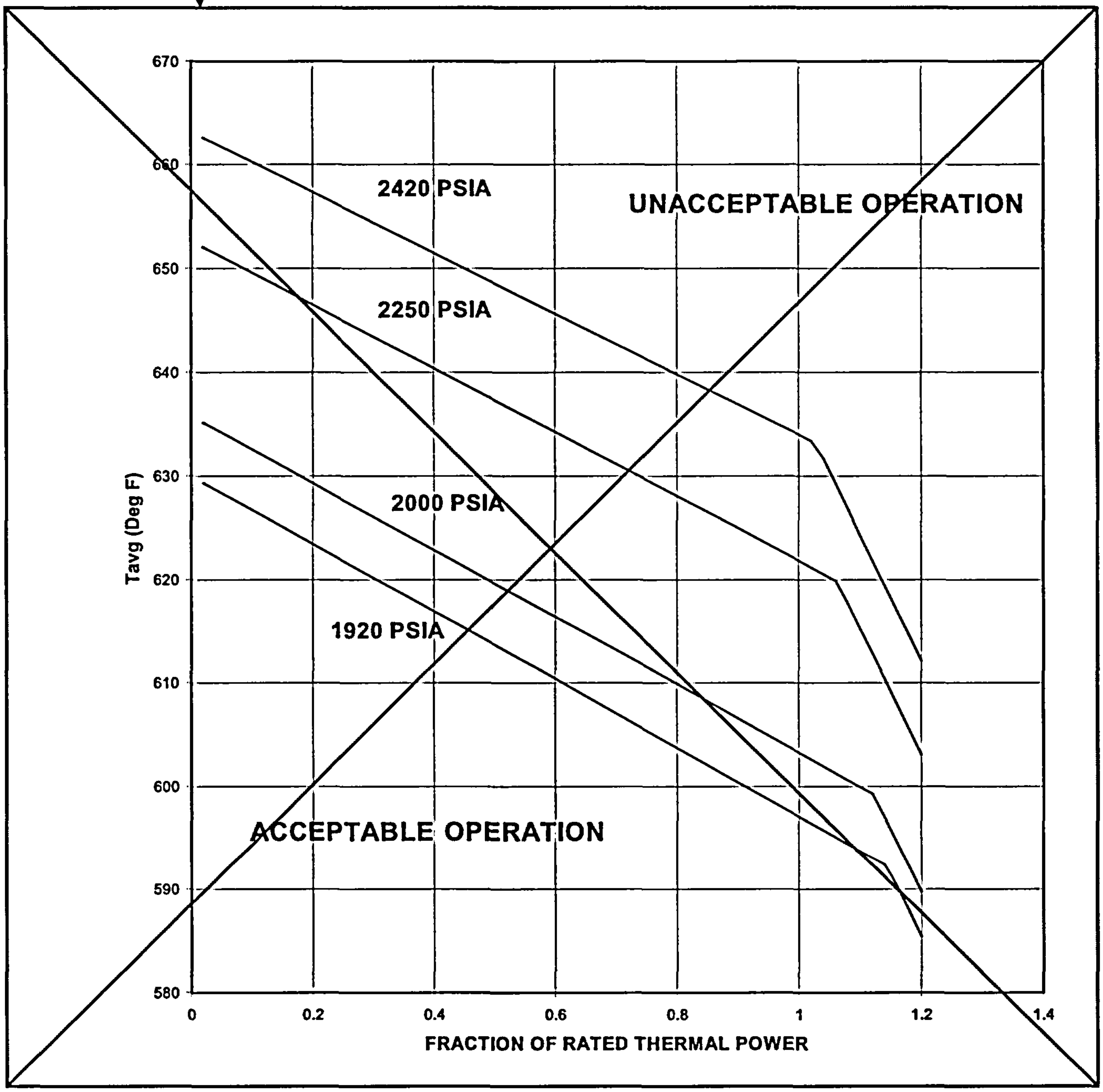

Figure 4.1-5
REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION
(Technical Specification Safety Limit 2.1.1)

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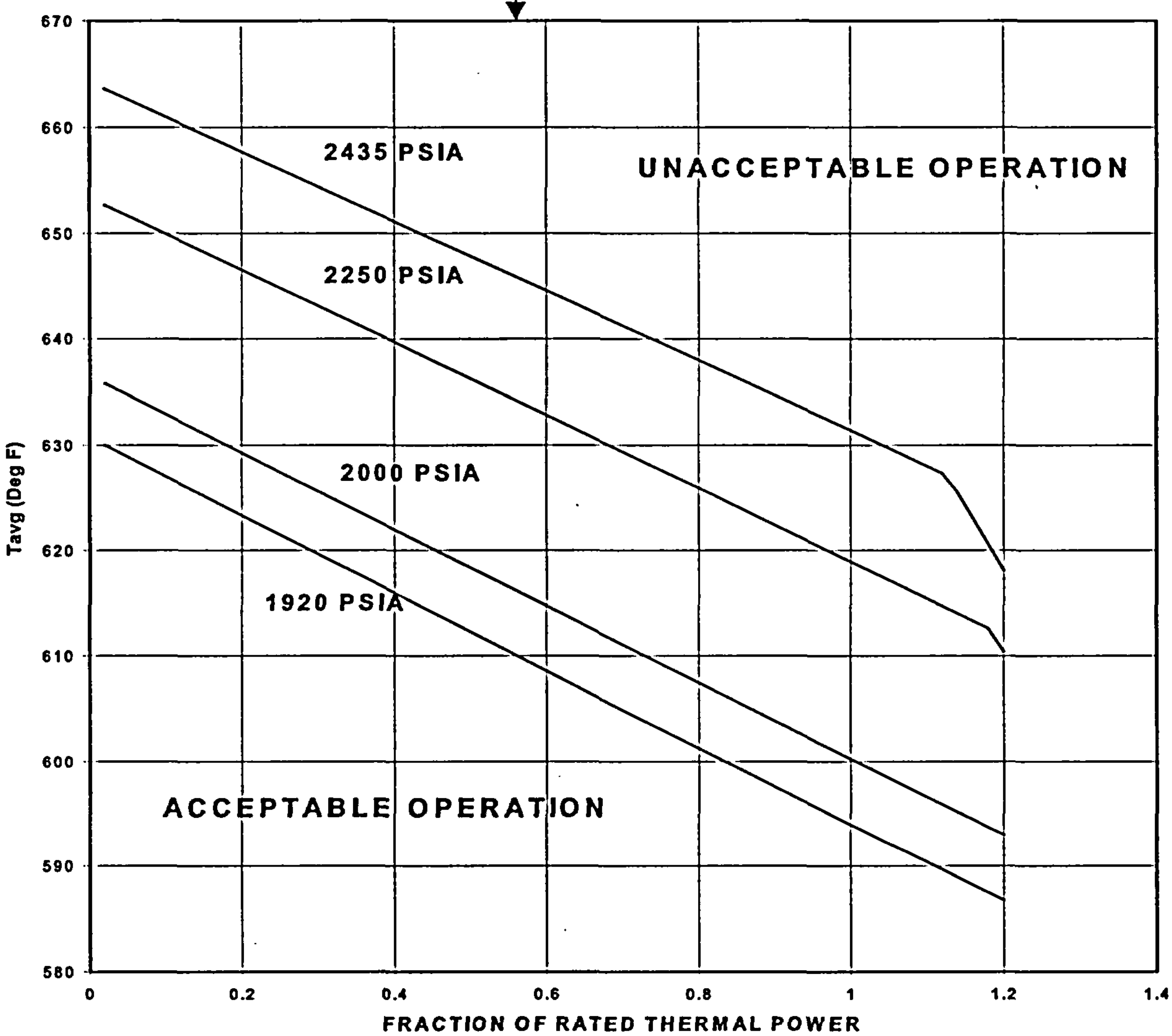

Table 4.1-1 (Page 1 of 2)
W(Z) Values

Exclusion Zone	Axial Point	Elevation (Ft)	150 MWD/MTU	3000 MWD/MTU	10000 MWD/MTU	18000 MWD/MTU
x	61	0.00	1.0000	1.0000	1.0000	1.0000
x	60	0.20	1.0000	1.0000	1.0000	1.0000
x	59	0.40	1.0000	1.0000	1.0000	1.0000
x	58	0.60	1.0000	1.0000	1.0000	1.0000
x	57	0.80	1.0000	1.0000	1.0000	1.0000
x	56	1.00	1.0000	1.0000	1.0000	1.0000
x	55	1.20	1.0000	1.0000	1.0000	1.0000
x	54	1.40	1.0000	1.0000	1.0000	1.0000
x	53	1.60	1.0000	1.0000	1.0000	1.0000
	52	1.80	1.3782	1.3227	1.2640	1.2794
	51	2.00	1.3543	1.3023	1.2457	1.2616
	50	2.20	1.3296	1.2829	1.2267	1.2430
	49	2.40	1.3026	1.2630	1.2074	1.2237
	48	2.60	1.2802	1.2425	1.1877	1.2041
	47	2.80	1.2668	1.2220	1.1683	1.1840
	46	3.00	1.2562	1.2032	1.1518	1.1646
	45	3.20	1.2449	1.1904	1.1423	1.1586
	44	3.40	1.2334	1.1821	1.1385	1.1576
	43	3.60	1.2240	1.1735	1.1353	1.1553
	42	3.80	1.2163	1.1640	1.1315	1.1581
	41	4.00	1.2098	1.1573	1.1275	1.1638
	40	4.20	1.2037	1.1540	1.1246	1.1682
	39	4.40	1.1967	1.1497	1.1234	1.1718
	38	4.60	1.1890	1.1445	1.1221	1.1742
	37	4.80	1.1805	1.1389	1.1201	1.1756
	36	5.00	1.1713	1.1327	1.1176	1.1757
	35	5.20	1.1619	1.1264	1.1134	1.1740
	34	5.40	1.1504	1.1176	1.1135	1.1736
	33	5.60	1.1432	1.1113	1.1218	1.1796
	32	5.80	1.1514	1.1187	1.1355	1.1947
	31	6.00	1.1608	1.1293	1.1524	1.2096

Note: Top and Bottom 15% Excluded

Table 4.1-1 (Page 2 of 2)
W(Z) Values

Exclusion Zone	Axial Point	Elevation (Ft)	150 MWD/MTU	3000 MWD/MTU	10000 MWD/MTU	18000 MWD/MTU
	30	6.20	1.1669	1.1417	1.1697	1.2215
	29	6.40	1.1726	1.1527	1.1855	1.2319
	28	6.60	1.1769	1.1626	1.1999	1.2402
	27	6.80	1.1797	1.1714	1.2129	1.2468
	26	7.00	1.1812	1.1786	1.2242	1.2526
	25	7.20	1.1813	1.1841	1.2336	1.2583
	24	7.40	1.1822	1.1904	1.2410	1.2621
	23	7.60	1.1827	1.1963	1.2462	1.2632
	22	7.80	1.1811	1.2003	1.2491	1.2621
	21	8.00	1.1780	1.2024	1.2496	1.2585
	20	8.20	1.1732	1.2031	1.2476	1.2526
	19	8.40	1.1667	1.2015	1.2429	1.2443
	18	8.60	1.1583	1.1978	1.2356	1.2337
	17	8.80	1.1501	1.1971	1.2268	1.2213
	16	9.00	1.1485	1.2035	1.2259	1.2056
	15	9.20	1.1544	1.2163	1.2348	1.1926
	14	9.40	1.1580	1.2296	1.2410	1.1976
	13	9.60	1.1622	1.2414	1.2504	1.2087
	12	9.80	1.1680	1.2528	1.2653	1.2184
	11	10.00	1.1726	1.2648	1.2817	1.2278
	10	10.20	1.1736	1.2753	1.2965	1.2369
x	9	10.40	1.0000	1.0000	1.0000	1.0000
x	8	10.60	1.0000	1.0000	1.0000	1.0000
x	7	10.80	1.0000	1.0000	1.0000	1.0000
x	6	11.00	1.0000	1.0000	1.0000	1.0000
x	5	11.20	1.0000	1.0000	1.0000	1.0000
x	4	11.40	1.0000	1.0000	1.0000	1.0000
x	3	11.60	1.0000	1.0000	1.0000	1.0000
x	2	11.80	1.0000	1.0000	1.0000	1.0000
x	1	12.00	1.0000	1.0000	1.0000	1.0000

Note: Top and Bottom 15% Excluded

Table 4.1-2
$F_Q(Z)$ Penalty Factor

Cycle Burnup (MWD/MTU)	$F_Q(Z)$ Penalty Factor
All Burnups	1.02

Note: The Penalty Factor, to be applied to $F_Q(Z)$ in accordance with Technical Specification Surveillance Requirement 4.2.2.3, is the maximum factor by which $F_Q(Z)$ is expected to increase over a 39 Effective Full Power Day (EFPD) interval (surveillance interval of 31 EFPD plus the maximum allowable extension not to exceed 25% of the surveillance interval per Technical Specification Surveillance Requirement 4.0.2) starting from the burnup at which the $F_Q(Z)$ was determined.

B.3.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs."

B.3.4 AXIAL FLUX DIFFERENCE (AFD) MONITOR ALARM

Surveillance of the AFD verifies that the AFD, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. During operation above ~~15~~50% RATED THERMAL POWER, when the AFD monitor alarm is inoperable, additional surveillance criteria is required by the Licensing Requirements Manual beyond the surveillance criteria required by the Technical Specifications to detect operation outside of the limits.~~target band and to compute the penalty deviation time before corrective action is required. The logged values of the AFD are assumed to exist for the preceding time interval in order for the operator to compute the cumulative penalty deviation time.~~

B.3.5 QUADRANT POWER TILT RATIO (QPTR) MONITOR ALARM

Surveillance of the QPTR verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. During operation above 50% RATED THERMAL POWER, when the QPTR monitor alarm is inoperable, additional surveillance criteria is required by the Licensing Requirements Manual beyond the surveillance criteria required by the Technical Specifications to detect any relatively slow changes in QPTR. For those causes of core power tilt that occur quickly (e.g., a dropped rod), there are other indications of abnormality that prompt a verification of core power tilt.

B.3.8 LEADING EDGE FLOW METER (Continued)

This surveillance is performed every 24 hours when power is above ~~15~~50%. The NIS excore power range channel indications are renormalized if they are not found to be within ±2% of the calorimetric measurement. This ±2% requirement for renormalization is distinct from the allowance for calorimetric uncertainty, and these allowances are handled as independent contributions to determine the maximum power assumed in design basis accident analyses.

The plant may then be run for the next 24-hour period using this normalized NIS indication. Although calorimetric power indication may be monitored continuously, it is not required to be consulted again until the required daily calorimetric comparisons of NIS indication are performed.

The surveillance requirement to perform planned maintenance and inspections every 18 months is based upon the manufacturer's recommendations, and is consistent with the surveillance intervals specified for similar electronic apparatus.

Additional guidance for determining steady-state THERMAL POWER is taken from the NRC Inspection Manual; Inspection Procedure 61706; C/N 86-036, 07/14/1986; "Core Thermal Power Evaluation"; step 03.02.d, and is described in the BVPS Operating Manual.

Attachment C-2

Beaver Valley Power Station, Unit No. 2
Proposed Licensing Requirements Manual Changes

License Amendment Request No. 182

The following is a list of the affected pages:

Page
v
vi
3.4-1
4.1-1
4.1-2
4.1-4
4.1-6
4.1-7
4.1-10
B.3-1
B.3-3

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LICENSING REQUIREMENTS MANUAL (LRM)
INDEX

LICENSING REQUIREMENTS MANUAL (LRM) INDEX

3.4 Axial Flux Difference (AFD) Monitor Alarm

LICENSING REQUIREMENT SURVEILLANCES

LRS 3.4.1 This surveillance is only required to be performed when the AFD monitor alarm is inoperable and power is above ~~15~~50% RATED THERMAL POWER. ~~Assume logged values of the AFD exist during the preceding time interval.~~ Monitor and log the indicated AFD for each OPERABLE channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter. ~~Monitor the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours after restoring the AFD monitor alarm to OPERABLE status.~~ LRS 1.2.3 is not applicable.

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4.1 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report provides the cycle specific parameter limits developed in accordance with the NRC approved methodologies specified in Technical Specification Administrative Control 6.9.5.

Specification 3.1.3.5 Shutdown Rod Insertion Limits

The Shutdown rods shall be withdrawn to at least 225 steps.*

Specification 3.1.3.6 Control Rod Insertion Limits

Control Banks A and B shall be withdrawn to at least 225 steps.*

Control Banks C and D shall be limited in physical insertion as shown in Figure 4.1-1.*

Specification 3.2.1 Axial Flux Difference
The Axial Flux Difference (AFD) acceptable operation limits are provided in Figure 4.1-2.
~~NOTE: The target band is ±7% about the target flux from 0% to 100% RATED THERMAL POWER.~~

~~The indicated Axial Flux Difference:~~

~~a. Above 90% RATED THERMAL POWER shall be maintained within the ±7% target band about the target flux difference.~~

~~b. Between 50% and 90% RATED THERMAL POWER is within the limits shown on Figure 4.1-2.~~

~~c. Below 50% RATED THERMAL POWER may deviate outside the target band.~~

Specification 3.2.2 Heat Flux Hot Channel Factor - $F_Q(Z)$ ~~and F_{xy}~~ Limits

The Heat Flux Hot Channel Factor - $F_Q(Z)$ variable is defined by:

$$\cancel{F_Q(Z) \le \frac{CF_Q}{P} * K(Z) \quad \text{for } P > 0.5}$$

$$F_Q(Z) \le \left[\frac{CFQ}{P}\right] * K(Z) \quad \text{for } P > 0.5$$

$$\cancel{F_Q(Z) \le \frac{CF_Q}{0.5} * K(Z) \quad \text{for } P \le 0.5}$$

$$F_Q(Z) \le \left[\frac{CFQ}{0.5}\right] * K(Z) \quad \text{for } P \le 0.5$$

Where: ~~CF_Q~~ $F_Q^{RTP}(z)$ = ~~2.3~~2.4 $\qquad P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

K(Z) = the function obtained from Figure 4.1-3.

*As indicated by the group demand counter

~~The F_{xy} limits [$F_{xy}(L)$] for RATED THERMAL POWER within specific core planes shall be:~~

~~$F_{xy}(L) = F_{xy}(RTP)\ (1 + PF_{XY} * (1-P))$~~

~~Where: For all core planes containing D-Bank:~~

~~$F_{xy}(RTP) \leq 1.71$~~

~~For unrodded core planes:~~

~~$F_{xy}(RTP) \leq 1.76$ from 1.8 ft. elevation to 2.3 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.80$ from 2.3 ft. elevation to 3.7 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.83$ from 3.7 ft. elevation to 5.8 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.84$ from 5.8 ft. elevation to 7.4 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.81$ from 7.4 ft. elevation to 9.0 ft. elevation~~

~~$F_{xy}(RTP) \leq 1.72$ from 9.0 ft. elevation to 10.2 ft. elevation~~

~~$PF_{xy} = 0.2$~~

$$\cancel{P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}}$$

~~Figure 4.1-4 provides the maximum total peaking factor times relative power ($F_Q^T * P_{rel}$) as a function of axial core height during normal core operation.~~

$$F_Q^C(Z) = F_Q^M(Z) * 1.0815$$

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

The Heat Flux Hot Channel Factor - $F_Q(Z)$ limit is defined by:

$$F_Q(Z) \leq \left[\frac{CFQ}{P * W(Z)}\right] * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \left[\frac{CFQ}{0.5 * W(Z)}\right] * K(Z) \qquad \text{for } P \leq 0.5$$

W(Z) values are provided in Table 4.1-1.

The $F_Q(Z)$ penalty function, applied when the analytic $F_Q(Z)$ function changes by more than 2% in a month, is provided in Table 4.1-2.

Specification 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$

$$F^N_{\Delta H} \le CF_{\Delta H} * (1 + PF_{\Delta H}\ (1 - P))$$

Where: $CF_{\Delta H} = 1.62$

$PF_{\Delta H} = 0.3$

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

Replace with Insert C2-1.

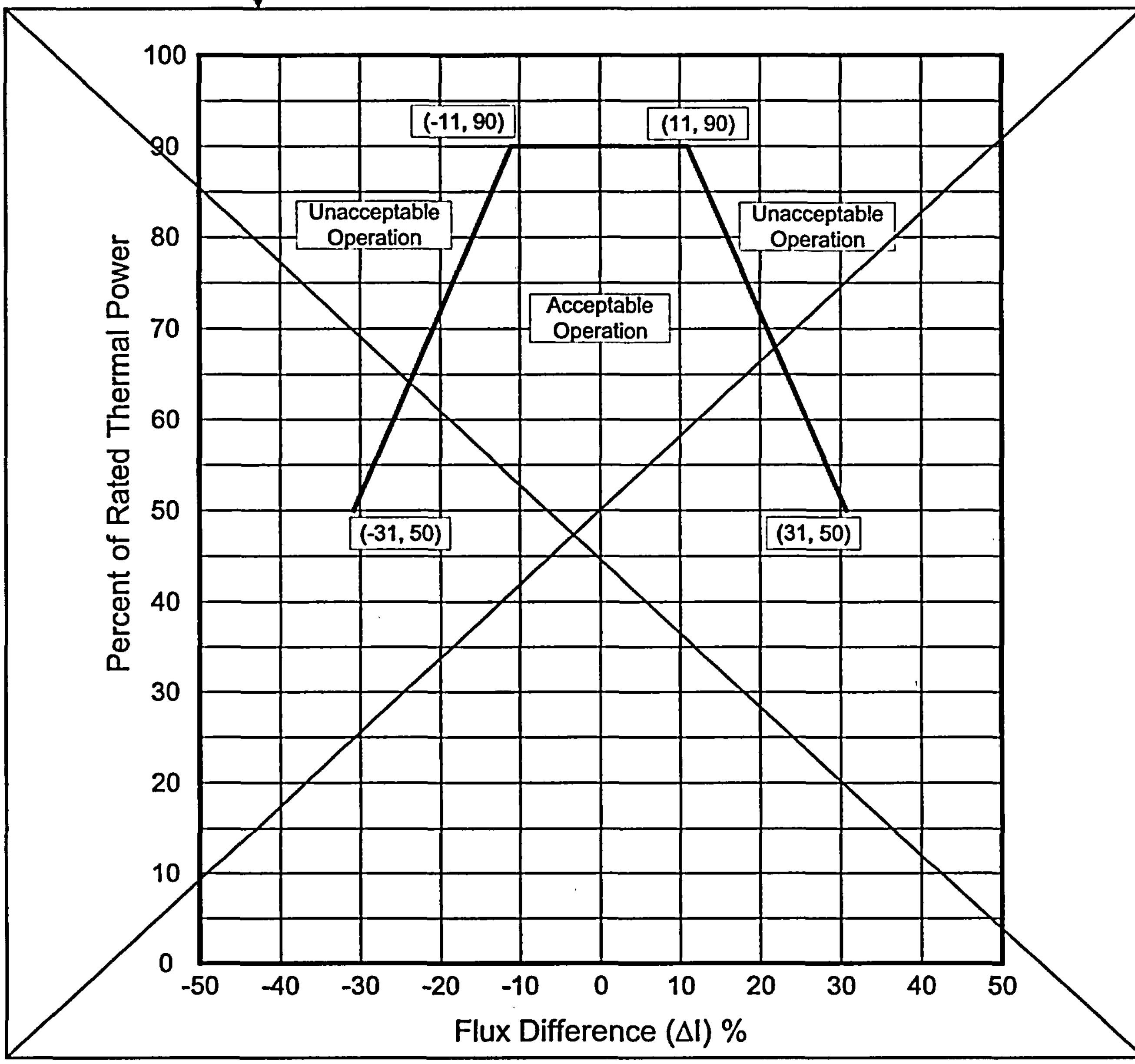


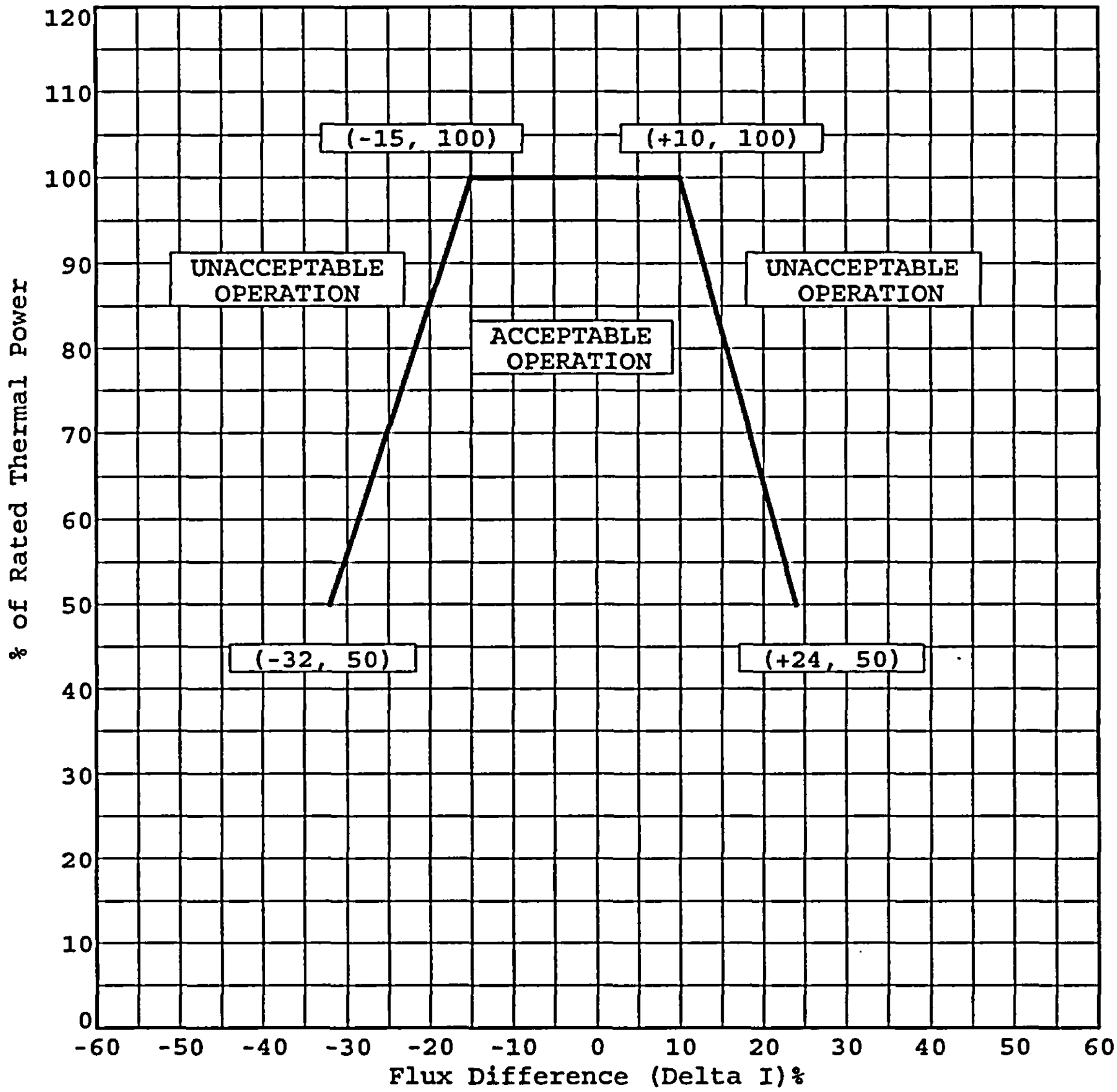
FIGURE 4.1-2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF

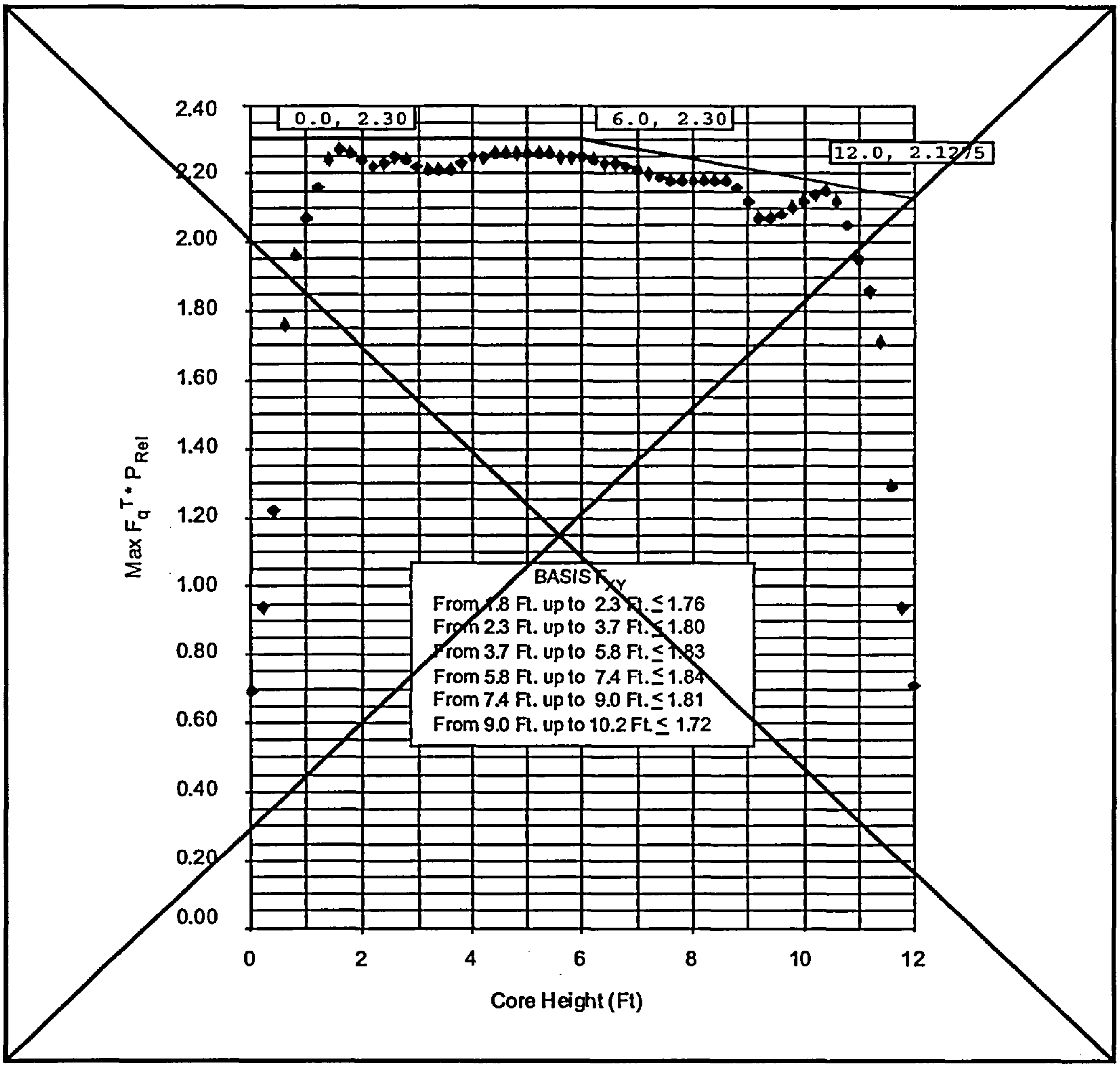
PERCENT OF RATED THERMAL POWER FOR RAOC

Provided for Information Only.

Insert C2-1.



Provided for Information Only.



~~FIGURE 4.1-4~~

~~MAXIMUM (F$_Q$T*Prel) VS. AXIAL CORE HEIGHT~~

~~DURING NORMAL OPERATION~~

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This page contains changes, shown double underlined, associated with LAR 173.

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Specification 3.3.1.1 Reactor Trip System Instrumentation Setpoints, Table 3.3-1 Table Notations A and B

Overtemperature ΔT Setpoint Parameter Values:

Parameter	Value
Overtemperature ΔT reactor trip setpoint	K1 ≤ ~~1.311~~1.239
Overtemperature ΔT reactor trip setpoint Tavg coefficient	K2 ≥ 0.0183/°F
Overtemperature ΔT reactor trip setpoint pressure coefficient	K3 ≥ ~~0.00082~~0.001/psia
Tavg at RATED THERMAL POWER	T' ≤ ~~576.2~~580.0°F
Nominal pressurizer pressure	P' ≥ 2250 psia
Measured reactor vessel ΔT lead/lag time constants (* The response time is toggled off to meet the analysis value of zero.)	τ_1 ~~≥ 8~~ = 0 sec* τ_2 ~~≤ 3~~ = 0 sec*
Measured reactor vessel ΔT lag time constant	τ_3 ≤ ~~0~~6 sec
Measured reactor vessel average temperature lead/lag time constants	τ_4 ≥ 30 sec τ_5 ≤ 4 sec
Measured reactor vessel average temperature lag time constant	τ_6 ≤ ~~0~~2 sec

f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

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(i) For q_t - q_b between ~~-32~~48% and ~~+11~~10%, $f_1(\Delta I)$ = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

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(ii) For each percent that the magnitude of q_t - q_b exceeds ~~-32~~48%, the ΔT Trip Setpoint shall be automatically reduced by ~~1.46~~4.67% of its value at RATED THERMAL POWER; and

2.52

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(iii) For each percent that the magnitude of q_t - q_b exceeds ~~+11~~10%, the ΔT Trip Setpoint shall be automatically reduced by ~~1.56~~1.47% of its value at RATED THERMAL POWER.

Replace with Insert C2-2.

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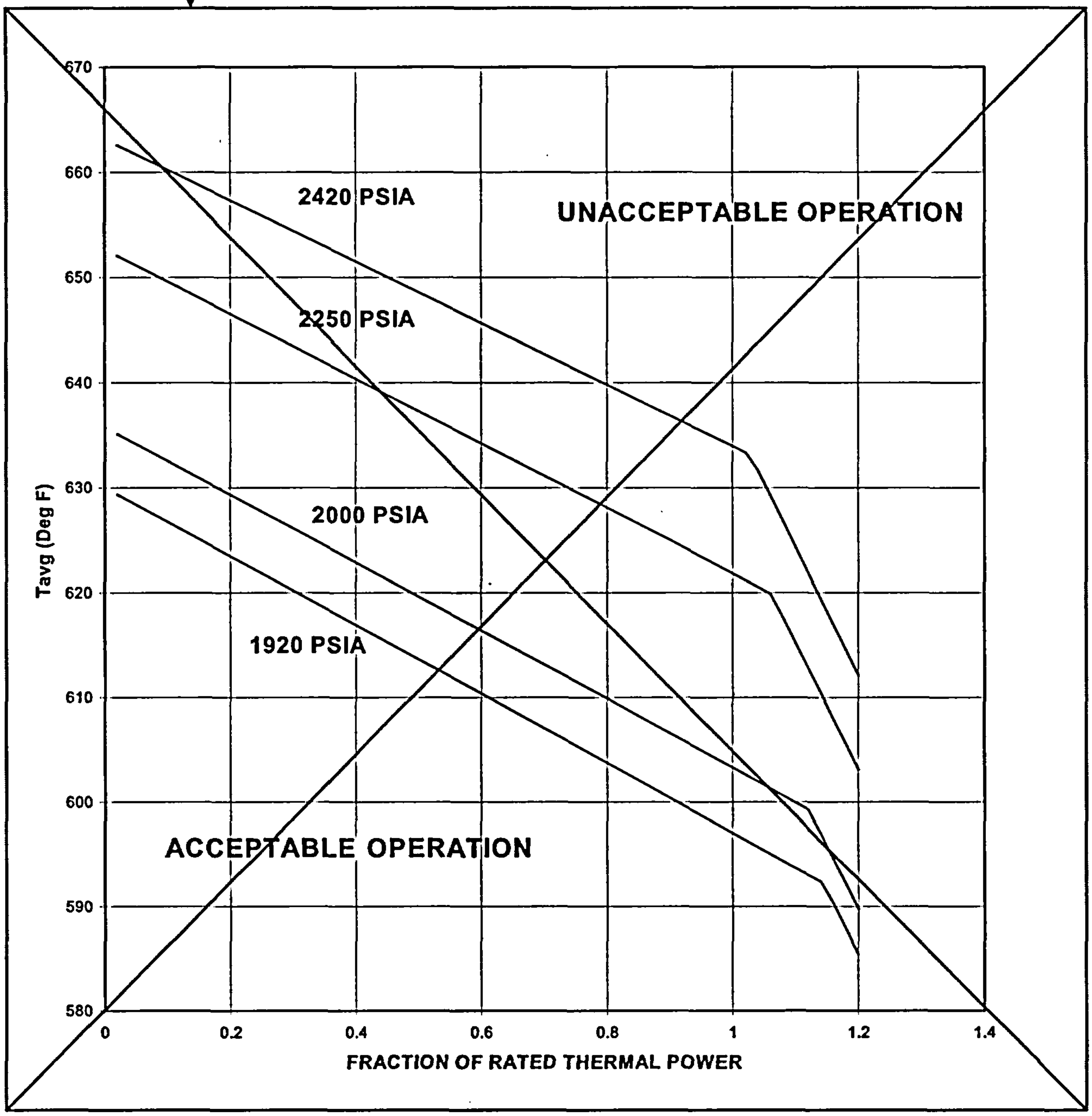

Figure 4.1-5
REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION
(Technical Specification Safety Limit 2.1.1)

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Insert C2-2.

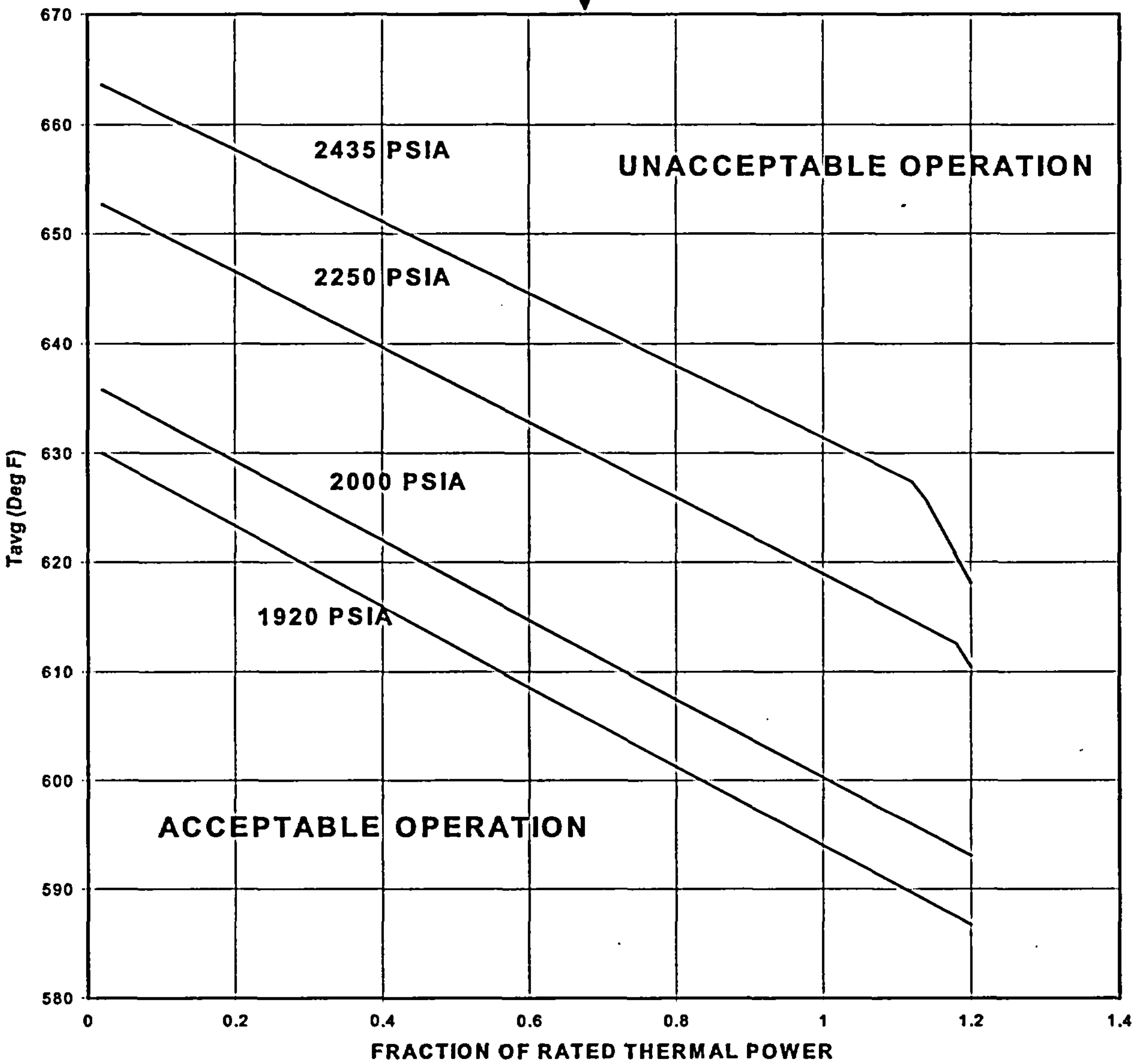

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Table 4.1-1 (Page 1 of 2)
W(Z) Values

Exclusion Zone	Axial Point	Elevation (Ft)	150 MWD/MTU	3000 MWD/MTU	10000 MWD/MTU	18000 MWD/MTU
x	61	0.00	1.0000	1.0000	1.0000	1.0000
x	60	0.20	1.0000	1.0000	1.0000	1.0000
x	59	0.40	1.0000	1.0000	1.0000	1.0000
x	58	0.60	1.0000	1.0000	1.0000	1.0000
x	57	0.80	1.0000	1.0000	1.0000	1.0000
x	56	1.00	1.0000	1.0000	1.0000	1.0000
x	55	1.20	1.0000	1.0000	1.0000	1.0000
x	54	1.40	1.0000	1.0000	1.0000	1.0000
x	53	1.60	1.0000	1.0000	1.0000	1.0000
	52	1.80	1.3782	1.3227	1.2640	1.2794
	51	2.00	1.3543	1.3023	1.2457	1.2616
	50	2.20	1.3296	1.2829	1.2267	1.2430
	49	2.40	1.3026	1.2630	1.2074	1.2237
	48	2.60	1.2802	1.2425	1.1877	1.2041
	47	2.80	1.2668	1.2220	1.1683	1.1840
	46	3.00	1.2562	1.2032	1.1518	1.1646
	45	3.20	1.2449	1.1904	1.1423	1.1586
	44	3.40	1.2334	1.1821	1.1385	1.1576
	43	3.60	1.2240	1.1735	1.1353	1.1553
	42	3.80	1.2163	1.1640	1.1315	1.1581
	41	4.00	1.2098	1.1573	1.1275	1.1638
	40	4.20	1.2037	1.1540	1.1246	1.1682
	39	4.40	1.1967	1.1497	1.1234	1.1718
	38	4.60	1.1890	1.1445	1.1221	1.1742
	37	4.80	1.1805	1.1389	1.1201	1.1756
	36	5.00	1.1713	1.1327	1.1176	1.1757
	35	5.20	1.1619	1.1264	1.1134	1.1740
	34	5.40	1.1504	1.1176	1.1135	1.1736
	33	5.60	1.1432	1.1113	1.1218	1.1796
	32	5.80	1.1514	1.1187	1.1355	1.1947
	31	6.00	1.1608	1.1293	1.1524	1.2096

Note: Top and Bottom 15% Excluded

Table 4.1-1 (Page 2 of 2)
W(Z) Values

Exclusion Zone	Axial Point	Elevation (Ft)	150 MWD/MTU	3000 MWD/MTU	10000 MWD/MTU	18000 MWD/MTU
	30	6.20	1.1669	1.1417	1.1697	1.2215
	29	6.40	1.1726	1.1527	1.1855	1.2319
	28	6.60	1.1769	1.1626	1.1999	1.2402
	27	6.80	1.1797	1.1714	1.2129	1.2468
	26	7.00	1.1812	1.1786	1.2242	1.2526
	25	7.20	1.1813	1.1841	1.2336	1.2583
	24	7.40	1.1822	1.1904	1.2410	1.2621
	23	7.60	1.1827	1.1963	1.2462	1.2632
	22	7.80	1.1811	1.2003	1.2491	1.2621
	21	8.00	1.1780	1.2024	1.2496	1.2585
	20	8.20	1.1732	1.2031	1.2476	1.2526
	19	8.40	1.1667	1.2015	1.2429	1.2443
	18	8.60	1.1583	1.1978	1.2356	1.2337
	17	8.80	1.1501	1.1971	1.2268	1.2213
	16	9.00	1.1485	1.2035	1.2259	1.2056
	15	9.20	1.1544	1.2163	1.2348	1.1926
	14	9.40	1.1580	1.2296	1.2410	1.1976
	13	9.60	1.1622	1.2414	1.2504	1.2087
	12	9.80	1.1680	1.2528	1.2653	1.2184
	11	10.00	1.1726	1.2648	1.2817	1.2278
	10	10.20	1.1736	1.2753	1.2965	1.2369
x	9	10.40	1.0000	1.0000	1.0000	1.0000
x	8	10.60	1.0000	1.0000	1.0000	1.0000
x	7	10.80	1.0000	1.0000	1.0000	1.0000
x	6	11.00	1.0000	1.0000	1.0000	1.0000
x	5	11.20	1.0000	1.0000	1.0000	1.0000
x	4	11.40	1.0000	1.0000	1.0000	1.0000
x	3	11.60	1.0000	1.0000	1.0000	1.0000
x	2	11.80	1.0000	1.0000	1.0000	1.0000
x	1	12.00	1.0000	1.0000	1.0000	1.0000

Note: Top and Bottom 15% Excluded

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Table 4.1-2
$F_Q(Z)$ Penalty Factor

Cycle Burnup (MWD/MTU)	$F_Q(Z)$ Penalty Factor
All Burnups	1.02

Note: The Penalty Factor, to be applied to $F_Q(Z)$ in accordance with Technical Specification Surveillance Requirement 4.2.2.3, is the maximum factor by which $F_Q(Z)$ is expected to increase over a 39 Effective Full Power Day (EFPD) interval (surveillance interval of 31 EFPD plus the maximum allowable extension not to exceed 25% of the surveillance interval per Technical Specification Surveillance Requirement 4.0.2) starting from the burnup at which the $F_Q(Z)$ was determined.

B.3.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs."

B.3.4 AXIAL FLUX DIFFERENCE (AFD) MONITOR ALARM

Surveillance of the AFD verifies that the AFD, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. During operation above ~~15~~50% RATED THERMAL POWER, when the AFD monitor alarm is inoperable, additional surveillance criteria is required by the Licensing Requirements Manual beyond the surveillance criteria required by the Technical Specifications to detect operation outside of the limits.~~target band and to compute the penalty deviation time before corrective action is required. The logged values of the AFD are assumed to exist for the preceding time interval in order for the operator to compute the cumulative penalty deviation time.~~

B.3.5 QUADRANT POWER TILT RATIO (QPTR) MONITOR ALARM

Surveillance of the QPTR verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. During operation above 50% RATED THERMAL POWER, when the QPTR monitor alarm is inoperable, additional surveillance criteria is required by the Licensing Requirements Manual beyond the surveillance criteria required by the Technical Specifications to detect any relatively slow changes in QPTR. For those causes of core power tilt that occur quickly (e.g., a dropped rod), there are other indications of abnormality that prompt a verification of core power tilt.

B.3.8 LEADING EDGE FLOW METER (Continued)

The Applicability Statement applies when performing calorimetric power measurements during MODE 1 operations at steady-state conditions above 2652 MWt. The Operating License limits the maximum steady state power to 2689 MWt when calorimetric heat balance measurements are made daily using the LEFM.

If the LEFM is not OPERABLE during the interval between required calorimetric heat balance measurements, plant operation may continue at ≤ 2689 MWt steady-state, using the existing Nuclear Instrumentation System (NIS) indication until the next required performance of the daily power calorimetric surveillance is due.

If the LEFM remains inoperable at the time that the next required calorimetric heat balance measurement is due, plant operation may continue at ≤ 2652 MWt steady-state, by making calorimetric measurements using feedwater flow venturis and Resistance Temperature Detector (RTD) indications. The requirement to reduce power within one hour is based upon comparison to similar action statements in the technical specifications. The increase in likelihood that the NIS will need renormalizing after 25 hours compared to after 24 hours is considered negligible. A Note, designated by "*", is added to the Licensing Requirement to denote a difference between power measurements obtained when using the feedwater flow venturis and the LEFM. An indication of 2652 MWt from the LEFM is equivalent to an indication of 2612 MWt from the feedwater flow venturis.

It is preferable that the daily heat balance calculations be made using the subroutine on the plant computer system (PCS). If the PCS is unavailable, a manual calculation that accounts for steam generator blowdown is acceptable, and may be performed in lieu of using the PCS.

This surveillance is performed every 24 hours when power is above ~~15~~50%. The NIS excore power range channel indications are renormalized if they are not found to be within ±2% of the calorimetric measurement. This ±2% requirement for renormalization is distinct from the allowance for calorimetric uncertainty, and these allowances are handled as independent contributions to determine the maximum power assumed in design basis accident analyses.

The plant may then be run for the next 24-hour period using this normalized NIS indication. Although calorimetric power indication may be monitored continuously, it is not required to be consulted again until the required daily calorimetric comparisons of NIS indication are performed.

The surveillance requirement to perform planned maintenance and inspections every 18 months is based upon the manufacturer's recommendations, and is consistent with the surveillance intervals specified for similar electronic apparatus.

Additional guidance for determining steady-state THERMAL POWER is taken from the NRC Inspection Manual; Inspection Procedure 61706; C/N 86-036, 07/14/1986; "Core Thermal Power Evaluation"; step 03.02.d, and is described in the BVPS Operating Manual.