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Constellation Energy
R.E. Ginna Nuclear Power Plant

April 29, 2005

Ms. Donna M. Skay
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Subject: License Amendment Request Regarding Adoption of Relaxed Axial Offset Control (RAOC)
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Ms. Skay:

In accordance with the provisions of 10 CFR 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) is submitting a request for a license amendment to modify the Technical Specifications (TS) for the R.E. Ginna Nuclear Power Plant.

The proposed amendment would revise the TS to allow the use of the Relaxed Axial Offset Control (RAOC) methodology for certain Power Distribution Limits. The use of the RAOC methodology permits reduced operator actions to maintain compliance with power distribution control TS. The proposed changes to use RAOC are being requested to accommodate the planned power uprate. These changes are consistent with applicable RAOC requirements specified in Revision 3 to NUREG-1431, Standard Technical Specifications Westinghouse Plants, with the exception of one surveillance in LCO 3.2.4, Quadrant Power Tilt Ratio, which will be maintained consistent with the current TS.

It has been determined that this amendment application does not involve a significant hazard consideration as determined by 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

Enclosure 1 provides a description and assessment of the proposed changes. Enclosure 2 provides the existing TS pages marked up to show the proposed changes. Enclosure 3 provides revised (clean) TS pages. Enclosure 4 provides the existing TS Bases pages marked up to reflect

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the proposed changes (for information only). Changes to the TS Bases will be provided in a future update in accordance with the Bases Control Program. Enclosure 5 provides a list of regulatory commitments associated with this license amendment request.

Approval of this amendment application is requested by April 30, 2006 to provide adequate time to prepare for implementation. Once approved, the amendment will be implemented prior to startup from the fall 2006 refueling outage.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated New York State Official.

If you have any questions regarding this submittal, please contact George Wrobel, Nuclear Safety and Licensing at (585) 771-3535.

Very truly yours,

Mary G. Korsnick

Mary G. Korsnick

Enclosures:
1. Evaluation of Proposed Change
2. Proposed Technical Specification Changes (markup)
3. Revised Technical Specification Pages (retyped)
4. Marked-up Copy of Technical Specification Bases
5. List of Regulatory Commitments

STATE OF NEW YORK :
: TO WIT:
COUNTY OF WAYNE :

I, Mary G. Korsnick, being duly sworn, state that I am Vice President - R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this response on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Mary G. Korsnick

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Monroe, this 29 day of April, 2005.

WITNESS my Hand and Notarial Seal: Michalene A. Bunts
Notary Public

My Commission Expires:

1-11-2007
Date

MICHALENE A BUNTS
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Registration No. 01BU6018576
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ENCLOSURE 1
R.E. Ginna Nuclear Power Plant

Description and Assessment of Proposed Change

Subject: Revision to Technical Specifications 3.2.1, 3.2.3, 3.2.4, 3.3.1 and 5.6.5 to Adopt Relaxed Axial Offset Control Methodologies for Power Distribution Limits

1.0 DESCRIPTION

This letter is a request to amend Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. The proposed changes will revise the Operating License 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 (TS) and to increase the ability to return to power after a plant trip or transient while still maintaining margin to safety limits under all operating conditions.

The changes proposed to TS 3.2.1, Heat Flux Hot Channel Factor $F_Q(Z)$; 3.2.3, Axial Flux Difference; and 5.6.5, Core Operating Limits Report (COLR) are being made to adopt the RAOC calculational procedure of the Standard Technical Specifications (STS) (NUREG-1431, "Standard Westinghouse Technical Specifications Westinghouse Plants"). Changes to TS 3.2.4, Quadrant Power Tilt Ratio (QPTR) are made to provide necessary consistency with the changes made to TS 3.2.1 and TS 3.2.3. Changes to TS 3.3.1 are made to accommodate the change to the RAOC methodologies by revising the form of the $f(\Delta I)$ penalty for overtemperature ΔT and for overpower ΔT (initially set to 0).

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 FQ - Surveillance Technical Specification. The NRC has found this WCAP acceptable for referencing in license applications.

Approval of the amendment is requested by April 30, 2006 to provide adequate time to prepare for implementation. Once approved, the amendment will be implemented prior to startup from the fall 2006 refueling outage.

2.0 PROPOSED CHANGES

This request proposes to modify the Ginna TS by (1) replacing existing specification 3.2.1, Heat Flux Hot Channel ($F_Q(Z)$) methodologies with specification 3.2.1B, Heat Flux Hot Channel ($F_Q(Z)$) from STS that is based on RAOC methodologies; (2) replacing existing specification 3.2.3, Axial Flux Difference (AFD) with specification 3.2.3B, Axial Flux Difference (AFD) from STS that is based on RAOC methodologies and to relocate the requirements for the AFD monitor alarm from the TS; (3) modifying existing specification 3.2.4, Quadrant Power Tilt Ratio to accommodate the changes made to specifications 3.2.1 and 3.2.3 and to relocate the requirements for the QPTR alarm monitor from the TS; (4) modifying existing specification 3.3.1 to accommodate the use of RAOC methodologies and (5) revise the listing of analysis methodologies contained in TS 5.6.5, Core Operating Limits Report (COLR) to include the references for the RAOC methodology. These changes are necessary to accommodate the planned power uprate and are supported by analyses performed at the planned uprated power level.

Accordingly, the analyses at the planned uprated power level bound operation at the current power level.

An item-by-item list of changes for the proposed changes to specifications 3.2.1 and 3.2.3 is not provided since the existing specifications are being replaced in their entirety with the appropriate STS specifications that are applicable for RAOC methodologies except for the portions of the existing specification 3.2.3 that are associated with the AFD monitor alarm. The existing requirements for the AFD monitor alarm are being relocated to the Technical Requirements Manual (TRM). The STS RAOC specifications 3.2.1B and 3.2.3B are being adopted with no technical deviations from the STS requirements. Editorial and presentation changes are made to the STS specifications for consistency with the Ginna TS.

The proposed changes revise 3.2.4 Quadrant Power Tilt Ration (QPTR) as follows:

(a) The LCO currently states "The QPTR monitor alarm shall be OPERABLE and QPTR shall be ≤ 1.02."

The LCO is being revised to state "The QPTR shall be ≤ 1.02." The requirement in the LCO for the QPTR monitor alarm is being removed from the TS and relocated to the Technical Requirements Manual (TRM).

(b) Required Action A.1 currently states "Limit THERMAL POWER to ≥ 3% below RTP for each 1% of QPTR > 1.00."

Required Action A.1 is being revised to state "Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00."

(c) The Completion Time for Required Action A.1 currently states "2 hours."

The Completion Time for Required Action A.1 is being revised to state "2 hours after each QPTR determination."

(d) Required Action A.2 currently states "Perform SR 3.2.4.2 and limit THERMAL POWER to ≥ 3% below RTP for each 1% of QPTR > 1.00."

Required Action A.2 is being revised to state "Determine QPTR."

(e) Required Action A.3 currently states "Perform SR 3.2.1.1 and SR 3.2.2.1."

Required Action A.3 is being revised to state "Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1."

(f) The first Completion Time for Required Action A.3 currently states "24 hours."

The first Completion Time for Required Action A.3 is being revised to state "24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1."

(g) Required Action A.5 currently states "Normalize excore detector instrumentation to eliminate tilt."

Required Action A.5 is being revised to state "Normalize excore detectors to restore QPTR to within limit."

(h) A second note to Required Action A.5 is being added that states "Required Action A.6 shall be completed whenever Required Action A.5 is performed."

(i) The Completion Time for Required Action A.5 currently states "Prior to increasing THERMAL POWER above the limit of Required Actions A.1 and A.2."

The Completion Time for Required Action A.5 is being revised to state "Prior to increasing THERMAL POWER above the limit of Required Action A.1."

(j) The three existing notes for Required Action A.6 are being replaced with one note that states "Perform Required Action A.6 only after Required Action A.5 is completed."

(k) Required Action A.6 currently states "Perform SR 3.2.1.1 and SR 3.2.2.1."

Required Action A.6 is being revised to state "Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1."

(l) The first Completion Time for Required Action A.6 currently states "Within 24 hours after reaching RTP."

The first Completion Time for Required Action A.6 is being revised to state "Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1."

(m) The second Completion Time for Required Action A.6 is being deleted.

(n) Action C is being removed from the TS and relocated to the Technical Requirements Manual (TRM).

(o) SR 3.2.4.1 is being removed from the TS and relocated to the Technical Requirements Manual (TRM). Existing SR 3.2.4.2 is renumbered as SR 3.2.4.1.

(p) The first note to existing SR 3.2.4.2 currently states, "With one power range channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR."

The first note to existing SR 3.2.4.2 is being revised to state "With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR."

(q) The second note to existing SR 3.2.4.2 currently states "With one power range channel inoperable and THERMAL POWER ≥ 75% RTP, perform SR 3.2.1.2 and SR 3.2.2.2."

The second note to existing SR 3.2.4.2 is being revised to state "SR 3.2.4.2 may be performed in lieu of this Surveillance."

(r) Existing SR 3.2.4.3 is being removed from the TS.

(s) A new SR 3.2.4.2 with a 24 hour frequency is being added that states "Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1."

This new SR includes a Note that states "Not required to be performed until 24 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP."

The proposed changes revise 3.3.1 Reactor Trip System (RTS) Instrumentation as follows:

(a) Table 3.3.1-1 function 6, Overpower ΔT, currently includes SR 3.3.1.3 and SR 3.3.1.6 as surveillance requirements applicable to this function.

SR 3.3.1.3 and SR 3.3.1.6 are being deleted from the list of surveillance requirements applicable to Table 3.3.1-2 function 6, Overpower ΔT.

(b) Note 1 to Table 3.3.1-1 currently includes the equation for Overtemperature ΔT. The equation states,

'Overtemperature $\Delta T \leq \Delta T_0 \{K_1 + K_2 (P-P') - K_3 (T-T') [(1+\tau_1 s) / (1+\tau_2 s)] - f(\Delta I)\}$'

The equation for Overtemperature ΔT is being revised to modify the $f(\Delta I)$ term. The revised equation now states,

'Overtemperature $\Delta T \leq \Delta T_0 \{K_1 + K_2 (P-P') - K_3 (T-T') [(1+\tau_1 s) / (1+\tau_2 s)] - f_1(\Delta I)\}$'

(c) Note 1 to Table 3.3.1-1 currently includes two (2) equations for the $f(\Delta I)$ penalty for Overtemperature ΔT. The equations currently state,

"$f(\Delta I) = 0$ when $q_t - q_b$ is $\leq$ [*]% RTP
$f(\Delta I) =$ [*] $\{(q_t - q_b) -$ [*]$\}$ when $q_t - q_b$ is $>$ [*]% RTP'

The two (2) equations for $f(\Delta I)$ are being replaced with the following three (3) equations for $f_1(\Delta I)$:

'$f_1(\Delta I) =$ [*] $\{$[*] $- (q_t - q_b)\}$ when $q_t - q_b \leq$ [*]% RTP
$f_1 (\Delta I) = 0$ % of RTP when [*] % RTP $< q_t - q_b \leq$ [*]% RTP
$f_1 (\Delta I) =$ [*] $\{(q_t - q_b) -$ [*]$\}$ when $q_t - q_b >$ [*]% RTP'

(d) Note 2 to Table 3.3.1-1 currently includes the equation for Overpower ΔT. The equation states,

'Overpower $\Delta T = \Delta T_o \{K_4 - K_5 (T-T') - K_6 [(\tau_3 s T) / (\tau_3 s+1)] - f(\Delta I)\}$'

The equation for Overpower ΔT is being revised to modify the $f(\Delta I)$ term. The revised equation now states,

'Overpower $\Delta T = \Delta T_o \{K_4 - K_5 (T-T') - K_6 [(\tau_3 s T) / (\tau_3 s+1)] - f_2(\Delta I\ \}$'

(e) Note 2 to Table 3.3.1-1 currently includes two (2) equations for the $f(\Delta I)$ penalty for Overpower ΔT

The equations currently state,

'$f(\Delta I) =$ [*] when $q_t - q_b \leq$ [*] %RTP
$f(\Delta I) =$ [*] $\{(q_t - q_b) -$ [*]$\}$ when $q_t - q_b >$ [*] %RTP

The two (2) equations for $f(\Delta I)$ are being replaced with the following one (1) equation for $f_2(\Delta I)$

$f_2(\Delta I) =$ [*]

The proposed changes revise 5.6.5 Core Operating Limits Report (COLR) as follows:

(t) 5.6.5.b item 3 currently states'WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974. (Methodology for LCO 3.2.3.)

5.6.5.b item 3 is being revised to state"WCAP-10216-P-A, Rev . 1A, "Relaxation of Constant Axial Offset Control / FQ Surveillance Technical Specification," February 1994. (Methodology for LCO 3.2.1 and LCO 3.2.3.)"

Changes to the TS Bases are provided in markup form as Enclosure 4 (for information only). There are no associated Bases for TS 5.6.5. The changes to the Bases include necessary revisions that incorporate the proposed revised Specification verbiage, revise descriptive information to reflect plant specific terminology, wording preferences and design information. The proposed TS changes and associated Bases changes are consistent with STS.

3.0 BACKGROUND

Axial power distribution control at Ginna is currently achieved by the Constant Axial Offset Control (CAOC) methodology. This methodology was developed and described in WCAP-8385 and WCAP-8403. This strategy assures peaking factors and departure from nucleate boiling (DNB) remains below the accident analysis limits. The CAOC methodology does this by maintaining the axial power distribution within a band of +5%, -5% ΔI, for Ginna 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 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 (i.e., normal operation) influences the possible power shapes and could result in violations of the kw/ft limit during Condition II transients (i.e., Faults of moderate frequency).

An F_Q surveillance requirement based upon the most limiting F_Q at each core elevation is presently incorporated into TS 3.2.1, Heat Flux Hot Channel Factor. The CAOC methodology is presently incorporated into TS 3.2.3, Axial Flux Difference. TS 3.2.4, Quadrant Power Tilt Ratio (QPTR) refers to the TS 3.2.1 $F_Q(Z)$ surveillance requirements. Application of the RAOC and F_Q surveillance methodologies requires the alteration of these TS. A change to TS 5.6.5, CORE OPERATING LIMITS REPORT (COLR), is also required to provide the methodology change.

4.0 TECHNICAL ANALYSIS

The proposed changes to the overtemperature ΔT equation and the $f(\Delta I)$ equations used in the determination of the overtemperature ΔT function and the proposed changes to the overpower ΔT equation and the $f(\Delta I)$ equations used in the determination of the overpower ΔT function are necessary to accommodate the change to the RAOC methodologies and are based on NRC approved analytical methods (WCAP-8745, "Design Basis for the Thermal Overpower Delta T and Thermal Overtemperature Delta T Trip Functions," March 1977). These changes are consistent with NUREG-1431, Revision 3. NUREG-1431, Revision 3 has been approved for use by the NRC.

The overpower ΔT function does not require a $f(\Delta I)$ input to the function so the value for $f_2(\Delta I)$ will be zero (0) in the COLR for this parameter. SR 3.3.1.3 and SR 3.3.1.6 are surveillances associated with determination of $f(\Delta I)$. These SRs are no longer necessary for function 6, the overpower ΔT function, since $f_2(\Delta I)$ will be set to zero (0) for the overpower ΔT function.

The current TS 3.2.1, Heat Flux Hot Channel Factor and TS 3.2.3, Axial Flux Difference (AFD) are being replaced in their entirety, except for the portions of the existing specification 3.2.3 that are associated with the AFD monitor. The existing requirements for the AFD monitor alarm are being relocated to the Technical Requirements Manual (TRM). The revised specification 3.2.1 and 3.2.3 are consistent with the corresponding specifications in NUREG-1431, Rev. 3 that are appropriate for use with the RAOC methodology. These NUREG-1431, Revision 3, RAOC specific specifications have been approved by the NRC for use in this application. Changes to the TRM are subject to the requirements of 10 CFR 50.59. The TRM revision process and 10 CFR 50.59 requirements provide adequate control for any changes to these AFD monitor alarm requirements.

The current TS 3.2.4, Quadrant Power Tilt (QPTR) is being revised to relocate requirements for the QPTR monitor alarm from the TS to the TRM. TS 3.2.4 is also being revised to accommodate the adoption of the RAOC for TS 3.2.1 and TS 3.2.3. The changes to TS 3.2.4 are consistent with NUREG-1431, Revision 3 except for SR 3.2.4.2, which will be maintained consistent with the current TS requirements. This amendment request proposes to require the performance of SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1, at a frequency of 24 hours, in lieu of the standard SR 3.2.4.2. The SR 3.2.4.2 specified in the NUREG is based on a capability to use two sets of four thimble locations with quarter core symmetry. The symmetric thimble locations could be used to generate symmetric thimble "tilt" which could then be compared to a reference symmetric thimble "tilt," from the most recent full core flux map to generate an incore QPTR. Ginna does not have two sets of four thimble locations with quarter core symmetry which are necessary to generate a partial flux map and thus would have to generate a full core flux map. The proposed surveillance and 24 hour frequency for SR 3.2.4.2 reflects the additional time necessary to generate the full core flux map and is consistent with that specified for the comparable requirement (existing SR 3.2.4.3) in the current TS. The proposed SR 3.2.4.2 is only required to be performed when input from one of more power range neutron flux channel are inoperable with THERMAL POWER > 75% RTP. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing a full core flux map provides an accurate means for ensuring that the core power distribution is consistent with the safety analyses. The proposed Frequency of 24 hours is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt. NUREG-1431, Revision 3 has been approved for use by the NRC. Changes to the TRM are subject to the requirements of 10 CFR 50.59. The TRM revision process and 10 CFR 50.59 requirements provide adequate control for any changes to these QPTR monitor alarm requirements.

The proposed changes to TS 5.6.5 are being made to be consistent with the RAOC methodology and the STS. A reference to WCAP-10216-P-A, Revision 1A is being added because it is applicable to the RAOC methodology. The reference to WCAP-8385 is being deleted because it is applicable to the CAOC methodology, not the RAOC methodology.

The implementation of RAOC and F_Q surveillance methodologies have been previously developed and approved by the NRC and documented in WCAP-10216-P-A, Rev. 1A. 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_Q surveillance requirement that currently exist.

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 with a high probability.

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 TS 3.2.1 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 as described in TS 3.1.4, "Rod Group Alignment Limits."

(b) Control rod groups are sequenced with overlapping groups as described in TS 3.1.6, "Control Bank Insertion Limits."

(c) The rod insertion limits of TS 3.1.5, "Shutdown Bank Insertion Limit" and TS 3.1.6, "Control Bank Insertion Limits" are maintained.

(d) The axial power distribution, expressed in terms of Axial Flux Difference, is maintained within the limits as described in TS 3.2.3, "Axial Flux Difference (AFD)."

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, BU), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$, is met. The power factor, W(Z,BU), 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

- 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.

Revised overtemperature ΔT and overpower ΔT setpoints were developed during the analysis supporting the planned extended power uprate (EPU) to ensure that DNBR design criteria and the fuel temperature design basis will continue to be met. These setpoints are applicable to the change to the RAOC and F_Q surveillance methodologies. The $f(\Delta I)$ function was 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 revise the positive wing and add a 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. Changes were also made to the overpower ΔT $f(\Delta I)$ function for the RAOC and F_Q surveillances which removed all $f(\Delta I)$ penalties. The penalty will be shown as zero. The implementation of RAOC has been explicitly included in the non-LOCA EPU analyses such that the conclusions presented in the UFSAR remain valid. The supporting analyses performed for the EPU bounds operation at the current power level.

4.2 LOCA and LOCA-Related Evaluations

The change from CAOC and the current F_Q surveillance requirement to the RAOC and $F_Q(Z)$ surveillance methodologies has been analyzed in the LOCA safety analyses performed in support of the planned extended power uprate (EPU) program.

The RAOC and $F_Q(Z)$ surveillance methodologies do not affect the normal plant operating parameters except for those changed for the EPU (i.e., the axial flux difference, F_Q, the overtemperature ΔT trip setpoint, and the overpower ΔT trip setpoint 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 projected impact of the RAOC Methodology has been analyzed in both the large-break and small-break LOCA analyses. All related core design parameters will be checked against the LOCA analyses' limits on a cycle-specific basis for reload cycles which utilize the RAOC methodology.

4.3 Core Design Evaluation

The changes from CAOC and the current F_Q surveillance requirement to RAOC and $F_Q(Z)$ surveillance methodologies have been evaluated for impact upon the Ginna core design. Consistent with the approved RAOC methodology, the Condition I axial power shapes were analyzed to demonstrate compliance with the revised 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 EPU analyses were made based on these evaluations. Revised overtemperature ΔT and overpower ΔT setpoints were developed during the EPU analysis and applicable to the RAOC and $F_Q(Z)$ surveillance methodologies to ensure that DNBR design criteria will continue to be met.

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 internal pressure of the fuel rods and the cladding transient stress and transient strain. Fuel performance limits were demonstrated to be able to be met under RAOC operation. Compliance with the safety analysis assumptions are 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 Ginna while still meeting corresponding core design bases and limits.

4.4 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 except for those changed for the EPU (i.e., the axial flux difference, F_Q, the overtemperature ΔT trip setpoint and the overpower ΔT trip setpoint parameters), protection system actuation, the safeguard system actuation, or any other plant capability important to the mitigation of a Non-LOCA or LOCA accident. Potential changes to normal plant operating parameters associated with the EPU will be addressed separately in the EPU submittal. The supporting analyses performed for the EPU bounds operation at the current power level.

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

Ginna LLC 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. Do the proposed changes 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 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 configurations, 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. Do the proposed changes 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 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. Do the proposed changes 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. The appropriate acceptance criteria for the various analyses and evaluations will continue to be met.

The projected impact associated with the implementation of RAOC on peak cladding temperature (PCT) has been been incorporated into the LOCA analyses for the planned extended power uprate. It has determined that implementation of RAOC at the extended power uprate power level does not result in a significant reduction in a margin of safety. The analysis performed for EPU bounds operation at the current power level.

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

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

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

A review of 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants 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.

GDC 10 Reactor Design states, "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."

UFSAR Discussion/Impact

The UFSAR contains analyses of accidents for the Axial Flux Difference parameter. The most important (limiting) Condition II events are the uncontrolled bank withdrawal, cooldown and boration/dilution accidents. The most important (limiting) Condition III and IV events are the loss of flow accident and LOCA, respectively. 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 accordance with 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

Ginna LLC has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration; and

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite; and

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

7.0 REFERENCES

None

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
----NOTE---- Required Action B.4 shall be completed whenever this Condition is entered.	B.1	Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	<u>AND</u>		
B. $F_Q^W(Z)$ not within limits.	B.2	Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	<u>AND</u>		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	<u>AND</u>		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

NOTE

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.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 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 31 EFPD thereafter

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.2.1.2	---NOTE--- If measurements indicate that the maximum over z [$F_Q^C(Z)$ / K(Z)] has increased since the previous evaluation of $F_Q^C(Z)$: 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 SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the maximum over z [$F_Q^C(Z)$ / K(Z)] has not increased. --- Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 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 <u>AND</u> 31 EFPD thereafter

Replace this spec except for AFD monitor alarm requirements with Spec 3.2.3B from NUREG-1431 including mark up changes.

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

Relocate to TRM.

LCO 3.2.3 ~~The AFD monitor alarm shall be OPERABLE~~ and AFD:

a. Shall be maintained within the target band about the target flux difference with THERMAL POWER ≥ 90% RTP. The target band is specified in the COLR.

b. May deviate outside the target band with THERMAL POWER < 90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

- NOTE -

1. The AFD shall be considered outside the target band when the average of four OPERABLE excore channels indicate AFD to be outside the target band. If one excore detector is out of service, the remaining three detectors shall be used to derive the average.

2. Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with THERMAL POWER ≥ 50% RTP, and AFD outside the target band.

3. Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with THERMAL POWER > 15% RTP and < 50% RTP, and AFD outside the target band.

4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	THERMAL POWER ≥ 90% RTP. AND AFD not within the target band.	A.1	Restore AFD to within target band.	15 minutes
B.	Required Action and associated Completion Time of Condition A not met.	B.1	Reduce THERMAL POWER to < 90% RTP.	15 minutes
C.	THERMAL POWER < 90% RTP and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. OR THERMAL POWER < 90% RTP and ≥ 50% RTP with AFD not within the target band and not within the acceptable operation limits.	C.1	Reduce THERMAL POWER to < 50% RTP.	30 minutes
D.	THERMAL POWER ≥ 90% RTP. AND AFD monitor alarm inoperable.	D.1	Perform SR 3.2.3.2.	Once every 15 minutes
E.	THERMAL POWER < 90% RTP. AND AFD monitor alarm inoperable.	E.1	Perform SR 3.2.3.3.	Once every 1 hour

Relocate To TRM

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.3.1	Verify AFD monitor is OPERABLE.	12 hours
SR 3.2.3.2	- NOTE - 1. Only required to be performed if AFD monitor alarm is inoperable when THERMAL POWER ≥ 90% RTP. 2. Assume logged values of AFD exist during the preceding 24 hour time interval if actual AFD values are not available. Verify AFD is within limits and log AFD for each OPERABLE excore channel.	Relocate to TRM Once within 15 minutes and every 15 minutes thereafter
SR 3.2.3.3	- NOTE - 1. Only required to be performed if AFD monitor alarm is inoperable when THERMAL POWER < 90% RTP. 2. Assume logged values of AFD exist during the preceding 24 hour time interval if actual AFD values are not available. Verify AFD is within limits and log AFD for each OPERABLE excore channel.	Once within 1 hour and every 1 hour thereafter
SR 3.2.3.4	Update target flux difference.	Once within 31 EFPD after each refueling AND 31 EFPD thereafter

SURVEILLANCE		FREQUENCY
SR 3.2.3.5	- NOTE - The initial target flux difference after each refueling may be determined from design predictions. Determine, by measurement, the target flux difference.	Once within 31 EFPD after each refueling AND 92 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3~~B~~ AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC) Methodology)~~

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

---NOTE---
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR) <Relocate to TRM>

LCO 3.2.4 The ~~QPTR monitor alarm~~ shall be OPERABLE and QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 ~~Limit~~ Reduce THERMAL POWER ~~to~~ from ≥ 3% ~~below~~ RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	AND	
	A.2 ~~Perform SR 3.2.4.2~~ Determine QPTR ~~and limit THERMAL POWER to ≥ 3% below RTP for each 1% of QPTR > 1.00.~~	Once per 12 hours
	AND	
	A.3 Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 AND Once per 7 days thereafter
	AND	
	A.4 Reevaluate safety analyses and confirm results remain valid for the duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND	

CONDITION	REQUIRED ACTION	COMPLETION TIME
2. Required Action A.6 shall be completed whenever Required Action A.5 is performed.	A.5 1. - NOTE - Perform Required Action A.5 only after Required Action A.4 is completed. Normalize excore ~~detector instrumentation to eliminate tilt.~~ detectors to restore QPTR to within limit. AND	Prior to increasing THERMAL POWER above the limit of Required Action~~s~~ A.1 ~~and A.2~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
Perform Required Action A.6 only after Required Action A.5 is completed.	A.6 ----- NOTE ----- 1. ~~Only required to be performed if the cause of the QPTR alarm is not associated with inoperable QPTR instrumentation.~~ 2. ~~Required Action A.6 must be completed when Required Action A.5 is completed and Note 1, above, does not apply.~~ 3. ~~Only one of the Completion Times, whichever becomes applicable first, must be met.~~ ------------------ Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after ~~reaching RTP~~ achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 ~~OR~~ ~~Within 48 hours after increasing THERMAL POWER above the limits of Required Actions A.1 and A.2~~
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours
C. QPTR monitor alarm inoperable.	C.1 Perform SR 3.2.4.3. OR	Once within 24 hours and every 24 hours thereafter

Relocate to TRM

<Relocate to TRM>

CONDITION	REQUIRED ACTION	COMPLETION TIME
	C.2 ~~Perform SR 3.2.1.2 and SR 3.2.2.2.~~	~~Once within 24 hours and every 24 hours thereafter~~

SURVEILLANCE REQUIREMENTS

<Relocate to TRM>

SURVEILLANCE		FREQUENCY
~~SR 3.2.4.1~~	Verify QPTR monitor alarm is OPERABLE.	12 hours
SR 3.2.4.~~2~~ 1	- NOTE - 1. With ~~one power range~~ channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. With one power range channel inoperable and THERMAL POWER ≥ 75% RTP, perform SR 3.2.1.2 and SR 3.2.2.2. Verify QPTR is within limit by calculation.	7 days
~~SR 3.2.4.3~~	~~- NOTE -~~ ~~1. Only required to be performed if the QPTR monitor alarm is inoperable.~~ ~~2. With one power range channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR.~~ ~~3. With one power range channel inoperable and THERMAL POWER ≥ 75% RTP, perform SR 3.2.1.2 and SR 3.2.2.2.~~ ~~Verify QPTR is within limit by calculation.~~	Once within 24 hours and every 24 hours thereafter

≤

input from one Power Range Neutron flux

SR 3.2.4.2 may be performed in lieu of this surveillance

Insert A

Insert A page 3.2.4-4

SR 3.2.4.2	------------------------NOTE------------------------------ Not required to be performed until 24 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. ---	
	Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	24 hours

Table 3.3.1-1
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	LIMITING SAFETY SYSTEM SETTINGS[a]
6. Overpower ΔT	1, 2	4	D,G	SR 3.3.1.1 ~~SR 3.3.1.3~~ ~~SR 3.3.1.6~~ SR 3.3.1.7 SR 3.3.1.10	Refer to Note 2
7. Pressurizer Pressure					
a. Low	1[g]	4	K,L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 1791.3 psig
b. High	1, 2	3	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 2396.2 psig
8. Pressurizer Water Level-High	1, 2	3	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 96.47%
9. Reactor Coolant Flow-Low					
a. Single Loop	1[h]	3 per loop	M,O	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 89.86%
b. Two Loops	1[i]	3 per loop	K,L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 89.86%
10. Reactor Coolant Pump (RCP) Breaker Position					
a. Single Loop	1[h]	1 per RCP	N,O	SR 3.3.1.11	NA
b. Two Loops	1[i]	1 per RCP	K,L	SR 3.3.1.11	NA

Table 3.3.1-1 (Note 1)
Overtemperature ΔT

- NOTE -

The Overtemperature ΔT Function Limiting Safety System Setting is defined by:

$$\text{Overtemperature } \Delta T \leq \Delta T_0 \{K_1 + K_2 (P-P') - K_3 (T-T') [(1+\tau_1 s) / (1+\tau_2 s)] - f(\Delta I)\}$$

$f_1(\Delta I)$

Where:

ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, °F.

P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, psig.

K_1 is the Overtemperature ΔT reactor trip setpoint, [*].
K_2 is the Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, [*]/psi.
K_3 is the Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, [*]/°F.

τ_1 is the measured lead time constant, [*] seconds.
τ_2 is the measured lag time constant, [*] seconds.

$f(\Delta I)$ is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

$f(\Delta I) = 0$ % of RTP when $q_t - q_b \leq$ [*]% RTP [*] % RTP
$f(\Delta I) = [*] \{(q_t - q_b) - [*]\}$ when $q_t - q_b >$ [*]% RTP

$F_1(\Delta I)$

* These values denoted with [*] are specified in the COLR.

$F_1(\Delta I) = [*] \{[*] - (q_t - q_b)\}$ when $q_t - q_b \leq [*]$ RTP

Table 3.3.1-1 (Note 2)
Overpower ΔT

- NOTE -

1 The Overpower ΔT Function Limiting Safety System Setting is defined by:

$$\text{Overpower } \Delta T \leq \Delta T_0 \{K_4 - K_5 (T-T') - K_6 [(\tau_3 sT) / (\tau_3 s+1)] - \cancel{f(\Delta I)}\}$$

$f_2(\Delta I)$

Where:

ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, °F.

1 K_4 is the Overpower ΔT reactor trip setpoint, [*].
1 K_5 is the Overpower ΔT reactor trip heatup setpoint penalty coefficient which is:
[*]/°F for T < T' and;
[*]/°F for T ≥ T'.
1 K_6 is the Overpower ΔT reactor trip thermal time delay setpoint penalty which is:
[*]/°F for increasing T and;
[*]/°F for decreasing T.

τ_3 is the measured impulse/lag time constant, [*] seconds.

~~f(ΔI) is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.~~

1 ~~f(ΔI) = [*] when $q_t - q_b$ is ≤ [*]% RTP~~

~~f(ΔI) = [*] {($q_t - q_b$) - [*]} when $q_t - q_b$ is > [*]% RTP~~

$f_2(\Delta I) = [*]$

1 * These values denoted with [*] are specified in the COLR.

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:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
   (Methodology for LCO 3.1.1, LCO 3.1.3, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.9.1.)

2. WCAP-13677-P-A, "10 CFR 50.46 Evaluation Model Report: WCOBRA/TRAC Two-Loop Upper Plenum Injection Model Updates to Support ZIRLO™ Cladding Option," February 1994.
   (Methodology for LCO 3.2.1.)

Insert A

3. ~~WCAP-8385,~~ "Power ~~Distribution~~ Control ~~and Load~~ Following Procedures - ~~Topical~~ Report," September 1974.
   (Methodology for LCO 3.2.3.)

LCO 3.2.1 and

4. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995.
   (Methodology for LCO 3.2.1.)

5. WCAP 11397-P-A, "Revised Thermal Design Procedure," April 1989.
   (Methodology for LCO 3.4.1 when using RTDP.)

6. WCAP-10054-P-A and WCAP-10081-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
   (Methodology for LCO 3.2.1.)

7. WCAP-10924-P-A, Volume 1, Revision 1, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation Responses to NRC Questions," and Addenda 1,2,3, December 1988.
   (Methodology for LCO 3.2.1.)

8. WCAP-10924-P-A, Volume 2, Revision 2, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addendum 1, December 1988.
   (Methodology for LCO 3.2.1.)

9. WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation, Addendum 4: Model Revisions," March 1991.
   (Methodology for LCO 3.2.1.)

Insert A page 5.6-3

WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control / FQ Surveillance Technical Specification," February 1994.

ENCLOSURE 2
R.E. Ginna Nuclear Power Plant

Proposed Technical Specification Changes (markup)

Replace this spec with spec 3.2.1 B from NUREG1431 including markup changes.

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

LCO 3.2.1 $F_Q(Z)$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_Q(Z)$ not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q(Z)$ exceeds limit.	15 minutes
	<u>AND</u>	
	A.2 Reduce AFD acceptable operation limits ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	8 hours
	<u>AND</u>	
	A.3 Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	72 hours
	<u>AND</u>	
	A.4 Reduce Overpower ΔT and Overtemperature ΔT trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	72 hours
	<u>AND</u>	
	A.5 Perform SR 3.2.1.1 or SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

MAKE same changes on all pages

3.2 POWER DISTRIBUTION LIMITS

3.2.1B Heat Flux Hot Channel Factor ($F_Q(Z)$ ~~(RAOC-W(Z) Methodology)~~)

LCO 3.2.1B $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
---NOTE--- Required Action A.4 shall be completed whenever this Condition is entered.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
A. $F_Q^C(Z)$ not within limit.	<u>AND</u>	
	A.2 Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	<u>AND</u>	
	A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	<u>AND</u>	
	A.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

Make same changes on all pages

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.1.1	Verify measured values of $F_Q(Z)$ are within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter
SR 3.2.1.2	- NOTE - Only required to be performed if one power range channel is inoperable with THERMAL POWER ≥ 75% RTP. Verify measured values of $F_Q(Z)$ are within limits specified in the COLR.	Once within 24 hours and every 24 hours thereafter

ENCLOSURE 3
R.E. Ginna Nuclear Power Plant

Revised Technical Specification Pages (retyped)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
- NOTE - Required Action A.4 shall be completed whenever this Condition is entered A. $F_Q^C(Z)$ not within limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	<u>AND</u>		
	A.2	Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	<u>AND</u>		
	A.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	<u>AND</u>		
	A.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

CONDITION	REQUIRED ACTION		COMPLETION TIME
- NOTE - Required Action B.4 shall be completed whenever this Condition is entered. B. $F_Q^W(Z)$ not within limits.	B.1	Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	AND		
	B.2	Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	AND		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	AND		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

- NOTE -

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.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 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 31 EFPD thereafter

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	- NOTE - If measurements indicate that the maximum over z $[F_Q^C(Z) / K(Z)]$ has increased since the previous evaluation of $F_Q^C(Z)$: 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 SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the maximum over z $[F_Q^C(Z) / K(Z)]$ has not increased. - Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 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 <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits spcified in the COLR.

- NOTE -

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

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	AND	
	A.2 Determine QPTR	Once per 12 hours
	AND	
	A.3 Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 AND Once per 7 days thereafter
	AND	
	A.4 Reevaluate safety analyses and confirm results remain valid for the duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND	

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.5	------------------- - NOTE - 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed -------------------	
		Normalize excore detectors to restore QPTR to within limit.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
	A.6	------------------- - NOTE - Perform Required Action A.6 only after Required Action A.5 is completed. -------------------	
		Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time of Condition A not met.	B.1	Reduce THERMAL POWER to $\leq$ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	- NOTE - 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	- NOTE - Not required to be performed until 24 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	24 hours

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable. OR Two source range channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. As required by Required Action A.1 and referenced by Table 3.3.1-1.	B.1 Restore channel to OPERABLE status.	48 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Initiate action to fully insert all rods.	6 hours
	AND	
	C.3 Place Control Rod Drive System in a condition incapable of rod withdrawal.	7 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action A.1 and referenced by Table 3.3.1-1.	D.1	- NOTE - The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. Place channel in trip.	 6 hours
E. As required by Required Action A.1 and referenced by Table 3.3.1-1.	E.1	Reduce THERMAL POWER to < 5E-11 amps.	2 hours
	<u>OR</u>		
	E.2	- NOTE - Required Action E.2 is not applicable when: a. Two channels are inoperable, or b. THERMAL POWER is < 5E-11 amps. Increase THERMAL POWER to ≥ 8% RTP.	 2 hours
F. As required by Required Action A.1 and referenced by Table 3.3.1-1.	F.1	Open RTBs and RTBBs upon discovery of two inoperable channels.	Immediately upon discovery of two inoperable channels
	<u>AND</u>		
	F.2	Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>		
	F.3	Restore channel to OPERABLE status.	48 hours

CONDITION		REQUIRED ACTION		COMPLETION TIME
G.	Required Action and associated Completion Time of Condition D, E, or F is not met.	G.1	Be in MODE 3.	6 hours
H.	As required by Required Action A.1 and referenced by Table 3.3.1-1.	H.1	Restore at least one channel to OPERABLE status upon discovery of two inoperable channels.	1 hour from discovery of two inoperable channels
		<u>AND</u>		
		H.2	Suspend operations involving positive reactivity additions.	Immediately
		<u>AND</u>		
		H.3	Restore channel to OPERABLE status.	48 hours
I.	Required Action and associated Completion Time of Condition H not met.	I.1	Initiate action to fully insert all rods.	Immediately
		<u>AND</u>		
		I.2	Place the Control Rod Drive System in a condition incapable of rod withdrawal.	1 hour
J.	As required by Required Action A.1 and referenced by Table 3.3.1-1.	J.1	Suspend operations involving positive reactivity additions.	Immediately
		<u>AND</u>		
		J.2	Perform SR 3.1.1.1.	12 hours <u>AND</u> Once per 12 hours thereafter

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. As required by Required Action A.1 and referenced by Table 3.3.1-1.	K.1 ----- - NOTE - The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip.	6 hours
L. Required Action and associated Completion Time of Condition K not met.	L.1 Reduce THERMAL POWER to < 8.5% RTP.	6 hours
M. As required by Required Action A.1 and referenced by Table 3.3.1-1.	M.1 ----- - NOTE - The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip.	6 hours
N. As required by Required Action A.1 and referenced by Table 3.3.1-1.	N.1 Restore channel to OPERABLE status.	6 hours
O. Required Action and associated Completion Time of Condition M or N not met.	O.1 Reduce THERMAL POWER to < 50% RTP.	6 hours
P. As required by Required Action A.1 and referenced by Table 3.3.1-1.	P.1 ----- - NOTE - The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip.	6 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
Q. Required Action and Associated Completion Time of Condition P not met.	Q.1	Reduce THERMAL POWER to < 50% RTP.	6 hours
	AND		
	Q.2.1	Verify Steam Dump System is OPERABLE.	7 hours
		OR	
	Q.2.2	Reduce THERMAL POWER to < 8% RTP.	7 hours
R. As required by Required Action A.1 and referenced by Table 3.3.1-1.	R.1	- NOTE - One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE.	
		Restore train to OPERABLE status.	6 hours
S. As required by Required Action A.1 and referenced by Table 3.3.1-1.	S.1	Verify interlock is in required state for existing plant conditions.	1 hour
	OR		
	S.2	Declare associated RTS Function channel(s) inoperable.	1 hour

CONDITION	REQUIRED ACTION		COMPLETION TIME
T. As required by Required Action A.1 and referenced by Table 3.3.1-1.	T.1	- NOTE - 1. One train may be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. 2. One RTB may be bypassed for up to 6 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. Restore train to OPERABLE status.	1 hour
U. As required by Required Action A.1 and referenced by Table 3.3.1-1.	U.1	Restore at least one trip mechanism to OPERABLE status upon discovery of two RTBs with inoperable trip mechanisms.	1 hour from discovery of two inoperable trip mechanisms
	<u>AND</u>		
	U.2	Restore trip mechanism to OPERABLE status.	48 hours
V. Required Action and associated Completion Time of Condition R, S, T, or U not met.	V.1	Be in MODE 3.	6 hours
W. As required by Required Action A.1 and referenced by Table 3.3.1-1.	W.1	Restore at least one trip mechanism to OPERABLE status upon discovery of two RTBs with inoperable trip mechanisms.	1 hour from discovery of two inoperable trip mechanisms
	<u>AND</u>		

CONDITION	REQUIRED ACTION		COMPLETION TIME
	W.2	Restore trip mechanism or train to OPERABLE status.	48 hours
X. Required Action and associated Completion Time of Condition W not met.	X.1	Initiate action to fully insert all rods.	Immediately
	<u>AND</u>		
	X.2	Place the Control Rod Drive System in a Condition incapable of rod withdrawal.	1 hour

SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	- NOTE - Required to be performed within 12 hours after THERMAL POWER is ≥ 50% RTP. Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output and adjust if calorimetric power is > 2% higher than indicated NIS power.	24 hours
SR 3.3.1.3	- NOTE - 1. Required to be performed within 7 days after THERMAL POWER is ≥ 50% RTP but prior to exceeding 90% RTP following each refueling and if the Surveillance has not been performed within the last 31 EFPD. 2. Performance of SR 3.3.1.6 satisfies this SR. Compare results of the incore detector measurements to NIS AFD and adjust if absolute difference is ≥ 3%.	31 effective full power days (EFPD)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.6	- NOTE - Not required to be performed until 7 days after THERMAL POWER is ≥ 50% RTP, but prior to exceeding 90% RTP following each refueling. Calibrate excore channels to agree with incore detector measurements.	92 EFPD
SR 3.3.1.7	- NOTE - Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entering MODE 3. Perform COT.	92 days
SR 3.3.1.8	- NOTE - 1. Not required for power range and intermediate range instrumentation until 4 hours after reducing power < 6% RTP. 2. Not required for source range instrumentation until 4 hours after reducing power < 5E-11 amps. Perform COT.	92 days
SR 3.3.1.9	- NOTE - Setpoint verification is not required. Perform TADOT.	92 days

SURVEILLANCE		FREQUENCY
SR 3.3.1.10	- NOTE - Neutron detectors are excluded. Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.11	Perform TADOT.	24 months
SR 3.3.1.12	- NOTE - Setpoint verification is not required. Perform TADOT.	Prior to reactor startup if not performed within previous 31 days
SR 3.3.1.13	Perform COT.	24 months

Table 3.3.1-1
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	LIMITING SAFETY SYSTEM SETTINGS[(a)]
1. Manual Reactor Trip	1, 2, 3[(b)], 4[(b)], 5[(b)]	2	B,C	SR 3.3.1.11	NA
2. Power Range Neutron Flux					
a. High	1, 2	4	D,G	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10	≤ 112.27% RTP
b. Low	1[(c)], 2	4	D,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	≤ 29.28% RTP
3. Intermediate Range Neutron Flux	1[(c)], 2	2	E,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	(d)
4. Source Range Neutron Flux	2[(e)]	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	(d)
	3[(b)], 4[(b)], 5[(b)]	2	H,I	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	(d)
	3[(f)], 4[(f)], 5[(f)]	1	J	SR 3.3.1.1 SR 3.3.1.10	NA
5. Overtemperature ΔT	1, 2	4	D,G	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10	Refer to Note 1
6. Overpower ΔT	1, 2	4	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	Refer to Note 2

Table 3.3.1-1
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	LIMITING SAFETY SYSTEM SETTINGS(a)
7. Pressurizer Pressure					
a. Low	1(g)	4	K,L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 1791.3 psig
b. High	1, 2	3	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 2396.2 psig
8. Pressurizer Water Level-High	1, 2	3	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 96.47%
9. Reactor Coolant Flow-Low					
a. Single Loop	1(h)	3 per loop	M,O	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 89.86%
b. Two Loops	1(i)	3 per loop	K,L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 89.86%
10. Reactor Coolant Pump (RCP) Breaker Position					
a. Single Loop	1(h)	1 per RCP	N,O	SR 3.3.1.11	NA
b. Two Loops	1(i)	1 per RCP	K,L	SR 3.3.1.11	NA
11. Undervoltage-Bus 11A and 11B	1(g)	2 per bus	K,L	SR 3.3.1.9 SR 3.3.1.10	(d)
12. Underfrequency-Bus 11A and 11B	1(g)	2 per bus	K,L	SR 3.3.1.9 SR 3.3.1.10	≥ 57.5 HZ

Table 3.3.1-1
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	LIMITING SAFETY SYSTEM SETTINGS[(a)]
13. Steam Generator (SG) Water Level-Low Low	1, 2	3 per SG	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 13.88%
14. Turbine Trip					
a. Low Autostop Oil Pressure	1[(k)(l)]	3	P,Q	SR 3.3.1.10 SR 3.3.1.12	(d)
b. Turbine Stop Valve Closure	1[(k)(l)]	2	P,Q	SR 3.3.1.12	NA
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1, 2	2	R,V	SR 3.3.1.11	NA

Table 3.3.1-1
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	LIMITING SAFETY SYSTEM SETTINGS[a]
16. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2[e]	2	S,V	SR 3.3.1.10 SR 3.3.1.13	≥ 5E-11 amp
b. Low Power Reactor Trips Block, P-7	1[g]	4 (power range only)	S,V	SR 3.3.1.10 SR 3.3.1.13	≤ 8.0% RTP
c. Power Range Neutron Flux, P-8	1[h]	4	S,V	SR 3.3.1.10 SR 3.3.1.13	≤ 49.0% RTP
d. Power Range Neutron Flux, P-9	1[i]	4	S,V	SR 3.3.1.10 SR 3.3.1.13	≤ 50.0% RTP
	1[k]	4	S,V	SR 3.3.1.10 SR 3.3.1.13	≤ 8.0% RTP
e. Power Range Neutron Flux, P-10	1[c], 2	4	S,V	SR 3.3.1.10 SR 3.3.1.13	≥ 6.0% RTP
17. Reactor Trip Breakers[m]	1, 2 3[b], 4[b], 5[b]	2 trains 2 trains	T,V W,X	SR 3.3.1.4 SR 3.3.1.4	NA NA
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1, 2 3[b], 4[b], 5[b]	1 each per RTB 1 each per RTB	U,V W,X	SR 3.3.1.4 SR 3.3.1.4	NA NA
19. Automatic Trip Logic	1, 2 3[b], 4[b], 5[b]	2 trains 2 trains	R,V W,X	SR 3.3.1.5 SR 3.3.1.5	NA NA

(a) A channel is OPERABLE when both of the following conditions are met:

1. The absolute difference between the as-found Trip Setpoint (TSP) and the previous as-left TSP is within the COT Acceptance Criteria. The COT Acceptance Criteria is defined as:

   $|\text{as-found TSP} - \text{previous as-left TSP}| \leq \text{COT uncertainty}$

   The COT uncertainty shall not include the calibration tolerance.

2. The as-left TSP is within the established calibration tolerance band about the nominal TSP. The nominal TSP is the desired setting and shall not exceed the Limiting Safety System Setting (LSSS). The LSSS and the established calibration tolerance band are defined in accordance with the Ginna Instrument Setpoint Methodology. The channel is considered operable even if the as-left TSP is non-conservative with respect to the LSSS provided that the as-left TSP is within the established calibration tolerance band.

(b) With Control Rod Drive (CRD) System capable of rod withdrawal or all rods not fully inserted.

(c) THERMAL POWER < 6% RTP.

(d) UFSAR Table 7.2-3.

(e) Both Intermediate Range channels < 5E-11 amps.

(f) With CRD System incapable of withdrawal and all rods fully inserted. In this condition, the Source Range Neutron Flux function does not provide a reactor trip, only indication.

(g) THERMAL POWER ≥ 8.5% RTP.

(h) THERMAL POWER ≥ 50% RTP.

(i) THERMAL POWER ≥ 8.5% RTP and Reactor Coolant Flow-Low (Single Loop) trip Function blocked.

(j) THERMAL POWER ≥ 8.5% RTP and RCP Breaker Position (Single Loop) trip Function blocked.

(k) THERMAL POWER > 8% RTP, and either no circulating water pump breakers closed, or condenser vacuum ≤ 20".

(l) THERMAL POWER ≥ 50% RTP, 1 of 2 circulating water pump breakers closed, and condenser vacuum > 20".

(m) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (Note 1)
Overtemperature ΔT

- NOTE -

The Overtemperature ΔT Function Limiting Safety System Setting is defined by:

$$\text{Overtemperature } \Delta T \leq \Delta T_0 \{K_1 + K_2 (P-P') - K_3 (T-T') [(1+\tau_1 s) / (1+\tau_2 s)] - f_1(\Delta I)\}$$

Where:

ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, °F.

P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, psig.

K_1 is the Overtemperature ΔT reactor trip setpoint, [*].
K_2 is the Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, [*]/psi.
K_3 is the Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, [*]/°F.

τ_1 is the measured lead time constant, [*] seconds.
τ_2 is the measured lag time constant, [*] seconds.

f(ΔI) is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

$f_1(\Delta I) = [*] \{[*] - (q_t - q_b)\}$	when $q_t - q_b \leq$ [*]% RTP
$f_1(\Delta I)$ = 0% of RTP	when [*] % RTP < $q_t - q_b \leq$ [*]% RTP
$f_1(\Delta I) = [*] \{(q_t - q_b) - [*]\}$	when $q_t - q_b >$ [*]% RTP

* These values denoted with [*] are specified in the COLR.

Table 3.3.1-1 (Note 2)
Overpower ΔT

- NOTE -

The Overpower ΔT Function Limiting Safety System Setting is defined by:

Overpower $\Delta T \leq \Delta T_0 \{K_4 - K_5 (T-T') - K_6 [(\tau_3 sT) / (\tau_3 s+1)] - f_2(\Delta I)\}$

Where:

ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, °F.

K_4 is the Overpower ΔT reactor trip setpoint, [*].
K_5 is the Overpower ΔT reactor trip heatup setpoint penalty coefficient which is:
[*]/°F for $T < T'$ and;
[*]/°F for $T \geq T'$.
K_6 is the Overpower ΔT reactor trip thermal time delay setpoint penalty which is:
[*]/°F for increasing T and;
[*]/°F for decreasing T.

τ_3 is the measured impulse/lag time constant, [*] seconds.

$f_2(\Delta I)$ = [*]

* These values denoted with [*] are specified in the COLR.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Deleted

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring activities for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the plant shall be submitted 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 plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

The following administrative requirements apply to the 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,	"Safety Limits (SLs)";
LCO 3.1.1,	"SHUTDOWN MARGIN (SDM)";
LCO 3.1.3,	"MODERATOR TEMPERATURE COEFFICIENT (MTC)";
LCO 3.1.5,	"Shutdown Bank Insertion Limit";
LCO 3.1.6,	"Control Bank Insertion Limits";
LCO 3.2.1,	"Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2,	"Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3,	"AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.3.1,	"Reactor Protection System (RPS) Instrumentation";
LCO 3.4.1,	"RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1,	"Boron Concentration."

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:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
(Methodology for 2.1, LCO 3.1.1, LCO 3.1.3, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.9.1.)

2. WCAP-13677-P-A, "10 CFR 50.46 Evaluation Model Report: WCOBRA/TRAC Two-Loop Upper Plenum Injection Model Updates to Support ZIRLO™ Cladding Option," February 1994.
(Methodology for LCO 3.2.1.)

3. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Control / FQ Surveillance Technical Specification," February 1994.
(Methodology for LCO 3.2.1 and LCO 3.2.3.)

4. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995.
(Methodology for LCO 3.2.1.)

5. WCAP 11397-P-A, "Revised Thermal Design Procedure," April 1989.
(Methodology for LCO 3.4.1 when using RTDP.)

6. WCAP-10054-P-A and WCAP-10081-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
(Methodology for LCO 3.2.1.)

7. WCAP-10924-P-A, Volume 1, Revision 1, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation Responses to NRC Questions," and Addenda 1,2,3, December 1988.
(Methodology for LCO 3.2.1.)

8. WCAP-10924-P-A, Volume 2, Revision 2, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addendum 1, December 1988.
(Methodology for LCO 3.2.1.)

9. WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation, Addendum 4: Model Revisions," March 1991.
(Methodology for LCO 3.2.1.)

10. WCAP-8745, "Design Basis for the Thermal Overpower Delta T and Thermal Overtemperature Delta T Trip Functions," March 1977.
(Methodology for LCO 3.3.1.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The following administrative requirements apply to the PTLR:

a. RCS pressure and temperature limits for heatup, cooldown, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"

b. The power operated relief valve lift settings required to support the Low Temperature Overpressure Protection (LTOP) System, and the LTOP enable temperature shall be established and documented in the PTLR for the following:

   LCO 3.4.6, "RCS Loops - MODE 4";

   LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";

   LCO 3.4.10, "Pressurizer Safety Valves"; and

   LCO 3.4.12, "LTOP System."

c. The analytical methods used to determine the RCS pressure and temperature and LTOP limits shall be those previously reviewed and approved by the NRC in NRC letter, "R.E. Ginna - Acceptance for Referencing of Pressure Temperature Limits Report, Revision 2 (TAC No. M96529)," dated November 28, 1997. Specifically, the methodology is described in the following documents:

   1. Letter from R.C. Mecredy, Rochester Gas and Electric Corporation (RG&E), to Document Control Desk, NRC, Attention: Guy S. Vissing, "Application for Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) Administrative Controls Requirements," Attachment VI, September 29, 1997, as supplemented by letter from R.C. Mecredy, RG&E, to Guy S. Vissing, NRC, "Corrections to Proposed Low Temperature Overpressure Protection System Technical Specification," October 8, 1997.

   2. WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Sections 1 and 2, January, 1996.

d. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for revisions or supplement thereto.

ENCLOSURE 4
R.E. Ginna Nuclear Power Plant

Marked-up Copy of Technical Specification Bases

initial conditions of the safety analyses (Ref. 5) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

2.1.1

Figure COLR-5 shows an example of the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is greater than or equal to the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. Each of the curves of Figure COLR-5 has three distinct slopes. Working from left to right, the first slope ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid such that overtemperature ΔT indication remains valid. The second slope ensures that the hot leg steam quality remains ≤ 15%. The final slope ensures that DNBR is always ≥ ~~1.4~~ safety analysis DNBR limit

The reactor core SLs are established to preclude violation of the following fuel design criteria:

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

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

BASES

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 of the core (Z).

$F_Q(Z)$ 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 adjusted for uncertainty. Therefore, $F_Q(Z)$ is a measure of the peak pellet power within the reactor core.

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. Therefore, these LCOs preserve core limits on a continuous basis.

$F_Q(Z)$ is sensitive to 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. Measurements are generally taken with the core at or near steady state conditions. With the measured three dimensional power distributions, it is possible to determine a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include variations in the value of $F_Q(Z)$, which are present during a nonequilibrium situation such as load following when the plant changes power level to match grid demand peaks and valleys.

Core monitoring and control under transient conditions (i.e., Condition 1 events as described in Reference 1) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion, Sequence and Overlap Limits.

Replace this Bases section with Bases B3.2.1B from NUREG-1431 including markup changes

APPLICABLE SAFETY ANALYSES

Limits on $F_Q(Z)$ preclude core power distributions that violate the following fuel design criteria:

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

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 2);

c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 3); and

d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN (SDM) with the highest worth control rod stuck fully withdrawn (Ref. 4).

Limits on $F_Q(Z)$ ensure that the value of the total peaking factor assumed as an initial condition 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.

The $F_Q(Z)$ limits provided in the COLR are based on the limits used in the LOCA analysis. $F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ assumed in safety analyses for other accidents because of the requirements set forth in 10 CFR 50.46 (Ref. 2) and ECCS model development in accordance with the required features of the ECCS evaluation models provided in 20 CFR 50, Appendix K (Ref. 5). Therefore, this LCO provides conservative limits for other accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

The $F_Q(Z)$ shall be maintained within the limits of the relationships provided in the COLR.

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 (Refs. 6 and 7).

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 $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produce 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 while 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 neither sufficient stored energy in the fuel nor sufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its limit maintains an acceptable absolute power density. 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.

A.2

When core peaking factors are sufficiently high that LCO 3.2.1 does not permit operation at RTP, the acceptable operation limits for AFD are reduced. The acceptable operation limits are reduced 1% for each 1% by which $F_Q(Z)$ exceeds its limit. For example, if the measured $F_Q(Z)$ exceeds the limit by 3% and the acceptable operation limits for AFD are ± 11% at 90% RTP and ± 31% at 50% RTP, then the revised AFD Acceptable Operation Limits would be ± 8% at 90% RTP and ± 28% at 50% RTP. This ensures a near constant maximum linear heat rate in units of kilowatts per foot at the acceptable operation limits. The Completion Time of 8 hours for the change in setpoints is sufficient, considering the small likelihood of a severe transient in this relatively short time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its specified limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions since this trip setpoint helps protect reactor core safety limits. This reduction shall be made as follows, given an $F_Q(Z)$ limit of 2.32, a measured $F_Q(Z)$ of 2.4, and a Power Range Neutron Flux-High setpoint of 108%, the Power Range Neutron Flux-High setpoint must be reduced by at least 3.4% to 104.6%. The Completion Time of 72 hours is sufficient, considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Reduction in the Overpower ΔT and Overtemperature ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions since these trip setpoints help protect reactor core safety limits. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.5

Verification that $F_Q(Z)$ has been restored to within its limit by performing SR 3.2.1.1 or SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If the Required Actions of A.1 through A.5 cannot be 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 allowed 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 3.2.1.1

Verification that $F_Q(Z)$ is within its limit involves increasing the measured values of $F_Q(Z)$ to allow for manufacturing tolerance and measurement uncertainties and then making a comparison with the limits. These limits are provided in the COLR. Specifically, the measured value of the Heat Flux Hot Channel Factor ($F^M Q$) is increased by 3% to account for fuel manufacturing tolerances and by 5% for flux map measurement uncertainty for a full core flux map using the movable incore detector flux mapping system. This procedure is equivalent to increasing the directly measured values of $F_Q(Z)$ by 8.15% before comparing with LCO limits.

Performing the Surveillance in MODE 1 prior to THERMAL POWER exceeding 75% RTP after each refueling ensures that $F_Q(Z)$ is within limit when RTP is achieved and provides confirmation of the nuclear design and the fuel loading pattern.

The Frequency of 31 EFPD is adequate for monitoring the change of power distribution with core burnup because the power distribution changes relatively slowly for this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F_Q(Z)$ limit cannot be exceeded for any significant period of time.

SR 3.2.1.2

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables.

With an NIS power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.1.2 at a Frequency of 24 hours provides an accurate alternative means for ensuring that F_Q remains within limits and the core power distribution is consistent with the safety analyses. A Frequency of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map.

This Surveillance is modified by a Note, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is ≥ 75% RTP.

REFERENCES

1. ~~American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.~~
2. ~~10 CFR 50.46.~~
3. ~~UFSAR, Section 15.4.5.1.~~
4. ~~Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10, 1967.~~
5. ~~10 CFR 50, Appendix K.~~
6. ~~UFSAR, Section 15.6.4.1.~~
7. ~~UFSAR, Section 15.6.4.2.~~

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1~~B~~ Heat Flux Hot Channel Factor ($F_Q(Z)$) ~~(RAOC-W(Z) Methodology)~~

BASES

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~~ adjusted for uncertainty. 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 3.2.3, "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 3.1.6, "Control Bank 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. Insert A

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.

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~~ Bank insertion, sequence, and overlap limits.

change on all pages

Insert A page B 3.2.4-1

Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping.

BASES

APPLICABLE SAFETY ANALYSES

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

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1), departure from nucleate boiling

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 (Ref. 2), and 200

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 (Ref. 3).

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.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents

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

LCO

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

$F_Q(Z) \leq (CFQ / P)\ K(Z)$ for $P > 0.5$

$F_Q(Z) \leq (CFQ / 0.5)\ K(Z)$ for $P \leq 0.5$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

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

P = THERMAL POWER / RTP

BASES

LCO (continued)

(2.60)

For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of [2.32] and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1.

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

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

$$F^C_Q(Z) = F^M_Q(Z)\,[1.0815]$$

(1.03)

where [1.0815] is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty (1.05) (Ref. 4).

Insert A

$F^C_Q(Z)$ 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.

The expression for $F^W_Q(Z)$ is:

$$F^W_Q(Z) = F^C_Q(Z)\,W(Z)$$

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 $F^C_Q(Z)$ 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 $F^C_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required and if $F^W_Q(Z)$ 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.

Insert A page B 3.2.1-3

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.

BASES

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.

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 Required 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.

(completion of applicable surveillance)

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 Required Action A.1. The maximum allowable Power Range Neutron Flux - High trip setpoints initially determined by Required 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.

(completion of applicable surveillance)

BASES

ACTIONS (continued)

A.3

Reduction in the Overpower ΔT trip setpoints (value of K_4) 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 Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required 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.

(completion of applicable surveillance)

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition A is modified by a Note that requires Required Action A.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

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 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 (Ref. 5).

insert A

Insert A Page B 3.2.1-5

The percent that Fq(Z) exceeds its transient limit is calculated based on the following expression :

$$\left\{\left(\begin{matrix}\text{maximum}\\\text{over z}\end{matrix}\left[\frac{F_Q^C(Z)*W(z)}{\frac{CFQ}{P}*K(z)}\right]\right)-1\right\}*100\ \text{for}\ P>0.5$$

$$\left\{\left(\begin{matrix}\text{maximum}\\\text{over z}\end{matrix}\left[\frac{F_Q^C(Z)*W(z)}{\frac{CFQ}{0.5}*K(z)}\right]\right)-1\right\}*100\ \text{for}\ P\leq 0.5$$

BASES

ACTIONS (continued)

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 Required 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 Required Action B.1.

B.3

Reduction in the Overpower ΔT trip setpoints value of K_4 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 Required Action B.1.

B.4

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by Required Action B.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

BASES

ACTIONS (continued)

Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition ~~A~~ B is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

C.1

If Required Actions A.1 through A.4 or 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 allowed 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 3.2.1.1 and SR 3.2.1.2 are modified by a Note. 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 Frequency conditions 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. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, 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 Frequency condition, together with the Frequency 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 Frequency conditions, it is possible to

BASES

SURVEILLANCE REQUIREMENTS (continued)

increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency 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 3.2.1.1

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] (Ref. 4). $F_Q^C(Z)$ 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.

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 Frequency 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).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2

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 61 ~~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.

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 a Note that 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 [$F_Q^C(Z)$ / 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 (~~Ref. 5~~)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~
~~WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control and F_Q Surveillance Technical Specification," February 1994, or other appropriate plant specific methodology, is to be listed in the COLR description in the Administrative Controls Section 5.0 to address the methodology used to derive this factor.~~

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.

The Surveillance Frequency 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 Frequency 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.

REFERENCES

1. 10 CFR 50.46 ~~(1974)~~.
2. ~~Regulatory Guide 1.77, Rev. 0, May 1974.~~ UFSAR 15.4.5.43
3. ~~10 CFR 50, Appendix A, GDC 26.~~ Atomic Industrial Forum (AIF) GDC-29, Issued for comment July 10, 1967
4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
5. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification," February 1994.

Insert A

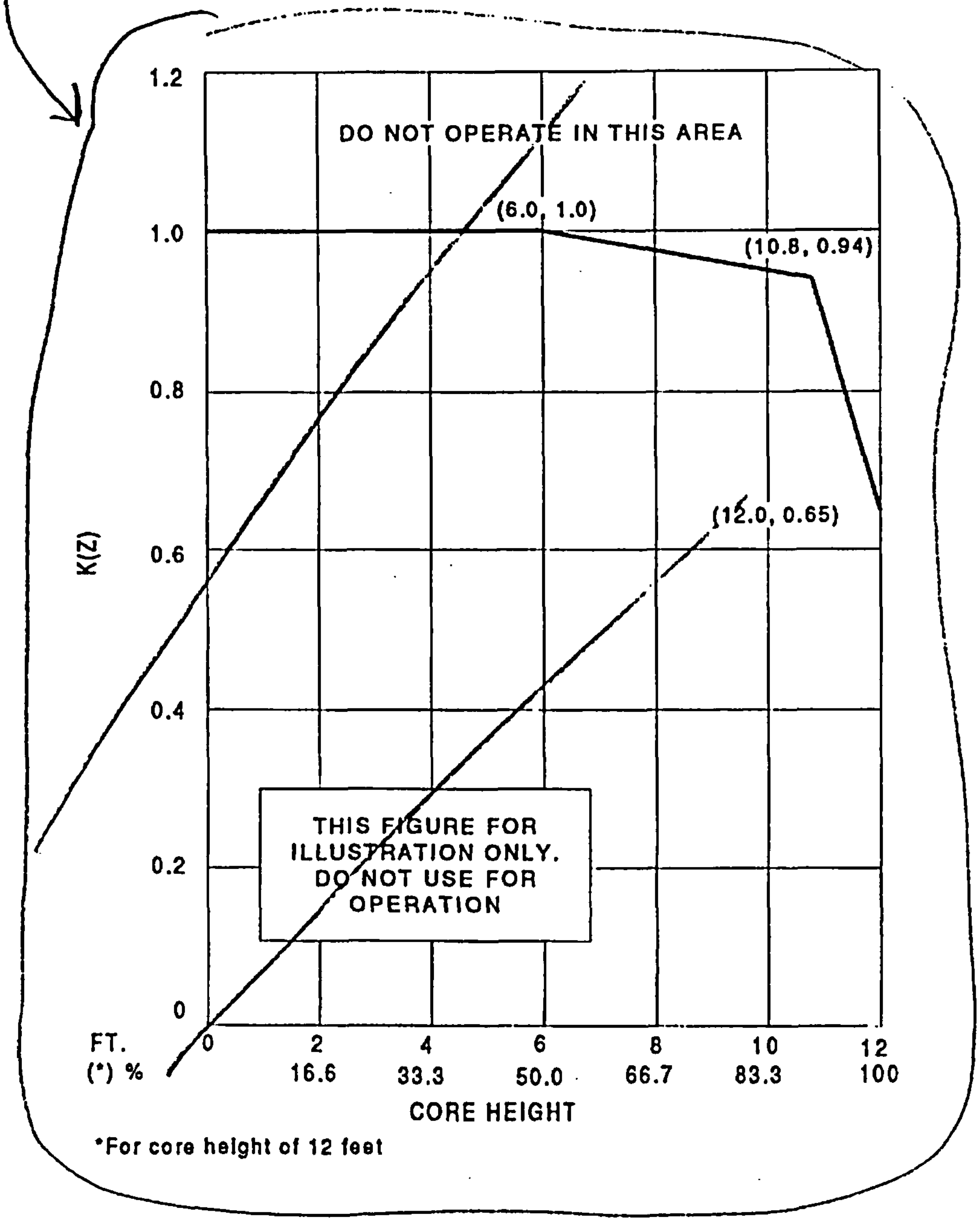
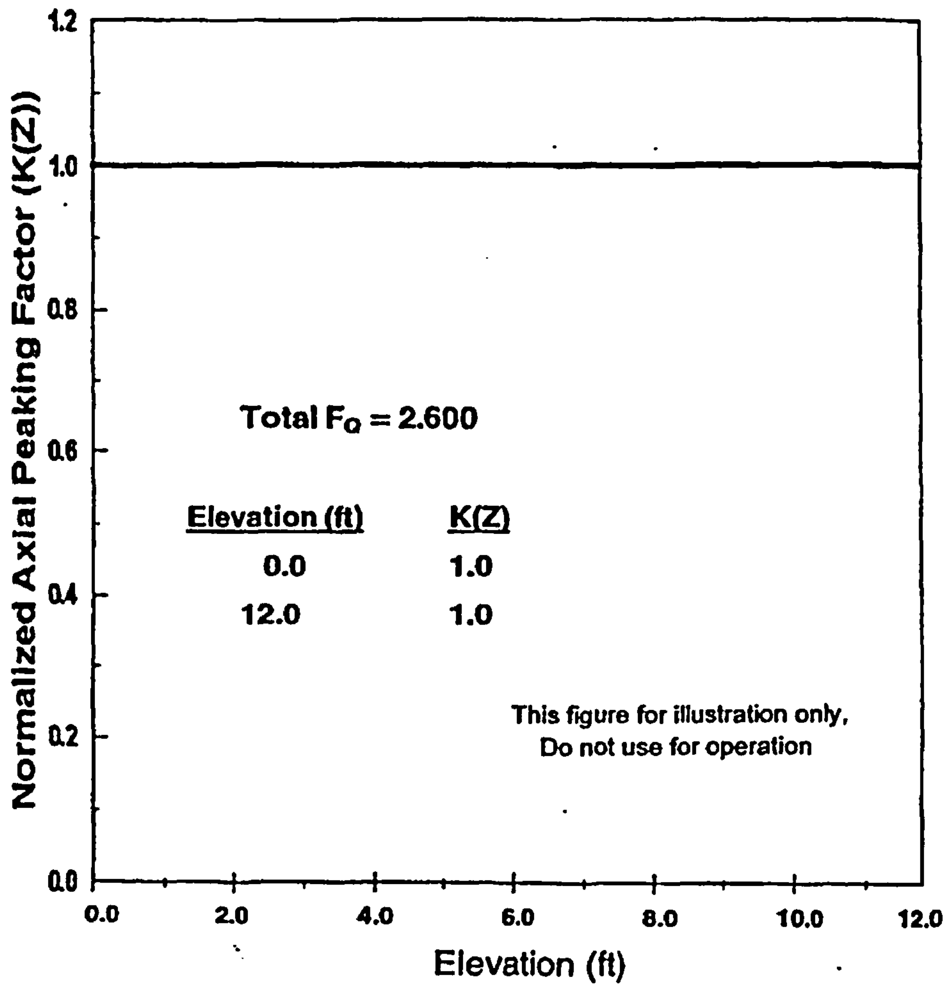


Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

Insert A page B 3 2 1-11



Replace this Bases section with Bases B 3.2.3B from NUREG-1431 including MU changes.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the 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.

The operating scheme used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during plant maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QUADRANT POWER TILT RATIO (QPTR) LCOs limit the radial component of the peaking factors.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to 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 concentrations. 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 CAOC methodology (Ref. 1) entails:

a. Establishing an envelope of allowed power shapes and power densities;

b. Devising an operating strategy for the cycle that maximizes plant flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure 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 limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events (Ref. 2). This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the loss of coolant accident. The most significant Condition 3 event is the loss of flow accident. The most significant Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement.

LCO

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

Signals are available to the operator to help define the power profile from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom excore neutron detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector 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.

With THERMAL POWER ≥ 90% RTP (i.e., Part A of this LCO), the AFD must be kept within the target band about the target flux difference. The target band is provided in the COLR. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated. With THERMAL POWER < 90% RTP, the AFD may be outside the target band provided that the deviation time is restricted.

It is intended that the plant is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part B of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours when > 15% RTP, is allowed during which the plant may be operated outside of the target band but within the acceptable operation limits provided in the COLR. The cumulative penalty time is the sum of penalty times as calculated by Notes 2 and 3 of this LCO.

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The reduced penalty deviation time accumulation rate reflects this reduced significance. With THERMAL POWER ≤ 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the Plant Process Computer System (PPCS) is nominally once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time. The inoperability of this monitor requires independent verification that AFD remains within limit and that the peaking factors assumed in the accident analyses remain valid.

This LCO is modified by four Notes. The first Note states the conditions necessary for declaring the AFD outside of the target band. The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup. The average of the four OPERABLE excore detectors is used to determine when AFD is outside the target band. If one excore detector is out of service, the remaining three detectors are used to derive the average AFD. The second and third Notes describe how the cumulative penalty deviation time is calculated. The second Note states that with THERMAL POWER ≥ 50% RTP the penalty deviation time is accumulated at the rate of 1 minute for each 1 minute of power operation with AFD outside the target band. The third Note states that with THERMAL POWER > 15% RTP and < 50% RTP the penalty deviation time is accumulated at the rate of 0.5 minutes for each 1 minute of power operation with AFD outside the target band. The cumulative penalty time is the sum of penalty times from Notes 2 and 3 of this LCO. The fourth Note addresses AFD outside of the target band during surveillances. For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 EFPDs.

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

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 1). Above 15% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. Also, low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP. The value of the AFD in these conditions does not affect the consequences of the design basis events.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If Required Action A.1 is not completed with the required Completion Time of 15 minutes, the axial xenon distribution starts to become skewed. Reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes to reduce THERMAL POWER to < 90% RTP allows for a controlled reduction in power without allowing the plant to remain in an unanalyzed condition for an extended period of time.

C.1

This Required Action must be implemented with THERMAL POWER < 90% RTP but ≥ 50% RTP if either the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. Reducing THERMAL POWER to < 50% RTP will put the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits. The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

D.1

When the AFD monitor alarm is inoperable and THERMAL POWER is ≥ 90% RTP, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time at a frequency of every 15 minutes to ensure that the plant does not operate in an unanalyzed condition. A Completion Time of 15 minutes is adequate to ensure that the AFD is within its limits at high THERMAL POWER levels and is consistent with the Completion Time for restoring AFD to within limits (Condition A).

E.1

When the AFD monitor alarm is inoperable and THERMAL POWER is < 90% RTP, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time at a frequency of every hour to ensure that the plant does not operate in an unanalyzed condition. A Completion Time of 1 hour is adequate since the AFD may deviate from the target band for up to 1 hour using the methodology of Notes 2 and 3 of this LCO to calculate the cumulative penalty deviation time before corrective action is required.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This SR is the verification that the AFD monitor is OPERABLE. This is normally accomplished by introducing a signal into the plant process computer to verify control room annunciation of AFD not within the target band. The Frequency of 12 hours is sufficient to ensure OPERABILITY of the AFD monitor since under normal plant operation, the AFD is not expected to significantly change.

SR 3.2.3.2

The AFD is monitored on a continuous basis using the Plant Process Computer System (PPCS) that has an AFD monitor alarm. The PPCS determines the 1 minute average of the OPERABLE excore detector outputs and provides an alarm message and a main control board annunciator immediately if the average AFD is outside the target band and then re-alarms when the cumulative penalty deviation time reaches 15 minute intervals within the previous 24 hours. The computer also sends an alarm message when the cumulative penalty deviation time is ≥ 1 hour within the previous 24 hours. This alarm message does not clear until the cumulative penalty deviation time is < 1 hour within the previous 24 hours.

With the AFD monitor alarm inoperable, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at ≥ 90% RTP, the AFD measurement is monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. The AFD should be monitored and logged more frequently during periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.2 is modified by two Notes. The first Note states that this surveillance is only required to be performed when the AFD monitor alarm is inoperable with THERMAL POWER ≥ 90% RTP. The second Note states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time if AFD values cannot be obtained from the PPCS. Inoperability of the alarm does not necessarily prevent the actual AFD values from being available (e.g., from the computer logs or hand logs). AFD values for the preceding 24 hours can be obtained from the hourly PPCS printouts or hand logs.

SR 3.2.3.3

The AFD is monitored on a continuous basis using the PPCS that has an AFD monitor alarm. The PPCS determines the 1 minute average of the OPERABLE excore detector outputs and provides an alarm message and a main control board annunciator immediately if the average AFD is outside the target band and then re-alarms when the cumulative penalty deviation time reaches 15 minute intervals within the previous 24 hours. The computer also sends an alarm message when the cumulative penalty deviation time is ≥ 1 hour within the previous 24 hours. This alarm message does not clear until the cumulative penalty deviation time is < 1 hour within the previous 24 hours.

With the AFD monitor alarm inoperable, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at < 90% RTP, but > 15% RTP, the AFD measurement is monitored at a Surveillance Frequency of 1 hour to ensure that the AFD is within its limits. The Frequency of 1 hour is adequate since the AFD may deviate from the target band for up to 1 hour using the methodology of Notes 2 and 3 of this LCO to calculate the cumulative penalty deviation time before corrective action is required. The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.3 is modified by two Notes. The first Note states that this surveillance is only required to be performed when the AFD monitor alarm is inoperable with THERMAL POWER < 90% RTP. The second Note states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time if AFD values cannot be obtained from the PPCS. Inoperability of the alarm does not necessarily prevent the actual AFD values from being available (e.g., from the computer logs or hand logs). AFD values for the preceding 24 hours can be obtained from the hourly PPCS printouts or hand logs.

SR 3.2.3.4

This Surveillance requires that the target flux difference be updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup.

There are two methods by which this update can be completed. The first method requires measuring the target flux difference in accordance with SR 3.2.3.5. This measurement may be obtained using incore or excore instrumentation. The second method involves interpolation between measured and predicted values. The nuclear design report provides predicted values for target flux difference at various cycle burnups. The difference between the last measured value and the predicted value at the same burnup is applied to the predicted value at the burnup where the target flux difference update is required. This revised predicted value can then be used to determine the updated value of the target flux difference.

SR 3.2.3.5

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of once within 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference to the value measured at steady state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference values that account for changes in incore-excore calibrations that may have occurred in the interim.

This SR is modified by a Note that allows the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.
2. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
3. UFSAR, Section 7.7.2.6.4.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3B AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC) Methodology)~~

BASES

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 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.

~~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 (Ref. 1). 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.~~

BASES

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 (Ref. ~~2~~ 1) 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 (Ref. 2).

The limits on the AFD ensure 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 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 important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident. The most important Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents ~~simulated~~ assumed 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 (Ref. 3). 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.

BASES

LCO (continued)

The AFD limits are provided in the COLR. Figure B 3.2.3~~B~~-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.

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.

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 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 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. ~~WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.~~

American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973

2. ~~R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10217(NP), June 1983.~~

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

3. FSAR, Chapter ~~[15]~~ 7.7.2.6.4

INSERT A

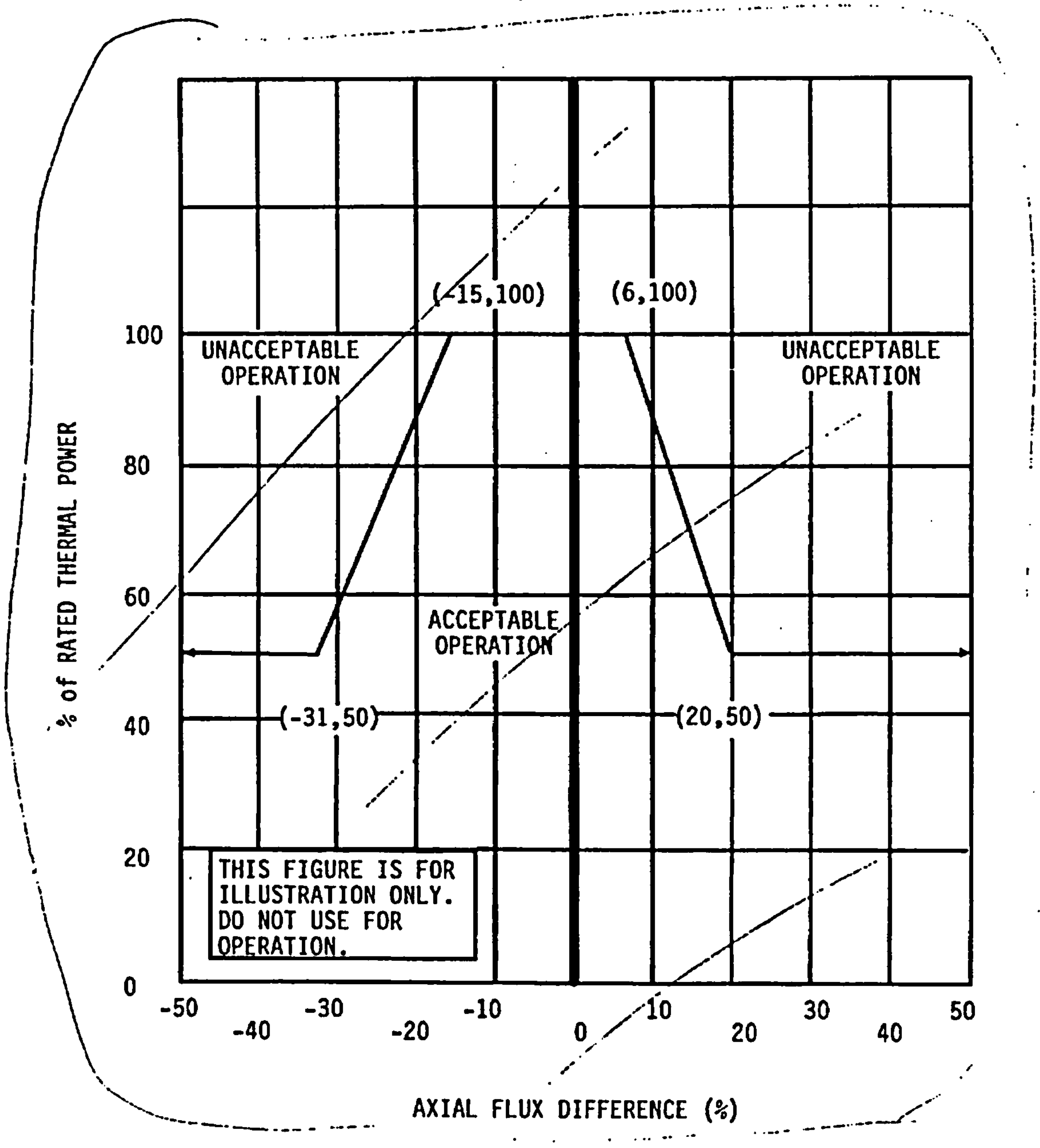


Figure B 3.2.3B-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

Insert A page B 3.2.3-4

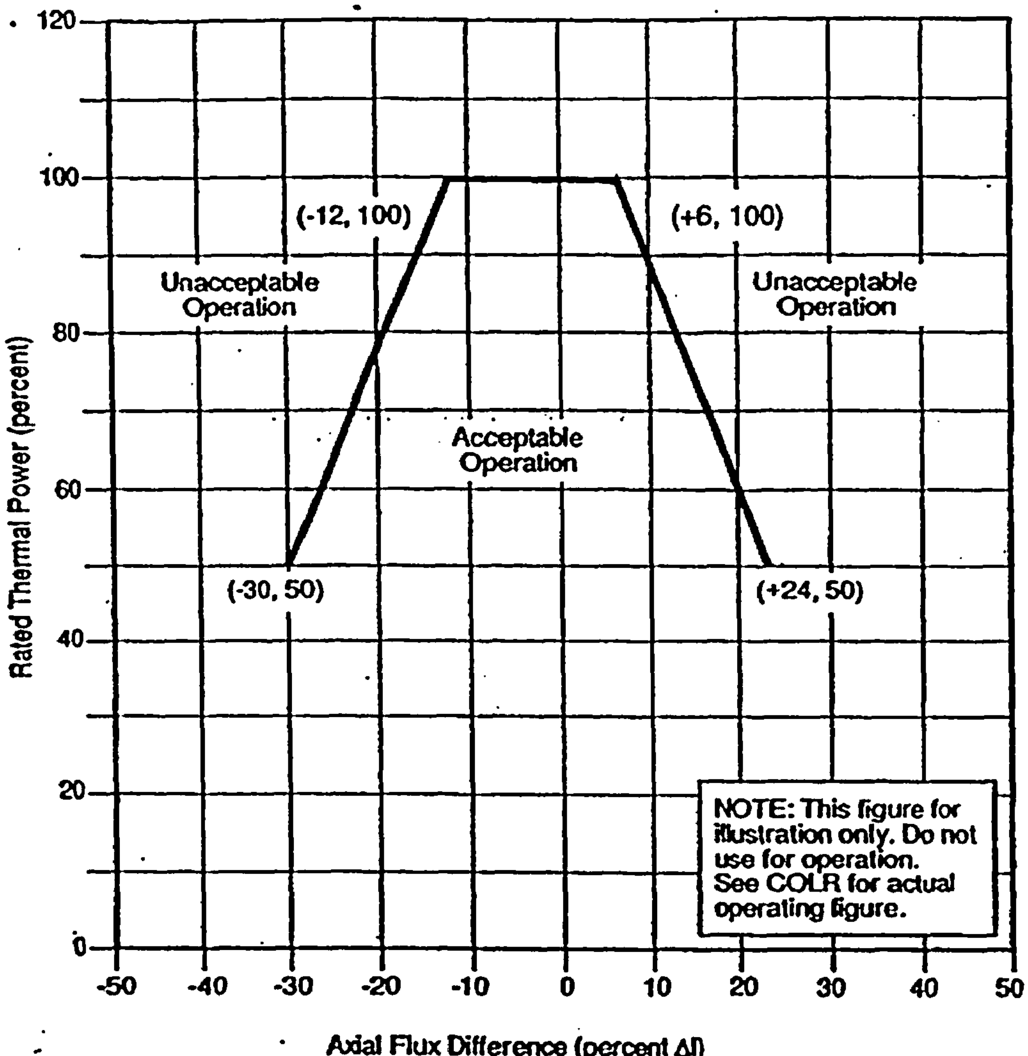


Figure B 3.2.3-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR 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. Quadrant Power Tilt is a core tilt that is measured with the use of the excore power range flux detectors. A core tilt is defined as the ratio of maximum to average quadrant power. The QPTR is defined as the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants. Limiting the QPTR prevents radial xenon oscillations and will indicate any core asymmetries.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," and LCO 3.1.6, "Control Bank 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 fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

Limits on QPTR preclude core power distributions that violate the following fuel design criteria:

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

b. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 2); 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 (Ref. 3).

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

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

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

The QPTR satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36

LCO

Relocate to TRM

The QPTR monitor ~~alarm shall be OPERABLE and~~ QPTR shall be maintained at or below the limit of 1.02.

~~QPTR is monitored on an automatic basis using the Plant Process Computer System (PPCS) that has a QPTR monitor alarm. The PPCS determines from the excore detector outputs the ratio of the highest average nuclear power in any quadrant to the average of nuclear power in the four quadrants and provides an alarm message if the QPTR is above the 1.02 limit.~~

The QPTR limit of 1.02, above which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of ~~1.025~~ 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged. ~~However, the additional QPTR of 0.005 is provided for margin in the LCO.~~

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits assumed in the safety analyses.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is neither sufficient stored energy in the fuel nor sufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply below 50% RTP, but allow progressively higher peaking factors as THERMAL POWER decreases below 50% RTP.

ACTIONS

A.1

after each QPTR determination

With the QPTR exceeding its limit, limiting THERMAL POWER to ≥ 3% below RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. A further increase in ~~the QPTR~~ requires a lower limit to THERMAL POWER in accordance with Required Action A.2.

Insert A

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR in accordance with SR 3.2.4.2 once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER must be limited accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

(as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$ and $F^N_{\Delta H}$)

Insert B

after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their

Insert A page B 3.2.4-3

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in the QPTR would require power reductions within 2 hours of QPTR determination (completion of applicable surveillance), if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

Insert B page B 3.2.4-3

Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping.

limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

Insert A

~~If the QPTR has exceeded the 1.02 limit and the verification of $F^N_{\Delta H}$ and $F_Q(Z)$ shows that safety requirements are met, the excore detectors are normalized to eliminate the indicated tilt prior to increasing THERMAL POWER to above the limit of Required Actions A.1 and A.2. This is done to detect any subsequent significant changes in QPTR and to provide a meaningful QPTR alarm.~~

~~Required Action A.5 is modified by a Note that states that the indicated tilt is not eliminated until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). It is necessary to verify that the core power distribution is acceptable prior to adjusting the excore detectors to eliminate the indicated tilt and increasing power to ensure that the plant is not operating in an unanalyzed condition. This Note is intended to prevent any ambiguity about the required sequence of actions.~~

equilibrium condition at

A.6

Once

~~After~~ the flux tilt is normalized to eliminate the indicated tilt (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours after reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but it increases slowly, then the peaking factor surveillances must be performed within 48 hours ~~of the time when the ascent to power was begun~~. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action(s) A.1 ~~and A.2~~ while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$,

after increasing THERMAL POWER above the limit of Required Action A.1

a Note that

Required Action A.6 is modified by ~~three Notes~~. The first Note states that it is not necessary to perform Required Action A.6 if the cause of the QPTR alarm is associated with instrumentation alignment. The intent of this Note is to clarify that the core power distribution does not have to be re-verified if the QPTR alarm is only due to the instrumentation (i.e., the excore detectors) being out of adjustment and not due to an anomaly within the core. The second Note states that the peaking factor surveillances ~~are not required until~~ after the excore detectors have been normalized to eliminate the indicated tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are adjusted to eliminate the indicated tilt and the core returned to power. The third Note states that only one of the following Completion Times, whichever becomes applicable first, must be met. The intent of this Note is to clearly indicate that the first Completion Time to become applicable is the Completion Time which must be met to satisfy Required Action A.6.

May only be done

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the plant must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ≤ 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

<Relocate to TRM>

C.1 and C.2

When the QPTR monitor alarm is inoperable the QPTR must be verified within limits at a frequency of every 24 hours to ensure that the plant does not operate in an unanalyzed condition. When THERMAL POWER is ≥ 75% RTP and one power range channel is inoperable, QPTR cannot be adequately measured using the excore detectors. In this situation a flux map must be completed to verify that the core power distribution is consistent with the safety analyses. A Completion Time of 24 hours is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt and provides sufficient time to stabilize the plant and perform a flux map when necessary. The Completion Time of 24 hours is also consistent with the Frequency of SR 3.2.4.3 with one inoperable power range channel since these channels provide input into the QPTR monitor.

SURVEILLANCE REQUIREMENTS

~~SR 3.2.4.1~~

(Relocate to TRM)

This SR is the verification that the QPTR monitor is OPERABLE. This is normally accomplished by introducing a signal into the PPCS to verify control room annunciation of QPTR not within limit. The Frequency of 12 hours is sufficient to ensure OPERABILITY of the QPTR monitor since under normal plant operation, QPTR is not expected to significantly change.

SR 3.2.4.~~2~~ 1

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days ~~when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.~~

takes into account other information & alarms available in the control room

SR 3.2.4.~~2~~ 1 is modified by two Notes. ~~The first~~ Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and one power range channel is inoperable. ~~The second Note states that SR 3.2.1.2 and SR 3.2.2.2 should be performed if THERMAL POWER is ≥ 75% RTP and one power range channel is inoperable. The intent of this Note is to clarify that when one power range channel is inoperable and THERMAL POWER is ≥ 75% RTP, a full core flux map should be performed to verify the core power distribution instead of using the three OPERABLE power range channels to verify QPTR. At or above 75% RTP with one power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is~~

Note 2 allows performance of SR 3.2.4.2 in lieu of SR3.2.4.1

Insert A

decreased. Performing a full core flux map provides an accurate alternative means for ensuring that $F_Q(Z)$ and $F^N_{\Delta H}$ remain within limits and the core power distribution is consistent with the safety analyses.

SR 3.2.4.3 (2)

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits when the QPTR alarm is inoperable. The Frequency of 24 hours is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

Relocate to TRM

SR 3.2.4.3 is modified by three Notes. The first Note states that the surveillance is only required to be performed if the QPTR monitor alarm is inoperable. This surveillance requires a more frequent verification that the QPTR is within limit since the monitor alarm is inoperable. The second Note allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and one power range channel is inoperable. The third Note states that SR 3.2.1.2 and SR 3.2.2.2 should be performed if THERMAL POWER is ≥ 75% RTP and one power range channel is inoperable. The intent of this Note is clarify that when one power range channel is inoperable and THERMAL POWER is ≥ 75% RTP, a full core flux map should be performed to verify the core power distribution instead of using the three OPERABLE power range channels to verify QPTR. At or above 75% RTP with one power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing a full core flux map provides an accurate alternative means for ensuring that $F_Q(Z)$ and $F^N_{\Delta H}$ remain within limits and the core power distribution is consistent with the safety analyses.

Insert B

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.4.5.
3. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10, 1967.

Insert A page B 3.2.4-7

For those causes of quadrant power tilt that occur quickly (e. g., a dropped rod), there typically are other indications of abnormality that prompt a verification of the core power tilt.

Insert B page B 3.2.4-7

This Surveillance is modified by a Note, which states that it is not required until 24 hours after the input from one or more Power Range Neutron Flux channel is inoperable and the THERMAL POWER is > 75% RTP. With the input from a NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased.

When one NIS power range channel input is inoperable and THERMAL POWER is > 75% RTP, a full core flux map should be performed to verify the core power distribution instead of using the three OPERABLE power range channel inputs to verify QPTR by performing SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1, at a Frequency of 24 hours. Performing a full core flux map provides an accurate alternative means for ensuring that $F_Q(Z)$ and $F^N_{\Delta H}$ remain within limits and the core power distribution is consistent with the safety analyses.

- ~~axial power distribution f(ΔI) - the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.~~

The Overpower ΔT trip Function is calculated in two channels for each loop as described in Note 2 of Table 3.3.1-1. A reactor trip occurs if the Overpower ΔT trip setpoint is reached in two-out-of-four channels. Since the temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent an unnecessary reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only MODES where enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function is not required to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

7. Pressurizer Pressure

The same sensors (PT-429, PT-430, and PT-431) provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip with the exception that the Pressurizer Pressure-Low and Overtemperature ΔT trips also receive input from PT-449. Since the Pressurizer Pressure channels are also used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function.

SR 3.3.1.2

This SR compares the calorimetric heat balance calculation to the NIS Power Range Neutron Flux-High channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is still OPERABLE but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is then declared inoperable.

This SR is modified by a Note which states that this Surveillance is required to be performed within 12 hours after power is ≥ 50% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

This SR compares the incore system to the NIS channel output every 31 effective full power days (EFPD). If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is then declared inoperable. This surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function ~~and Overpower ΔT Function~~.

This SR is modified by two Notes. Note 1 clarifies that the Surveillance is required to be performed within 7 days after THERMAL POWER is ≥ 50% RTP but prior to exceeding 90% RTP following each refueling and if it has not been performed within the last 31 EFPD. Note 2 states that performance of SR 3.3.1.6 satisfies this SR since it is a more comprehensive test.

The Frequency of every 31 EFPD is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

SR 3.3.1.4

This SR is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS of the RTB, and the RTB Undervoltage and Shunt Trip Mechanisms. This test shall verify OPERABILITY by actuation of the end devices.

The test shall include separate verification of the undervoltage and shunt trip mechanisms except for the bypass breakers which do not require separate verification since no capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.11. However, the bypass breaker test shall include a local shunt trip. This test must be performed on the bypass breaker prior to placing it in service to take the place of a RTB.

The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.

<u>SR 3.3.1.5</u>

This SR is the performance of an ACTUATION LOGIC TEST on the RTS Automatic Trip Logic every 31 days on a STAGGERED TEST BASIS. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.

<u>SR 3.3.1.6</u>

This SR is a calibration of the excore channels to the incore channels every 92 EFPD. If the measurements do not agree, the excore channels are still OPERABLE but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are then declared inoperable. This surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function ~~and Overpower ΔT Function~~.

This SR has been modified by a Note stating that this Surveillance is required to be performed within 7 days after THERMAL POWER is ≥ 50% RTP but prior to exceeding 90% RTP following each refueling.

The Frequency of 92 EFPD is adequate based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

<u>SR 3.3.1.7</u>

This SR is the performance of a COT every 92 days for the following RTS functions:

- Power Range Neutron Flux-High;
- Source Range Neutron Flux (in MODE 3, 4, or 5 with CRD System capable of rod withdrawal or all rods not fully inserted);

B 3.4 REACTOR COOLANT SYSTEMS (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the departure from nucleate boiling (DNB) design criterion will be met for each of the transients analyzed.

The design method employed to meet the DNB design criterion for fuel assemblies is the Revised Thermal Design Procedure (RTDP). With the RTDP methodology, uncertainties in plant operating parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit departure from nucleate boiling ratio (DNBR) values are determined in order to meet the DNB design criterion.

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The RTDP design limit DNBR ~~values are~~ 1.24 ~~and 1.23~~ for the typical and thimble cells ~~respectively~~ for fuel analyses with the WRB-1 correlation for the 422V+ and OFA fuel.

Additional DNBR margin is maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility. The safety analysis DNBR value ~~is 1.40~~ are 1.40 and 1.34 for the 422V+ and OFA fuel, respectively, for the typical and thimble cells.

For the WRB-1 correlation, the 95/95 DNBR correlation limit is 1.17. The W-3 DNB correlation is used where the primary DNBR correlation is not applicable. The WRB-1 correlation was developed based on mixing vane data and therefore is only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 to 1000 psia, the W-3 correlation limit is 1.45. For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30.

ENCLOSURE 5
R.E. Ginna Nuclear Power Plant

List of Regulatory Commitments

The following table identifies those actions committed to by R.E. Ginna Nuclear Power Plant, LLC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENT	DUE DATE
Once approved, the amendment will be implemented prior to startup from the fall 2006 refueling outage.	Prior to startup from the fall 2006 refueling outage.
All related core design parameters will be checked against the LOCA analyses' limits on a cycle-specific basis for reload cycles which utilize the RAOC methodology.	Prior to startup from the fall 2006 refueling outage.