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ComEd

February 15, 2000

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Request for Amendment to Technical Specifications for Byron and Braidwood Stations to Implement the Best Estimate Analyzer for Core Operations Nuclear Power Distribution Monitoring System

References: (1) WCAP-12472-P-A, "BEACON – Core Monitoring and Operations Support System," August 1994.

(2) Letter from A. C. Thadani, (NRC) to N. J. Liparulo (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-12472-P, 'BEACON: Core Monitoring and Operations Support System,'" February 16, 1994.

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

(4) Letter from A. C. Thadani (NRC) to N. J. Liparulo (Westinghouse), "Acceptance for Referencing of Revised Version of Licensing Topic Report WCAP-10216-P, Rev. 1, 'Relaxation of Constant Axial Offset Control – F_Q Surveillance Technical Specification,'" November 26, 1993.

A001

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes allow us to use the Westinghouse core monitoring and support system known as Best Estimate Analyzer for Core Operations Nuclear (BEACON).

The BEACON Power Distribution Monitoring System (PDMS) has been developed to improve the operational support for Pressurized Water Reactors (PWRs). It is an advanced core monitoring and support package, which uses current instrumentation in conjunction with a fully analytical methodology for on-line generation of three-dimensional core power distributions. The system provides core monitoring, core measurement reduction, core analysis and follow, and core predictions. The justification to implement BEACON is provided in WCAP-12472-P-A, "BEACON – Core Monitoring and Operations Support System," August 1994 (Reference 1). This justification has been approved by the NRC in Reference 2. As part of the planned implementation of BEACON, we are utilizing Relaxed Axial Offset Control (RAOC) methodology for determining the Axial Flux Difference. The analytical method is provided in WCAP-10216-P-A, Revision 1, "Relaxation of Constant Axial Offset Control – F_Q Surveillance Technical Specification," February 1994 (Reference 3). This methodology was been approved by the NRC in Reference 4.

We request approval of these proposed changes by January 31, 2001, to support procedural changes and work planning prior to the Byron Station, Unit 2, Spring 2001 refueling outage.

The license amendment request is subdivided as follows.

1. Attachment A gives a description and safety analysis for the proposed changes.
2. The marked-up TS pages are included in Attachments B-1 and B-2 for Byron and Braidwood Stations, respectively. The clean copy TS pages are included in Attachments B-3 and B-4 for Byron and Braidwood Stations, respectively. The clean copy of the Technical Requirements Manual (TRM) pages showing the proposed changes are included for informational purposes in Attachments B-5 and B-6 for Byron and Braidwood Stations, respectively. The clean copy TS Bases pages incorporating the proposed changes are included for informational purposes in Attachments B-7 and B-8 for Byron and Braidwood Stations, respectively. The changes to the Core Operating Limits Report (COLR) that result from the proposed TS changes are included in Attachments B-9 and B-10 for Byron and Braidwood Stations, respectively.
3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91(a)(1), and provides information supporting a finding of no significant hazard consideration using the standards in 10 CFR 50.92(c).
4. Attachment D provides information supporting an Environmental Assessment.

The Plant Operations Review Committee and the Nuclear Safety Review Board in accordance with the Quality Assurance Program have reviewed these changes.

ComEd is notifying the State of Illinois of this license amendment request by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this submittal, please contact Ms. Kelly M. Root at (630) 663-7292.

Respectfully,

R. M. Krich
Vice President – Regulatory Services

Attachments: Attachment B-1 - Marked-up TS Pages for Byron Station
Attachment B-2 - Marked-up TS Pages for Braidwood Station
Attachment B-3 - Clean Copy TS Pages for Byron Station
Attachment B-4 - Clean Copy TS Pages for Braidwood Station
Attachment B-5 - Clean Copy TRM Pages for Byron Station
Attachment B-6 - Clean Copy TRM Pages for Braidwood Station
Attachment B-7 - Clean Copy TS Bases Pages for Byron Station
Attachment B-8 - Clean Copy TS Bases Pages for Braidwood Station
Attachment B-9 - COLR Changes for Byron Station
Attachment B-10 - COLR Changes for Braidwood Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	**Docket Numbers**
BRAIDWOOD STATION, UNITS 1 and 2	)	**STN 50-456 and STN 50-457**
AND	)	
BYRON STATION, UNITS 1 and 2	)	**STN 50-454 and STN 50-455**

SUBJECT: **Request for Amendment to Technical Specifications for Byron and Braidwood Stations to Implement the Best Estimate Analyzer for Core Operations Nuclear Power Distribution Monitoring System**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Vice President - Regulatory Services

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 15 day of February, 2000.

OFFICIAL SEAL
JEFFERY A BATARA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:02/24/02

(OFFICIAL SEAL)

Notary Public

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company is requesting a change to Appendix A, Technical Specifications (TS) of Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77, for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, respectively.

The proposed changes allow us to use the Westinghouse core monitoring and support system known as Best Estimate Analyzer for Core Operations Nuclear (BEACON). The BEACON Power Distribution Monitoring System (PDMS) has been developed to improve the operational support for Pressurized Water Reactors (PWRs). It is an advanced core monitoring and support package, which uses current instrumentation in conjunction with a fully analytical methodology for on-line generation of three-dimensional (3-D) core power distributions. The system provides core monitoring, core measurement reduction, core analysis and follow, and core predictions. The topical report WCAP-12472-P-A, "BEACON – Core Monitoring and Operations Support System," August 1994 (Reference 1) was approved by the NRC by letter dated February 16, 1994 (Reference 7).

As part of the planned implementation of BEACON, we are utilizing Relaxed Axial Offset Control (RAOC) methodology (Reference 2) for determining the Axial Flux Difference (AFD). This analytical method was provided in topic report WCAP-10216-P-A, Revision 1, "Relaxation of Constant Axial Offset Control – F_Q Surveillance Technical Specification," February 1994, which was approved by the NRC by letter dated November 26, 1993 (Reference 8), and is already referenced in the Byron and Braidwood Stations' TS 5.6.5. A change from Constant Axial Offset Control (CAOC) to RAOC is an option contained in WCAP-12472-P-A.

We are proposing changes to the following Byron and Braidwood Stations' TS:

- TS 3.1.4, Rod Group Alignment Limits;
- TS 3.1.7, Rod Position Indication;
- TS 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$);
- TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$);
- TS 3.2.3, AXIAL FLUX DIFFERENCE (AFD);
- TS 3.2.4, QUADRANT POWER TILT RATIO (QPTR);
- TS 3.3.1, Reactor Trip System (RTS) Instrumentation; and
- TS 5.6.5, CORE OPERATING LIMITS REPORT (COLR).

We propose to add the following TS to the Byron and Braidwood Stations' TS:

- TS 3.2.5, Departure From Nucleate Boiling Ratio (DNBR).

The associated TS Bases proposed changes are included for information only for the following Byron and Braidwood Stations' TS Bases:

TS B 3.1.4, Rod Group Alignment Limits;
TS B 3.1.7, Rod Position Indication;
TS B 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$);
TS B 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$);
TS B 3.2.3, AXIAL FLUX DIFFERENCE (AFD);
TS B 3.2.4, QUADRANT POWER TILT RATIO (QPTR);
TS B 3.2.5, Departure from Nucleate Boiling Ratio (DNBR); and
TS B 3.3.1, Reactor Trip System (RTS) Instrumentation.

The associated Technical Requirements Manual (TRM) proposed changes are included for information only for the following Byron and Braidwood Stations' TRM:

TRM 3.3.a, Movable Incore Detectors. (Note: The Movable Incore Detection System TS was relocated to the TRM during the conversion to the Improved Standard TS (ISTS).); and
TRM 3.3.h, Power Distribution Monitoring System (PDMS) Instrumentation.

In addition, the Byron and Braidwood Stations' Core Operating Limits Report (COLR) proposed changes are included for information only.

The proposed changes are discussed in Section E of this Attachment. The marked-up TS pages are included in Attachments B-1 and B-2 for the Byron and Braidwood Stations, respectively. The clean copy TS pages are included in Attachments B-3 and B-4 for the Byron and Braidwood Stations, respectively. The clean copy TRM pages are included for informational purposes in Attachments B-5 and B-6 for the Byron and Braidwood Stations, respectively. The clean copy TS Bases pages are included for informational purposes in Attachments B-7 and B-8 for the Byron and Braidwood Stations, respectively. The changes to the COLR that result from the proposed TS changes are included for information only in Attachments B-9 and B-10 for the Byron and Braidwood Stations, respectively.

B. DESCRIPTION OF CURRENT REQUIREMENTS

Below is a description of the requirements for the current TS identified in Section A of this Attachment.

TS 3.1.4, Rod Group Alignment Limits

TS 3.1.4 requires that all shutdown and control rods be Operable and individual indicated rod positions be within 12 steps of their group step counter demand position in Modes 1 and 2.

With one or more rod(s) inoperable or with more than one rod not within alignment limit, Shutdown Margin (SDM) must be verified within the limits specified in the COLR or boration must be initiated to restore SDM to within the limit within one hour, and the unit must be in Mode 3 within six hours. With one rod not within alignment limits, SDM must be verified within the limits specified in the COLR or boration must be initiated to restore SDM to within the limit within one hour, and Thermal Power must be reduced to ≤ 75% Rated Thermal Power (RTP) within two hours, and SDM must be verified within the limits specified in the COLR once per 12 hours, and Surveillance Requirement (SR) 3.2.1.1 and SR 3.2.2.1 for verifying F^C_Q (Z) and F^W_Q(Z) must be performed within 72 hours, and safety analyses must be re-evaluated and results confirmed to remain valid for duration of operation under these conditions within five days; otherwise the unit must be in Mode 3 within six hours.

TS 3.1.7, Rod Position Indication

TS 3.1.7 requires that the Digital Rod Position Indication (DRPI) System and the Demand Position Indication System be Operable in Modes 1 and 2.

With one DRPI per group inoperable for one or more groups, the position of the rods with inoperable DRPIs must be verified by using the movable incore detectors once per eight hours, or Thermal Power must be reduced to ≤ 50% RTP within eight hours. With one or more rods with inoperable DRPIs having been moved in excess of 24 steps in one direction since the last determination of the rod's position, action must be initiated to verify the position of the rods with inoperable DRPIs by using the movable incore detectors immediately, or Thermal Power must be reduced to ≤ 50% RTP within eight hours. With one demand position indicator per bank inoperable for one or more banks, all DRPIs for the affected bank(s) must be verified Operable by administrative means once per eight hours and the most withdrawn rod and the least withdrawn rod of the affected bank(s) must be verified ≤ 12 steps apart once per eight hours, or Thermal Power must be reduced to ≤ 50% RTP within eight hours. Otherwise, the unit must be in Mode 3 within six hours.

TS 3.2.1, Heat Flux Hot Channel Factor (F_Q(Z))

TS 3.2.1 requires that F_Q(Z), as approximated by F^C_Q (Z) (i.e., an approximation for F_Q(Z) when the reactor is at steady state power at which the incore flux map was taken) and F^W_Q(Z) (i.e., a cycle dependent function that accounts for power distribution transients encountered during normal operation), be within the limits specified in the COLR in Mode 1.

With F^C_Q (Z) not within the limit, Thermal Power must be reduced ≥ 1% RTP for each 1% F^C_Q (Z) exceeds the limit within 15 minutes (i.e., TS Required Action A.1), and Power Range Neutron Flux-High trip setpoints must be reduced ≥ 1% for each 1% F^C_Q (Z) exceeds the limit within 72 hours, and Overpower Delta Temperature (ΔT) trip setpoints must be reduced ≥ 1% for each 1% F^C_Q (Z) exceeds the limit within 72 hours, and SR 3.2.1.1 and SR 3.2.1.2 for verifying F^C_Q (Z) and F^W_Q(Z) must be performed prior to exceeding the Thermal Power limit of TS Required Action A.1. With F^W_Q(Z) not within the limits, Thermal Power must be reduced ≥ 1% RTP for each 1% F^W_Q(Z) exceeds the limit within four hours (i.e., TS Required Action B.1), and Power Range Neutron Flux-High trip setpoints must be reduced ≥ 1% for each 1% F^W_Q(Z) exceeds the limit within 72 hours, and Overpower ΔT trip setpoints must be reduced ≥ 1% for each 1% F^W_Q(Z) exceeds the limit within 72 hours, and SR 3.2.1.1 and SR 3.2.1.2 must be performed prior to exceeding the Thermal Power limit of TS Required Action B.1. Otherwise, the unit must be in Mode 2 within six hours.

TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

TS 3.2.2 requires that $F^N_{\Delta H}$ be within the limits specified in the COLR in Mode 1.

With $F^N_{\Delta H}$ not within the limits, Thermal Power must be reduced to < 50% RTP within four hours, and SR 3.2.2.1 must be performed within 24 hours, and Power Range Neutron Flux-High trip setpoints must be reduced to ≤ 55% RTP within 72 hours, and SR 3.2.2.1 must be performed prior to exceeding 50% RTP and prior to exceeding 75% RTP and 24 hours after reaching ≥ 95% RTP. Otherwise, the unit must be in Mode 2 within six hours.

TS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

TS 3.2.3 requires that AFD be maintained with the target band specified in the COLR about the target flux difference in Mode 1 with Thermal Power > 15% RTP. AFD may deviate outside the target band with Thermal Power < 90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits specified in the COLR and cumulative penalty deviation time is ≤ one hour during the previous 24 hours. AFD may deviate outside the target band with Thermal Power < 50% RTP.

With Thermal Power ≥ 90% RTP and AFD not within the target band, AFD must be restored to within target band within 15 minutes; otherwise Thermal Power must be reduced to < 90% RTP within 15 minutes. With Thermal Power < 90% RTP and ≥ 50% RTP with cumulative penalty deviation time > one hour during the previous 24 hours or Thermal Power < 90% RTP and ≥ 50% RTP with AFD not within the acceptable operation limits, Thermal Power must be reduced to < 50% RTP within 30 minutes; otherwise Thermal Power must be reduced to < 15% RTP within nine hours.

TS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

TS 3.2.4 requires that the QPTR be ≤ 1.02 in Mode 1 with Thermal Power > 50% RTP.

With QPTR not within limit. Thermal Power must be reduced ≥ 3% from RTP for each 1% of QPTR > 1.00 within two hours after each QPTR determination (i.e., TS Required Action A.1). QPTR must be determined and Thermal Power must be reduced ≥ 3% from RTP for each 1% of QPTR > 1.00 once per 12 hours. SR 3.2.1.1 and SR 3.2.2.1 must be performed within 24 hours after achieving equilibrium conditions from a Thermal Power reduction in accordance with TS Required Action A.1 and once per seven days thereafter. Safety analyses must be re-evaluated and results confirmed to remain valid for duration of operation under this condition prior to exceeding the Thermal Power limit of TS Required Action A.1, and the excore neutron flux detectors must be normalized to restore QPTR to within limits prior to exceeding the Thermal Power limits of TS Required Action A.1. SR 3.2.1.1 and SR 3.2.2.1 must be performed with 24 hours after achieving equilibrium conditions at

RTP not to exceed 48 hours after exceeding the Thermal Power limit of TS Required Action A.1. Otherwise, Thermal Power must be reduced to ≤ 50% RTP within four hours.

TS 3.3.1, Reactor Trip System (RTS) Instrumentation

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during transients, including Anticipated Operational Occurrences (AOOs), and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The RTS functions to maintain the Safety Limits during all AOOs and mitigates the consequences of Design Basis Accidents (DBAs) in all Modes in which the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis takes credit for most RTS trip Functions. RTS Functions not specifically credited in the accident analysis are qualitatively credited in the safety analysis and the NRC approved licensing basis for the unit. These RTS Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS Functions that were credited in the accident analysis.

The TS Limiting Condition for Operation (LCO) requires all instrumentation performing an RTS Function to be Operable when the unit status is within the TS Applicability. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

TS 5.6.5, CORE OPERATING LIMITS REPORT (COLR)

TS 5.6.5 lists the analytical methods used to determine the core operating limits in the COLR previously reviewed and approved by the NRC for the Byron and Braidwood Stations.

C. BASES FOR THE CURRENT REQUIREMENTS

TS 3.1.4, Rod Group Alignment Limits

The Operability (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on Operability ensure that upon a reactor trip, the assumed negative reactivity will be available and will be inserted. The Operability requirements (i.e., trippability to meet SDM) are separate from the alignment requirements, which ensure that the Rod Cluster Control Assemblies

(RCCAs) and banks maintain the correct power distribution and rod alignment. The rod Operability requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move rods (e.g., rod urgent failures), but do not impact trippability, do not result in rod inoperability provided proper alignment is maintained.

TS 3.1.7, Rod Position Indication

The Operability, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess Operability and misalignment.

LCO 3.1.7 specifies that the DRPI System and the Bank Demand Position Indication System be Operable for each control rod. For the control rod position indicators to be Operable the following requirements must be met:

a. The DRPI System indicates within 12 steps of the group step counter demand position;

b. The DRPI System has no failed coils; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis that specified control rod group insertion limits.

These requirements ensure that control rod position indication during power operation and Physics Tests is accurate, and that design assumptions are not challenged.

Operability of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

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

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

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

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

This TS 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 TS limits, reduction of the Thermal Power is required.

TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

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

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. $F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR. The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a Departure from Nucleate Boiling (DNB). The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the plant safety analyses.

TS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

The purpose of this TS is to establish limits on the values of the AFD in order to limit the axial (i.e., vertical) 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 the axial power distribution control.

The operating scheme used to control the axial power distribution uses the CAOC Methodology. This methodology 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 unit maneuvers.

The shape of the power profile in the axial 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 from the Nuclear Instrumentation System (NIS) excore neutron flux detectors. Separate signals are taken from the top and bottom excore detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore neutron flux detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as % flux or % ΔI.

With Thermal Power ≥ 90% RTP (i.e., Part "a" of this TS LCO), the AFD must be kept within the target band about the target flux difference. With the AFD outside the target band with Thermal Power ≥ 90% RTP, the assumptions of the accident analyses may be violated.

It is intended that the unit 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 TS LCO), a one hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR.

For Thermal Power levels > 15% RTP and < 50% RTP (i.e., Part "c" of this TS LCO), deviations of the AFD outside of the target band are less significant. The accumulation of 0.5 minute penalty deviation time per one minute of actual time outside the target band 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 cumulative penalty time is the sum of penalty times identified in Parts "a," "b," and "c" of this TS LCO.

TS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

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.

The QPTR limit of 1.02, at 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 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged.

TS 3.3.1, Reactor Trip System (RTS) Instrumentation

TS 3.3.1, Condition D, applies to the Power Range Neutron Flux High Function. The NIS power range detectors provide input to the Rod Control System and the Steam Generator Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition within six hours. This results in a partial trip condition requiring only one-out-of-three logic for actuation. In addition to placing the inoperable channel in the tripped condition, Thermal Power must be reduced to ≤ 75% RTP within 12 hours.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within six hours and the QPTR monitored once every 12 hours in accordance with SR 3.2.4.2. Calculating QPTR every 12 hours compensates for the potential lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels > 75% RTP. The six hour Completion Time and the 12 hour Frequency are consistent with TS 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

As an alternative to the above actions, the plant must be placed in an operational Mode where this Function is no longer required to be Operable. Twelve hours are allowed to place the plant in Mode 3. This is a reasonable time, based on operating experience, to reach Mode 3 from full power in an orderly manner and without challenging plant systems. If TS Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 (i.e., if the LCO and associated Actions are not met, an associated Action is not provided, or if directed by the associated Actions, the unit must be placed in a Mode or other specified condition in which the LCO is not applicable) must be entered.

TS Required Action D.2.2 has been modified by a Note that only requires SR 3.2.4.2 to be performed if the Power Range Neutron Flux input to QPTR becomes inoperable. Failure of a component in the Power Range Neutron Flux Channel that renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using this movable incore detectors once per 12 hours may not be necessary. TS Required Action D.2.2 is duplicative of the requirements of SR 3.2.4.2.

TS 5.6.5, CORE OPERATING LIMITS REPORT (COLR)

TS 5.6.5 lists the analytical methods used to determine the core operating limits in the COLR previously reviewed and approved by the NRC for the Byron and Braidwood Stations.

D. NEED FOR REVISION OF THE REQUIREMENTS

In the present PWR core monitoring methodology, there is no direct margin assessment on a continuous basis. The current reactor core design and surveillance methodologies impact the capabilities for increased duties on the fuel. Current fuel cycles contain design margins to assure safe core operation under steady state and transient conditions due to the operator's inability to continuously monitor the core power distribution. These margins far exceed the actual operational requirements.

The proposed changes allow us to use the Westinghouse core monitoring and support system known as BEACON. The BEACON PDMS has been developed to improve the operational support for PWRs. It is an advanced core monitoring and support package, which uses current instrumentation in conjunction with a fully analytical methodology for on-line generation of 3-D core power distributions. The system provides core monitoring, core measurement reduction, core analysis and follow, and core predictions. The PDMS maintains an on-line 3-D nodal model that is continuously updated to reflect the current plant operation conditions. The nodal solution method used by the PDMS is the same as the NRC approved Westinghouse Advanced Nodal Code (ANC) core design code (Reference 3). The utilization of 3-D nodal calculations to generate the reference power distribution overcomes the shortcomings associated with previous methods, which used various reconstruction functions to generate the power distribution. The Core Exit Thermocouples (CETCs) and excore neutron flux detectors are used with the reference 3-D power distribution to determine the measured power distribution. By coupling the measured 3-D power distribution with an on-line evaluation, actual core margins are better understood. The PDMS provides an understanding of operating and design margins to address strategic fuel cycle changes. The BEACON methodology (Reference 1) would allow for changes in the core design methods and provide for more optimized core loading patterns. Additionally, the BEACON methodology significantly improves the quality of the surveillance process since it uses a depleted model to match the actual operational profile. The PDMS continuously monitors the limiting $F_Q(Z)$, $F^N_{\Delta H}$, and DNBR and enhances operational flexibility (i.e., replaces the current AFD and QPTR limits). The NRC approved the WCAP in February 1994.

Implementation of the PDMS at the Byron and Braidwood Stations does not replace, eliminate, or modify existing plant instrumentation. The PDMS software runs on a workstation connected to the plant process computer. The PDMS combines input from currently installed plant instrumentation and design data generated each fuel cycle. Together, this data provides a means to monitor power distribution limits continuously and to alert the operator when limits are being approached.

As part of the proposed implementation of BEACON, we are planning to utilize RAOC methodology for determining AFD. This analytical method was provided in WCAP-10216-P-A, Revision 1, and approved by the NRC in November, 1993 (Reference 8), and is already referenced in the Byron and Braidwood Stations' TS 5.6.5. A change from CAOC to RAOC is an option contained in WCAP-12472-P-A.

E. DESCRIPTION OF THE PROPOSED CHANGES

The TS discussed in WCAP-12472-P-A are applicable to an application of PDMS in which the core monitoring application of the BEACON methodology is in use continuously. This situation will exist for the application of the BEACON methodology to Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. In this application, if PDMS meets specified Operability requirements, it is used to generate detailed power distribution information and comparisons to core limits on a continuous basis and to supply that information to the operator.

Byron and Braidwood Stations converted to the ISTS, NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 1, in February 1999. Therefore, the TS markups contained in WCAP-12472-P-A, which are based on NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Draft Revision 5, are not directly applicable to the Byron and Braidwood Stations. Many changes to the WCAP-12472-P-A Specifications would be expected when converted to NUREG-1431, Revision 1, format. These differences from the WCAP-12472-P-A Specifications are not specifically detailed and justified here.

The following changes to the Byron and Braidwood Stations' TS are proposed, and where the change differs in technical nature from the WCAP-12472-P-A Specifications, these differences are discussed.

TS 3.1.4, Rod Group Alignment Limits:

a. The Completion Time for TS Required Action B.2 has been revised from "2 hours" to "2 hours from discovery of Condition B (i.e., one rod not within alignment limits) concurrent with inoperability of Power Distribution Monitoring System (PDMS)." This change will only require Thermal Power to be reduced to ≤ 75% RTP within two hours with one rod not within alignment limit when PDMS is inoperable. Reduction of power to 75% RTP when PDMS is inoperable, ensures that local Linear Heat Rate (LHR) increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. The Completion Time of two hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the RTS. This change is necessary in the event that PDMS becomes inoperable more than two hours after entering Condition B. in which case the Completion Time for TS Required Action B.2 would have already expired Therefore, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both: a. One rod is not within alignment limits. and b PDMS is inoperable. Discovering one rod not within alignment limits coincident with PDMS inoperable results in starting the Completion Time clock for the Required Action.

   During power operation when PDMS is Operable, LHR is measured continuously. Therefore, a reduction of power to 75% RTP is not necessary to ensure that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. This change is reflected in the WCAP-12472-P-A presentation of Specification 3.1.3.1 Action b.3.b).

b. Required Actions B.4 and B.5 for performing SR 3.2.1.1 and SR 3.2.2.1, respectively, are editorially combined into TS Required Action B.4, and TS Required

Action B.6 renumbered to B.5, and are modified to state that $F_Q(Z)$ and $F^N_{\Delta H}$ shall be "determined" within 72 hours of entering Condition B rather than stating specific Surveillances for $F_Q(Z)$ and $F^N_{\Delta H}$. This presentation prevents having separate Required Actions for performing Surveillances for $F_Q(Z)$ and $F^N_{\Delta H}$ depending upon the Operability status of PDMS (i.e., perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1 with PDMS inoperable, and perform SR 3.2.1.3, SR 3.2.1.4, and SR 3.2.2.2 with PDMS Operable). This change is reflected in the WCAP-12472-P-A presentation of Specification 3.1.3.1 Action b.3.a).

c. Byron and Braidwood Stations' current TS Required Action D.2, for more than one rod not within alignment limits, requires the unit to be in Mode 3 within six hours. The NRC Safety Evaluation (SE), dated February 16, 1994 (Reference 7), Section 2.0, "Evaluation," page 6, discusses an approved change to TS 3.1.3.1. In part, the NRC approved that the "... required shutdown for more than one rod inoperable but trippable is extended from six hours to 72 hours because of available information on power distribution." This change was originally reflected in WCAP-12472, Section 7, "Technical Specification Modification," (i.e., TS 3.1.3.1, Action b.2), but when this Section was resubmitted to the NRC on November 4, 1992, the change was inadvertently omitted. Since PDMS can adequately monitor the critical operating parameters with rod misalignments, this change does not represent a significant impact on safety. Therefore, TS Required Action C.3 was added to allow 72 hours to "Restore rod(s) to within alignment limit," with a Note stating, "Only required to be performed when PDMS is OPERABLE."

   Required Action C.2 retains the current requirement for being in Mode 3 within six hours with more than one rod not within alignment limits. The Completion Time for TS Required Action C.2 has been revised from "6 hours" to "6 hours from discovery of Condition C concurrent with inoperability of PDMS." This change will only require the unit to be in Mode 3 with more than one rod not within alignment limits when PDMS is inoperable. The allowed Completion Time is reasonable, based on operating experience, for reaching Mode 3 from full power conditions in an orderly manner and without challenging plant systems. This change is necessary in the event that PDMS becomes inoperable more than six hours after entering Condition C, in which case the Completion Time for TS Required Action C.2 would have already expired. Therefore, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both: a. More than one rod is not within alignments limit. and b PDMS is inoperable. Discovering more than one rod not within alignment limits coincident with PDMS inoperable results in starting the Completion Time clock for the Required Action.

d. In revising the current TS Required Action D.2 (i.e., Be in Mode 3 within six hours), Condition D and associated Required Actions are renumbered as "C." Similarly, the current Condition C and associated Required Actions are renumbered as "D." Condition D has been revised to state, "Required Action and associated Completion Time of Condition B or TS Required Action C.3 not met." The addition of "Required Action C.3" is necessary so that with more than one rod not within alignment limits when PDMS is Operable, if the rod(s) is(are) not restored to within alignment limits within 72 hours, the unit is required to be in Mode 3 within six hours.

TS 3.1.7, Rod Position Indication

The requirements for using the movable incore detectors to verify the position of the rods with inoperable DRPIs in TS Required Action A.1 (i.e., one DRPI per group inoperable for one or more groups) and TS Required Action B.1 (i.e., one or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the determination of the rod's position) are relocated to the Bases. The Bases contains the details associated with verifying the position of the rods with inoperable DRPIs depending upon the Operability of PDMS. The change to TS 3.1.7 allows the use of the Operable PDMS or the movable incore detectors for verifying the position of the rod with an inoperable DRPI. This change is reflected in the NRC SE, dated February 16, 1994 (Reference 7), Section 2.0, "Evaluation," page 6, Item (b). It is also reflected in WCAP-12472-P-A, Section 7, "Technical Specification Modification," TS 3.1.3.2, Action a.1.

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

a. Required Actions A.4 and B.4 for performing SR 3.2.1.1 and SR 3.2.1.2 are deleted. In the current TS these Required Actions are only explicitly stating the implicit requirement, i.e., $F_Q(Z)$ must be within limits before the Thermal Power limit imposed by Required Actions can be exceeded. $F_Q(Z)$ is only determined by the performance of SRs 3.2.1.1 and 3.2.1.2, and once verified to be within limit, the Condition is exited and the TS Required Action A.1 or B.1 limit is no longer in effect. With the changes being made to adopt PDMS, the system continuously monitors for compliance with the $F_Q(Z)$ limit. The same rules of usage apply, i.e., $F_Q(Z)$ must be within limits by performing flux mapping SRs as before or by utilizing an Operable PDMS, which is reflected in new alternate SRs discussed below before the Thermal Power limit imposed by TS Required Action A.1 or B.1 can be exceeded. Explicitly stating this in TS Required Action A.4 and B.4 is unnecessary, and furthermore, would add increased unnecessary complexity to now have to also address the PDMS-based options (i.e., PDMS Operable or PDMS inoperable).

b. SRs 3.2.1.3 and 3.2.1.4. Note 2 to SR 3.2.1.1, and Note 3 to SR 3.2.1.2 are added to address the $F_Q(Z)$ monitoring requirements of an Operable PDMS. Note that the current TS 3.2.1 Note applicable to all SRs is editorially moved to each of the existing SRs (i.e., SRs 3.2.1.1 and 3.2.1.2) as Note 1, since this Note will not apply to the new PDMS-based SRs The added Note (i.e., Note 2 to SR 3.2.1.1 and Note 3 to SR 3.2.1.2) states, "Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.1.3 (SR 3.2.1.4) satisfies the initial performance of this SR after declaring PDMS inoperable." This Note removes the requirement to utilize incore flux mapping to determine $F_Q(Z)$, and furthermore allows a reasonable period of time (i.e., 12 hours) to perform these SRs in the event PDMS is declared inoperable. This will avoid an immediate failure to perform a Surveillance when PDMS initially becomes inoperable. When PDMS becomes inoperable, rather than having to determine $F^C_Q(Z)$ and $F^W_Q(Z)$ using the movable incore detectors within 12 hours, this Note also allows credit to be taken for the PDMS determination of $F_Q(Z)$ using the last SR when PDMS was Operable. While this allowance is not reflected in WCAP-12472-P-A, it is reasonable based on previous monitoring provided by the Operable PDMS and the minimal probability of significant changes in core reactivity during this time.

Note that WCAP-12472-P-A presented the PDMS-based determination of $F_Q(Z)$ within the new, separate, TS for PDMS. This presentation revised the Applicability of the $F_Q(Z)$ TS (i.e., TS 3.2.2 in the WCAP) to be only when PDMS was inoperable. The Byron and Braidwood Stations' presentation results in identical requirements, except for the Surveillance Frequency as discussed below, and reflect a presentation preference. This is consistent with the NRC SE, dated February 16, 1994 (Reference 7), Section 2.0, "Evaluation," page 8. The evaluation of the WCAP proposed BEACON TS, identifies that the actions are "consistent with current practice, and are therefore acceptable." Since the Byron and Braidwood Stations' Required Actions for $F_Q(Z)$, and for $F^N_{\Delta H}$ as discussed below, differ from the Required Actions for DNBR, a revised format is appropriate.

WCAP-12472-P-A proposed the Frequency for determining power distribution and reactivity control parameters and limits (i.e., as presented in the WCAP proposed TS SR 4.2.6) as once per eight hours. Since the current non-PDMS nominal Surveillance Frequencies for this monitoring varies from once per seven days (i.e., for AFD and QPTR) to once per 31 Effective Full Power Days (EFPD) (i.e., for $F_Q(Z)$ and $F^N_{\Delta H}$), and since the BEACON design provides a continuous monitoring capability, the proposed once per eight hours is deemed unnecessarily restrictive. Given the design provides for continuous monitoring, the requirement to periodically log the values as a TS requirement should not be required any more frequently than in current TS. As such a seven day Frequency is provided.

TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

a. Required Actions A.2 and A.4 are modified to state that $F^N_{\Delta H}$ shall be "determined," rather than stating a specific SR for $F^N_{\Delta H}$ (i.e., perform SR 3.2.2.1). This presentation prevents having separate Required Actions for performing the Surveillance for $F^N_{\Delta H}$ depending upon the Operability status of PDMS (i.e., perform SR 3.2.2.1 with PDMS inoperable and perform SR 3.2.2.2 with PDMS Operable). This change is reflected in the WCAP-12472-P-A presentation of TS 3.2.3 Action b.

b. SR 3.2.2.2 and the Note to SR 3.2.2.1 are added to address the $F^N_{\Delta H}$ monitoring requirements of an Operable PDMS The added SR 3.2.2.1 Note states, "Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.2.2 satisfies the initial performance of this SR after declaring PDMS inoperable." This Note removes the requirement to utilize incore flux mapping to determine $F^N_{\Delta H}$, and furthermore allows a reasonable period of time (i.e., 12 hours) to perform the SR in the event PDMS is declared inoperable. This will avoid an immediate failure to perform a surveillance when PDMS initially becomes inoperable. When PDMS becomes inoperable, rather than having to determine $F^N_{\Delta H}$ using the movable incore detectors within 12 hours, this Note also allows credit to be taken for the PDMS determination of $F^N_{\Delta H}$ using the last SR when PDMS was Operable. While this allowance is not reflected in WCAP-12472, it is reasonable based on previous monitoring provided by the Operable PDMS and the minimal probability of significant changes in core reactivity during this time.

Note that WCAP-12472-P-A presented the PDMS-based determination of $F^N_{\Delta H}$ within the new, separate, TS for PDMS. This presentation revised the Applicability of the $F^N_{\Delta H}$ TS (i.e., TS 3.2.2 in the WCAP) to be only when PDMS was inoperable. The Byron and Braidwood Stations' presentation results in identical requirements, except for the Surveillance Frequency as discussed below, and reflects a presentation preference. This is consistent with the NRC SE, dated February 16, 1994 (Reference 7), Section 2.0, "Evaluation," page 8. The evaluation of the WCAP proposed "BEACON Specification," identifies that the actions are "consistent with current practice, and are therefore acceptable." Since the Byron and Braidwood Stations' actions for $F^N_{\Delta H}$ and $F_Q(Z)$, as discussed above, differ from the actions for DNBR, a revised format is appropriate.

WCAP-12472-P-A proposed the frequency for determining power distribution and reactivity control parameters and limits as presented in the WCAP proposed TS SR 4.2.6 as once per eight hours. Since the current non-PDMS nominal Surveillance Frequencies for this monitoring varies from once per seven days for AFD and QPTR to once per 31 EFPD for $F_Q(Z)$ and $F^N_{\Delta H}$, and since the BEACON design provides a continuous monitoring capability, the proposed once per eight hours is deemed unnecessarily restrictive. Given the design providing continuous monitoring, the requirement to periodically log the values as a TS requirement should not be required any more frequently than in current TS. As such a seven day Frequency is provided.

TS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

a. TS 3.2.3 is changed to conform to RAOC methodology. As part of the implementation of PDMS, the Byron and Braidwood Stations are utilizing RAOC methodology for determining AFD. This analytical method was provided in WCAP-10216-P-A, Revision 1 (Reference 2) and approved by the NRC (Reference 8). This WCAP reference is included in the Byron and Braidwood Stations' TS 5.6.5 Section b.11, "COLR." This is consistent with WCAP-12472, Section 7, "Technical Specification Modification, Specification 3.3.2."

b. The Applicability is modified to add, "... when Power Distribution Monitoring System (PDMS) is inoperable." In the event that PDMS becomes inoperable, the plant must revert to the AFD TS. When PDMS is Operable, PDMS is directly monitoring the key power distribution parameters (i.e., $F_Q(Z)$, $F^N_{\Delta H}$, and DNBR) continuously. This direct monitoring capability eliminates the need to monitor the indirect indicators (i.e., AFD and QPTR) used in the current TS. This change is reflected in the WCAP-12472-P-A presentation of TS 3.2.1.

c. A Note stating, "Not required to be performed until 12 hours after declaring PDMS inoperable," is added to SR 3.2.3.1 to allow a reasonable period of time (i.e., 12 hours) to perform this SR in the event PDMS is declared inoperable. This will avoid an immediate failure to perform a surveillance when PDMS initially becomes inoperable. While this allowance is not reflected in WCAP-12472, it is reasonable based on previous monitoring provided by the Operable PDMS and the minimal probability of significant changes in core reactivity during this time.

The NUREG-1431, Revision 1, Frequency for verifying AFD within limits for each Operable excore channel is every seven days and "Once within one hour and every one hour thereafter with the AFD monitor alarm inoperable." As justified in the conversion to ISTS, the Byron and Braidwood Stations relocated the conditional surveillance requirement (i.e., once within one hour and every one hour thereafter with the AFD monitor alarm inoperable) to the TRM. Therefore, consistent with the current SR 3.2.3.1, the Frequency for the new SR 3.2.3.1 remains as seven days.

d. The NUREG-1431, Revision 1, LCO for the RAOC methodology states, "The AFD in % flux difference units shall be maintained within the limits specified in the COLR." The unit for AFD, i.e., % flux difference units, was deleted. This amount of detail is not commonly contained in TS and is more appropriate for the Bases. The LCO Section of the Bases for TS 3.2.3 will state, "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."

TS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

a. The Applicability is modified to add, "... when Power Distribution Monitoring System (PDMS) is inoperable." In the event that PDMS becomes inoperable, the plant must revert to the QPTR TS. 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. With PDMS Operable, precise radial power distribution measurements are obtained continuously and therefore QPTR limits are not required. When PDMS is Operable, PDMS is directly monitoring the key power distribution parameters (i.e., $F_Q(Z)$, $F^N_{\Delta H}$, and DNBR) continuously. This direct monitoring capability eliminates the need to monitor the indirect indicators (i.e., AFD and QPTR) used in the current TS. This change is reflected in the WCAP-12472-P-A presentation of TS 3.2.1.

b. SR 3.2.1.2 for verifying $F^W_Q(Z)$ is within limits is added to the list of SRs that are required to be performed in TS Required Action A.3 and TS Required Action A.6 with QPTR not within limits. Since $F_Q(Z)$ is approximated by $F^C_Q(Z)$ (i.e., SR 3.2.1.1) and $F^W_Q(Z)$ (i.e., SR 3.2.1.2), SR 3.2.1.2 is added for completeness. This change is unrelated to PDMS implementation.

c. A Note stating, "Not required to be performed until 12 hours after declaring PDMS inoperable," is added to SR 3.2.4.1 and SR 3.2.4.2 to allow a reasonable period of time (i.e., 12 hours) to perform these SRs in the event PDMS is declared inoperable. This will avoid an immediate failure to perform a surveillance when PDMS initially becomes inoperable. While this allowance is not reflected in WCAP-12472-P-A, it is reasonable based on previous monitoring provided by the Operable PDMS and the minimal probability of significant changes in core reactivity during this time.

TS 3.2.5, Departure From Nucleate Boiling Ratio (DNBR)

a. PDMS monitors the key normal operation reactor parameters (i.e., $F_Q(Z)$, $F^N_{\Delta H}$, and DNBR) continuously during normal operation. This direct monitoring capability eliminates the need for the current indirect indicators (i.e., AFD and QPTR). Of these key parameters in the current TS, PDMS results in explicit and continuous confirmation that the reactor core is operating within true normal operation safety limits and provides more flexible plant operations. This change is reflected in the WCAP-12472-P-A presentation of TS 3.2.6.

b. WCAP-12472-P-A proposed the frequency for determining power distribution and reactivity control parameters and limits, presented in the WCAP proposed TS SR 4.2.6, as once per eight hours. Since the current non-PDMS nominal Surveillance Frequencies for this monitoring varies from once per seven days for AFD and QPTR to once per 31 EFPD for $F_Q(Z)$ and $F^N_{\Delta H}$, and since the BEACON design provides a continuous monitoring capability, the proposed once per eight hours is deemed unnecessarily restrictive. Given the design providing continuous monitoring, the requirement to periodically log the values as a TS requirement should not be required any more frequently than in current TS. As such a seven day Frequency is provided.

TS 3.3.1, Reactor Trip System (RTS) Instrumentation

Required Actions D.1.2 and D.2.2, for one inoperable Power Range Neutron Flux-High channel, have been deleted since they are already adequately addressed by the QPTR TS (i.e., LCO 3.2.4, and SRs 3.2.4.1 and 3.2.4.2). With the input to QPTR from one Power Range Neutron Flux Channel inoperable with Thermal Power > 75% RTP, SR 3.2.4.2 verifies QPTR ≤ 1.02 using the movable incore detectors, thereby compensating for the potential lost monitoring capability due to the inoperable Power Range Neutron Flux Channel and allows continued operation at power levels > 75% RTP (i.e., the equivalent of TS Required Action D.2.2). In addition, the TS 3.3.1 Required Actions D.1.2 and D.2.2 may be too conservative in the instance where the Power Range Neutron Flux Channel's inoperability affects only the RTS function, but does not affect the input to QPTR. Only when one Power Range Neutron Flux Channel does not provide accurate input to QPTR does the QPTR TS impose limitations on Thermal Power or increased performance of QPTR determination. Therefore, the proposed Required Actions for the inoperability of an RTS Power Range Neutron Flux Channel address only the RTS safety function. Any inoperability in the input to QPTR, which may or may not be concurrent with an RTS inoperability, is adequately addressed in SRs 3.2.4.1 and SR 3.2.4.2.

TS 5.6.5, CORE OPERATING LIMITS REPORT (COLR)

TS 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)," was added to the list of TS for which core operating limits are documented in the COLR.

TS 5.6.5 lists the analytical methods used to determine the core operating limits previously reviewed and approved by the NRC. WCAP-12472-P-A was added to this list of analytical methods.

These changes are consistent with WCAP-12472.

TRM

Similar to the Movable Incore Detectors, PDMS Instrumentation does not meet the selection criteria set forth in 10 CFR 50.36(c)(2)(ii). Therefore, the PDMS Instrumentation requirements are contained in the TRM along with necessary changes to the TRM Specification for Movable Incore Detectors.

The justification for not including PDMS Instrumentation in the TS is outlined in the screening below.

The purpose of this screening is to determine if the structures, systems, or components associated with PDMS Instrumentation are required to be contained in the TS. This screening is done in accordance with the requirements contained in 10 CFR 50.36(c)(2)(ii). A TS LCO must be established for each item meeting one or more of the following criteria:

(A) **Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.**

PDMS Instrumentation is not associated with monitoring of any aspect of the reactor coolant pressure boundary.

(B) **A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.**

The limits for the power distribution parameters $F_Q(Z)$, $F^N_{\Delta H}$, and DNBR are operating restrictions, which ensure that all analyzed DBAs remain valid. These limits are included in the TS for the Byron and Braidwood Stations. The PDMS Instrumentation, however, as a processor provides the capability to monitor these parameters at more frequent intervals than are currently required by TS. Additionally, these limits can be determined independent of the Operability of PDMS. Therefore, PDMS Instrumentation is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) **A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.**

PDMS Instrumentation provides the capability to monitor core power distribution parameters at more frequent intervals than is currently required by TS. PDMS Instrumentation does not change any of the key safety parameter limits or levels of margin as considered in the reference design basis evaluations. The demonstrated adherence to these standards and criteria precludes new risks to components and systems that could introduce a new type of accident. All design and performance criteria will continue to be met and no new failure modes or limiting single failure mechanisms have been created nor will the core operate in excess of pertinent design basis operating limits for the key safety parameters. The PDMS Instrumentation has no functions or actuations that mitigate any DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

PDMS Instrumentation provides the capability to monitor power distribution parameters at more frequent intervals than is currently required by TS. PDMS Instrumentation is a system utilized to monitor the core power distribution and has no impact on the results or consequences of any DBA or transient analysis. Implementation of PDMS does not create the possibility of a malfunction of equipment important to safety different from any already evaluated in the Byron and Braidwood Stations' Updated Final Safety Analysis Report (UFSAR). The demonstrated adherence to these standards and criteria precludes new risks to components and systems that could introduce a new type of a malfunction of equipment important to safety. Therefore it has no impact on public health and safety.

The evaluation completed above indicates that PDMS Instrumentation does not meet any of the criteria for inclusion in the TS.

The WCAP-12472-P-A TS for BEACON System Instrumentation (i.e., Table 3.3.3-12 of TS 3.3.3.12) contained "Pressurizer Pressure" as a required input to the PDMS Instrumentation. We have removed the "Pressurizer Pressure" input from the list of the PDMS Instrumentation Functions (i.e., TRM Table T3.3.h-1) required for PDMS Operability since this variable does not impact either the Operability of PDMS or its uncertainty evaluations. If actual pressurizer pressure deviates outside of a preset operating pressure band, PDMS will default to the nominal pressurizer pressure. (i.e., 2235 psig). Therefore, a loss of the pressurizer pressure input would not result in a PDMS inoperability.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The proposed changes allow us to use the Westinghouse core monitoring and support system known as BEACON. The PDMS has been developed to improve the operational support for PWRs. It is an advanced core monitoring and support package, which uses current instrumentation in conjunction with a fully analytical methodology for on-line generation of 3-D core power distributions, i.e., SPNOVA (Reference 5). SPNOVA is based on two group-diffusion equations and is used, for example, to provide data reduction of incore flux maps, core parameter (e.g., fuel burnup, xenon, etc.) analysis and follow, and core prediction (e.g., critical conditions for startup). Its primary role in BEACON, however, is to generate detailed power distribution information and comparisons to core limits on a continuous basis and to supply that information to the operator. SPNOVA is calibrated periodically using the incore neutron flux measurement system, i.e., the Movable Incore Detection (MID) System, to provide details of the power distribution, and is calibrated continuously using the CETCs for radial updating and using the excore neutron flux detectors for axial updating. The MID System information is also used to calibrate the CETCs and the excore neutron flux detectors. The SPNOVA code was updated in Reference 4 to include the same neutronics solution module as the ANC design code (Reference 3).

WCAP-12472-P-A, "BEACON – Core Monitoring and Operations Support System," was issued in August 1994 (Reference 1) and approved by the NRC on February 16, 1994 (Reference 7). Reference 1 describes the system, the methodologies involved, the calibration processes, the uncertainties to be associated with the determined power distributions, xenon transient and criticality analysis, and the necessary TS changes for PDMS Operable and inoperable conditions. As part of the proposed implementation of BEACON, we are planning on utilizing RAOC methodology for determining AFD. This analytical method was previously provided in WCAP-10216-P-A, Revision 1 (Reference 2) and approved by the NRC on November 26, 1993 (Reference 8), and is included in the Byron and Braidwood Stations' TS 5.6.5, Section b.11. RAOC methodology is used during the reload cycle design process to analyze and verify the applicable RAOC based AFD space by evaluating the available power margins for the particular conditions of the core through the measurement of $F_Q(Z)$, $F^N_{\Delta H}$, and DNBR.

The PDMS software runs on a workstation connected to the plant process computer. PDMS combines input from currently installed plant instrumentation and design data generated each cycle. The process computer continuously monitors information generated by PDMS and provides means to alert the operator when core operating limits are being approached. The proposed implementation of the PDMS at the site does not impose any replacements, elimination, or modification of existing plant instrumentation.

BEACON provides a greatly improved continuous on-line power distribution measurement and display, limit surveillance, and operation prediction information system. No new instrumentation or calculation system other than interface systems and integration analysis is introduced. As discussed in the Technical Evaluation Report (TER) that is included in WCAP-12472-P-A, the system review has concluded that BEACON is acceptable for performing core monitoring and operations support functions for Westinghouse PWRs subject to the conditions summarized below.

1) In the cycle-specific applications of BEACON, the power peaking uncertainties $U_{\Delta H}$ and U_Q must provide 95 % probability upper tolerance limits at the 95 % confidence level (i.e., Section 3.3 of the TER).

2) In order to insure that the assumptions made in the BEACON uncertainty analysis remain valid, the generic uncertainty components may require reevaluation when BEACON is applied to plant or core designs that differ

sufficiently to have a significant impact on the WCAP-12472-P-A database (i.e., Section 3.4 of the TER).

We have analyzed the above conditions and have found them satisfied for all core monitoring and operations support functions that will be performed by the BEACON system for the Byron and Braidwood Stations. The following discussion summarizes this conclusion.

1) In the cycle-specific applications of BEACON, the power peaking uncertainties $U_{\Delta H}$ and U_Q must provide 95 % probability upper tolerance limits at the 95 % confidence level.

There is no change to the BEACON core monitoring methodology applicable to the Byron and Braidwood Stations.

For the Byron and Braidwood Stations' applications, determination of the BEACON uncertainty components is consistent with the methods discussed in WCAP-12472-P-A.

The uncertainties to be applied to the BEACON power distribution measurements are calculated differently than those applied to the traditional flux map systems because BEACON uses a more comprehensive set of instrumentation.

The uncertainty in the BEACON power peaking resulting from errors in the SPNOVA model calibration and CETC calibration is determined using a Monte Carlo error propagation technique. In this analysis, the BEACON three-step calibration, model update and power distribution update procedure is simulated. The SPNOVA model and CETC calibration factors are subjected to random variations, based on their uncertainties, and the resulting variations in the BEACON power distribution are used to determine the 95% probability upper tolerance limit on the assembly power for the approximately twenty highest powered assemblies.

The analysis is performed for a range of operating conditions including off-normal power distributions and extended calibration intervals. A typical set of CETC uncertainties is used together with a relatively large tolerance factor, which results in substantial smoothing of the CETC measurements. The upper tolerance limit on the assembly power peaking factor is calculated and found to increase as the square root of the CETC uncertainty.

The $F^N_{\Delta H}$ and $F_Q(Z)$ uncertainties are determined by a statistical combination of the assembly peaking factor, axial peaking factor, calibration interval, inoperable detector, and local power peaking component uncertainties. The $F^N_{\Delta H}$ and $F_Q(Z)$ uncertainties are continuously updated by the PDMS for actual operating conditions. With the PDMS inoperable, a set of uncertainty constants to be applied is listed in the COLR.

The uncertainties in DNBR calculations are considered in all items that may significantly affect the calculated DNBR values. For the applicable DNBR limit of 1.4 (i.e., the safety analysis limit used in the Byron and Braidwood Station Revised

Thermal Design Procedure analyses), only a random portion of the plant operating parameter uncertainties can be included in the statistical combination of the effect of the uncertainties for the Design Limit (DL) DNBR. Instrumentation bias is treated as a direct DNBR penalty.

To distinguish between an uncertainty and a bias, for the purpose of statistical determination of DNBR limits, an uncertainty would be some distribution of probable values that the plant could see about a defined mean. The definition of the distribution would depend upon the parameter, how it is affected in operation, and the instrumentation system. A bias, on the other hand, is an adjustment of the mean. It occurs in a plant analysis when there is a known, or suspected, uniformly observed change in the measurement of a parameter. Biases are applied directly to analysis input. The purpose of both is to arrive at a most probable core condition for the plant with respect to DNBR determination, then to set a limit on DNBR through the establishment of uncertainties and biases such that the probability that DNB will not occur on the most limiting fuel rod is at a 95% confidence level.

The accuracy of the BEACON analysis decreases as the calibration intervals increase and the power distribution diverges from the reference power shape. In order to minimize BEACON uncertainty, the reference power distribution is updated every 15 minutes, or when significant changes occur in the AFD or reactor power.

Power Distribution Measurement Uncertainty with PDMS Operable

$U_{F\Delta H}$ Power Distribution Measurement Uncertainty to be applied to $F^N_{\Delta H}$ shall be calculated by the following formula:

$$U_{F\Delta H} = 1.0 + \frac{U_{\Delta H}}{100.0}$$

where:

$U_{\Delta H}$ = Uncertainty for enthalpy rise as defined in Reference 1.

U_{FQ} Power Distribution Measurement Uncertainty to be applied to $F_Q(Z)$ shall be calculated by the following formula:

$$U_{FQ} = \left(1.0 + \frac{U_Q}{100.0}\right) \bullet U_e$$

where:

U_Q = Uncertainty for $F_Q(Z)$ as defined in Reference 1.
U_e = Engineering uncertainty factor = 1.03, or as defined in the COLR.

Note: With the PDMS Operable, the Power Distribution Measurement Uncertainties $U_{F\Delta H}$ and U_{FQ} are continuously calculated by PDMS and applied to the PDMS Monitoring Function.

Power Distribution Measurement Uncertainty with PDMS inoperable:

$U_{F\Delta H}$ Power Distribution Measurement Uncertainty to be applied to $F^N_{\Delta H}$ shall be calculated by the following formula:

$$U_{F\Delta H} = U_{F\Delta Hm}$$

where:

$U_{F\Delta Hm}$ = Base $F^N_{\Delta H}$ measurement uncertainty = 1.04, or as defined in COLR.

U_{FQ} Power Distribution Measurement Uncertainty to be applied to $F_Q(Z)$, shall be calculated by the following formula:

$$U_{FQ} = U_{qu} \bullet U_e$$

where:

U_{qu} = Base $F_Q(Z)$measurement uncertainty = 1.05, or as defined in the COLR.

U_e = Engineering uncertainty factor = 1.03, or as defined in the COLR.

Note: With the PDMS inoperable, the Power Distribution Measurement Uncertainties $U_{F\Delta H}$ and U_{FQ} are specified in the COLR and shall be applied by hand calculations.

2) **In order to insure that the assumptions made in the BEACON uncertainty analysis remain valid, the generic uncertainty components may require reevaluation when BEACON is applied to plant or core designs that differ sufficiently to have a significant impact on the WCAP-12472-P-A database.**

For the Byron and Braidwood Stations' applications, the cycle prior to the installation of BEACON will be examined to establish reference uncertainties for use by BEACON. For this examination, a BEACON model is compared to actual operating data (e.g., flux maps, thermocouple maps, etc.) from the reactor for that cycle. This process will reevaluate the generic uncertainty components to account for plant or core designs that differ sufficiently to have a significant impact on the WCAP-12472-P-A database. This reevaluation process is applicable for mixed fuel in the reactor from multiple vendors, as well.

PDMS requires information on current plant and core conditions in order to determine the core power distribution using the core peaking factor measurement and measurement uncertainty methodology described in Reference 1. The core and plant condition information is used as input to the continuous core power distribution measurement software that continuously and automatically determines the current peaking factor values. The core power distribution calculation software provides the measured peaking factor values at nominal one minute intervals.

In order for PDMS to accurately determine the core peaking factor value, the continuous core power distribution measurement software requires accurate information measured by the plant instrumentation (e.g. current reactor power level, average reactor vessel inlet temperature, control bank positions, power range detector calibrated voltage values, measured temperatures from a minimum number and distribution of Operable CETCs).

The individual uncertainty components in the BEACON monitored power peaking are discussed in detail in WCAP-12472-P-A, Section 5.7. These components are grouped into three categories, (i.e. a) generic components, b) plant/cycle specific components, and c) input related to the plant operating conditions). The core instrumentation in particular can have different characteristics from plant to plant and cycle to cycle. Therefore, the uncertainties are generated on a plant specific basis with a confirmation performed each cycle. Additionally, PDMS continuously updates the uncertainty depending on the reactor operating conditions and the time since the last calibration constant update. The equations and constants to be used to determine the applicable measurement uncertainties to be applied to the core peaking factors determined by PDMS in the event that PDMS is inoperable are defined in the COLR.

Each plant/cycle specific application of PDMS requires a reevaluation of all information that provides an input to the PDMS software. This reevaluation is a calculational process of adjusting the data to subtle differences in the data storage and data structure between ANC and BEACON. ANC and BEACON use exactly the same solution techniques. Additionally, the reload design and plant/cycle specific information (e. g. COLR information, instrumentation data, RCCA data) has to be updated. The reevaluation process also generates the cycle specific PDMS constants (i.e., reference model), which includes the initial calculated calibration information. Upon initial plant startup following refueling, the PDMS uses the calculated calibration information (i.e., data set). The calculated calibration data set is programmed in the PDMS to conservatively calculate the core peaking factor values.

The proposed TS changes ensure that the BEACON uncertainties are applicable to the instrumentation that BEACON is using In addition, the use of the BEACON methodology to continuously monitor the power distribution and power peaking in the core allows the reactor to be operated without the imposition of the AFD bands and QPTR limit which are traditionally required for Westinghouse reactors. When the PDMS is Operable, precise radial and power distribution measurements are made continuously and therefore the AFD and QPTR limits are not required.

Safety Analysis Methods

The peaking factors are used in both the Large Break LOCA and Small Break LOCA analysis, Non-LOCA analysis, and the Operational Transient Analysis. Since no limits are changed due to implementation of PDMS and RAOC, no transient re-analysis is required.

Each accident analysis addressed in the Byron and Braidwood Stations' UFSAR will be examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC-approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination will be performed in accordance with the requirements of 10 CFR 50.59, "Changes, tests and experiments."

The RAOC and the F_Q SR changes have been provided in WCAP-10216-P-A, Revision 1 (Reference 2) and approved by the NRC on November 26, 1993 (Reference 8) in the conversion to ISTS, the Byron and Braidwood Stations converted from the $F_{XY}(Z)$ methodology to the $F_Q(Z)W(Z)$ methodology. As a result, during the conversion to ISTS, WCAP-10216-P-A, Revision 1, was added to TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," as an analytical method previously reviewed and approved by the NRC. As part of the implementation of BEACON, the Byron and Braidwood Stations are converting from the CAOC methodology to the RAOC methodology (i.e., refer to TS 3.2.3 for AFD). RAOC has been developed for relaxing the current constraints on axial power distribution control. This widens the allowed ΔI versus Power operating space (i.e., AFD band) relative to the CAOC operation particularly at reduced power levels while ensuring that safety considerations are satisfied. This is achieved by examination of a wide range of possible xenon distributions and the possible range of axial power distributions associated with each xenon distribution in both normal operation and accident conditions. Each power shape generated is examined to see if LOCA limits are met or exceeded. The result of this examination is a ΔI band as a function of power which meets the LOCA limits. The power shapes within this range are then examined to ascertain whether they meet the thermal-hydraulic constraints imposed by the Loss of Flow Accident (LOFA), and the limits are revised accordingly.

The effect of the widened ΔI band on the consequences of the anticipated transients discussed in WCAP-10216-P-A, Revision 1, was evaluated. The analysis consists of choosing initial power distributions from the allowed ΔI versus Power domain, including the entire domain, and performing the transient calculation with each distribution. The axial power shapes are preserved from each "snapshot" in the event, and core peaking factors are synthesized by the standard procedure. The results are examined for violations of peak power density and DNB limits. Finally, the Overtemperature ΔT and Overpower ΔT trip setpoints are verified for the investigated ΔI band.

For the Byron and Braidwood Stations, the RAOC analysis for the PDMS inoperable AFD bands will be performed on cycle specific basis and controlled through ComEd's reload design evaluation process (i.e., Safety Parameter Interaction List (SPIL)). There are no changes to the existing limits listed in either the SPIL document or to the Overtemperature ΔT and Overpower ΔT trip setpoint equations.

The implementation of PDMS and RAOC does not involve a reduction in a margin of safety. The reload cores will be designed to operate within established safety analysis acceptance limits and therefore will maintain safety margins. This includes the Byron and Braidwood Stations' UFSAR transients and subsequent reload specific analyses and evaluations performed in accordance with the NRC approved methodologies. Therefore, the margins of safety, as defined in the Bases of the TS and the UFSAR, are not impacted or reduced.

1) LOCA Evaluation

The proposed changes have been evaluated for impact upon the LOCA safety analyses. The LOCA and LOCA-related accident analyses remain valid for the PDMS and RAOC implementation. Neither of these methodologies affects the existing safety analysis limits. Neither of these methodologies affect the normal plant operating parameters, the safeguards systems actuation, the accident mitigation capabilities important to LOCA, and the assumptions used in the LOCA-related accidents; or create conditions more limiting than those assumed in these analyses.

2) Non-LOCA Evaluations

The effect of the non-LOCA events, as a result of the PDMS and RAOC implementation, 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 Overpower ΔT setpoint is designed to ensure plant operation within the fuel temperature design basis and is unaffected by the change to RAOC. There is no change to the Overtemperature ΔT and Overpower ΔT setpoints specified in the TS due to the implementation of PDMS and RAOC.

3) Containment Integrity Evaluations

The implementation of PDMS and RAOC does not adversely affect the short and long term LOCA mass and energy releases and/or the main steamline break mass and energy release containment analyses. Neither PDMS nor RAOC methodology affects the normal plant operating parameters, system actuations, accident mitigating capabilities, or assumptions important to the containment analyses, or create conditions more limiting than those assumed in these analyses. Therefore, the conclusions presented in the UFSAR remain valid with respect to the containment.

4) Radiological Evaluations

The implementation of PDMS and RAOC does not affect the radiological consequences or the post-LOCA hydrogen generation. Since the inputs to the dose analyses do not change, those previously reported in the UFSAR bound the accident doses. Therefore, the consequences to the public resulting from any accident previously evaluated in the UFSAR have not increased.

5) Mechanical Component and Systems Evaluations

The implementation of PDMS and RAOC does not directly or indirectly involve mechanical component hardware considerations. Direct effects as well as indirect effects on safety-related equipment have been considered. Indirect effects include activities that involve non-safety related equipment that may affect safety-related equipment. Component hardware considerations include overall component integrity, sub-component integrity and the adequacy of component supports during all plant conditions. This evaluation has determined that the implementation of PDMS and RAOC does not alter the design, material, construction standards, function or method of performance of any safety-related equipment.

The implementation of PDMS and RAOC does not affect the integrity of any plant auxiliary fluid system or the ability of any system to perform its intended function.

Changes to the Design Methods

There are no changes to the design methods as a result of the implementation of the BEACON methodology or RAOC methodology.

Control Room Alarms

The PDMS receives inputs from process variables via the plant computer. The primary process variables that are monitored are rod position, nuclear power, Reactor Coolant System cold leg temperatures, and core exit temperatures. The PDMS calculates the margin available between the actual plant power distribution parameters and the corresponding parameters that are considered in the safety analysis. These parameters may also be determined manually by obtaining flux maps using the MID System.

The process computer monitors the main PDMS processes and generates an annunciator alarm in the Main Control Room (MCR) when predetermined limits are exceeded.

The following alarms are provided in the MCR based on the power distribution limits calculated by PDMS:

PDMS Warning

Generated when any of the calculated power distribution limit margins are less than or equal to a cycle specific warning setpoint defined in the COLR (i.e., 2% warning setpoint).

PDMS Alarm

Generated when any of the calculated power distribution limit margins are 0% or less (i.e., 0% alarm setpoint).

PDMS Inoperable

Generated when PDMS is inoperable. The PDMS software includes functions, which automatically determine whether the required instrumentation data is available. If this data information is not available or is inadequate to allow the core power distribution to be determined accurately, the reactor operator is automatically alerted that the PDMS is inoperable. Additionally, the reactor operator is also automatically alerted that the PDMS is inoperable if the PDMS software experiences trouble that affects the ability of the PDMS to generate the current core power distribution, or determine the validity of the plant input and core condition information.

G. IMPACT ON THE PREVIOUS SUBMITTALS

We have reviewed the proposed changes regarding impact on any previous submittals. The Expanded COLR license amendment request submitted on December 22, 1999, and the BEACON license amendment request both contain revisions to TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," and TS 5.6.5, "CORE OPERATING LIMITS REPORT

(COLR)." The changes to TS 3.3.1 and TS 5.6.5 associated with the Expanded COLR license amendment request are not incorporated in these proposed changes. However, they do not invalidate any assumptions or conclusions in this submittal.

H. SCHEDULE NEEDS

If found acceptable, we request approval of the proposed TS change by August 31, 2000, to support plant modifications, procedural changes and work planning prior to the Byron Station, Unit 1, and Braidwood Station, Unit 2, Fall 2000 refueling outages.

I. SUMMARY

The BEACON system provides the capability for accurate and continuous core monitoring. It uses current plant instrumentation in conjunction with a fully analytical methodology to generate on-line 3-D core power distributions. The NRC, as documented in WCAP-12472-P-A, has accepted the BEACON methodology and its impact on TS. The TS changes proposed in this submittal will not adversely impact the safe operation of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. No safety-related equipment, safety function, or plant operations will be altered as a result of these proposed changes. The applicable UFSAR limits will be maintained and the TS will continue to require operation within the core operational limits calculated by NRC approved methodologies. These proposed changes will control the cycle-specific parameters within the acceptance criteria and assure conformance to 10 CFR 50.36, "Technical specifications," by using the approved methodology specified in TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)." The COLR will document the specific parameter limits resulting from the unit/cycle specific reload design calculations, including mid-cycle or other revisions to parameter values. Any changes to the COLR will be made in accordance with the provisions of 10 CFR 50.59, "Changes, tests and experiments." From cycle to cycle, the COLR will be revised such that the appropriate core operating limits for the applicable cycle apply.

J. REFERENCES

1. WCAP-12472-P-A, "BEACON Core Monitoring and Support System," August 1994.

2. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," Revision 1, February 1994.

3. WCAP-10965-P-A, "ANC-A Westinghouse Advanced Nodal Computer Code," December 1985.

4. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Process Improvement to the Westinghouse Neutronics Code System," NSD-NRC-96-4679, March 29, 1996.

5. WCAP-12394-P-A, "SPNOVA – A Multidimensional Static and Transient Computer Program for PWR Analyses," June 1991.

6. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

7. Letter from A. C. Thadani, (NRC) to N. J. Liparulo (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-12472-P, 'BEACON: Core Monitoring and Operations Support System,'" February 16, 1994.

8. Letter from A. C. Thadani (NRC) to N. J. Liparulo (Westinghouse), "Acceptance for Referencing of Revised Version of Licensing Topic Report WCAP-10216-P, Rev. 1, 'Relaxation of Constant Axial Offset Control – F_Q Surveillance Technical Specification,'" November 26, 1993.

Att B-1 / B-2
m/u pages

ATTACHMENT B-1

MARKED-UP TS PAGES
FOR BYRON STATION, UNITS 1 AND 2

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

<u>AND</u>

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	A.2	Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One rod not within alignment ~~limits~~ limit.	B.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours from discovery of Condition B concurrent with inoperability of Power Distribution Monitoring System (PDMS)
	AND		
	B.3	Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	AND		
	B.4	~~Perform SR 3.2.1.1.~~ Determine Heat Flux Hot Channel Factor ($F_Q(Z)$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$).	72 hours
	~~AND~~		
	~~B.5~~	~~Perform SR 3.2.2.1.~~	~~72 hours~~
	AND		
	B.~~6~~ 5	Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D ~~C~~. Required Action and associated Completion Time of Condition B or Required Action C.3 not met.	D ~~C~~.1 Be in MODE 3.	6 hours
C ~~B~~. More than one rod not within alignment limit.	C ~~B~~.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	OR	
	C ~~B~~.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
Insert 3.1.4-3A	~~B.2~~ ~~Be in MODE 3~~.	~~6 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days

(continued)

3.1.4 Rod Group Alignment Limits

Insert 3.1.4-3A:

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
	C.2	Be in MODE 3.	6 hours from discovery of Condition C concurrent with inoperability of PDMS
	AND		
	C.3	---------NOTE--------- Only required to be performed when PDMS is OPERABLE. ---------------------	
		Restore rod(s) to within alignment limit.	72 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 550°F; and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each inoperable DRPI and each demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable DRPIs ~~by using movable incore detectors~~. __OR__ A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1 Initiate action to verify the position of the rods with inoperable DRPIs ~~by using movable incore detectors~~. __OR__ B.2 Reduce THERMAL POWER to ≤ 50% RTP.	Immediately 8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One demand position indicator per bank inoperable for one or more banks.	C.1.1 Verify by administrative means all DRPIs for the affected bank(s) are OPERABLE.	Once per 8 hours
	AND	
	C.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected bank(s) are ≤ 12 steps apart.	Once per 8 hours
	OR	
	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Prior to criticality after each removal of the reactor head.

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 limit~~s~~ specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
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
	<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
	<u>AND</u>		
	A.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
	~~AND~~		
	~~A.4~~	~~Perform SR 3.2.1.1 and SR 3.2.1.2~~.	~~Prior to exceeding the THERMAL POWER limit of Required Action A.1~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. $F_Q^W(Z)$ not within limit~~s~~.	B.1	Reduce THERMAL POWER ≥ 1% RTP 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% $F_Q^W(Z)$ exceeds limit.	72 hours
	AND		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	~~AND~~		
	~~B.4~~	~~Perform SR 3.2.1.1 and SR 3.2.1.2~~.	~~Prior to increasing THERMAL POWER above the limit of Required Action B.1~~
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SR 3.2.1.1 Note 1 and SR 3.2.1.2 Note 1

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 Insert 3.2.1-3A	Verify $F_Q^C(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling 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 Effective Full Power Days (EFPD) thereafter

(continued)

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

Insert 3.2.1-3A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	-------------------NOTES------------------- 1. 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. 2. Not required to be performed until 12 hours after declaring Power Distribution Monitoring System (PDMS) inoperable. Performance of SR 3.2.1.3 satisfies the initial performance of this SR after declaring PDMS inoperable. ---	

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
Insert 3.2.1-4A SR 3.2.1.2	------------------NOTE------------------ 5 2. If $F_Q^W(Z)$ measurements indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ 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 specified in the COLR; 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 $\left[\frac{F_C^C(Z)}{K(Z)}\right]$ has not increased. --- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND (continued)

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

Insert 3.2.1-4A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-------------------NOTES------------------- 1. 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. 2. ... 3. Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.1.4 satisfies the initial performance of this SR after declaring PDMS inoperable. ---	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^M(Z)$ was last verified <u>AND</u> 31 EFPD thereafter

Insert 3.2.1-5A → SR 3.2.1.3 and SR 3.2.1.4

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

Insert 3.2.1-5A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. -- Verify $F_Q^C(Z)$ is within limit specified in the COLR.	7 days
SR 3.2.1.4	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. -- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the ~~limits~~ limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Actions A.2 and A.4 must be completed whenever Condition A is entered. ---------------------- $F^N_{\Delta H}$ not within limit.	A.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND		
	A.2	~~Perform SR 3.2.2.1~~. Determine $F^N_{\Delta H}$.	24 hours
	AND		
	A.3	Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.	72 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	--------NOTE-------- THERMAL POWER does not have to be reduced to comply with this Required Action. -------------------- ~~Perform SR 3.2.2.1~~. Determine $F^N_{\Delta H}$.	Prior to exceeding 50% RTP AND Prior to exceeding 75% RTP AND 24 hours after reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
Insert 3.2.2-3A		
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Prior to exceeding 75% RTP after each refueling <u>AND</u> 31 Effective Full Power Days thereafter

Insert 3.2.2-3B → SR 3.2.2.2

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

Insert 3.2.2-3A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	----------NOTE---------- Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.2.2 satisfies the initial performance of this SR after declaring PDMS inoperable. ----------------------------	

Insert 3.2.2-3B:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	----------NOTE---------- Only required to be performed when PDMS is OPERABLE. ---------------------------- Verify $F^N_{\Delta H}$ is within limit specified in the COLR.	7 days

Replaced by NUREG-1431, Revision 1, RAOC Technical Specification with exceptions noted in markup.

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. 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.

-----NOTES-----

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

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, provided AFD is maintained within acceptable operation limits.

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.	---------NOTE--------- Required Action C.1 must be completed whenever Condition C is entered. ---------------------- 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 acceptable operation limits.	C.1	Reduce THERMAL POWER to < 50% RTP.	30 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ---------NOTE--------- Required Action D.1 must be completed whenever Condition D is entered. ---------------------- Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to < 15% RTP.	9 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel	7 days
SR 3.2.3.2	Update target flux difference.	Once within 31 Effective Full Power Days (EFPD) after each refueling AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.3.3	-----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 <u>AND</u> 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~~.~~ when Power Distribution Monitoring System (PDMS) is inoperable

ACTIONS

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

SURVEILLANCE REQUIREMENTS

Insert 3.2.3-1A

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days ~~AND~~ ~~Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable~~

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

Insert 3.2.3-1A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	-------------------NOTE------------------- Not required to be performed until 12 hours after declaring PDMS inoperable. --	

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.~~ RTP when Power Distribution Monitoring System (PDMS) is inoperable

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
	<u>AND</u>		
	A.2	Determine QPTR and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>		
	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 <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to exceeding the THERMAL POWER limit of Required Action A.1
	AND		
	A.5	---------NOTES--------- 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 OPTR to within limits.	Prior to exceeding the THERMAL POWER limits of Required Action A.1
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.6	---------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. --------------------------- , SR 3.2.1.2, Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after exceeding the THERMAL POWER limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	Insert 3.2.4-4A -----NOTES----- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channel inputs can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --------------- Verify QPTR is ≤ 1.02 by calculation.	7 days
SR 3.2.4.2	Insert 3.2.4-4B -----NOTES----- 1. Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. --------------- Verify QPTR is ≤ 1.02 using the movable incore detectors.	12 hours

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

Insert 3.2.4-4A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	-----------------NOTES----------------- 1. ... 2. ... 3. Not required to be performed until 12 hours after declaring PDMS inoperable. ---	

Insert 3.2.4-4B:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.2	-----------------NOTES----------------- 1. ... 2. Not required to be performed until 12 hours after declaring PDMS inoperable. ---	

Insert ITS 3.2.5

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Departure from Nucleate Boiling Ratio (DNBR)

LCO 3.2.5 DNBR shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP when Power Distribution Monitoring System (PDMS) is OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DNBR not within limit.	A.1 Restore DNBR to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify DNBR is within limit specified in the COLR.	7 days

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 or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or train(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2 Be in MODE 3.	54 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	------------NOTE------------ While this LCO is not met for Function 18, 19, or 20 in MODE 5, making the Rod Control System capable of rod withdrawal is not permitted. ----------------------------	
	C.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	C.2.1 Initiate action to fully insert all rods.	48 hours
	<u>AND</u>	
	C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One Power Range Neutron Flux–High channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. ---------------------------------	
	D.1~~.1~~ Place channel in trip.	6 hours
	~~AND~~	
	~~D.1.2 Reduce THERMAL POWER to ≤ 75% RTP.~~	~~12 hours~~
	~~OR~~	
	~~D.2.1 Place channel in trip.~~	~~6 hours~~
	~~AND~~	
	~~------------NOTE------------ Only required to be performed when the Power Range Neutron Flux input to QPTR is inoperable. ---------------------------------~~	
	~~D.2.2 Perform SR 3.2.4.2.~~	~~Once per 12 hours~~
	OR	
	D.~~3~~2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------	
	E.1 Place channel in trip.	6 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to $<$ P-6.	2 hours
	<u>OR</u>	
	F.2 Increase THERMAL POWER to $>$ P-10.	2 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	G.2 Reduce THERMAL POWER to $<$ P-6.	2 hours
H. One Source Range Neutron Flux channel inoperable.	H.1 Suspend operations involving positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. Two Source Range Neutron Flux channels inoperable.	I.1 Open Reactor Trip Breakers (RTBs).	Immediately
J. One Source Range Neutron Flux channel inoperable.	J.1 Restore channel to OPERABLE status.	48 hours
	OR	
	J.2.1 Initiate action to fully insert all rods.	48 hours
	AND	
	J.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
K. One channel inoperable.	-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.	
	K.1 Place channel in trip.	6 hours
	OR	
	K.2 Reduce THERMAL POWER to < P-7.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One Turbine Trip channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---------------------------------	
	L.1 Place channel in trip.	6 hours
	<u>OR</u>	
	L.2 Reduce THERMAL POWER to < P-8.	12 hours
M. One train inoperable.	------------NOTE------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------------------------------	
	M.1 Restore train to OPERABLE status.	6 hours
	<u>OR</u>	
	M.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One RTB train inoperable.	-----NOTES----- 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 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. -----	
	N.1 Restore train to OPERABLE status.	1 hour
	<u>OR</u>	
	N.2 Be in MODE 3.	7 hours
O. One or more channels inoperable.	O.1 Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>	
	O.2 Be in MODE 3.	7 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
P. One or more channels inoperable.	P.1	Verify interlock is in required state for existing unit conditions.	1 hour
	OR		
	P.2	Be in MODE 2.	7 hours
Q. One trip mechanism inoperable for one RTB.	Q.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	OR		
	Q.2	Be in MODE 3.	54 hours

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.2	-------------------NOTES------------------- 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. --- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	 24 hours
SR 3.3.1.3	-------------------NOTES------------------- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Only required to be performed with THERMAL POWER > 15% RTP. --- Compare results of the incore ~~detector~~ measurements to NIS AFD.	 Prior to exceeding 75% RTP after each refueling AND 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.4	-------------------NOTE------------------- This Surveillance must be performed on the RTBB prior to placing the bypass breaker in service. -- 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 24 hours after THERMAL POWER is ≥ 75% RTP. -- 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 entry into MODE 3 -- Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	---NOTE--- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. Perform COT.	---NOTE--- Only required when not performed within previous 92 days Prior to reactor startup <u>AND</u> Four hours after reducing power below P-10 for power and intermediate instrumentation <u>AND</u> Four hours after reducing power below P-6 for source range instrumentation <u>AND</u> Every 92 days thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	-----NOTE----- Verification of setpoint is not required. Perform TADOT.	92 days
SR 3.3.1.10	-----NOTE----- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months
SR 3.3.1.13	-----NOTE----- Verification of setpoint is not required. Perform TADOT	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	-----NOTE------ Only required when not performed within previous 31 days -------------- Prior to reactor startup
SR 3.3.1.15	-------------------NOTE------------------- Neutron detectors are excluded from response time testing. -- Verify RTS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	1(b), 2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 6.2% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Source Range Block Permissive) interlock

(d) Below the P-6 (Source Range Block Permissive) interlock

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Overtemperature ΔT	1,2	4	E	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 SR 3.3.1.15	Refer to Note 1 (Page 3.3.1-18)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3.1-19)
8. Pressurizer Pressure					
a. Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1875 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2393 psig
9. Pressurizer Water Level – High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.5% of instrument span
10. Reactor Coolant Flow – Low (per loop)	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.3% of loop minimum measured flow
11. Reactor Coolant Pump (RCP) Breaker Position (per train)	1(e)	1	K	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) [illegible]

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Undervoltage RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4920 V
13. Underfrequency RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.05 Hz
14. Steam Generator (SG) Water Level - Low Low (per SG)					
a. Unit 1	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.1% of narrow range instrument span
b. Unit 2	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 34.8% of narrow range instrument span
15. Turbine Trip					
a. Emergency Trip Header Pressure (per train)	1(f)	3	L	SR 3.3.1.10 SR 3.3.1.14	≥ 910 psig
b. Turbine Throttle Valve Closure (per train)	1(f)	4	L	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	M	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Trip System Interlocks					
a. Source Range Block Permissive, P-6	2(d)	2	O	SR 3.3.1.11 SR 3.3.1.12	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7					
(1) P-10 Input	1	3	P	SR 3.3.1.11 SR 3.3.1.12	NA
(2) P-13 Input	1	2	P	SR 3.3.1.10 SR 3.3.1.12	NA
c. Power Range Neutron Flux, P-8	1	3	P	SR 3.3.1.11 SR 3.3.1.12	≤ 32.1% RTP
d. Power Range Neutron Flux, P-10	1,2	3	O	SR 3.3.1.11 SR 3.3.1.12	≥ 7.9% RTP and ≤ 12.1% RTP
e. Turbine Impulse Pressure, P-13	1	2	P	SR 3.3.1.10 SR 3.3.1.12	≤ 12.1% turbine power
18. Reactor Trip Breakers (RTBs)(g)	1,2	2 trains	N	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	Q	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	1 each per RTB	C	SR 3.3.1.4	NA
20. Automatic Trip Logic	1,2	2 trains	M	SR 3.3.1.5	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(d) Below the P-6 (Source Range Block Permissive) interlocks.

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

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)}\left[T \frac{1}{(1+\tau_6 s)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured Reactor Coolant System (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, ≤ 588.4°F.

P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, = 2235 psig.

K_1 = 1.325	K_2 = 0.0297/°F	K_3 = 0.00181/psig
τ_1 = 8 sec	τ_2 = 3 sec	τ_3 ≤ 2 sec
τ_4 = 33 sec	τ_5 = 4 sec	τ_6 ≤ 2 sec

$f_1(\Delta I)$ = $-3.35\{24 + (q_t - q_b)\}$ when $q_t - q_b < -24\%$ RTP
0% of RTP when -24% RTP $\le q_t - q_b \le 10\%$ RTP
$4.11\{(q_t - q_b) - 10\}$ when $q_t - q_b > 10\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \leq \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left[\frac{1}{1+\tau_6 s}\right] T - K_6 \left[T \frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

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, ≤ 588.4°F.

K_4 = 1.072 K_5 = 0.02/°F for increasing T_{avg} K_6 = 0.00245/°F when T > T"
0/°F for decreasing T_{avg} 0/°F when T ≤ T"

τ_1 = 8 sec τ_2 = 3 sec τ_3 ≤ 2 sec
τ_6 ≤ 2 sec τ_7 = 10 sec

$f_2(\Delta I)$ = 0 for all ΔI.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

LCO 3.1.1. "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3. "Moderator Temperature Coefficient";
LCO 3.1.5. "Shutdown Bank Insertion Limits";
LCO 3.1.6. "Control Bank Insertion Limits";
LCO 3.1.8. "PHYSICS TESTS Exceptions - MODE 2";
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)"; ~~and~~
LCO 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)";
LCO 3.9.1. "Boron Concentration"; and

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 Evaluations Methodology," July 1985.

2. ~~WCAP-8385, "Power Distribution Control and Load Following Procedures Topical Report," September 1974.~~ WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods," July 1983.

4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," July 1990.

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. WCAP-9220-P-A. "Westinghouse ECCS Evaluation Model-1981 Version." February 1982.

7. WCAP-9561-P-A. Add. 3. "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model." July 1986.

8. WCAP-10266-P-A. "The 1981 Version of Westinghouse Evaluation Model using BASH Code." March 1987. including Addendum 1 "Power Shape Sensitivity Studies." Revision 2-P-A. dated December 15. 1987. and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements." Revision 2. Dated May 1988.

9. WCAP-10079-P-A. "NOTRUMP. A Nodal Transient Small Break and General Network Code." August 1985.

10. WCAP-10054-P-A. "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code." August 1985.

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

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: and

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

ATTACHMENT B-2

MARKED-UP TS PAGES
FOR BRAIDWOOD STATION, UNITS 1 AND 2

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One rod not within alignment limit~~s~~.	B.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>		
	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours from discovery of Condition B concurrent with inoperability of Power Distribution Monitoring System (PDMS)
	<u>AND</u>		
	B.3	Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	<u>AND</u>		
	B.4	~~Perform SR 3.2.1.1.~~ Determine Heat Flux Hot Channel Factor ($F_Q(Z)$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$).	72 hours
	~~AND~~		
	~~B.5~~	~~Perform SR 3.2.2.1.~~	~~72 hours~~
	<u>AND</u>		
	B.~~6~~ 5	Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D ~~C~~. Required Action and associated Completion Time of Condition B or Required Action C.3 not met.	D ~~C~~.1 Be in MODE 3.	6 hours
C ~~D~~. More than one rod not within alignment limit.	C ~~D~~.1.1 Verify SDM is within the limits specified in the COLR. OR C ~~D~~.1.2 Initiate boration to restore required SDM to within limit. AND Insert 3.1.4-3A ~~D.2 Be in MODE 3.~~	1 hour 1 hour ~~6 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days

(continued)

3.1.4 Rod Group Alignment Limits

Insert 3.1.4-3A:

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
	C.2	Be in MODE 3.	6 hours from discovery of Condition C concurrent with inoperability of PDMS
	AND		
	C.3	--------NOTE-------- Only required to be performed when PDMS is OPERABLE. -------------------- Restore rod(s) to within alignment limit.	72 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 550°F$; and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each inoperable DRPI and each demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable DRPIs ~~by using movable incore detectors~~.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1 Initiate action to verify the position of the rods with inoperable DRPIs ~~by using movable incore detectors~~.	Immediately
	OR	
	B.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

FOR INFORMATION ONLY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One demand position indicator per bank inoperable for one or more banks.	C.1.1 Verify by administrative means all DRPIs for the affected bank(s) are OPERABLE.	Once per 8 hours
	AND	
	C.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected bank(s) are ≤ 12 steps apart.	Once per 8 hours
	OR	
	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Prior to criticality after each removal of the reactor head.

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 limit~~s~~ specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
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
	AND		
	A.2	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
	AND		
	A.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
	~~AND~~		
	~~A.4~~	~~Perform SR 3.2.1.1 and SR 3.2.1.2.~~	~~Prior to exceeding the THERMAL POWER limit of Required Action A.1~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. $F_Q^W(Z)$ not within limit~~s~~.	B.1	Reduce THERMAL POWER ≥ 1% RTP 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% $F_Q^W(Z)$ exceeds limit.	72 hours
	AND		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	~~AND~~		
	~~B.4~~	~~Perform SR 3.2.1.1 and SR 3.2.1.2~~.	~~Prior to increasing THERMAL POWER above the limit of Required Action B.1~~
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SR 3.2.1.1 Note 1 and SR 3.2.1.2 Note 1

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 specified in the COLR. Insert 3.2.1-3A	Prior to exceeding 75% RTP after each refueling 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 Effective Full Power Days (EFPD) thereafter

(continued)

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

Insert 3.2.1-3A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	-----------------NOTES----------------- 1. 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. 2. Not required to be performed until 12 hours after declaring Power Distribution Monitoring System (PDMS) inoperable. Performance of SR 3.2.1.3 satisfies the initial performance of this SR after declaring PDMS inoperable. ---	

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
Insert 3.2.1-4A → SR 3.2.1.2	-------------------NOTES------------------- 2. If $F_Q^W(Z)$ measurements indicate that the $$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right],$$ 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 specified in the COLR; 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 $$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ has not increased. --- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND (continued)

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

Insert 3.2.1-4A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-------------------NOTES------------------- 1. 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. 2. ... 3. Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.1.4 satisfies the initial performance of this SR after declaring PDMS inoperable. --	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

Insert 3.2.1-5A → SR 3.2.1.3 and SR 3.2.1.4

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

Insert 3.2.1-5A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. -- Verify $F_Q^C(Z)$ is within limit specified in the COLR.	 7 days
SR 3.2.1.4	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. -- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	 7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limit~~s~~ specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Actions A.2 and A.4 must be completed whenever Condition A is entered. ------------------------ $F^N_{\Delta H}$ not within limit.	A.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND		
	A.2	~~Perform SR 3.2.2.1.~~ Determine $F^N_{\Delta H}$.	24 hours
	AND		
	A.3	Reduce Power Range Neutron Flux—High trip setpoints to ≤ 55% RTP.	72 hours
	AND		

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
Insert 3.2.2-3A →		
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limit~~s~~ specified in the COLR.	Prior to exceeding 75% RTP after each refueling <u>AND</u> 31 Effective Full Power Days thereafter

Insert 3.2.2-3B → SR 3.2.2.2

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

Insert 3.2.2-3A:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	------------NOTE------------ Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.2.2 satisfies the initial performance of this SR after declaring PDMS inoperable. ----------------------------	

Insert 3.2.2-3B:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	------------NOTE------------ Only required to be performed when PDMS is OPERABLE. ---------------------------- Verify $F^N_{\Delta H}$ is within limit specified in the COLR.	7 days

Replaced by NUREG-1431, Revision 1, RAOC Technical Specification with exceptions noted in markup

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. 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.

---NOTES---

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.
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, provided AFD is maintained within acceptable operation limits.

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. ----------NOTE---------- Required Action C.1 must be completed whenever Condition C is entered. -------------------------- 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 acceptable operation limits.	C.1	Reduce THERMAL POWER to < 50% RTP.	30 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ----------NOTE---------- Required Action D.1 must be completed whenever Condition D is entered. ---------------------------- Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to < 15% RTP.	9 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days
SR 3.2.3.2	Update target flux difference.	Once within 31 Effective Full Power Days (EFPD) after each refueling <u>AND</u> 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.3.3	-------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 <u>AND</u> 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~~.~~ when Power Distribution Monitoring System (PDMS) is inoperable

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
Insert 3.2.3-1A		
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days ~~AND~~ ~~Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable~~

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

Insert 3.2.3-1A:

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.3.1	------------------NOTE------------------ Not required to be performed until 12 hours after declaring PDMS inoperable. --	

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 when Power Distribution Monitoring System (PDMS) is inoperable.

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
	<u>AND</u>		
	A.2	Determine QPTR and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>		
	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 <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to exceeding the THERMAL POWER limit of Required Action A.1
	AND		
	A.5	---NOTES--- 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 limits.	Prior to exceeding the THERMAL POWER limits of Required Action A.1
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.6 -----NOTE----- Perform Required Action A.6 only after Required Action A.5 is completed. ---------- , SR 3.2.1.2, Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after exceeding the THERMAL POWER limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channel inputs can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. Insert 3.2.4-4A ------------------------------- Verify QPTR is ≤ 1.02 by calculation.	7 days
SR 3.2.4.2	----------NOTE(S)---------- 1. Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. Insert 3.2.4-4B ------------------------------- Verify QPTR is ≤ 1.02 using the movable incore detectors.	12 hours

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

Insert 3.2.4-4A:

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	------------------NOTES------------------ 1. ... 2. ... 3. Not required to be performed until 12 hours after declaring PDMS inoperable. --	

Insert 3.2.4-4B:

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.2	------------------NOTES------------------ 1. .... 2. Not required to be performed until 12 hours after declaring PDMS inoperable. --	

Insert ITS 3.2.5

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Departure from Nucleate Boiling Ratio (DNBR)

LCO 3.2.5 DNBR shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP when Power Distribution Monitoring System (PDMS) is OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DNBR not within limit.	A.1 Restore DNBR to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify DNBR is within limit specified in the COLR.	7 days

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 or more required channels or trains inoperable.	A.1	Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or train(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1	Restore channel to OPERABLE status.	48 hours
	<u>OR</u>		
	B.2	Be in MODE 3.	54 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	------------NOTE------------ While this LCO is not met for Function 18, 19, or 20 in MODE 5, making the Rod Control System capable of rod withdrawal is not permitted. ------------------------------	
	C.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	C.2.1 Initiate action to fully insert all rods.	48 hours
	<u>AND</u>	
	C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One Power Range Neutron Flux – High channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. ----------------------------	
	D.1~~.1~~ Place channel in trip.	6 hours
	AND	
	D.~~1~~.2 Reduce THERMAL POWER to ≤ 75% RTP.	12 hours
	OR	
	D.2.1 Place channel in trip.	6 hours
	AND	
	------------NOTE------------ Only required to be performed when the Power Range Neutron Flux input to QPTR is inoperable. ----------------------------	
	D.2.2 Perform SR 3.2.4.2.	Once per 12 hours
	OR	
	D.~~3~~ 2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. --------------------------------	
	E.1 Place channel in trip.	6 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours
	<u>OR</u>	
	F.2 Increase THERMAL POWER to > P-10.	2 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. One Source Range Neutron Flux channel inoperable.	H.1 Suspend operations involving positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. Two Source Range Neutron Flux channels inoperable.	I.1 Open Reactor Trip Breakers (RTBs).	Immediately
J. One Source Range Neutron Flux channel inoperable.	J.1 Restore channel to OPERABLE status.	48 hours
	OR	
	J.2.1 Initiate action to fully insert all rods.	48 hours
	AND	
	J.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
K. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------	
	K.1 Place channel in trip.	6 hours
	OR	
	K.2 Reduce THERMAL POWER to < P-7.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One Turbine Trip channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------------	
	L.1 Place channel in trip.	6 hours
	<u>OR</u>	
	L.2 Reduce THERMAL POWER to < P-8.	12 hours
M. One train inoperable.	------------NOTE------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------------------------------	
	M.1 Restore train to OPERABLE status.	6 hours
	<u>OR</u>	
	M.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One RTB train inoperable.	------------NOTES------------ 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 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. -----------------------------	
	N.1 Restore train to OPERABLE status.	1 hour
	<u>OR</u>	
	N.2 Be in MODE 3.	7 hours
O. One or more channels inoperable.	O.1 Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>	
	O.2 Be in MODE 3.	7 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
P. One or more channels inoperable.	P.1	Verify interlock is in required state for existing unit conditions.	1 hour
	OR		
	P.2	Be in MODE 2.	7 hours
Q. One trip mechanism inoperable for one RTB.	Q.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	OR		
	Q.2	Be in MODE 3.	54 hours

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-------------------NOTES------------------- 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. --- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours
SR 3.3.1.3	-------------------NOTES------------------- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Only required to be performed with THERMAL POWER > 15% RTP. --- Compare results of the incore ~~detector~~ measurements to NIS AFD.	Prior to exceeding 75% RTP after each refueling AND 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	------------------NOTE------------------ This Surveillance must be performed on the RTBB prior to placing the bypass breaker in service. -- 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 24 hours after THERMAL POWER is ≥ 75% RTP. -- 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 entry into MODE 3 -- Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.1.8 ------------------NOTE------------------ This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -- Perform COT.	-----NOTE----- Only required when not performed within previous 92 days -------------- Prior to reactor startup <u>AND</u> Four hours after reducing power below P-10 for power and intermediate instrumentation <u>AND</u> Four hours after reducing power below P-6 for source range instrumentation <u>AND</u> Every 92 days thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	-------------------NOTE------------------- Verification of setpoint is not required. --- Perform TADOT.	92 days
SR 3.3.1.10	-------------------NOTE------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. --- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months
SR 3.3.1.13	-------------------NOTE------------------- Verification of setpoint is not required. --- Perform TADOT.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	------NOTE------ Only required when not performed within previous 31 days ---------------- Prior to reactor startup
SR 3.3.1.15	-------------------NOTE------------------- Neutron detectors are excluded from response time testing. -- Verify RTS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.6% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 6.2% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Overtemperature ΔT	1,2	4	E	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 SR 3.3.1.15	Refer to Note 1 (Page 3.3.1-18)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3.1-19)
8. Pressurizer Pressure					
a. Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1875 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2393 psig
9. Pressurizer Water Level – High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.5% of instrument span
10. Reactor Coolant Flow – Low (per loop)	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.3% of loop minimum measured flow
11. Reactor Coolant Pump (RCP) Breaker Position (per train)	[illegible]	[illegible]	K	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Undervoltage RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4920 V
13. Underfrequency RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.06 Hz
14. Steam Generator (SG) Water Level - Low Low (per SG)					
a. Unit 1	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.1% of narrow range instrument span
b. Unit 2	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 34.8% of narrow range instrument span
15. Turbine Trip					
a. Emergency Trip Header Pressure (per train)	1(f)	3	L	SR 3.3.1.10 SR 3.3.1.14	≥ 910 psig
b. Turbine Throttle Valve Closure (per train)	1(f)	4	L	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	M	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor [illegible]

(f) Above the P-8 (Power Range Neut[illegible]

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Trip System Interlocks					
a. Source Range Block Permissive, P-6	2(d)	2	O	SR 3.3.1.11 SR 3.3.1.12	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7					
(1) P-10 Input	1	3	P	SR 3.3.1.11 SR 3.3.1.12	NA
(2) P-13 Input	1	2	P	SR 3.3.1.10 SR 3.3.1.12	NA
c. Power Range Neutron Flux, P-8	1	3	P	SR 3.3.1.11 SR 3.3.1.12	≤ 32.1% RTP
d. Power Range Neutron Flux, P-10	1,2	3	O	SR 3.3.1.11 SR 3.3.1.12	≥ 7.9% RTP and ≤ 12.1% RTP
e. Turbine Impulse Pressure, P-13	1	2	P	SR 3.3.1.10 SR 3.3.1.12	≤ 12.1% turbine power
18. Reactor Trip Breakers (RTBs)(g)	1,2	2 trains	N	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	Q	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	1 each per RTB	C	SR 3.3.1.4	NA
20. Automatic Trip Logic	1,2	2 trains	M	SR 3.3.1.5	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable [illegible] more rods not fully inserted.

(d) Below the P-6 (Source Range Block [illegible].

(g) Including any reactor trip bypass [illegible] and closed for bypassing an RTB.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[T \frac{1}{(1+\tau_6 s)} - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured Reactor Coolant System (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, ≤ 588.4°F.

P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, = 2235 psig.

K_1 = 1.325	K_2 = 0.0297/°F	K_3 = 0.00181/psig
τ_1 = 8 sec	τ_2 = 3 sec	τ_3 ≤ 2 sec
τ_4 = 33 sec	τ_5 = 4 sec	τ_6 ≤ 2 sec

$f_1(\Delta I)$ = $-3.35\{24 + (q_t - q_b)\}$ when $q_t - q_b < -24\%$ RTP
0% of RTP when -24% RTP $\le q_t - q_b \le 10\%$ RTP
$4.11\{(q_t - q_b) - 10\}$ when $q_t - q_b > 10\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left[\frac{1}{1+\tau_6 s}\right] T - K_6 \left[T \frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

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, ≤ 588.4°F.

$K_4 = 1.072$

$K_5 = 0.02/°F$ for increasing T_{avg}
$0/°F$ for decreasing T_{avg}

$K_6 = 0.00245/°F$ when $T > T''$
$0/°F$ when $T \le T''$

$\tau_1 = 8$ sec
$\tau_2 = 3$ sec
$\tau_3 \le 2$ sec
$\tau_6 \le 2$ sec
$\tau_7 = 10$ sec

$f_2(\Delta I) = 0$ for all ΔI.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

LCO 3.1.1. "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3. "Moderator Temperature Coefficient";
LCO 3.1.5. "Shutdown Bank Insertion Limits";
LCO 3.1.6. "Control Bank Insertion Limits";
LCO 3.1.8. "PHYSICS TESTS Exceptions - MODE 2";
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)"; ~~and~~
LCO 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)";
LCO 3.9.1. "Boron Concentration"; and

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 Evaluations Methodology," July 1985.

2. ~~WCAP-8385, "Power Distribution Control and Load Following Procedures-Topical Report," September 1974.~~ WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods," July 1983.

4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," July 1990.

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

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

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; and

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

ATTACHMENT B-3

CLEAN COPY TS PAGES
FOR BYRON STATION, UNITS 1 AND 2

~~3.1 REACTIVITY CONTROL SYSTEMS~~

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	A.2	Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One rod not within alignment limit.	B.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	OR	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours from discovery of Condition B concurrent with inoperability of Power Distribution Monitoring System (PDMS)
	AND	
	B.3 Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	AND	
	B.4 Determine Heat Flux Hot Channel Factor ($F_Q(Z)$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$).	72 hours
	AND	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.5	Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. More than one rod not within alignment limit.	C.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>		
	C.1.2	Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>		
	C.2	Be in MODE 3.	6 hours from discovery of Condition C concurrent with inoperability of PDMS
	<u>AND</u>		
	C.3	--------NOTE-------- Only required to be performed when PDMS is OPERABLE. -------------------- Restore rod(s) to within alignment limit.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B or Required Action C.3 not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 550°F, and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---------------------------------------NOTE---------------------------------------
Separate Condition entry is allowed for each inoperable DRPI and each demand position indicator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable DRPIs.	Once per 8 hours
	<u>OR</u>		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1	Initiate action to verify the position of the rods with inoperable DRPIs.	Immediately
	<u>OR</u>		
	B.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One demand position indicator per bank inoperable for one or more banks.	C.1.1 Verify by administrative means all DRPIs for the affected bank(s) are OPERABLE.	Once per 8 hours
	<u>AND</u>	
	C.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected bank(s) are ≤ 12 steps apart.	Once per 8 hours
	<u>OR</u>	
	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Prior to criticality after each removal of the reactor head.

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 limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
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
	<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
	<u>AND</u>	
	A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
B. $F_Q^W(Z)$ not within limit.	B.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	<u>AND</u>		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	------------------------------NOTES------------------------------ 1. 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. 2. Not required to be performed until 12 hours after declaring Power Distribution Monitoring System (PDMS) inoperable. Performance of SR 3.2.1.3 satisfies the initial performance of this SR after declaring PDMS inoperable. -- Verify $F^C_Q(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^C_Q(Z)$ was last verified AND 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	------------------NOTES------------------ 1. 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. 2. If $F_Q^W(Z)$ measurements indicate that the $$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ 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 specified in the COLR; 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 $$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ has not increased. --	
		(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued) -------------------NOTES------------------- 3. Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.1.4 satisfies the initial performance of this SR after declaring PDMS inoperable. --- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND Once within 12 hours after achieving equilibrium conditions after exceeding by ≥ 10% RTP the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	-----NOTE----- Only required to be performed when PDMS is OPERABLE. --------------- Verify $F_Q^C(Z)$ is within limit specified in the COLR.	7 days
SR 3.2.1.4	-----NOTE----- Only required to be performed when PDMS is OPERABLE. --------------- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Actions A.2 and A.4 must be completed whenever Condition A is entered. ------------------------ $F^N_{\Delta H}$ not within limit.	A.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND		
	A.2	Determine $F^N_{\Delta H}$.	24 hours
	AND		
	A.3	Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.	72 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 ----------NOTE---------- THERMAL POWER does not have to be reduced to comply with this Required Action. ------------------------------ Determine $F_{\Delta H}^{N}$.	Prior to exceeding 50% RTP AND Prior to exceeding 75% RTP AND 24 hours after reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	-------------------NOTE------------------- Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.2.2 satisfies the initial performance of this SR after declaring PDMS inoperable. -- Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Prior to exceeding 75% RTP after each refueling <u>AND</u> 31 Effective Full Power Days thereafter
SR 3.2.2.2	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. -- Verify $F^N_{\Delta H}$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD 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 when Power Distribution Monitoring System (PDMS) is inoperable.

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	-----NOTE----- Not required to be performed until 12 hours after declaring PDMS inoperable. --- Verify AFD is 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 when Power Distribution Monitoring System (PDMS) is inoperable.

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
	<u>AND</u>		
	A.2	Determine QPTR and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>		
	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 <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to exceeding the THERMAL POWER limit of Required Action A.1
	<u>AND</u>		
	A.5	---------NOTES--------- 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 limits.	Prior to exceeding the THERMAL POWER limits of Required Action A.1
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	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.	24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after exceeding the THERMAL POWER limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------------NOTES------------ 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channel inputs can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. 3. Not required to be performed until 12 hours after declaring PDMS inoperable. ----------------------------------- Verify QPTR is ≤ 1.02 by calculation.	7 days
SR 3.2.4.2	------------NOTES------------ 1. Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. 2. Not required to be performed until 12 hours after declaring PDMS inoperable. ----------------------------------- Verify QPTR is ≤ 1.02 using the movable incore detectors.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Departure from Nucleate Boiling Ratio (DNBR)

LCO 3.2.5 DNBR shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP when Power Distribution Monitoring System (PDMS) is OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DNBR not within limit.	A.1 Restore DNBR to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify DNBR is within limit specified in the COLR.	7 days

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 or more required channels or trains inoperable.	A.1	Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or train(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1	Restore channel to OPERABLE status.	48 hours
	<u>OR</u>		
	B.2	Be in MODE 3.	54 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	-------------NOTE------------- While this LCO is not met for Function 18, 19, or 20 in MODE 5, making the Rod Control System capable of rod withdrawal is not permitted. ------------------------------	
	C.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	C.2.1 Initiate action to fully insert all rods.	48 hours
	<u>AND</u>	
	C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
D. One Power Range Neutron Flux-High channel inoperable.	-------------NOTE------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. ------------------------------	
	D.1 Place channel in trip.	6 hours
	<u>OR</u>	
	D.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------NOTE------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------	
	E.1 Place channel in trip.	6 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours
	<u>OR</u>	
	F.2 Increase THERMAL POWER to > P-10.	2 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. One Source Range Neutron Flux channel inoperable.	H.1 Suspend operations involving positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. Two Source Range Neutron Flux channels inoperable.	I.1	Open Reactor Trip Breakers (RTBs).	Immediately
J. One Source Range Neutron Flux channel inoperable.	J.1	Restore channel to OPERABLE status.	48 hours
	<u>OR</u>		
	J.2.1	Initiate action to fully insert all rods.	48 hours
	<u>AND</u>		
	J.2.2	Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
K. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------		
	K.1	Place channel in trip.	6 hours
	<u>OR</u>		
	K.2	Reduce THERMAL POWER to < P-7.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One Turbine Trip channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ------------------------------	
	L.1 Place channel in trip.	6 hours
	<u>OR</u>	
	L.2 Reduce THERMAL POWER to < P-8.	12 hours
M. One train inoperable.	------------NOTE------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ------------------------------	
	M.1 Restore train to OPERABLE status.	6 hours
	<u>OR</u>	
	M.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One RTB train inoperable.	-------------NOTES------------- 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 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. -------------------------------	
	N.1 Restore train to OPERABLE status.	1 hour
	<u>OR</u>	
	N.2 Be in MODE 3.	7 hours
O. One or more channels inoperable.	O.1 Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>	
	O.2 Be in MODE 3.	7 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
P. One or more channels inoperable.	P.1	Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>		
	P.2	Be in MODE 2.	7 hours
Q. One trip mechanism inoperable for one RTB.	Q.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	<u>OR</u>		
	Q.2	Be in MODE 3.	54 hours

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.2	-----NOTES----- 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. ---------- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours
SR 3.3.1.3	-----NOTES----- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Only required to be performed with THERMAL POWER > 15% RTP. ---------- Compare results of the incore measurements to NIS AFD.	Prior to exceeding 75% RTP after each refueling AND 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	-------------------NOTE------------------- This Surveillance must be performed on the RTBB prior to placing the bypass breaker in service. -- 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 24 hours after THERMAL POWER is ≥ 75% RTP. -- Calibrate excore channels to agree with incore 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 entry into MODE 3 -- Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	----NOTE---- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ---------- Perform COT.	-----NOTE----- Only required when not performed within previous 92 days ---------- Prior to reactor startup AND Four hours after reducing power below P-10 for power and intermediate instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND Every 92 days thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	92 days
SR 3.3.1.10	-------------------NOTE------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months
SR 3.3.1.13	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14	-----NOTE----- Verification of setpoint is not required. --- Perform TADOT.	-----NOTE----- Only required when not performed within previous 31 days --- Prior to reactor startup
SR 3.3.1.15	-----NOTE----- Neutron detectors are excluded from response time testing. --- Verify RTS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 6.2% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Overtemperature ΔT	1.2	4	E	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 SR 3.3.1.15	Refer to Note 1 (Page 3.3.1-18)
7. Overpower ΔT	1.2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3.1-19)
8. Pressurizer Pressure					
a. Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1875 psig
b. High	1.2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2393 psig
9. Pressurizer Water Level – High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.5% of instrument span
10. Reactor Coolant Flow – Low (per loop)	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.3% of loop minimum measured flow
11. Reactor Coolant Pump (RCP) Breaker Position (per train)	1(e)	4	K	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) Interlock.

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Undervoltage RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4920 V
13. Underfrequency RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.08 Hz
14. Steam Generator (SG) Water Level - Low Low (per SG)					
a. Unit 1	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.1% of narrow range instrument span
b. Unit 2	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 34.8% of narrow range instrument span
15. Turbine Trip					
a. Emergency Trip Header Pressure (per train)	1(f)	3	L	SR 3.3.1.10 SR 3.3.1.14	≥ 910 psig
b. Turbine Throttle Valve Closure (per train)	1(f)	4	L	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	M	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock

(f) Above the P-8 (Power Range Neutron Flux) interlock

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Trip System Interlocks					
a. Source Range Block Permissive, P-6	2[d]	2	O	SR 3.3.1.11 SR 3.3.1.12	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7					
(1) P-10 Input	1	3	P	SR 3.3.1.11 SR 3.3.1.12	NA
(2) P-13 Input	1	2	P	SR 3.3.1.10 SR 3.3.1.12	NA
c. Power Range Neutron Flux, P-8	1	3	P	SR 3.3.1.11 SR 3.3.1.12	≤ 32.1% RTP
d. Power Range Neutron Flux, P-10	1,2	3	O	SR 3.3.1.11 SR 3.3.1.12	≥ 7.9% RTP and ≤ 12.1% RTP
e. Turbine Impulse Pressure, P-13	1	2	P	SR 3.3.1.10 SR 3.3.1.12	≤ 12.1% turbine power
18. Reactor Trip Breakers (RTBs)[g]	1,2	2 trains	N	SR 3.3.1.4	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	Q	SR 3.3.1.4	NA
	3[a], 4[a], 5[a]	1 each per RTB	C	SR 3.3.1.4	NA
20. Automatic Trip Logic	1,2	2 trains	M	SR 3.3.1.5	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(d) Below the P-6 (Source Range Block Permissive) interlocks.

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

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[T \frac{1}{(1+\tau_6 s)} - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured Reactor Coolant System (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, ≤ 588.4°F.

P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, = 2235 psig.

$K_1 = 1.325$	$K_2 = 0.0297/°F$	$K_3 = 0.00181/psig$
$\tau_1 = 8$ sec	$\tau_2 = 3$ sec	$\tau_3 \leq 2$ sec
$\tau_4 = 33$ sec	$\tau_5 = 4$ sec	$\tau_6 \leq 2$ sec

$f_1(\Delta I)$ = $-3.35\{24 + (q_t - q_b)\}$ when $q_t - q_b < -24\%$ RTP
0% of RTP when -24% RTP $\leq q_t - q_b \leq 10\%$ RTP
$4.11\{(q_t - q_b) - 10\}$ when $q_t - q_b > 10\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left[\frac{1}{1+\tau_6 s}\right] T - K_6 \left[T \frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

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, ≤ 588.4°F.

K_4 = 1.072

K_5 = 0.02/°F for increasing T_{avg}
0/°F for decreasing T_{avg}

K_6 = 0.00245/°F when T > T"
0/°F when T ≤ T"

τ_1 = 8 sec
τ_2 = 3 sec
τ_3 ≤ 2 sec
τ_6 ≤ 2 sec
τ_7 = 10 sec

$f_2(\Delta I)$ = 0 for all ΔI.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Moderator Temperature Coefficient";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
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 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)";
LCO 3.9.1, "Boron Concentration"; and

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 Evaluations Methodology," July 1985.

2. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods," July 1983.

4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," July 1990.

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

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

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; and

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

ATTACHMENT B-4

CLEAN COPY TS PAGES
FOR BRAIDWOOD STATION, UNITS 1 AND 2

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One rod not within alignment limit.	B.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>		
	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours from discovery of Condition B concurrent with inoperability of Power Distribution Monitoring System (PDMS)
	<u>AND</u>		
	B.3	Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	<u>AND</u>		
	B.4	Determine Heat Flux Hot Channel Factor ($F_Q(Z)$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$).	72 hours
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.5 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. More than one rod not within alignment limit.	C.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	OR	
	C.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
	C.2 Be in MODE 3.	6 hours from discovery of Condition C concurrent with inoperability of PDMS
	AND	
	C.3 --------NOTE--------- Only required to be performed when PDMS is OPERABLE. --------------------- Restore rod(s) to within alignment limit.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B or Required Action C.3 not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 550°F$, and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each inoperable DRPI and each demand position indicator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable DRPIs.	Once per 8 hours
	OR		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1	Initiate action to verify the position of the rods with inoperable DRPIs.	Immediately
	OR		
	B.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One demand position indicator per bank inoperable for one or more banks.	C.1.1	Verify by administrative means all DRPIs for the affected bank(s) are OPERABLE.	Once per 8 hours
	<u>AND</u>		
	C.1.2	Verify the most withdrawn rod and the least withdrawn rod of the affected bank(s) are ≤ 12 steps apart.	Once per 8 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Prior to criticality after each removal of the reactor head.

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 limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q^C(Z)$ not within limit.	A.1	Reduce THERMAL POWER $\geq$ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes
	AND		
	A.2	Reduce Power Range Neutron Flux–High trip setpoints $\geq$ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
	AND		
	A.3	Reduce Overpower ΔT trip setpoints $\geq$ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
B. $F_Q^W(Z)$ not within limit.	B.1	Reduce THERMAL POWER $\geq$ 1% RTP for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	<u>AND</u>		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	-------------------NOTES------------------- 1. 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. 2. Not required to be performed until 12 hours after declaring Power Distribution Monitoring System (PDMS) inoperable. Performance of SR 3.2.1.3 satisfies the initial performance of this SR after declaring PDMS inoperable. ---	
	Verify $F_Q^C(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling <u>AND</u> 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 <u>AND</u> 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-------------------NOTES------------------- 1. 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. 2. If $F_Q^W(Z)$ measurements indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)} \right]$ 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 specified in the COLR; 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 $\left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has not increased. --	
		(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued) -----NOTES----- 3. Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.1.4 satisfies the initial performance of this SR after declaring PDMS inoperable. --------------- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. --- Verify $F_Q^C(Z)$ is within limit specified in the COLR.	7 days
SR 3.2.1.4	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. --- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE--------- Required Actions A.2 and A.4 must be completed whenever Condition A is entered. ---------------------- $F^N_{\Delta H}$ not within limit.	A.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND		
	A.2	Determine $F^N_{\Delta H}$.	24 hours
	AND		
	A.3	Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.	72 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	---------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. --------------------- Determine $F^{N}_{\Delta H}$.	Prior to exceeding 50% RTP <u>AND</u> Prior to exceeding 75% RTP <u>AND</u> 24 hours after reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	-------------------NOTE------------------- Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.2.2 satisfies the initial performance of this SR after declaring PDMS inoperable. --	
	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND 31 Effective Full Power Days thereafter
SR 3.2.2.2	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. --	
	Verify $F^N_{\Delta H}$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3. The AFD 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 when Power Distribution Monitoring System (PDMS) is inoperable.

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	-----NOTE----- Not required to be performed until 12 hours after declaring PDMS inoperable. ---------- Verify AFD is 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 when Power Distribution Monitoring System (PDMS) is inoperable.

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
	<u>AND</u>		
	A.2	Determine QPTR and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>		
	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 <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
			(continued)

FOR INFORMATION ONLY

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to exceeding the THERMAL POWER limit of Required Action A.1
	AND	
	A.5 ---------NOTES--------- 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 limits.	Prior to exceeding the THERMAL POWER limits of Required Action A.1
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	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.	24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after exceeding the THERMAL POWER limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-----NOTES----- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channel inputs can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. 3. Not required to be performed until 12 hours after declaring PDMS inoperable. ----- Verify QPTR is ≤ 1.02 by calculation.	7 days
SR 3.2.4.2	-----NOTES----- 1. Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. 2. Not required to be performed until 12 hours after declaring PDMS inoperable. ----- Verify QPTR is ≤ 1.02 using the movable incore detectors.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Departure from Nucleate Boiling Ratio (DNBR)

LCO 3.2.5 DNBR shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP when Power Distribution Monitoring System (PDMS) is OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DNBR not within limit.	A.1 Restore DNBR to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify DNBR is within limit specified in the COLR.	7 days

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 or more required channels or trains inoperable.	A.1	Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or train(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1	Restore channel to OPERABLE status.	48 hours
	OR		
	B.2	Be in MODE 3.	54 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	------------NOTE------------ While this LCO is not met for Function 18, 19, or 20 in MODE 5, making the Rod Control System capable of rod withdrawal is not permitted. -----------------------------	
	C.1 Restore channel or train to OPERABLE status.	48 hours
	OR	
	C.2.1 Initiate action to fully insert all rods.	48 hours
	AND	
	C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
D. One Power Range Neutron Flux–High channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels -----------------------------	
	D.1 Place channel in trip.	6 hours
	OR	
	D.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------		
	E.1	Place channel in trip.	6 hours
	OR		
	E.2	Be in MODE 3.	12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1	Reduce THERMAL POWER to < P-6.	2 hours
	OR		
	F.2	Increase THERMAL POWER to > P-10.	2 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1	Suspend operations involving positive reactivity additions.	Immediately
	AND		
	G.2	Reduce THERMAL POWER to < P-6.	2 hours
H. One Source Range Neutron Flux channel inoperable.	H.1	Suspend operations involving positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. Two Source Range Neutron Flux channels inoperable.	I.1	Open Reactor Trip Breakers (RTBs).	Immediately
J. One Source Range Neutron Flux channel inoperable.	J.1	Restore channel to OPERABLE status.	48 hours
	<u>OR</u>		
	J.2.1	Initiate action to fully insert all rods.	48 hours
	<u>AND</u>		
	J.2.2	Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
K. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------		
	K.1	Place channel in trip.	6 hours
	<u>OR</u>		
	K.2	Reduce THERMAL POWER to < P-7.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One Turbine Trip channel inoperable.	-------------NOTE------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. -------------------------------	
	L.1 Place channel in trip.	6 hours
	OR	
	L.2 Reduce THERMAL POWER to < P-8.	12 hours
M. One train inoperable.	-------------NOTE------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. -------------------------------	
	M.1 Restore train to OPERABLE status.	6 hours
	OR	
	M.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One RTB train inoperable.	-----NOTES----- 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 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. ---------------	
	N.1 Restore train to OPERABLE status.	1 hour
	<u>OR</u>	
	N.2 Be in MODE 3.	7 hours
O. One or more channels inoperable.	O.1 Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>	
	O.2 Be in MODE 3.	7 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
P. One or more channels inoperable.	P.1	Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>		
	P.2	Be in MODE 2.	7 hours
Q. One trip mechanism inoperable for one RTB.	Q.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	<u>OR</u>		
	Q.2	Be in MODE 3.	54 hours

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.2	-----NOTES----- 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. --------------- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours
SR 3.3.1.3	-----NOTES----- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Only required to be performed with THERMAL POWER > 15% RTP. --------------- Compare results of the incore measurements to NIS AFD.	Prior to exceeding 75% RTP after each refueling AND 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.4	-------------------NOTE------------------- This Surveillance must be performed on the RTBB prior to placing the bypass breaker in service. -- 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 24 hours after THERMAL POWER is ≥ 75% RTP. -- Calibrate excore channels to agree with incore 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 entry into MODE 3. -- Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	-------------------NOTE------------------- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -- Perform COT.	-----NOTE----- Only required when not performed within previous 92 days --------------- Prior to reactor startup <u>AND</u> Four hours after reducing power below P-10 for power and intermediate instrumentation <u>AND</u> Four hours after reducing power below P-6 for source range instrumentation <u>AND</u> Every 92 days thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	92 days
SR 3.3.1.10	-------------------NOTE------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months
SR 3.3.1.13	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	-----NOTE------ Only required when not performed within previous 31 days -------------- Prior to reactor startup
SR 3.3.1.15	-------------------NOTE------------------- Neutron detectors are excluded from response time testing. -- Verify RTS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 6.2% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Overtemperature ΔT	1,2	4	E	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 SR 3.3.1.15	Refer to Note 1 (Page 3.3.1-18)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3.1-19)
8. Pressurizer Pressure					
a. Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1875 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2393 psig
9. Pressurizer Water Level – High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.5% of instrument span
10. Reactor Coolant Flow – Low (per loop)	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.3% of loop minimum measured flow
11. Reactor Coolant Pump (RCP) Breaker Position (per train)	[illegible]	[illegible]	K	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor [illegible]

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Undervoltage RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4920 V
13. Underfrequency RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.08 Hz
14. Steam Generator (SG) Water Level - Low Low (per SG)					
a. Unit 1	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.1% of narrow range instrument span
b. Unit 2	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 34.8% of narrow range instrument span
15. Turbine Trip					
a. Emergency Trip Header Pressure (per train)	1(f)	3	L	SR 3.3.1.10 SR 3.3.1.14	≥ 910 psig
b. Turbine Throttle Valve Closure (per train)	1(f)	4	L	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	M	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor [illegible]

(f) Above the P-8 (Power Range Neutron [illegible]

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Trip System Interlocks					
a. Source Range Block Permissive, P-6	2(d)	2	O	SR 3.3.1.11 SR 3.3.1.12	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7					
(1) P-10 Input	1	3	P	SR 3.3.1.11 SR 3.3.1.12	NA
(2) P-13 Input	1	2	P	SR 3.3.1.10 SR 3.3.1.12	NA
c. Power Range Neutron Flux, P-8	1	3	P	SR 3.3.1.11 SR 3.3.1.12	≤ 32.1% RTP
d. Power Range Neutron Flux, P-10	1,2	3	O	SR 3.3.1.11 SR 3.3.1.12	≥ 7.9% RTP and ≤ 12.1% RTP
e. Turbine Impulse Pressure, P-13	1	2	P	SR 3.3.1.10 SR 3.3.1.12	≤ 12.1% turbine power
18. Reactor Trip Breakers (RTBs)(g)	1,2	2 trains	N	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	Q	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	1 each per RTB	C	SR 3.3.1.4	NA
20. Automatic Trip Logic	1,2	2 trains	M	SR 3.3.1.5	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable [illegible] more rods not fully inserted.

(d) Below the P-6 (Source Range Block [illegible]

(g) Including any reactor trip bypass [illegible] racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[T \frac{1}{(1+\tau_6 s)} - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured Reactor Coolant System (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, ≤ 588.4°F.

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure, = 2235 psig.

K_1 = 1.325	K_2 = 0.0297/°F	K_3 = 0.00181/psig
τ_1 = 8 sec	τ_2 = 3 sec	τ_3 ≤ 2 sec
τ_4 = 33 sec	τ_5 = 4 sec	τ_6 ≤ 2 sec

$f_1(\Delta I)$ = $-3.35\{24 + (q_t - q_b)\}$ when $q_t - q_b < -24\%$ RTP
0% of RTP when -24% RTP $\le q_t - q_b \le 10\%$ RTP
$4.11\{(q_t - q_b) - 10\}$ when $q_t - q_b > 10\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \leq \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left[\frac{1}{1+\tau_6 s}\right] T - K_6 \left[T \frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

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, ≤ 588.4°F.

K_4 = 1.072

K_5 = 0.02/°F for increasing T_{avg}
0/°F for decreasing T_{avg}

K_6 = 0.00245/°F when T > T"
0/°F when T ≤ T"

τ_1 = 8 sec
τ_2 = 3 sec
τ_3 ≤ 2 sec
τ_6 ≤ 2 sec
τ_7 = 10 sec

$f_2(\Delta I)$ = 0 for all ΔI.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Moderator Temperature Coefficient";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
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 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)";
LCO 3.9.1, "Boron Concentration"; and

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 Evaluations Methodology," July 1985.

2. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods," July 1983.

4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," July 1990.

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

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

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; and

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

Att B-5/B-6
TRM pgs

ATTACHMENT B-5

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FOR BYRON STATION, UNITS 1 AND 2

3.3 INSTRUMENTATION

3.3.a Movable Incore Detectors

TLCO 3.3.a The Movable Incore Detection System shall be OPERABLE with:

1. ≥ 75% of the detector thimbles,
2. ≥ 2 detector thimbles per core quadrant, and
3. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

---NOTE---

Only ≥ 50% of the detector thimbles are required for Power Distribution Monitoring System (PDMS) calibrations after the initial PDMS calibration following each refueling.

APPLICABILITY: When the Movable Incore Detection System is used for:

1. Recalibration of the Excore Neutron Flux Detection System,
2. Calibration of the PDMS,
3. Monitoring normalized symmetric power distribution, or
4. Measurement of $F^{N}_{\Delta H}$, $F^{C}_{Q}(Z)$, and $F^{W}_{Q}(Z)$.

ACTIONS

-----NOTE-----
TLCO 3.0.c is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Movable Incore Detection System inoperable.	A.1 Suspend use of the Movable Incore Detection System data for applicable recalibration, measurement, or monitoring.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.3.a.1 Normalize each detector output when required for: a. Recalibration of the Excore Neutron Flux Detection System. b. Calibration of the PDMS. c. Monitoring normalized symmetric power distribution, or d. Measurement of $F^N_{\Delta H}$, $F^C_Q(Z)$, and $F^W_Q(Z)$.	24 hours

3.3 INSTRUMENTATION

3.3.h Power Distribution Monitoring System (PDMS) Instrumentation

TLCO 3.3.h The PDMS Instrumentation for each Function in Table T3.3.h-1 shall be OPERABLE.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP when PDMS is OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Restore required channel to OPERABLE status.	4 hours
B. PDMS inoperable for reasons other than Condition A. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Declare PDMS inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.h.1	Perform CHANNEL CHECK for each required instrumentation channel.	12 hours
TSR 3.3.h.2	------------NOTE------------ Neutron detectors are excluded from CHANNEL CALIBRATION. ---------------------------- Perform CHANNEL CALIBRATION for each required instrumentation channel.	18 months
TSR 3.3.h.3	Perform PDMS calibration.	Prior to declaring PDMS OPERABLE after each refueling <u>AND</u> 31 Effective Full Power Days (EFPD) thereafter with the Core Exit Thermocouple (CETC) chess knight move pattern not satisfied <u>AND</u> 180 EFPD thereafter with the CETC chess knight move pattern satisfied

Table T3.3.h-1 (Page 1 of 1)
Power Distribution Monitoring System Instrumentation

FUNCTION	REQUIRED CHANNELS
1. Power Range Neutron Flux Monitors	3
2. Reactor Coolant System (RCS) Cold Leg Temperature (Wide Range T_c)	2[a]
3. Reactor Power	1[b]
4. Control Bank Position (per bank)	1[c]
5. Core Exit Temperature	17 with ≥ 2 per core quadrant

(a) T_c shall be from the same RCS loop and core quadrant as an OPERABLE Power Range Neutron Flux Monitor.

(b) Either calorimetric power, the average power of the power range neutron flux monitors, or the average power of the ΔT channels.

(c) Either the Digital Rod Position Indication (DRPI) System or the Demand Position Indication System.

ATTACHMENT B-6

CLEAN COPY TRM PAGES
FOR BRAIDWOOD STATION, UNITS 1 AND 2

3.3 INSTRUMENTATION

3.3.a Movable Incore Detectors

TLCO 3.3.a The Movable Incore Detection System shall be OPERABLE with:

1. ≥ 75% of the detector thimbles,
2. ≥ 2 detector thimbles per core quadrant, and
3. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

---NOTE---
Only ≥ 50% of the detector thimbles are required for Power Distribution Monitoring System (PDMS) calibrations after the initial PDMS calibration following each refueling.

APPLICABILITY: When the Movable Incore Detection System is used for:

1. Recalibration of the Excore Neutron Flux Detection System.
2. Calibration of the PDMS.
3. Monitoring normalized symmetric power distribution, or
4. Measurement of $F^{N}_{\Delta H}$, $F^{C}_{Q}(Z)$, and $F^{W}_{Q}(Z)$.

ACTIONS

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

TLCO 3.0.c is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Movable Incore Detection System inoperable.	A.1 Suspend use of the Movable Incore Detection System data for applicable recalibration, measurement, or monitoring.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.a.1	Normalize each detector output when required for: a. Recalibration of the Excore Neutron Flux Detection System. b. Calibration of the PDMS. c. Monitoring normalized symmetric power distribution, or d. Measurement of $F_{\Delta H}^{N}$, $F_{Q}^{C}(Z)$, and $F_{Q}^{W}(Z)$.	24 hours

3.3 INSTRUMENTATION

3.3.h Power Distribution Monitoring System (PDMS) Instrumentation

TLCO 3.3.h The PDMS Instrumentation for each Function in Table T3.3.h-1 shall be OPERABLE.

APPLICABILITY: MODE 1 with THERMAL POWER > 25% RTP when PDMS is OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1	Restore required channel to OPERABLE status.	4 hours
B. PDMS inoperable for reasons other than Condition A. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1	Declare PDMS inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.h.1	Perform CHANNEL CHECK for each required instrumentation channel.	12 hours
TSR 3.3.h.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION for each required instrumentation channel.	18 months
TSR 3.3.h.3	Perform PDMS calibration.	Prior to declaring PDMS OPERABLE after each refueling AND 31 Effective Full Power Days (EFPD) thereafter with the Core Exit Thermocouple (CETC) chess knight move pattern not satisfied AND 180 EFPD thereafter with the CETC chess knight move pattern satisfied

Table T3.3.h-1 (Page 1 of 1)
Power Distribution Monitoring System Instrumentation

FUNCTION	REQUIRED CHANNELS
1. Power Range Neutron Flux Monitors	3
2. Reactor Coolant System (RCS) Cold Leg Temperature (Wide Range T_c)	2[(a)]
3. Reactor Power	1[(b)]
4. Control Bank Position (per bank)	1[(c)]
5. Core Exit Temperature	17 with ≥ 2 per core quadrant

(a) T_c shall be from the same RCS loop and core quadrant as an OPERABLE Power Range Neutron Flux Monitor.

(b) Either calorimetric power, the average power of the power range neutron flux monitors, or the average power of the ΔT channels.

(c) Either the Digital Rod Position Indication (DRPI) System or the Demand Position Indication System.

ATTACHMENT B-7

CLEAN COPY ITS BASES PAGES FOR BYRON STATION, UNITS 1 AND 2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution, and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod Cluster Control Assemblies (RCCAs), or rods, are moved by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

BASES

BACKGROUND (continued)

The 53 RCCAs are divided among four control banks and five shutdown banks. A bank of RCCAs consists of either one group, or, two groups that are moved in a staggered fashion to provide for precise reactivity control but which are always within one step of each other. Each of the control banks are divided into two groups, for a total of 25 control bank rods. Shutdown banks A and B are also divided into two groups, however, shutdown banks C, D and E have only one group each, for a total of 28 shutdown bank rods. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the fully withdrawn position, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

BASES

BACKGROUND (continued)

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch) but not very reliable because it is a demanded position indication, not an actual position indication. For example, if a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue (i.e., statically misaligned RCCA). This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case with control bank D inserted to its full power insertion limit and one RCCA fully withdrawn. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA fully withdrawn (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local Linear Heat Rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e. trippability to meet SDM) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move rods (e.g. rod urgent failures), but do not impact trippability, do not result in rod inoperability provided proper alignment.

BASES

LCO (continued)

The requirement to maintain individual indicated rod positions within 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements for MODE 6.

BASES

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value specified in the COLR, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating boration to restore SDM to within limit.

In this situation, SDM verification must account for the worth of the untrippable rod(s), as well as the rod of maximum worth.

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the unit must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1.1 and B.1.2

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." One hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

BASES

ACTIONS (continued)

However, in many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be partially inserted.

With a misaligned rod, SDM must be verified to be within limit (specified in the COLR) or boration must be initiated to restore SDM to within limit.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour.

The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration to restore SDM to within limit.

B.2, B.3, B.4, and B.5

For continued operation with a misaligned rod, THERMAL POWER must be reduced when Power Distribution Monitoring System (PDMS) is inoperable, SDM must periodically be verified within limits (specified in the COLR), hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP when PDMS is inoperable, ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 4). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. One rod is not within alignment limit; and

b. PDMS is inoperable.

BASES

ACTIONS (continued)

Discovering one rod not within alignment limit coincident with PDMS inoperable results in starting the Completion Time for the Required Action. During power operation when PDMS is OPERABLE, LHR is measured continuously. Therefore, a reduction of power to 75% RTP is not necessary to ensure that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation, at ≤ 75% RTP with PDMS inoperable and > 75% RTP with PDMS OPERABLE, with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain the core power distribution using the incore flux mapping system or PDMS and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Accident for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

Accident analyses (Ref. 3) requiring re-evaluation for continued operation with a misaligned rod include:

1. Increase in heat removal by the secondary system:

    a. Excessive increase in secondary steam flow,

    b. Inadvertent opening of a steam generator power operated relief or safety valve, and

    c. Steam system piping failure;

BASES

ACTIONS (continued)

2. Uncontrolled RCCA bank withdrawal at power:

3. RCCA misoperation:

   a. One or more dropped RCCAs within the same group.

   b. A dropped RCCA bank.

   c. Statically misaligned RCCA. and

   d. Withdrawal of a single RCCA:

4. RCCA ejection accidents: and

5. Loss of coolant accidents resulting from postulated piping breaks within the reactor coolant pressure boundary.

C.1.1 and C.1.2

More than one control rod becoming misaligned from its group average position is not expected. and has the potential to reduce SDM. Therefore. SDM (specified in the COLR) must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM. if necessary. requires increasing the RCS boron concentration to provide negative reactivity. as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable. based on the time required for potential xenon redistribution. the low probability of an accident occurring. and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored

BASES

ACTIONS (continued)

C.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement when PDMS is inoperable, the unit conditions may fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. More than one rod is not within alignment limit; and

b. PDMS is inoperable.

Discovering more than one rod not within alignment limit coincident with PDMS inoperable results in starting the Completion Time for the Required Action.

C.3

If more than one rod is found to be misaligned or becomes misaligned because of bank movement when PDMS is OPERABLE, operation may continue in Condition C for a period that should not exceed 72 hours. The allowed Completion Time is reasonable, based on the available information on power distributions (Ref 6) This Required Action is modified by a Note that requires the performance of Required Action C.3 only when PDMS is OPERABLE.

BASES

ACTIONS (continued)

D.1

When Required Actions of Condition B or C.3 cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. This frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable (e.g., as a result of excessive friction, mechanical interference, or rod control system failure), a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times once prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all Reactor Coolant Pumps (RCPs) operating and the average moderator temperature ≥ 550°F to ensure that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

This Surveillance is performed during a unit outage, due to conditions needed to perform the SR and the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

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

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

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

When Power Distribution Monitoring System (PDMS) is inoperable, $F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 Effective Full Power Days (EFPD). However, during power operation when PDMS is inoperable, 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 address directly and continuously measured process variables. During power operation when PDMS is OPERABLE, the linear power along the fuel rod with the highest integrated power is measured continuously and $F^N_{\Delta H}$ is determined continuously.

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design criterion for the Departure from Nucleate Boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

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

APPLICABLE SAFETY ANALYSES

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

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

b. During a large break Loss Of Coolant Accident (LOCA), Peak Cladding Temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design criterion is met for normal operation, operational transients, and any transients arising from events of moderate frequency. Refer to the Bases for LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits," for a discussion of the applicable Departure from Nucleate Boiling Ratio (DNBR) limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," LCO 3.2.2, LCO 3.2.3, LCO 3.2.4, and LCO 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)."

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

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

BASES

LCO

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

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

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the plant safety analyses.

The power multiplication factor in this equation provides margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

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

ACTIONS

A.1, A.2, A.3, and A.4

With $F^N_{\Delta H}$ exceeding its limit, Condition A is entered. $F^N_{\Delta H}$ may be restored to within its limits within 4 hours, through, for example, realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. If the value of $F^N_{\Delta H}$ is not restored to within its specified limit, THERMAL POWER must be reduced to $\leq$ 50% RTP in accordance with Required Action A.1. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses.

BASES

ACTIONS (continued)

However, the DNBR limit may be violated if a DNB limiting event occurs. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and is not likely to cause the DNBR limit to be violated in steady state operation. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the unit to remain in an unacceptable condition for an extended period of time.

Condition A is modified by a Note that requires that Required Actions A.2 and A.4 must be completed whenever Condition A is entered. Thus, even if $F^N_{\Delta H}$ is restored within the 4 hour time period of Required Action A.1, Required Action A.2 would nevertheless require another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1. Required Action A.4 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP.

Required Action A.2 requires the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level once the power level has been reduced to < 50% RTP per Required Action A.1. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by Action A.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map when PDMS is inoperable, perform the required calculations, and evaluate $F^N_{\Delta H}$.

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, Required Action A.3 requires the Power Range Neutron Flux—High trip setpoints be reduced to ≤ 55% RTP. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin.

ACTIONS (continued)

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.3 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

Required Action A.4 requires verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence. This ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced to comply with this Required Action.

B.1

If the Required Actions of A.1 through A.4 are not met within their associated Completion Times, the unit must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the unit in at least MODE 2 within 6 hours.

The 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 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

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

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

This Surveillance has been modified by a Note. The Note requires the measured value of $F^N_{\Delta H}$ be obtained from incore flux map results only when PDMS is inoperable. The Note modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable. and that the last performance of SR 3.2.2.2 prior to declaring PDMS inoperable satisfies the initial performance of this SR after declaring PDMS inoperable. If SR 3.2.2.1 were not performed within its specified Frequency, this Note allows 12 hours to verify $F^N_{\Delta H}$ is within limit using either the incore flux map results or by taking credit for the last performance of SR 3.2.2.2 when PDMS was OPERABLE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

The confirmation of the power distribution parameter, $F^N_{\Delta H}$, is an additional verification over the automated monitoring performed by PDMS. This assures that PDMS is functioning properly and that the core limits are met.

The Surveillance Frequency of 7 days takes into account other information and alarms available to the operator in the control room, and is adequate because $F^N_{\Delta H}$ is monitored by the process computer.

This Surveillance is modified by a Note that requires the performance of SR 3.2.2.2 for determining $F^N_{\Delta H}$ only when PDMS is OPERABLE.

REFERENCES

1. UFSAR, Section 15.4.8.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.

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 amount of axial power distribution skewing to either the top or bottom of the core when Power Distribution Monitoring System (PDMS) is inoperable. 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. When PDMS is OPERABLE, Peak Linear Heat Rate is measured continuously.

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 plant 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.

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) 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.

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 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 provide assurance that the thermal limits assumed in the accident analysis ($F^N_{\Delta H}$ and $F_Q(Z)$) are met. Thereby, the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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.

A Note modifies the LCO by stating the AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

The AFD limits are provided in the COLR. 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 with THERMAL POWER ≥ 50% RTP (i.e., when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis) when PDMS is inoperable.

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.

BASES

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 channels is within limits. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer.

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 limits.

A Note modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable. If SR 3.2.3.1 were not performed within its specified Frequency, this Note allows 12 hours to verify AFD is within limits.

REFERENCES

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

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

3. UFSAR, Section 7.7.1.3.1.

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.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, 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. When Power Distribution Monitoring System (PDMS) is OPERABLE, Peak Linear Heat Rate and the linear power along the fuel rod with the highest integrated power are measured continuously.

APPLICABLE SAFETY ANALYSES

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

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

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 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin with the highest worth control rod stuck fully withdrawn (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion, 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 limits on the QPTR provide assurance that the thermal limits assumed in the accident analysis ($F^N_{\Delta H}$ and $F_Q(Z)$) are met. Thereby, the QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at 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 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP when PDMS is inoperable to prevent core power distributions from exceeding the design limits.

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.

BASES

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% from 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.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reductions within 2 hours of QPTR determination, 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.

A.2

After completion of Required Action A.1, periodic monitoring provides a basis for maintaining the appropriate reduced power level. As such, a check of the QPTR is required once per 12 hours. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly, such that it is maintained at a reduced power level of 3% from RTP for each 1% by which QPTR exceeds 1.00.

Any of the Surveillance methods for determining QPTR may be used within the constraints for acceptability of the Surveillance (i.e., if the excore detectors are available, they should be used; if the excore detectors are not available, the moveable incore detectors may be used). A 12 hour Completion Time is sufficient because any additional change in QPTR should be relatively slow. Further, this Completion Time is consistent with the Frequency required for the Surveillances with an inoperable alarm or instrumentation

BASES

ACTIONS (continued)

A.3

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 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. The Completion Time takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the unit and perform a flux map. If these peaking factors are not within their 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 control 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.

BASES

ACTIONS (continued)

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limit prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the quadrant power tilt (QPT) is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (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 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 achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

BASES

ACTIONS (continued)

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (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 normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit 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.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

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 takes into account other information and alarms available to the operator in the control room.

This SR is modified by three Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1. Note 3 modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable. If SR 3.2.4.1 were not performed within its specified Frequency, this Note allows 12 hours to verify QPTR is within limits.

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

With input from an 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. The Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when input from one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations:

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of the radial core tilt to verify the QPTR can be used to confirm that QPTR is within limits.

With input from one NIS channel inoperable, the indicated tilt may be changed from the value indicated with input from all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is modified by two Notes. Note 1 states that it is not required to be performed until 12 hours after the input from one Power Range Neutron Flux channel is inoperable and the THERMAL POWER is > 75% RTP. Note 2 modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable. If SR 3.2.4.2 were not performed within its specified Frequency, this Note allows 12 hours to verify QPTR is ≤ 1.02 using the movable incore detectors.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Departure from Nucleate Boiling Ratio (DNBR)

BASES

BACKGROUND

The purpose of the limits on the value of DNBR determined by Power Distribution Monitoring System (PDMS) is to provide assurance of fuel integrity during Condition I (Normal Operation and Operational Transients) and Condition II (Faults of Moderate Frequency) events by providing the reactor operator with the information required to avoid exceeding the minimum Axial Power Shape Limiting DNBR ($DNBR_{APSL}$) in the core during normal operation and in short-term transients.

DNBR is defined as the ratio of the heat flux required to cause Departure from Nucleate Boiling (DNB) to the actual channel heat flux for given conditions.

During power operation when PDMS is inoperable, 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 within power distribution limits on a continuous basis.

During power operation when PDMS is OPERABLE, DNBR is determined continuously. Continuously monitoring the operation of the core significantly limits the adverse nature of power distribution initial conditions for transients. The core depletion status, xenon distribution, and soluble boron concentration restrict the possible power and reactivity transients. Continuously monitoring the power distribution allows the actual DNBR value to be maintained ≥ the $DNBR_{APSL}$ value specified in the COLR. $DNBR_{APSL}$ is the DNBR value determined to be the most sensitive to the core axial power distribution at the initial conditions of the limiting accident during the cycle-specific core reload design accident analysis process.

BASES

APPLICABLE SAFETY ANALYSES

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

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

The DNB safety analysis limit for a loss of forced reactor coolant flow accident (Ref. 1) is met by limiting DNBR to the 95/95 DNB design criterion of 1.4 using the WRB-2 Critical Heat Flux (CHF) correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience DNB. Maintaining the $DNBR_{APSL}$ value ≥ the DNBR value assumed in the safety and accident analyses ensures that the 95/95 DNB design criterion of 1.4 is met.

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. When PDMS is OPERABLE, this LCO and the following LCOs ensure this: LCO 3.1.6, LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," and LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$). When PDMS is inoperable, the following LCOs ensure this: LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4.

DNBR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

DNBR shall be maintained within the limit of the relationship specified in the COLR.

Maintaining DNBR ≥ $DNBR_{APSL}$ ensures the core operates within the limits assumed in the safety analyses. The $DNBR_{APSL}$ limit must be maintained to prevent core power distributions from exceeding the fuel design limits for DNBR.

Another limit on DNBR is provided in SL 2.1.1, "Reactor Core SLs." LCO 3.2.5 represents the initial conditions of the safety analysis which are far more restrictive than the Safety Limit (SL). Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

BASES

APPLICABILITY

The DNBR limit must be maintained in MODE 1 with THERMAL POWER ≥ 50% RTP when PDMS is OPERABLE to ensure DNB design criteria will be met in the event of an unplanned loss of forced coolant flow transient.

ACTIONS

A.1

Parameters affecting DNBR include Reactor Coolant System (RCS) pressure, RCS average temperature, RCS total flow rate, and Thermal Power. RCS pressure and RCS average temperature are controllable and measurable parameters. RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. With DNBR not within limit due to RCS pressure or RCS average temperature, action must be taken to restore these parameter(s). With DNBR not within limit due to the indicated RCS total flow rate, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of DNBR provides sufficient time to adjust unit parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If the value of DNBR is not restored to within its specified limit, THERMAL POWER must be reduced to < 50% RTP in accordance with Required Action B.1. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and is not likely to cause the DNBR limit to be violated in steady state operation. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore DNBR to within its limits without allowing the unit to remain in an unacceptable condition for an extended period of time.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The confirmation of the power distribution parameter, DNBR, is an additional verification over the automated monitoring performed by PDMS. This assures that PDMS is functioning properly and that the core limits are met.

The Surveillance Frequency of 7 days takes into account other information and alarms available to the operator in the control room, and is adequate because DNBR is monitored by the process computer.

REFERENCES

1. UFSAR, Chapter 15.

BASES

ACTIONS (continued)

A Note to the ACTIONS restricts the transition from MODE 5 with the Rod Control System not capable of rod withdrawal and all rods fully inserted, to MODE 5 with the Rod Control System capable of rod withdrawal or all rods not fully inserted for Functions 18, 19, and 20 while complying with the ACTIONS (i.e., while the LCO is not met). LCO 3.0.4 typically allows entry into MODES or other specified conditions in the Applicability while in MODE 5, however, the restriction of this Note is necessary to assure an OPERABLE RTS function prior to commencing operation with the Rod Control System capable of rod withdrawal or all rods not fully inserted.

D.1 and D.2

Condition D applies to the Power Range Neutron Flux—High Function.

The NIS power range detectors provide input to the Rod Control System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 7).

As an alternative to the above Action, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 7.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. UFSAR, Section 15.4.3.
5. UFSAR, Section 15.1.5.
6. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

BASES

BACKGROUND (continued)

Rod Cluster Control Assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The 53 RCCAs are divided among 4 control banks and 5 shutdown banks. A bank of RCCAs consists of either one group, or, two groups that are moved in a staggered fashion to provide for precise reactivity control but which are always within one step of each other. Each of the control banks are divided into two groups, for a total of 25 control bank rods. Shutdown banks A and B are also divided into two groups, however, shutdown banks C, D, and E have only one group each, for a total of 28 shutdown bank rods. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch) but not very reliable because it is a demanded position indication, not an actual position indication. For example, if a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

BASES

BACKGROUND (continued)

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. The DRPI System determines the actual position of each control bank and shutdown bank rod by using individual coils that are mounted concentrically along the outside boundaries of the rod drive pressure housings. Each control bank rod has 42 coil assemblies evenly spaced along its length at 3.75 inch (6 step) intervals from rod bottom to the fully withdrawn position. Each shutdown bank rod has 20 coil assemblies evenly spaced along its length at 3.75 inch intervals from rod bottom to 18 steps and from 210 steps to the fully withdrawn position, with a transition LED representing shutdown bank rod position between 18 steps and the fully withdrawn position. The coils magnetically sense the presence or absence of a rod drive shaft and send this information to two Data Cabinets located in the containment building. To prevent total loss of position indication due to a single failure, the outputs of every other coil are connected as inputs to one Data Cabinet, while the outputs of the remaining coils are connected to the other Data Cabinet. This division of coils and their respective cabinets is referenced as Data A and Data B coils/cabinets, and allows detection of rod position within the required band of ± 12 steps even with a complete failure of a set of coils.

Normal system accuracy is ± 4 steps (± 3 steps with an additional step added for coil placement and thermal expansion). If a data error occurs, the system is shifted to the "half accuracy" mode. As a rod is moved under "half accuracy" conditions, only every other LED will light (i.e., the LEDs associated with the operable data system) since the effective coil spacing is 7.5 inches (12 steps). Under "half accuracy" conditions with data A bad, the system accuracy is + 10 steps, - 4 steps. Under "half accuracy" conditions with data B bad, the system accuracy is + 4 steps, - 10 steps. Therefore, the normal indication accuracy of the DRPI System is ± 4 steps, and the maximum uncertainty is 10 steps. With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 22 steps.

BASES

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking, ejected rod worth, and with minimum SDM limits (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that the DRPI System and the Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE the following requirements must be met:

a. The DRPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits;"

b. The DRPI System has no failed coils; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

BASES

LCO (continued)

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable DRPI and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

BASES

ACTIONS (continued)

A.1

When one DRPI channel per group fails, the position of the rod can still be determined by use of the movable incore detectors or Power Distribution Monitoring System (PDMS). When PDMS is OPERABLE, the position of the rod may be determined from the difference between the measured core power distribution and the core power distribution expected to exist based on the position of the rod indicated by the group step counter demand position. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position will not cause core peaking factors to approach the core peaking factor limits.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

These Required Actions clarify that when one or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If immediate actions have not been initiated to verify the rod's position, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps.

BASES

ACTIONS (continued)

C.1.1 and C.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the DRPIs for the affected banks are OPERABLE and the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. This verification can be an examination of logs, administrative controls, or other information that shows that all DRPIs in the affected bank are OPERABLE.

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position will not cause core peaking to approach the core peaking factor limits. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.

This surveillance is performed prior to reactor criticality after each removal of the reactor head, since there is potential for unnecessary plant transients if the SR were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.

2. UFSAR, Chapter 15.

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 (Z) of the core.

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

During power operation when Power Distribution Monitoring System (PDMS) is inoperable, 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 within power distribution limits on a continuous basis. During power operation when PDMS is OPERABLE, PLHR is measured continuously.

$F_Q(Z)$ 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 when PDMS is inoperable. These measurements are generally taken with the core at or near equilibrium conditions. When PDMS is OPERABLE, $F_Q(Z)$ is determined continuously.

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.

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

a. During a large break Loss Of Coolant Accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

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 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 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).

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).

BASES

LCO

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

$$F_Q(Z) \le \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

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

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

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}.$$

For this facility, the actual values of F_Q^{RTP} and K(Z) are given in the COLR; however, F_Q^{RTP} is normally a number on the order of 2.50, and K(Z) is a function that looks like the one provided in Figure B 3.2.1-1.

$F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

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

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

where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

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

LCO (continued)

When PDMS is OPERABLE, $F_Q(Z)$ is determined continuously. Then,

$$F_Q^C(Z) = F_Q^M(Z) * U_{FQ}$$

where U_{FQ} is a factor that accounts for measurement uncertainty (Ref. 4) and engineering uncertainty defined in the COLR.

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

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

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

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

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

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 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

BASES

ACTIONS

A.1, A.2, and A.3

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ 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 unit to remain in an unacceptable condition for an extended period of time.

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.

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.

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 THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, maintains an acceptable absolute power density such that even if a transient occurred, core peaking factors are not exceeded.

BASES

ACTIONS (continued)

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds the 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 period and the preceding prompt reduction in THERMAL POWER 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 $F_Q^W(Z)$ exceeds the 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 period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.

C.1

If the Required Actions of A.1 through A.3, or B.1 through B.3, are not met within their associated Completion Times, the unit must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the unit in at least MODE 2 within 6 hours. The 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note (i.e.. Note 1) that 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 meaningful power distribution map can be obtained. These SRs are normally performed at > 40% RTP to provide core conditions as much like the full power conditions as possible (Ref. 5). 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 (and establishing equilibrium conditions) 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 $F_Q(Z)$ is 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 increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^C(Z)$. 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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. 6). $F_Q^C(Z)$ 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 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 analysis and because of the difficulty of making a precise measurement in these regions.

The Frequency of 31 Effective Full Power Days (EFPD) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the unit is operated in accordance with the Technical Specifications (TS).

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance has been modified by two Notes. Note 2 requires the measured value of $F_Q^C(Z)$ be obtained from incore flux map results only when PDMS is inoperable. Note 2 modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable, and that the last performance of SR 3.2.1.3 prior to declaring PDMS inoperable satisfies the initial performance of this SR after declaring PDMS inoperable. If SR 3.2.1.1 was not performed within its specified Frequency, this Note allows 12 hours to verify $F_Q^C(Z)$ is within limit using either the incore flux map results or by taking credit for the last performance of SR 3.2.1.3 when PDMS was OPERABLE.

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 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

This Surveillance has been modified by three Notes. Note 2 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

$$\text{maximum over } z \quad \left[\frac{F_Q^C(Z)}{K(Z)} \right] ,$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of the factor of 1.02 or by an appropriate factor specified in the COLR (Ref. 7), 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.

Note 3 requires the measured value of $F_Q^W(Z)$ be obtained from incore flux map results only when PDMS is inoperable. Note 3 modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable, and that the last performance of SR 3.2.1.4 prior to declaring PDMS inoperable satisfies the initial performance of this SR after declaring PDMS inoperable. If SR 3.2.1.2 were not performed within its specified Frequency, this Note allows 12 hours to verify $F_Q^W(Z)$ is within limit using either the incore flux map results or by taking credit for the last performance of SR 3.2.1.4 when PDMS was OPERABLE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 unit is operated in accordance with the TS. to preclude adverse peaking factors between 31 day surveillances.

SR 3.2.1.3

The confirmation of the power distribution parameter. $F_Q^C(Z)$. is an additional verification over the automated monitoring performed by PDMS. This assures that PDMS is functioning properly and that the core limits are met.

The Surveillance Frequency of 7 days takes into account other information and alarms available to the operator in the control room. and is adequate because $F_Q(Z)$ is monitored by the process computer.

This Surveillance is modified by a Note that requires the performance of SR 3.2.1.3 for determining $F_Q^C(Z)$ only when PDMS is OPERABLE

SR 3.2.1.4

The confirmation of the power distribution parameter. $F_Q^W(Z)$. is an additional verification over the automated monitoring performed by PDMS. This assures that PDMS is functioning properly and that the core limits are met.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency of 7 days takes into account other information and alarms available to the operator in the control room, and is adequate because $F_Q(Z)$ is monitored by the process computer.

This Surveillance is modified by a Note that requires the performance of SR 3.2.1.4 for determining $F_Q^W(Z)$ only when PDMS is OPERABLE.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.
5. ANSI/ANS-19.6.1-1985, "Reload Startup Physics Test for Pressurized Water Reactors," December 13, 1985.
6. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
7. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification," February 1994.

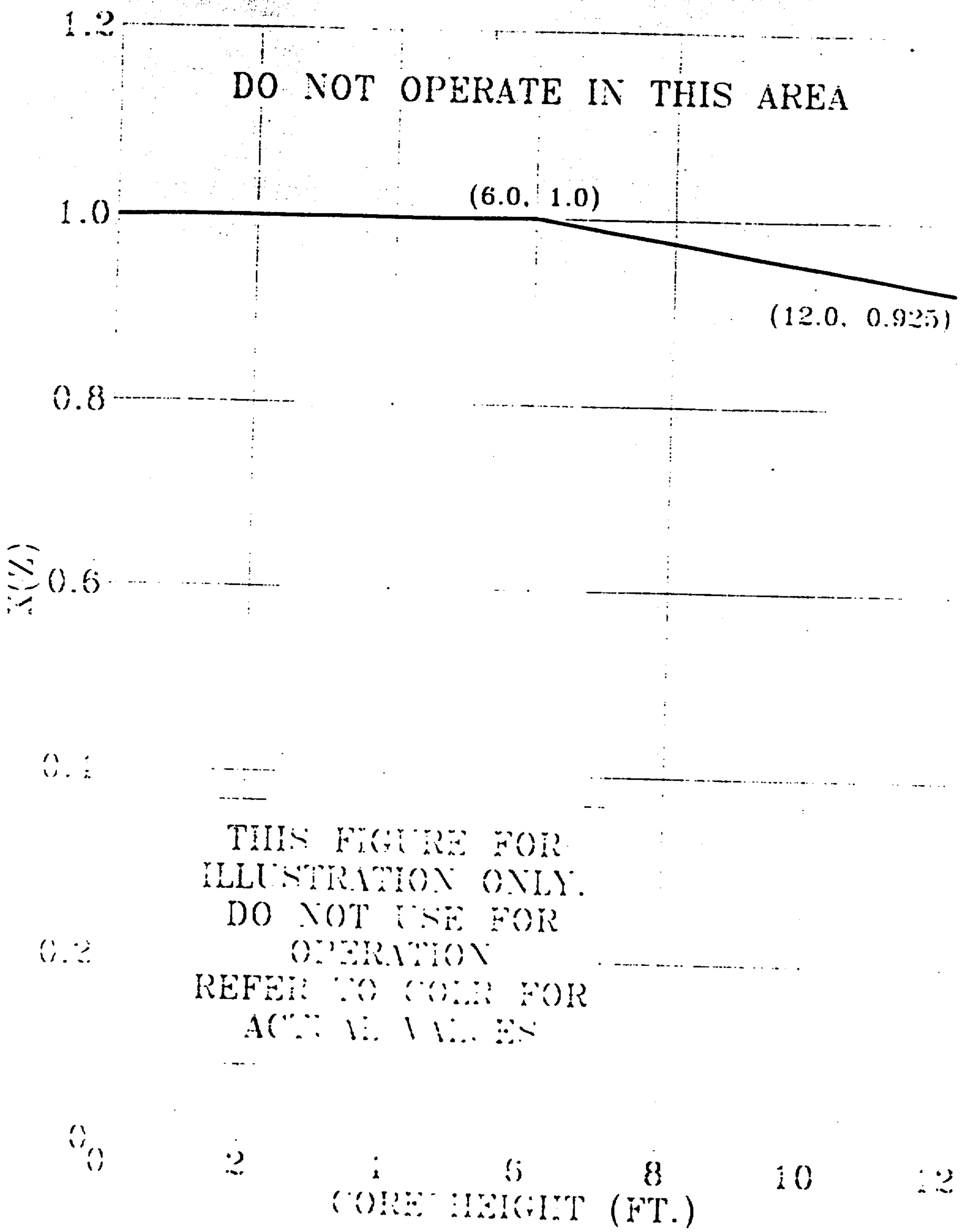

Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a function of Core Height

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BASES

ACTIONS (continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux – Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Power Range Neutron Flux – High Positive Rate;
- Power Range Neutron Flux – High Negative Rate;
- Pressurizer Pressure – High; and
- SG Water Level – Low Low.

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

BASES

ACTIONS (continued)

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 2 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The provisions of LCO 3.0.4 allow entry into a MODE or other specified condition in the Applicability as directed by the Required Actions. Therefore, a MODE change is permitted with one channel inoperable whenever Required Action F.2 is used. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

BASES

ACTIONS (continued)

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint. the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE. the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6. the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

H.1

Condition H applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2. below the P-6 setpoint. With the unit in this Condition. below P-6. the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable. operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE. core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

BASES

ACTIONS (continued)

I.1

Condition I applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition.

J.1 and J.2

Condition J applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour, are justified in Reference 7.

BASES

ACTIONS (continued)

K.1 and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low;
- RCP Breaker Position;
- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint. These Functions do not have to be OPERABLE below the P-7 setpoint. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

BASES

ACTIONS (continued)

L.1 and L.2

Condition L applies to Turbine Trip on Emergency Trip Header Pressure or on Turbine Throttle Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-8 setpoint within the next 6 hours. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

M.1 and M.2

Condition M applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action M.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours (Required Action M.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action M.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

BASES

ACTIONS (continued)

N.1 and N.2

Condition N applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 results in Action C entry while RTB(s) are inoperable.

The Required Actions have been modified by two Notes. Note 1 allows one channel to be bypassed for up to 2 hours for surveillance testing, provided the other channel is OPERABLE. Note 2 allows one RTB to be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE. The 2 hour time limit is justified in Reference 7.

O.1 and O.2

Condition O applies to the P-6 and P-10 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition by observation of the associated permissive annunciator window within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

BASES

ACTIONS (continued)

P.1 and P.2

Condition P applies to the P-7, P-8, and P-13 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition by observation of the associated permissive annunciator window within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging plant systems.

Q.1 and Q.2

Condition Q applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

With the unit in MODE 3, Action C would apply to any inoperable RTB trip mechanism. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 2 hours for the reasons stated under Condition N.

BASES

ACTIONS (continued)

The Completion Time of 48 hours for Required Action Q.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure, thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It 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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output prior to exceeding 75% RTP after each refueling and every 31 Effective Full Power days (EFPD) thereafter. 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 declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%. Note 2 clarifies that the Surveillance is required only if reactor power is > 15% RTP.

The Frequency of once prior to exceeding 75% RTP following each refueling outage considers that the core may be changed during a refueling outage such that the previous comparison, prior to the refueling outage, is no longer completely valid. The Frequency also considers that the comparison accuracy increases with power level such that the comparison is preferred to be performed at as high a power level as possible. An initial performance at ≤ 75% RTP provides a verification prior to attaining full power.

The Frequency of every 31 EFPD is adequate. It 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

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.13. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

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

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, 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 adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to agree with the incore measurements. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is ≥ 75% RTP and that 24 hours is allowed for performing the first surveillance after reaching 75% RTP.

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

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days. A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the calculated normal uncertainties consistent with the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis (Ref. 7) when applicable.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours, this Surveillance must be performed prior to 4 hours after entry into MODE 3.

The Frequency of 92 days is justified in Reference 7.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days of the Frequencies prior to reactor startup and four hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "4 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the unit remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT every 92 days, as justified in Reference 7.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the plant specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the calculated normal uncertainties consistent with the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP, and obtaining detector plateau curves, evaluating those curves, and comparing the curves to the manufacturer's data. The CHANNEL CALIBRATION for the source range, intermediate range, and power range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a COT of RTS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, and the SI Input from ESFAS. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip Mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of Turbine Trip Functions. This TADOT is performed prior to reactor startup. A Note states that this Surveillance is required if it has not been performed once within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the Turbine Trip Function is OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.15

SR 3.3.1.15 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in the UFSAR, Section 7.2 (Ref 9). Individual component response times are not modeled in the analyses.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state.

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate UFSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. Reference 8 provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

BASES

SURVEILLANCE REQUIREMENTS (continued)

As appropriate, each channel's response must be verified every 18 months on a STAGGERED TEST BASIS. Testing of the final actuation devices is included in the testing. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.15 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. UFSAR, Chapter 7.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. IEEE-279-1971
5. Technical Requirements Manual.
6. WCAP-12523, "RTS/ESFAS Setpoint Methodology Study," October 1990.
7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
8. WCAP-13632, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," August 1995.
9. UFSAR, Section 7.2.
10. WCAP-12583, "Westinghouse Setpoint Methodology For Protection Systems, Byron/Braidwood Stations," May 1990.
11. ComEd NES-EIC-20.04, Revision 0, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy," October 14, 1997.

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ATTACHMENT B-8

CLEAN COPY ITS BASES PAGES FOR BRAIDWOOD STATION, UNITS 1 AND 2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution, and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod Cluster Control Assemblies (RCCAs), or rods, are moved by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

BASES

BACKGROUND (continued)

The 53 RCCAs are divided among four control banks and five shutdown banks. A bank of RCCAs consists of either one group, or, two groups that are moved in a staggered fashion to provide for precise reactivity control but which are always within one step of each other. Each of the control banks are divided into two groups, for a total of 25 control bank rods. Shutdown banks A and B are also divided into two groups, however, shutdown banks C, D and E have only one group each, for a total of 28 shutdown bank rods. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the fully withdrawn position, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

BASES

BACKGROUND (continued)

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch) but not very reliable because it is a demanded position indication, not an actual position indication. For example, if a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue (i.e., statically misaligned RCCA). This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case with control bank D inserted to its full power insertion limit and one RCCA fully withdrawn. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA fully withdrawn (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local Linear Heat Rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e. trippability to meet SDM) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move rods (e.g. rod urgent failures), but do not impact trippability, do not result in rod inoperability provided proper alignment.

BASES

LCO (continued)

The requirement to maintain individual indicated rod positions within 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements for MODE 6.

BASES

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value specified in the COLR, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating boration to restore SDM to within limit.

In this situation, SDM verification must account for the worth of the untrippable rod(s), as well as the rod of maximum worth.

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the unit must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1.1 and B.1.2

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." One hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

BASES

ACTIONS (continued)

However, in many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be partially inserted.

With a misaligned rod, SDM must be verified to be within limit (specified in the COLR) or boration must be initiated to restore SDM to within limit.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour.

The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration to restore SDM to within limit.

B.2, B.3, B.4, and B.5

For continued operation with a misaligned rod, THERMAL POWER must be reduced when Power Distribution Monitoring System (PDMS) is inoperable. SDM must periodically be verified within limits (specified in the COLR), hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP when PDMS is inoperable, ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 4). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. One rod is not within alignment limit; and

b. PDMS is inoperable.

BASES

ACTIONS (continued)

Discovering one rod not within alignment limit coincident with PDMS inoperable results in starting the Completion Time for the Required Action. During power operation when PDMS is OPERABLE, LHR is measured continuously. Therefore, a reduction of power to 75% RTP is not necessary to ensure that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation, at ≤ 75% RTP with PDMS inoperable and > 75% RTP with PDMS OPERABLE, with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain the core power distribution using the incore flux mapping system or PDMS and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Accident for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

Accident analyses (Ref. 3) requiring re-evaluation for continued operation with a misaligned rod include:

1. Increase in heat removal by the secondary system:
   a. Excessive increase in secondary steam flow,
   b. Inadvertent opening of a steam generator power operated relief or safety valve, and
   c. Steam system piping failure;

BASES

ACTIONS (continued)

2. Uncontrolled RCCA bank withdrawal at power;
3. RCCA misoperation:
    a. One or more dropped RCCAs within the same group,
    b. A dropped RCCA bank,
    c. Statically misaligned RCCA, and
    d. Withdrawal of a single RCCA;
4. RCCA ejection accidents; and
5. Loss of coolant accidents resulting from postulated piping breaks within the reactor coolant pressure boundary.

C.1.1 and C.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM (specified in the COLR) must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

BASES

ACTIONS (continued)

C.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement when PDMS is inoperable. the unit conditions may fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. More than one rod is not within alignment limit; and

b. PDMS is inoperable.

Discovering more than one rod not within alignment limit coincident with PDMS inoperable results in starting the Completion Time for the Required Action.

C.3

If more than one rod is found to be misaligned or becomes misaligned because of bank movement when PDMS is OPERABLE. operation may continue in Condition C for a period that should not exceed 72 hours. The allowed Completion Time is reasonable, based on the available information on power distributions (Ref. 6). This Required Action is modified by a Note that requires the performance of Required Action C.3 only when PDMS is OPERABLE.

BASES

ACTIONS (continued)

D.1

When Required Actions of Condition B or C.3 cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. This frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable (e.g., as a result of excessive friction, mechanical interference, or rod control system failure), a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times once prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all Reactor Coolant Pumps (RCPs) operating and the average moderator temperature ≥ 550°F to ensure that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

This Surveillance is performed during a unit outage, due to conditions needed to perform the SR and the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. UFSAR, Section 15.4.3.
5. UFSAR, Section 15.1.5.
6. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

BASES

BACKGROUND (continued)

Rod Cluster Control Assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The 53 RCCAs are divided among 4 control banks and 5 shutdown banks. A bank of RCCAs consists of either one group, or, two groups that are moved in a staggered fashion to provide for precise reactivity control but which are always within one step of each other. Each of the control banks are divided into two groups, for a total of 25 control bank rods. Shutdown banks A and B are also divided into two groups, however, shutdown banks C, D, and E have only one group each, for a total of 28 shutdown bank rods. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch) but not very reliable because it is a demanded position indication, not an actual position indication. For example, if a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

BASES

BACKGROUND (continued)

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. The DRPI System determines the actual position of each control bank and shutdown bank rod by using individual coils that are mounted concentrically along the outside boundaries of the rod drive pressure housings. Each control bank rod has 42 coil assemblies evenly spaced along its length at 3.75 inch (6 step) intervals from rod bottom to the fully withdrawn position. Each shutdown bank rod has 20 coil assemblies evenly spaced along its length at 3.75 inch intervals from rod bottom to 18 steps and from 210 steps to the fully withdrawn position, with a transition LED representing shutdown bank rod position between 18 steps and the fully withdrawn position. The coils magnetically sense the presence or absence of a rod drive shaft and send this information to two Data Cabinets located in the containment building. To prevent total loss of position indication due to a single failure, the outputs of every other coil are connected as inputs to one Data Cabinet, while the outputs of the remaining coils are connected to the other Data Cabinet. This division of coils and their respective cabinets is referenced as Data A and Data B coils/cabinets, and allows detection of rod position within the required band of ± 12 steps even with a complete failure of a set of coils.

Normal system accuracy is ± 4 steps (± 3 steps with an additional step added for coil placement and thermal expansion). If a data error occurs, the system is shifted to the "half accuracy" mode. As a rod is moved under "half accuracy" conditions, only every other LED will light (i.e., the LEDs associated with the operable data system) since the effective coil spacing is 7.5 inches (12 steps). Under "half accuracy" conditions with data A bad, the system accuracy is + 10 steps, - 4 steps. Under "half accuracy" conditions with data B bad, the system accuracy is + 4 steps, - 10 steps. Therefore, the normal indication accuracy of the DRPI System is ± 4 steps, and the maximum uncertainty is 10 steps. With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 22 steps.

BASES

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking, ejected rod worth, and with minimum SDM limits (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that the DRPI System and the Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE the following requirements must be met:

a. The DRPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits;"

b. The DRPI System has no failed coils; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

BASES

LCO (continued)

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable DRPI and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

BASES

ACTIONS (continued)

A.1

When one DRPI channel per group fails, the position of the rod can still be determined by use of the movable incore detectors or Power Distribution Monitoring System (PDMS). When PDMS is OPERABLE, the position of the rod may be determined from the difference between the measured core power distribution and the core power distribution expected to exist based on the position of the rod indicated by the group step counter demand position. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position will not cause core peaking factors to approach the core peaking factor limits.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

These Required Actions clarify that when one or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If immediate actions have not been initiated to verify the rod's position, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps.

BASES

ACTIONS (continued)

C.1.1 and C.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the DRPIs for the affected banks are OPERABLE and the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. This verification can be an examination of logs, administrative controls, or other information that shows that all DRPIs in the affected bank are OPERABLE.

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position will not cause core peaking to approach the core peaking factor limits. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.

This surveillance is performed prior to reactor criticality after each removal of the reactor head, since there is potential for unnecessary plant transients if the SR were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.

2. UFSAR, Chapter 15.

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 (Z) of the core.

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

During power operation when Power Distribution Monitoring System (PDMS) is inoperable, 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 within power distribution limits on a continuous basis. During power operation when PDMS is OPERABLE, PLHR is measured continuously.

$F_Q(Z)$ 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 when PDMS is inoperable. These measurements are generally taken with the core at or near equilibrium conditions. When PDMS is OPERABLE, $F_Q(Z)$ is determined continuously.

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.

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

a. During a large break Loss Of Coolant Accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

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 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 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).

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).

BASES

LCO

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

$$F_Q(Z) \le \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

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

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

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

For this facility, the actual values of F_Q^{RTP} and $K(Z)$ are given in the COLR; however, F_Q^{RTP} is normally a number on the order of 2.50, and $K(Z)$ is a function that looks like the one provided in Figure B 3.2.1-1.

$F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

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

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

where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

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

BASES

LCO (continued)

When PDMS is OPERABLE. $F_Q(Z)$ is determined continuously. Then.

$$F_Q^C(Z) = F_Q^M(Z) * U_{FQ}$$

where U_{FQ} is a factor that accounts for measurement uncertainty (Ref. 4) and engineering uncertainty defined in the COLR.

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

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

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

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

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

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 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, A.2, and A.3

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ 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 unit to remain in an unacceptable condition for an extended period of time.

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.

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.

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 THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, maintains an acceptable absolute power density such that even if a transient occurred, core peaking factors are not exceeded

BASES

ACTIONS (continued)

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds the 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 period and the preceding prompt reduction in THERMAL POWER 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 $F_Q^W(Z)$ exceeds the 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 period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.

C.1

If the Required Actions of A.1 through A.3, or B.1 through B.3, are not met within their associated Completion Times, the unit must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the unit in at least MODE 2 within 6 hours. The 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note (i.e.. Note 1) that 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 meaningful power distribution map can be obtained. These SRs are normally performed at > 40% RTP to provide core conditions as much like the full power conditions as possible (Ref. 5). 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 (and establishing equilibrium conditions) 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 $F_Q(Z)$ is 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 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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. 6). $F_Q^C(Z)$ 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 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 analysis and because of the difficulty of making a precise measurement in these regions.

The Frequency of 31 Effective Full Power Days (EFPD) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the unit is operated in accordance with the Technical Specifications (TS).

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance has been modified by two Notes. Note 2 requires the measured value of $F_Q^C(Z)$ be obtained from incore flux map results only when PDMS is inoperable. Note 2 modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable, and that the last performance of SR 3.2.1.3 prior to declaring PDMS inoperable satisfies the initial performance of this SR after declaring PDMS inoperable. If SR 3.2.1.1 was not performed within its specified Frequency, this Note allows 12 hours to verify $F_Q^C(Z)$ is within limit using either the incore flux map results or by taking credit for the last performance of SR 3.2.1.3 when PDMS was OPERABLE.

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 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

This Surveillance has been modified by three Notes. Note 2 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 $F_Q^W(Z)$ 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

$$\text{maximum over z} \quad \left[\frac{F_Q^C(Z)}{K(Z)} \right],$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of the factor of 1.02 or by an appropriate factor specified in the COLR (Ref. 7), 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.

Note 3 requires the measured value of $F_Q^W(Z)$ be obtained from incore flux map results only when PDMS is inoperable. Note 3 modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable, and that the last performance of SR 3.2.1.4 prior to declaring PDMS inoperable satisfies the initial performance of this SR after declaring PDMS inoperable. If SR 3.2.1.2 were not performed within its specified Frequency, this Note allows 12 hours to verify $F_Q^W(Z)$ is within limit using either the incore flux map results or by taking credit for the last performance of SR 3.2.1.4 when PDMS was OPERABLE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 unit is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

SR 3.2.1.3

The confirmation of the power distribution parameter, $F_Q^C(Z)$, is an additional verification over the automated monitoring performed by PDMS. This assures that PDMS is functioning properly and that the core limits are met.

The Surveillance Frequency of 7 days takes into account other information and alarms available to the operator in the control room, and is adequate because $F_Q(Z)$ is monitored by the process computer.

This Surveillance is modified by a Note that requires the performance of SR 3.2.1.3 for determining $F_Q^C(Z)$ only when PDMS is OPERABLE.

SR 3.2.1.4

The confirmation of the power distribution parameter, $F_Q^W(Z)$, is an additional verification over the automated monitoring performed by PDMS. This assures that PDMS is functioning properly and that the core limits are met.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency of 7 days takes into account other information and alarms available to the operator in the control room, and is adequate because $F_Q(Z)$ is monitored by the process computer.

This Surveillance is modified by a Note that requires the performance of SR 3.2.1.4 for determining $F_Q^W(Z)$ only when PDMS is OPERABLE.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.
5. ANSI/ANS-19.6.1-1985, "Reload Startup Physics Test for Pressurized Water Reactors," December 13, 1985.
6. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
7. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification," February 1994.

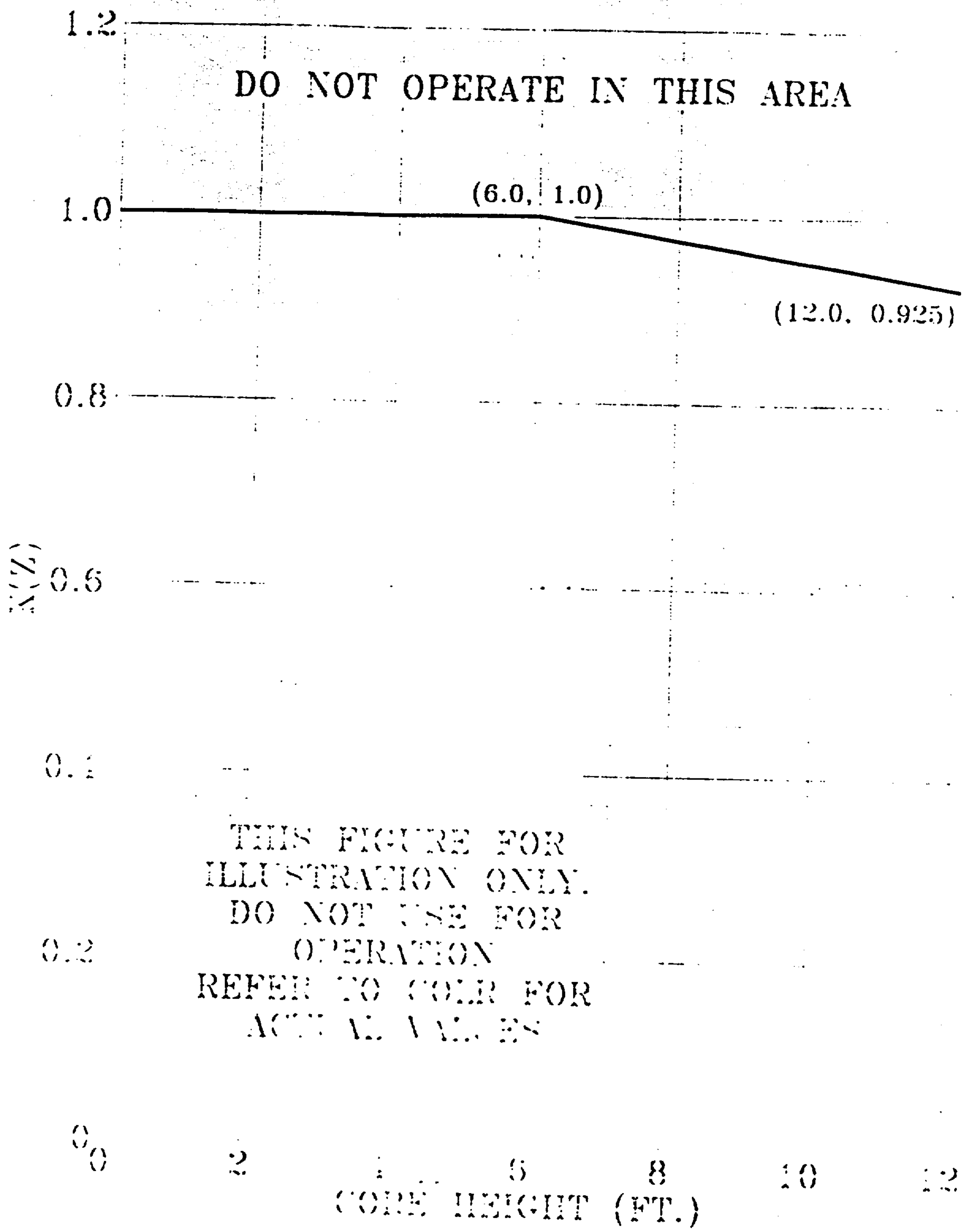

Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a function of Core Height

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

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

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

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

When Power Distribution Monitoring System (PDMS) is inoperable, $F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 Effective Full Power Days (EFPD). However, during power operation when PDMS is inoperable, 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 address directly and continuously measured process variables. During power operation when PDMS is OPERABLE, the linear power along the fuel rod with the highest integrated power is measured continuously and $F^N_{\Delta H}$ is determined continuously.

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design criterion for the Departure from Nucleate Boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

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

APPLICABLE SAFETY ANALYSES

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

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

b. During a large break Loss Of Coolant Accident (LOCA), Peak Cladding Temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design criterion is met for normal operation, operational transients, and any transients arising from events of moderate frequency. Refer to the Bases for LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits," for a discussion of the applicable Departure from Nucleate Boiling Ratio (DNBR) limits.

APPLICABLE SAFETY ANALYSES (continued)

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

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," LCO 3.2.2, LCO 3.2.3, LCO 3.2.4, and LCO 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)."

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

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

LCO

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

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

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the plant safety analyses.

The power multiplication factor in this equation provides margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

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

ACTIONS

A.1, A.2, A.3, and A.4

With $F^N_{\Delta H}$ exceeding its limit, Condition A is entered. $F^N_{\Delta H}$ may be restored to within its limits within 4 hours, through, for example, realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. If the value of $F^N_{\Delta H}$ is not restored to within its specified limit, THERMAL POWER must be reduced to ≤ 50% RTP in accordance with Required Action A.1. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses.

ACTIONS (continued)

However, the DNBR limit may be violated if a DNB limiting event occurs. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and is not likely to cause the DNBR limit to be violated in steady state operation. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the unit to remain in an unacceptable condition for an extended period of time.

Condition A is modified by a Note that requires that Required Actions A.2 and A.4 must be completed whenever Condition A is entered. Thus, even if $F^N_{\Delta H}$ is restored within the 4 hour time period of Required Action A.1, Required Action A.2 would nevertheless require another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1. Required Action A.4 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP.

Required Action A.2 requires the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level once the power level has been reduced to < 50% RTP per Required Action A.1. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by Action A.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map when PDMS is inoperable, perform the required calculations, and evaluate $F^N_{\Delta H}$.

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, Required Action A.3 requires the Power Range Neutron Flux-High trip setpoints be reduced to ≤ 55% RTP. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin.

BASES

ACTIONS (continued)

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.3 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

Required Action A.4 requires verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence. This ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced to comply with this Required Action.

B.1

If the Required Actions of A.1 through A.4 are not met within their associated Completion Times, the unit must be placed in a MODE in which the LCO requirements are not applicable. This is done by placing the unit in at least MODE 2 within 6 hours.

The 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 3.2.2.1

The value of $F_{\Delta H}^{N}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F_{\Delta H}^{N}$ from the measured flux distributions. The measured value of $F_{\Delta H}^{N}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F_{\Delta H}^{N}$ limit.

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

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

This Surveillance has been modified by a Note. The Note requires the measured value of $F_{\Delta H}^{N}$ be obtained from incore flux map results only when PDMS is inoperable. The Note modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable, and that the last performance of SR 3.2.2.2 prior to declaring PDMS inoperable satisfies the initial performance of this SR after declaring PDMS inoperable. If SR 3.2.2.1 were not performed within its specified Frequency, this Note allows 12 hours to verify $F_{\Delta H}^{N}$ is within limit using either the incore flux map results or by taking credit for the last performance of SR 3.2.2.2 when PDMS was OPERABLE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

The confirmation of the power distribution parameter. $F^N_{\Delta H}$. is an additional verification over the automated monitoring performed by PDMS. This assures that PDMS is functioning properly and that the core limits are met.

The Surveillance Frequency of 7 days takes into account other information and alarms available to the operator in the control room. and is adequate because $F^N_{\Delta H}$ is monitored by the process computer.

This Surveillance is modified by a Note that requires the performance of SR 3.2.2.2 for determining $F^N_{\Delta H}$ only when PDMS is OPERABLE.

REFERENCES

1. UFSAR. Section 15.4.8.
2. 10 CFR 50. Appendix A. GDC 26.
3. 10 CFR 50.46.

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 amount of axial power distribution skewing to either the top or bottom of the core when Power Distribution Monitoring System (PDMS) is inoperable. 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. When PDMS is OPERABLE, Peak Linear Heat Rate is measured continuously.

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 plant 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.

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) 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.

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 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 provide assurance that the thermal limits assumed in the accident analysis ($F^N_{\Delta H}$ and $F_Q(Z)$) are met. Thereby, the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

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.

A Note modifies the LCO by stating the AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

The AFD limits are provided in the COLR. 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 with THERMAL POWER ≥ 50% RTP (i.e., when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis) when PDMS is inoperable.

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.

BASES

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 channels is within limits. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer.

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 limits.

A Note modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable. If SR 3.2.3.1 were not performed within its specified Frequency, this Note allows 12 hours to verify AFD is within limits.

REFERENCES

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

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

3. UFSAR, Section 7.7.1.3.1.

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.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, 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. When Power Distribution Monitoring System (PDMS) is OPERABLE, Peak Linear Heat Rate and the linear power along the fuel rod with the highest integrated power are measured continuously.

APPLICABLE SAFETY ANALYSES

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

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

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 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition:

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 2): and

d. The control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin with the highest worth control rod stuck fully withdrawn (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion, 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 limits on the QPTR provide assurance that the thermal limits assumed in the accident analysis ($F^N_{\Delta H}$ and $F_Q(Z)$) are met. Thereby, the QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at 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 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP when PDMS is inoperable to prevent core power distributions from exceeding the design limits.

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.

BASES

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% from 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.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reductions within 2 hours of QPTR determination, 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.

A.2

After completion of Required Action A.1, periodic monitoring provides a basis for maintaining the appropriate reduced power level. As such, a check of the QPTR is required once per 12 hours. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly, such that it is maintained at a reduced power level of 3% from RTP for each 1% by which QPTR exceeds 1.00.

Any of the Surveillance methods for determining QPTR may be used within the constraints for acceptability of the Surveillance (i.e., if the excore detectors are available, they should be used; if the excore detectors are not available, the moveable incore detectors may be used). A 12 hour Completion Time is sufficient because any additional change in QPTR should be relatively slow. Further, this Completion Time is consistent with the Frequency required for the Surveillances with an inoperable alarm or instrumentation.

BASES

ACTIONS (continued)

A.3

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 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. The Completion Time takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the unit and perform a flux map. If these peaking factors are not within their 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 control 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.

BASES

ACTIONS (continued)

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limit prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the quadrant power tilt (QPT) is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (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 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 achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

BASES

ACTIONS (continued)

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (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 normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit 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.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

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 takes into account other information and alarms available to the operator in the control room.

This SR is modified by three Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1. Note 3 modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable. If SR 3.2.4.1 were not performed within its specified Frequency, this Note allows 12 hours to verify QPTR is within limits.

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

With input from an 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. The Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when input from one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of the radial core tilt to verify the QPTR can be used to confirm that QPTR is within limits.

With input from one NIS channel inoperable, the indicated tilt may be changed from the value indicated with input from all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is modified by two Notes. Note 1 states that it is not required to be performed until 12 hours after the input from one Power Range Neutron Flux channel is inoperable and the THERMAL POWER is > 75% RTP. Note 2 modifies the required performance of the Surveillance and states that this Surveillance is not required to be performed until 12 hours after declaring PDMS inoperable. If SR 3.2.4.2 were not performed within its specified Frequency, this Note allows 12 hours to verify QPTR is ≤ 1.02 using the movable incore detectors.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Departure from Nucleate Boiling Ratio (DNBR)

BASES

BACKGROUND

The purpose of the limits on the value of DNBR determined by Power Distribution Monitoring System (PDMS) is to provide assurance of fuel integrity during Condition I (Normal Operation and Operational Transients) and Condition II (Faults of Moderate Frequency) events by providing the reactor operator with the information required to avoid exceeding the minimum Axial Power Shape Limiting DNBR ($DNBR_{APSL}$) in the core during normal operation and in short-term transients.

DNBR is defined as the ratio of the heat flux required to cause Departure from Nucleate Boiling (DNB) to the actual channel heat flux for given conditions.

During power operation when PDMS is inoperable, 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 within power distribution limits on a continuous basis.

During power operation when PDMS is OPERABLE, DNBR is determined continuously. Continuously monitoring the operation of the core significantly limits the adverse nature of power distribution initial conditions for transients. The core depletion status, xenon distribution, and soluble boron concentration restrict the possible power and reactivity transients. Continuously monitoring the power distribution allows the actual DNBR value to be maintained ≥ the $DNBR_{APSL}$ value specified in the COLR. $DNBR_{APSL}$ is the DNBR value determined to be the most sensitive to the core axial power distribution at the initial conditions of the limiting accident during the cycle-specific core reload design accident analysis process.

BASES

APPLICABLE SAFETY ANALYSES

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

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

The DNB safety analysis limit for a loss of forced reactor coolant flow accident (Ref. 1) is met by limiting DNBR to the 95/95 DNB design criterion of 1.4 using the WRB-2 Critical Heat Flux (CHF) correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience DNB. Maintaining the $DNBR_{APSL}$ value ≥ the DNBR value assumed in the safety and accident analyses ensures that the 95/95 DNB design criterion of 1.4 is met.

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. When PDMS is OPERABLE, this LCO and the following LCOs ensure this: LCO 3.1.6, LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," and LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$). When PDMS is inoperable, the following LCOs ensure this: LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4.

DNBR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

DNBR shall be maintained within the limit of the relationship specified in the COLR.

Maintaining DNBR ≥ $DNBR_{APSL}$ ensures the core operates within the limits assumed in the safety analyses. The $DNBR_{APSL}$ limit must be maintained to prevent core power distributions from exceeding the fuel design limits for DNBR.

Another limit on DNBR is provided in SL 2.1.1, "Reactor Core SLs." LCO 3.2.5 represents the initial conditions of the safety analysis which are far more restrictive than the Safety Limit (SL). Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

BASES

APPLICABILITY

The DNBR limit must be maintained in MODE 1 with THERMAL POWER ≥ 50% RTP when PDMS is OPERABLE to ensure DNB design criteria will be met in the event of an unplanned loss of forced coolant flow transient.

ACTIONS

A.1

Parameters affecting DNBR include Reactor Coolant System (RCS) pressure, RCS average temperature, RCS total flow rate, and Thermal Power. RCS pressure and RCS average temperature are controllable and measurable parameters. RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. With DNBR not within limit due to RCS pressure or RCS average temperature, action must be taken to restore these parameter(s). With DNBR not within limit due to the indicated RCS total flow rate, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of DNBR provides sufficient time to adjust unit parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If the value of DNBR is not restored to within its specified limit, THERMAL POWER must be reduced to < 50% RTP in accordance with Required Action B.1. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and is not likely to cause the DNBR limit to be violated in steady state operation. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore DNBR to within its limits without allowing the unit to remain in an unacceptable condition for an extended period of time.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The confirmation of the power distribution parameter, DNBR, is an additional verification over the automated monitoring performed by PDMS. This assures that PDMS is functioning properly and that the core limits are met.

The Surveillance Frequency of 7 days takes into account other information and alarms available to the operator in the control room, and is adequate because DNBR is monitored by the process computer.

REFERENCES

1. UFSAR, Chapter 15.

BASES

ACTIONS (continued)

A Note to the ACTIONS restricts the transition from MODE 5 with the Rod Control System not capable of rod withdrawal and all rods fully inserted, to MODE 5 with the Rod Control System capable of rod withdrawal or all rods not fully inserted for Functions 18, 19, and 20 while complying with the ACTIONS (i.e., while the LCO is not met). LCO 3.0.4 typically allows entry into MODES or other specified conditions in the Applicability while in MODE 5, however, the restriction of this Note is necessary to assure an OPERABLE RTS function prior to commencing operation with the Rod Control System capable of rod withdrawal or all rods not fully inserted.

D.1 and D.2

Condition D applies to the Power Range Neutron Flux-High Function.

The NIS power range detectors provide input to the Rod Control System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 7).

As an alternative to the above Action, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 7.

BASES

ACTIONS (continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux – Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Power Range Neutron Flux – High Positive Rate;
- Power Range Neutron Flux – High Negative Rate;
- Pressurizer Pressure – High; and
- SG Water Level – Low Low.

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

BASES

ACTIONS (continued)

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 2 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The provisions of LCO 3.0.4 allow entry into a MODE or other specified condition in the Applicability as directed by the Required Actions. Therefore, a MODE change is permitted with one channel inoperable whenever Required Action F.2 is used. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

BASES

ACTIONS (continued)

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint. the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE. the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

H.1

Condition H applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

BASES

ACTIONS (continued)

I.1

Condition I applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition.

J.1 and J.2

Condition J applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour, are justified in Reference 7.

BASES

ACTIONS (continued)

K.1 and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low;
- RCP Breaker Position;
- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint. These Functions do not have to be OPERABLE below the P-7 setpoint. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

BASES

ACTIONS (continued)

L.1 and L.2

Condition L applies to Turbine Trip on Emergency Trip Header Pressure or on Turbine Throttle Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-8 setpoint within the next 6 hours. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

M.1 and M.2

Condition M applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action M.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours (Required Action M.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action M.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

BASES

ACTIONS (continued)

N.1 and N.2

Condition N applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 results in Action C entry while RTB(s) are inoperable.

The Required Actions have been modified by two Notes. Note 1 allows one channel to be bypassed for up to 2 hours for surveillance testing, provided the other channel is OPERABLE. Note 2 allows one RTB to be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE. The 2 hour time limit is justified in Reference 7.

O.1 and O.2

Condition O applies to the P-6 and P-10 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition by observation of the associated permissive annunciator window within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

BASES

ACTIONS (continued)

P.1 and P.2

Condition P applies to the P-7, P-8, and P-13 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition by observation of the associated permissive annunciator window within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging plant systems.

Q.1 and Q.2

Condition Q applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

With the unit in MODE 3, Action C would apply to any inoperable RTB trip mechanism. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 2 hours for the reasons stated under Condition N.

BASES

ACTIONS (continued)

The Completion Time of 48 hours for Required Action Q.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It 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. . .

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output prior to exceeding 75% RTP after each refueling and every 31 Effective Full Power days (EFPD) thereafter. 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 declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%. Note 2 clarifies that the Surveillance is required only if reactor power is > 15% RTP.

The Frequency of once prior to exceeding 75% RTP following each refueling outage considers that the core may be changed during a refueling outage such that the previous comparison, prior to the refueling outage, is no longer completely valid. The Frequency also considers that the comparison accuracy increases with power level such that the comparison is preferred to be performed at as high a power level as possible. An initial performance at ≤ 75% RTP provides a verification prior to attaining full power.

The Frequency of every 31 EFPD is adequate. It 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

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.13. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

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

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, 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 adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to agree with the incore measurements. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is ≥ 75% RTP and that 24 hours is allowed for performing the first surveillance after reaching 75% RTP.

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

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days. A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the calculated normal uncertainties consistent with the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis (Ref. 7) when applicable.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours, this Surveillance must be performed prior to 4 hours after entry into MODE 3.

The Frequency of 92 days is justified in Reference 7.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days of the Frequencies prior to reactor startup and four hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "4 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the unit remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT every 92 days, as justified in Reference 7.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the plant specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the calculated normal uncertainties consistent with the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP, and obtaining detector plateau curves, evaluating those curves, and comparing the curves to the manufacturer's data. The CHANNEL CALIBRATION for the source range, intermediate range, and power range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a COT of RTS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, and the SI Input from ESFAS. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip Mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of Turbine Trip Functions. This TADOT is performed prior to reactor startup. A Note states that this Surveillance is required if it has not been performed once within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the Turbine Trip Function is OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.15

SR 3.3.1.15 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in the UFSAR, Section 7.2 (Ref. 9). Individual component response times are not modeled in the analyses.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state.

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate UFSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. Reference 8 provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

BASES

SURVEILLANCE REQUIREMENTS (continued)

As appropriate, each channel's response must be verified every 18 months on a STAGGERED TEST BASIS. Testing of the final actuation devices is included in the testing. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.15 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. UFSAR, Chapter 7.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. IEEE-279-1971.
5. Technical Requirements Manual.
6. WCAP-12523, "RTS/ESFAS Setpoint Methodology Study," October 1990.
7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
8. WCAP-13632, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," August 1995.
9. UFSAR, Section 7.2.
10. WCAP-12583, "Westinghouse Setpoint Methodology For Protection Systems, Byron/Braidwood Stations," May 1990.
11. ComEd NES-EIC-20.04, Revision 0, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy," October 14, 1997.

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AH B-918-10
COLRs

ATTACHMENT B-9

COLR
FOR BYRON STATION, UNITS 1 AND 2

CORE OPERATING LIMITS REPORT (COLR) for BYRON UNIT X CYCLE X

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Byron Station Unit x Cycle x has been prepared in accordance with the requirements of Technical Specification 5.6.5 (ITS).

The Technical Specifications affected by this report are listed below:

LCO 3.1.1 Shutdown Margin (SDM)

LCO 3.1.3 Moderator Temperature Coefficient

LCO 3.1.4 Rod Group Alignment Limits

LCO 3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.6 Control Bank Insertion Limits

LCO 3.1.8 Physics Tests Exceptions – Mode 2

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 3.2.5 Departure of Nucleate Boiling Ratio (DNBR)

LCO 3.3.9 Boron Dilution Protection System (BDPS)

LCO 3.9.1 Boron Concentration

The portions of the Technical Requirements Manual affected by this report are listed below:

TRM TLCO 3.1.b Boration Flow Paths – Operating

TRM TLCO 3.1.d Charging Pumps – Operating

TRM TLCO 3.1.f Borated Water Sources – Operating

TRM TLCO 3.1.h Shutdown Margin (SDM) – MODE 1 and MODE 2 with keff ≥ 1.0

TRM TLCO 3.1.i Shutdown Margin (SDM) – MODE 5

TRM TLCO 3.1.j Shutdown and Control Rods

TRM TLCO 3.1.k Position Indication System – Shutdown (Special Test Exception)

TRM TLCO 3.3.h Power Distribution Monitoring System (PDMS) Instrumentation

CORE OPERATING LIMITS REPORT (COLR) for BYRON UNIT X CYCLE X

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits are applicable for the entire cycle unless otherwise identified. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 5.6.5.

2.1 Shutdown Margin (SDM)

The SDM limit for MODES 1, 2, 3, and 4 is:

2.1.1 The SDM shall be greater than or equal to 1.3% Δk/k (LCOs 3.1.1, 3.1.4, 3.1.5, 3.1.6, 3.1.8, and 3.3.9; TRM TLCOs 3.1.b, 3.1.d, 3.1.f, 3.1.h, and 3.1.j).

The SDM limits for MODE 5 are:

2.1.2.1 SDM shall be greater than or equal to 1.0% Δk/k (LCO 3.1.1)

2.1.2.2 SDM shall be greater than or equal to 1.3% Δk/k (LCO 3.3.9; TRM TLCO 3.1.i and 3.1.j)

2.2 Moderator Temperature Coefficient (LCO 3.1.3)

The Moderator Temperature Coefficient (MTC) limits are:

2.2.1 The BOL/ARO/HZP-MTC shall be less positive than $+3.6 \times 10^{-5}$ Δk/k/ F.

2.2.2 The EOL/ARO/HFP-MTC shall be less negative than -4.1×10^{-4} Δk/k/ F.

2.2.3 The EOL/ARO/HFP-MTC Surveillance limit at 300 ppm shall be less negative than or equal to -3.2×10^{-4} Δk/k/ F.

where: BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
HZP stands for Hot Zero Thermal Power
EOL stands for End of Cycle Life
HFP stands for Hot Full Thermal Power

2.3 Shutdown Bank Insertion Limit (LCO 3 1.5)

2.3.1 All shutdown banks shall be withdrawn to at least 228 steps.

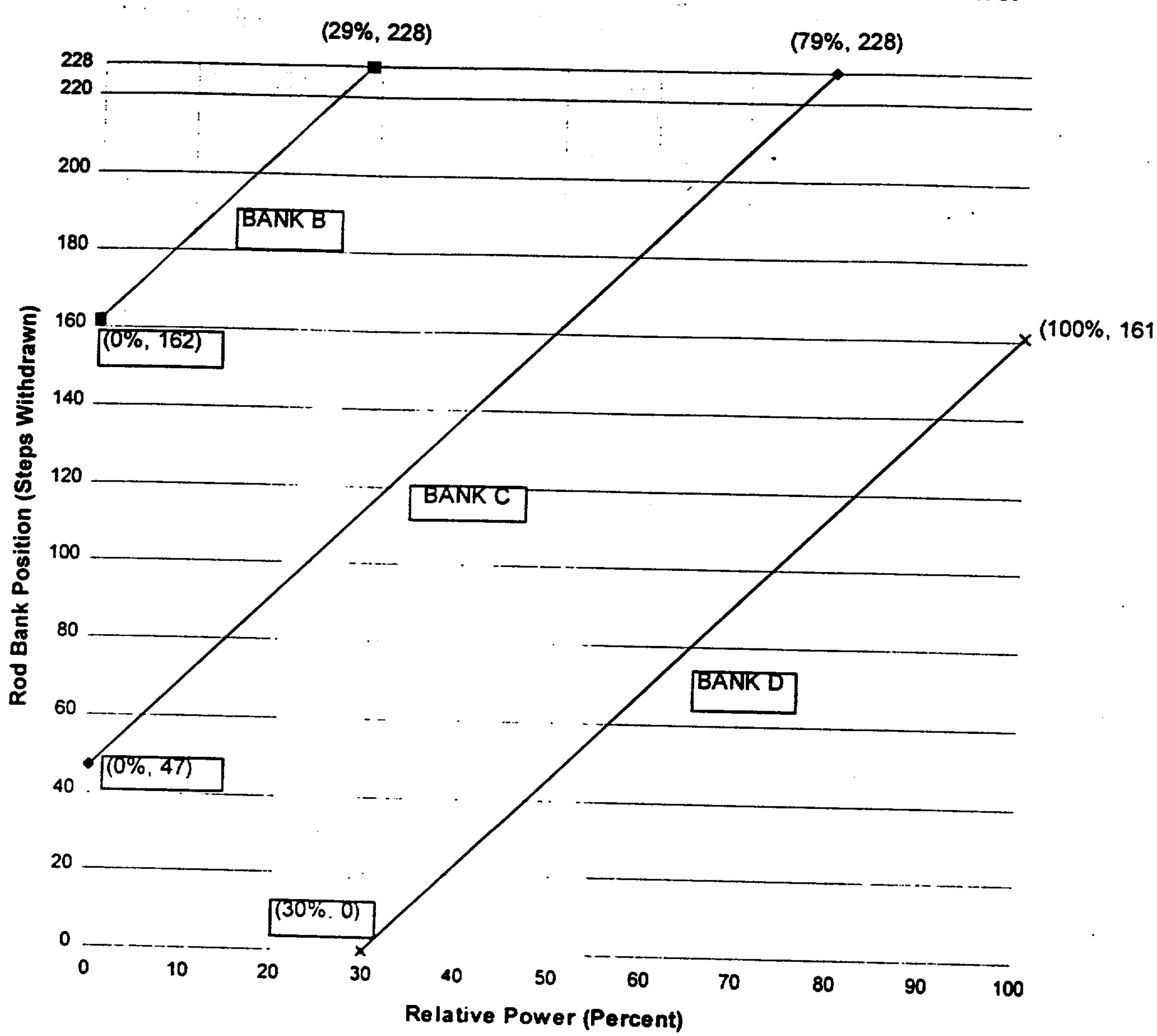
2.4 Control Bank Insertion Limits (LCO 3.1.6)

2.4.1 The control banks shall be limited in physical insertion as shown in Figure 2.4.1.

2.4.2 The control banks shall be operated in sequence by withdrawal of Bank A, Bank B, Bank C and Bank D. The control banks shall be sequenced in reverse order upon insertion.

2.4.3 The control banks shall be operated with a 115 step overlap.

Figure 2.4.1:
Control Bank Insertion Limits Versus Percent Rated Thermal Power



2.5 <u>Heat Flux hot Channel Factor $F_Q(Z)$</u> (LCO 3.2.1)

2.5.1

$$Fq(Z) \le \frac{Fq^{RTP}}{0.5} xK(Z) \text{ for } P \le 0.5$$

$$Fq(Z) \le \frac{Fq^{RTP}}{P} xK(Z) \text{ for } P > 0.5$$

where: P = the ratio of THERMAL POWER to RATED THERMAL POWER

F_q^{RTP} = 2.60

K(Z) is provided in Figure 2.5.1.

2.5.2 Uncertainty when PDMS is inoperable:

The uncertainty, U_{FQ}, to be applied to the Heat Flux Hot Channel Factor $F_Q(Z)$ shall be calculated by the following formula

$$U_{FQ} = U_{qu} \bullet U_e$$

where:

U_{qu} = Base FQ measurement uncertainty = 1.05

U_e = Engineering uncertainty factor = 1.03

2.5.3 W(Z) Values:

a) When PDMS is OPERABLE, W(Z) = 1.00000 for all axial points.

b) When PDMS is inoperable, W(Z) Values are provided in Figures 2.5.3.a through 2.5.3.c. The normal operation W(Z) values have been determined at burnups of 150, 8000 and 18800 MWD/MTU.

Table 2.5.3 shows the $F^C_Q(Z)$ penalty factors that are greater than 2% per 31 Effective Full Power Days (EFPD) These values shall be used to increase the $F^W_Q(Z)$ as per Surveillance Requirement 3.2.1.2. A 2% penalty factor shall be used at all cycle burnups that are outside the range of Table 2.5.3.

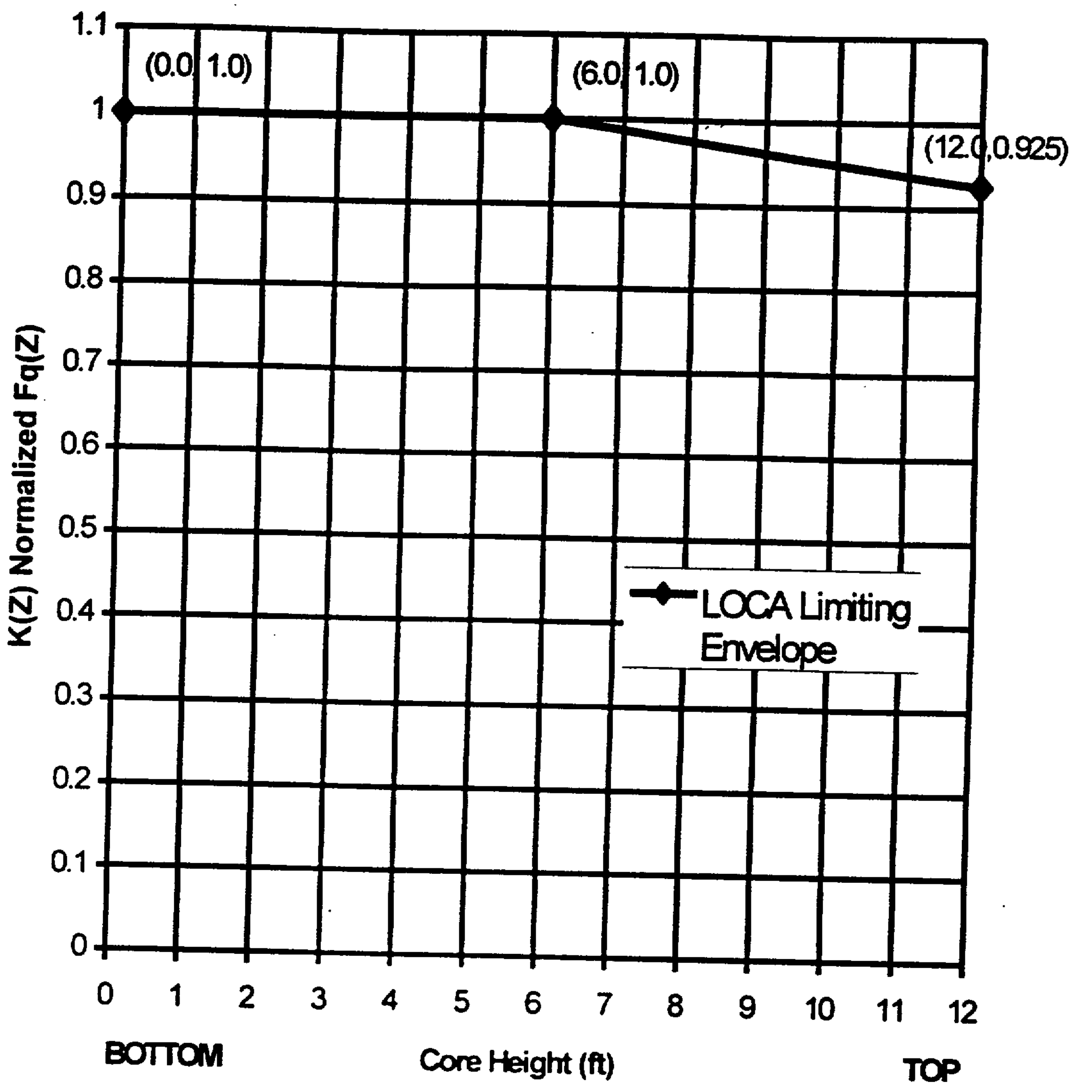
Multiplication Factor = 1.02

2.5.4 PDMS Alarms:

$F_Q(Z)$ Warning Setpoint = 2%

$F_Q(Z)$ Alarm Setpoint = 0%

Figure 2.5.1: K(Z) - Normalized Fq(Z) as a Function of Core Height



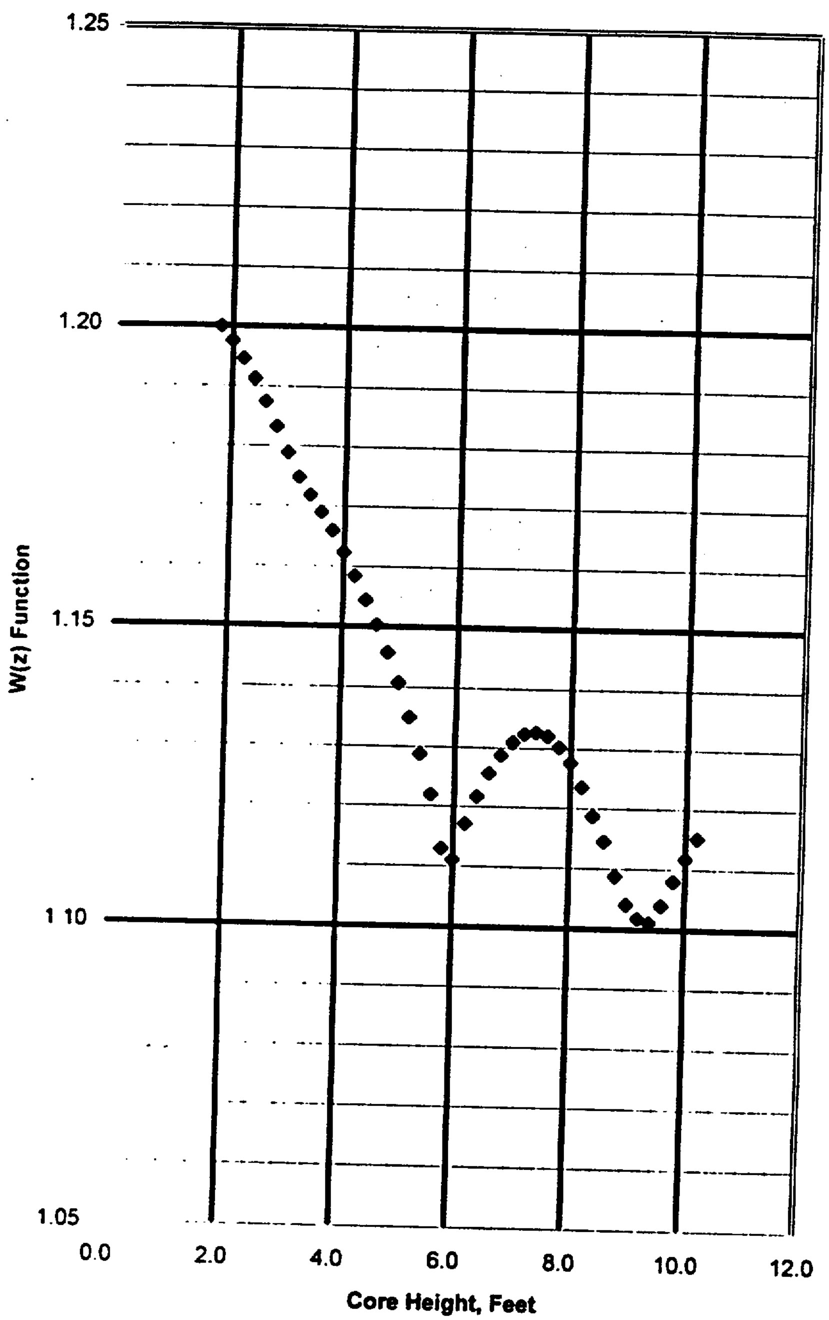
CORE OPERATING LIMITS REPORT (COLR) for BYRON UNIT X CYCLE X

Height Feet	Max W(z)
0.0	1.0000
0.2	1.0000
0.4	1.0000
0.6	1.0000
0.8	1.0000
1.0	1.0000
1.2	1.0000
1.4	1.0000
1.6	1.0000
1.8	1.1999
2.0	1.1975
2.2	1.1946
2.4	1.1912
2.6	1.1874
2.8	1.1832
3.0	1.1788
3.2	1.1747
3.4	1.1717
3.6	1.1689
3.8	1.1659
4.0	1.1623
4.2	1.1584
4.4	1.1544
4.6	1.1503
4.8	1.1457
5.0	1.1407
5.2	1.1349
5.4	1.1287
5.6	1.1219
5.8	1.1129
6.0	1.1110
6.2	1.1171
6.4	1.1216
6.6	1.1256
6.8	1.1287
7.0	1.1309
7.2	1.1323
7.4	1.1326
7.6	1.1320
7.8	1.1302
8.0	1.1276
8.2	1.1235
8.4	1.1187
8.6	1.1145
8.8	1.1087
9.0	1.1041
9.2	1.1018
9.4	1.1011
9.6	1.1040
9.8	1.1080
10.0	1.1117
10.2	1.1151
10.4	1.0000
10.6	1.0000
10.8	1.0000
11.0	1.0000
11.2	1.0000
11.4	1.0000
11.6	1.0000
11.8	1.0000
12.0	1.0000

Byron Unit X Cycle X

Figure 2.5.3.a

Summary of W(z) Function at 150 MWD/MTU
(Top and Bottom 15% Excluded per WCAP-10216)



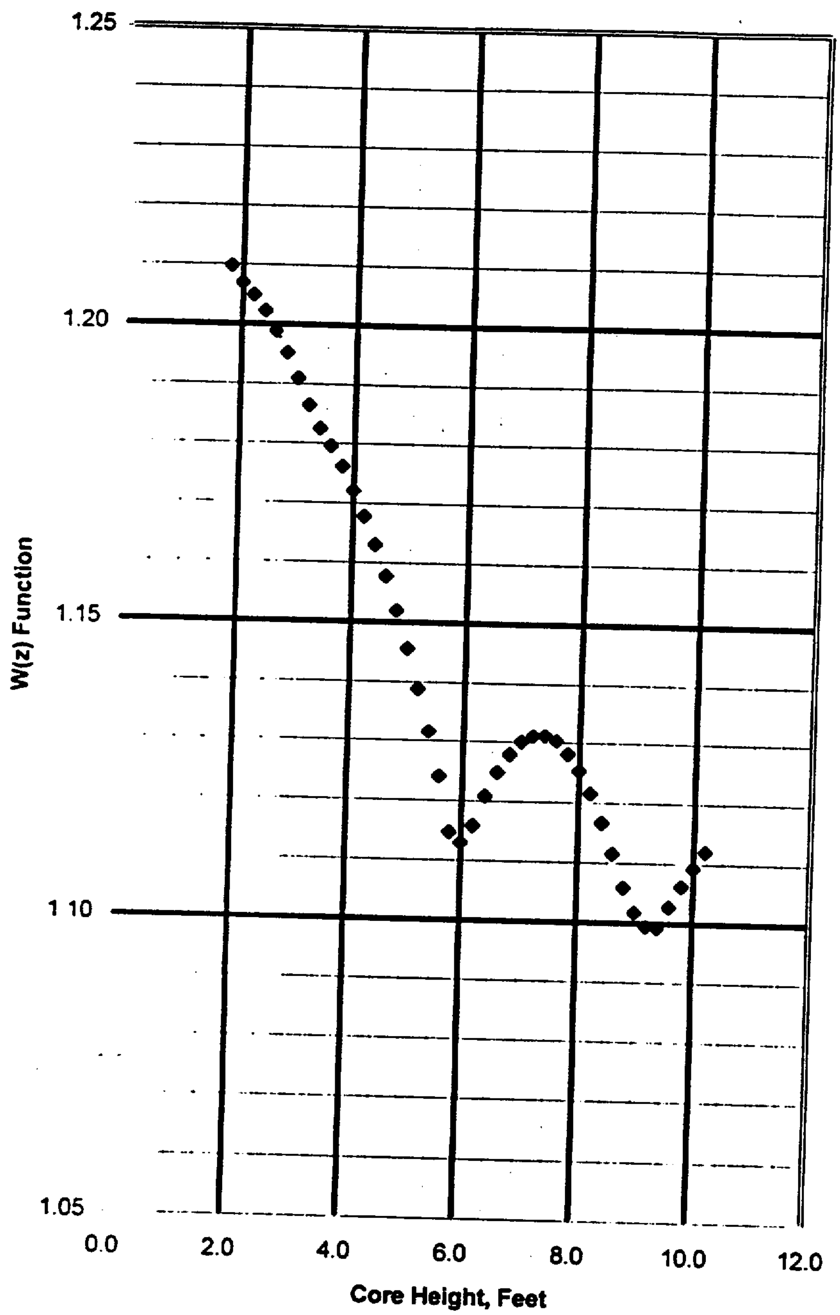
CORE OPERATING LIMITS REPORT (COLR) for BYRON UNIT X CYCLE X

Height Feet	Max W(z)
0.0	1.0000
0.2	1.0000
0.4	1.0000
0.6	1.0000
0.8	1.0000
1.0	1.0000
1.2	1.0000
1.4	1.0000
1.6	1.0000
1.8	1.2097
2.0	1.2069
2.2	1.2049
2.4	1.2022
2.6	1.1989
2.8	1.1952
3.0	1.1909
3.2	1.1864
3.4	1.1825
3.6	1.1795
3.8	1.1761
4.0	1.1720
4.2	1.1677
4.4	1.1630
4.6	1.1577
4.8	1.1519
5.0	1.1456
5.2	1.1388
5.4	1.1316
5.6	1.1240
5.8	1.1147
6.0	1.1129
6.2	1.1158
6.4	1.1208
6.6	1.1248
6.8	1.1279
7.0	1.1300
7.2	1.1311
7.4	1.1312
7.6	1.1303
7.8	1.1281
8.0	1.1253
8.2	1.1215
8.4	1.1166
8.6	1.1115
8.8	1.1059
9.0	1.1017
9.2	1.0994
9.4	1.0992
9.6	1.1027
9.8	1.1062
10.0	1.1092
10.2	1.1120
10.4	1.0000
10.6	1.0000
10.8	1.0000
11.0	1.0000
11.2	1.0000
11.4	1.0000
11.6	1.0000
11.8	1.0000
12.0	1.0000

Byron Unit X Cycle X

Figure 2.5.3.b

Summary of W(z) Function at 8000 MWD/MTU
(Top and Bottom 15% Excluded per WCAP-10216)



CORE OPERATING LIMITS REPORT (COLR) for BYRON UNIT X CYCLE X

Byron Unit X Cycle X

Figure 2.5.3.c

Summary of W(z) Function at 18800 MWD/MTU
(Top and Bottom 15% Excluded per WCAP-10216)

Height Feet	Max W(z)
0.0	1.0000
0.2	1.0000
0.4	1.0000
0.6	1.0000
0.8	1.0000
1.0	1.0000
1.2	1.0000
1.4	1.0000
1.6	1.0000
1.8	1.2188
2.0	1.2148
2.2	1.2099
2.4	1.2044
2.6	1.1982
2.8	1.1914
3.0	1.1846
3.2	1.1773
3.4	1.1705
3.6	1.1670
3.8	1.1642
4.0	1.1611
4.2	1.1577
4.4	1.1538
4.6	1.1495
4.8	1.1449
5.0	1.1396
5.2	1.1341
5.4	1.1276
5.6	1.1203
5.8	1.1210
6.0	1.1267
6.2	1.1319
6.4	1.1363
6.6	1.1397
6.8	1.1421
7.0	1.1435
7.2	1.1438
7.4	1.1430
7.6	1.1410
7.8	1.1386
8.0	1.1352
8.2	1.1304
8.4	1.1245
8.6	1.1178
8.8	1.1099
9.0	1.1034
9.2	1.1011
9.4	1.1015
9.6	1.1053
9.8	1.1088
10.0	1.1120
10.2	1.1149
10.4	1.0000
10.6	1.0000
10.8	1.0000
11.0	1.0000
11.2	1.0000
11.4	1.0000
11.6	1.0000
11.8	1.0000
12.0	1.0000

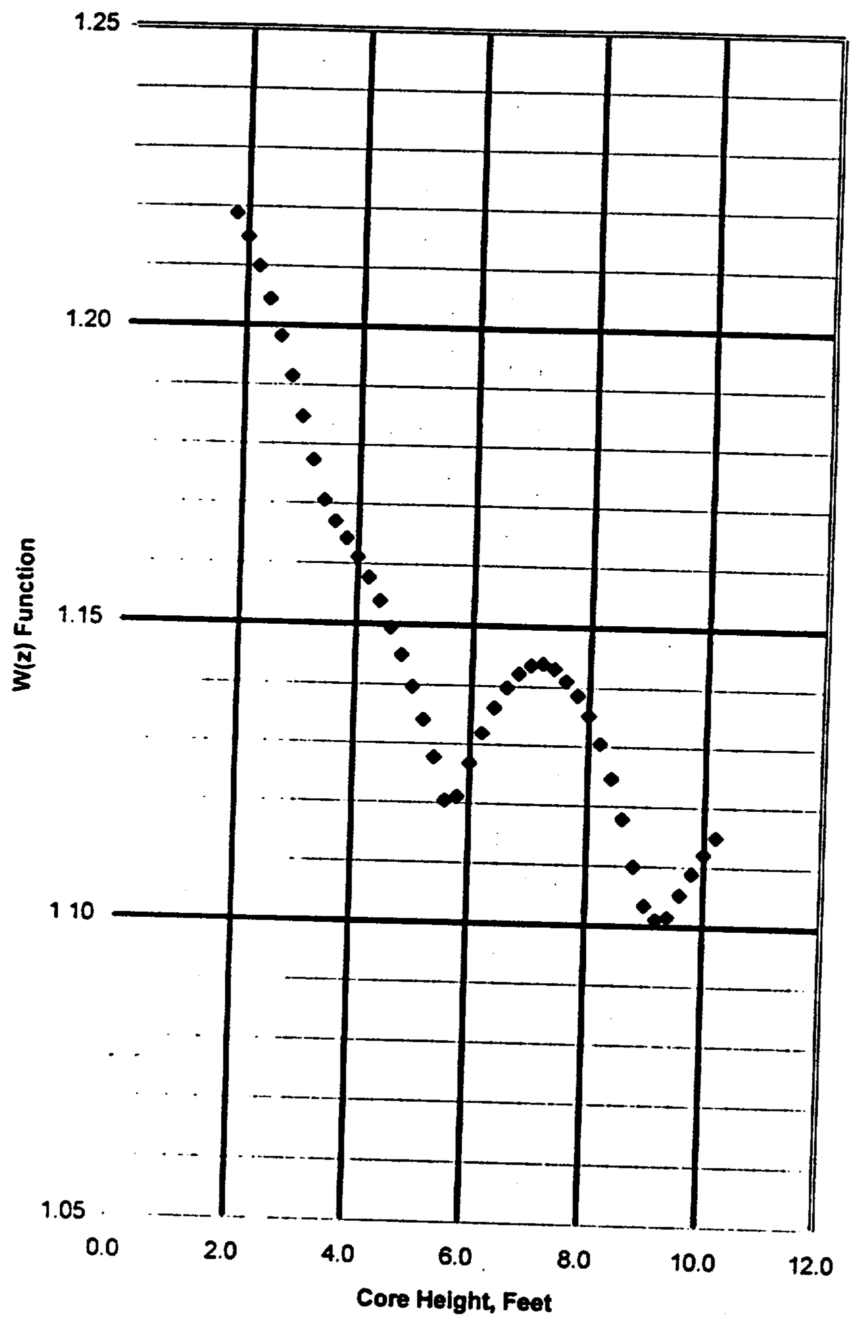


Table 2.5.3

Fq Margin Decreases in Excess of 2% per 31 EFPD

Cycle Burnup (MWD/MTU)	Max % Decrease in Fq Margin
150	3.62
800	4.21
1200	2.74
1800	2.21
2000	2.00

Note: All cycle burnups outside the range of the table shall use a 2% decrease in Fq margin for compliance with the 3.2.1.2 Surveillance Requirements.

2.6 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) (LCO 3.2.2)

2.6.1 $F^N_{\Delta H} \leq F^{RTP}_{\Delta H}[1.0 + PF_{\Delta H}(1.0 - P)]$

where: P = the ratio of THERMAL POWER to RATED THERMAL POWER
$F^{RTP}_{\Delta H} = 1.70$
$PF_{\Delta H} = 0.3$

2.6.2 Uncertainty when PDMS is inoperable:

The uncertainty, $U_{F\Delta H}$, to be applied to the Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}{}^N$ shall be calculated by the following formula

$U_{F\Delta H} = U_{F\Delta Hm}$

where:

$U_{F\Delta Hm}$ = Base FDH measurement uncertainty = 1.04

2.6.3 PDMS Alarms:

$F^N_{\Delta H}$ Warning Setpoint = 2%

$F^N_{\Delta H}$ Alarm Setpoint = 0%

2.7 Axial Flux Difference (AFD) (LCO 3.2.3)

2.7.1 When PDMS is OPERABLE, no AFD Acceptable Operation Limits are applicable.

2.7.2 When PDMS is inoperable, the AFD Acceptable Operation Limits are provided in Figure 2.7.2 or the latest valid PDMS Surveillance Report, whichever is more conservative.

2.8 Departure from Nucleate Boiling Ratio (DNBR) (LCO 3.2.5)

2.8.1 $DNBR_{APSL} = 1.4$

The $DNBR_{APSL}$ limit is applicable with THERMAL POWER ≥ 50% RTP when PDMS is OPERABLE.

2.8.2 PDMS Alarms:

DNBR Warning Setpoint = 2%

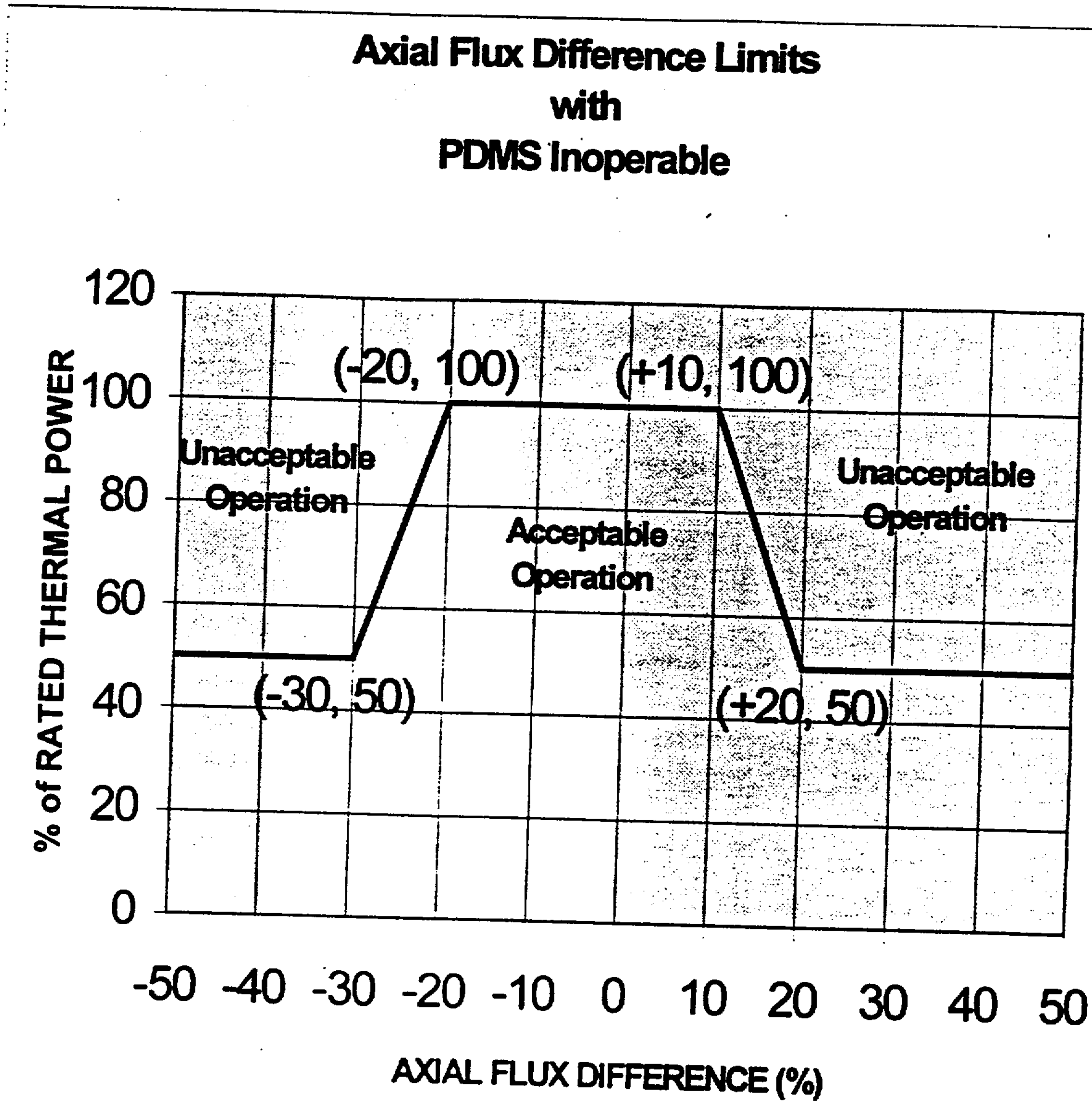
DNBR Alarm Setpoint = 0%

2.9 Boron Concentration (LCO 3.9.1)

2.9.1 The refueling boron concentration shall be greater than or equal to 2000 ppm.

2.9.2 The Reactor Coolant System boron concentration shall be greater than or equal to 1919 ppm to maintain adequate shutdown margin for Rod Drop Time Measurements. (TLCO 3.1.k)

Figure 2.7.2 Axial Flux Difference Limits as a Function of Rated Thermal Power



ATTACHMENT B-10

COLR
FOR BRAIDWOOD STATION, UNITS 1 AND 2

CORE OPERATING LIMITS REPORT (COLR) for BRAIDWOOD UNIT X CYCLE X

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Braidwood Station Unit x Cycle x has been prepared in accordance with the requirements of Technical Specification 5.6.5 (ITS).

The Technical Specifications affected by this report are listed below:

LCO 3.1.1 Shutdown Margin (SDM)

LCO 3.1.3 Moderator Temperature Coefficient

LCO 3.1.4 Rod Group Alignment Limits

LCO 3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.6 Control Bank Insertion Limits

LCO 3.1.8 Physics Tests Exceptions – Mode 2

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 3.2.5 Departure of Nucleate Boiling Ratio (DNBR)

LCO 3.3.9 Boron Dilution Protection System (BDPS)

LCO 3.9.1 Boron Concentration

The portions of the Technical Requirements Manual affected by this report are listed below:

TRM TLCO 3.1.b Boration Flow Paths – Operating

TRM TLCO 3.1.d Charging Pumps – Operating

TRM TLCO 3.1.f Borated Water Sources – Operating

TRM TLCO 3.1.h Shutdown Margin (SDM) – MODE 1 and MODE 2 with keff ≥ 1.0

TRM TLCO 3.1.i Shutdown Margin (SDM) – MODE 5

TRM TLCO 3.1.j Shutdown and Control Rods

TRM TLCO 3.1.k Position Indication System – Shutdown (Special Test Exception)

TRM TLCO 3.3.h Power Distribution Monitoring System (PDMS) Instrumentation

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits are applicable for the entire cycle unless otherwise identified. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 5.6.5.

2.1 Shutdown Margin (SDM)

The SDM limit for MODES 1, 2, 3, and 4 is:

2.1.1 The SDM shall be greater than or equal to 1.3% $\Delta k/k$ (LCOs 3.1.1, 3.1.4, 3.1.5, 3.1.6, 3.1.8, and 3.3.9; TRM TLCOs 3.1.b, 3.1.d, 3.1.f, 3.1.h, and 3.1.j).

The SDM limits for MODE 5 are:

2.1.2.1 SDM shall be greater than or equal to 1.0% $\Delta k/k$ (LCO 3.1.1)

2.1.2.2 SDM shall be greater than or equal to 1.3% $\Delta k/k$ (LCO 3.3.9; TRM TLCO 3.1.i and 3.1.j)

2.2 Moderator Temperature Coefficient (LCO 3.1.3)

The Moderator Temperature Coefficient (MTC) limits are:

2.2.1 The BOL/ARO/HZP-MTC shall be less positive than $+3.6 \times 10^{-5}$ $\Delta k/k/$ F.

2.2.2 The EOL/ARO/HFP-MTC shall be less negative than -4.1×10^{-4} $\Delta k/k/$ F.

2.2.3 The EOL/ARO/HFP-MTC Surveillance limit at 300 ppm shall be less negative than or equal to -3.2×10^{-4} $\Delta k/k/$ F.

where: BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
HZP stands for Hot Zero Thermal Power
EOL stands for End of Cycle Life
HFP stands for Hot Full Thermal Power

2.3 Shutdown Bank Insertion Limit (LCO 3 1 5)

2.3.1 All shutdown banks shall be withdrawn to at least 228 steps.

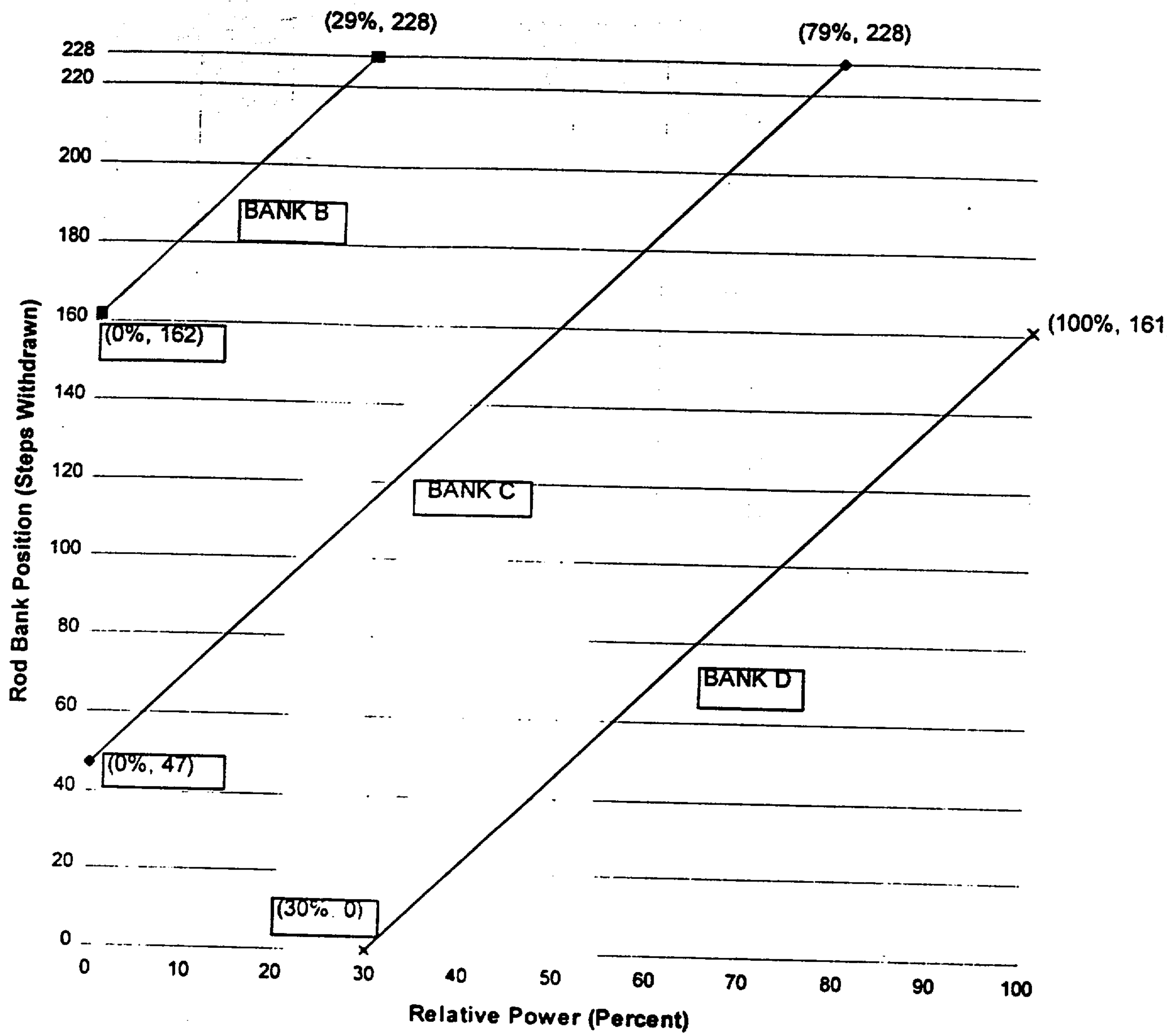
2.4 Control Bank Insertion Limits (LCO 3 1 6)

2.4.1 The control banks shall be limited in physical insertion as shown in Figure 2.4.1.

2.4.2 The control banks shall be operated in sequence by withdrawal of Bank A, Bank B, Bank C and Bank D. The control banks shall be sequenced in reverse order upon insertion.

2.4.3 The control banks shall be operated with a 115 step overlap.

Figure 2.4.1:
Control Bank Insertion Limits Versus Percent Rated Thermal Power



2.5 Heat Flux hot Channel Factor $F_Q(Z)$ (LCO 3.2.1)

2.5.1

$$Fq(Z) \leq \frac{Fq^{RTP}}{0.5} xK(Z) \text{ for } P \leq 0.5$$

$$Fq(Z) \leq \frac{Fq^{RTP}}{P} xK(Z) \text{ for } P > 0.5$$

where: P = the ratio of THERMAL POWER to RATED THERMAL POWER

F_q^{RTP} = 2.60

K(Z) is provided in Figure 2.5.1.

2.5.2 Uncertainty when PDMS is inoperable:

The uncertainty, U_{FQ}, to be applied to the Heat Flux Hot Channel Factor $F_Q(Z)$ shall be calculated by the following formula

$$U_{FQ} = U_{qu} \bullet U_e$$

where:

U_{qu} = Base FQ measurement uncertainty = 1.05

U_e = Engineering uncertainty factor = 1.03

2.5.3 W(Z) Values:

a) When PDMS is OPERABLE, W(Z) = 1.00000 for all axial points.

b) When PDMS is inoperable, W(Z) Values are provided in Figures 2.5.3.a through 2.5.3.c. The normal operation W(Z) values have been determined at burnups of 150, 8000 and 18800 MWD/MTU.

Table 2.5.3 shows the $F^C_Q(Z)$ penalty factors that are greater than 2% per 31 Effective Full Power Days (EFPD) These values shall be used to increase the $F^W_Q(Z)$ as per Surveillance Requirement 3.2.1.2. A 2% penalty factor shall be used at all cycle burnups that are outside the range of Table 2.5.3.

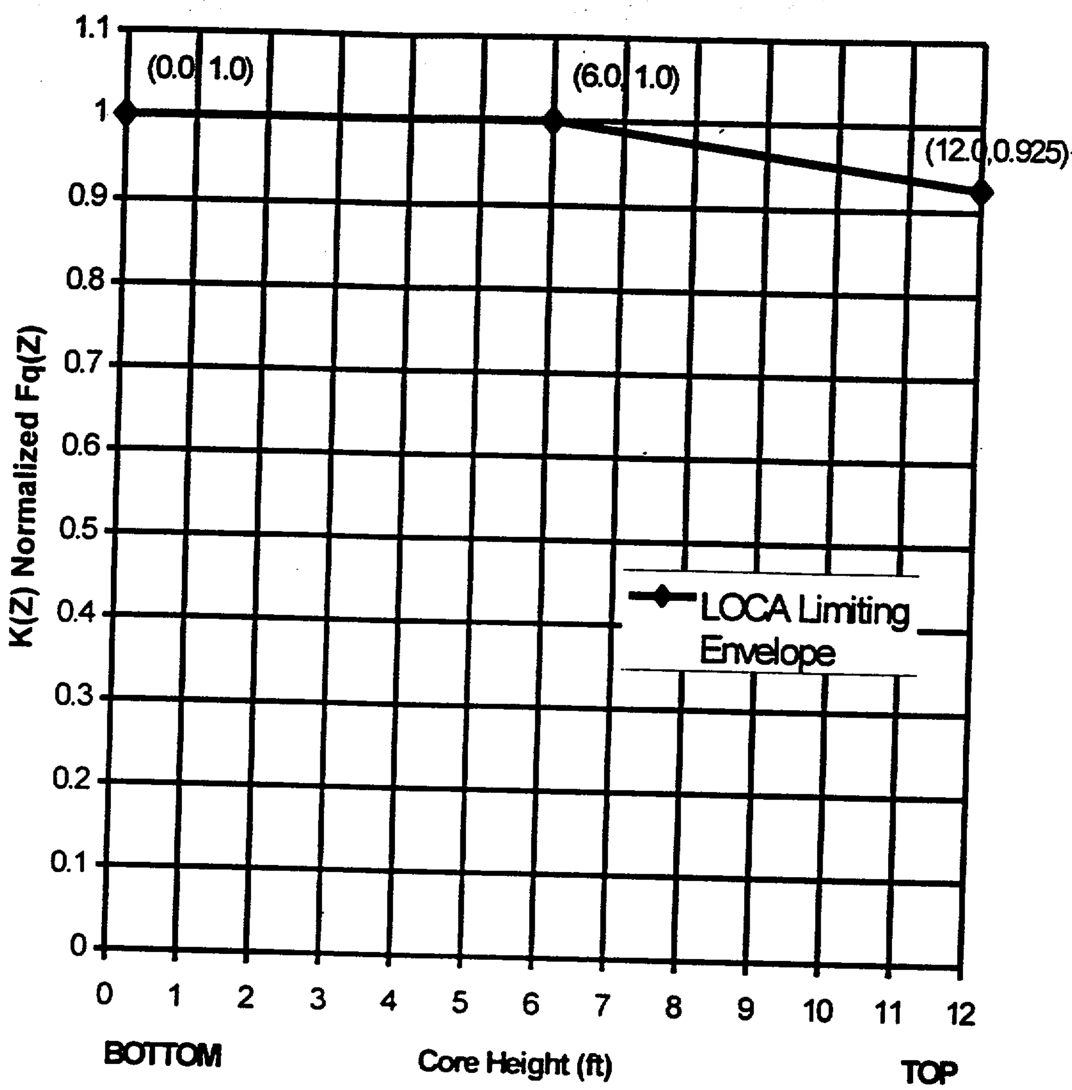
Multiplication Factor = 1.02

2.5.4 PDMS Alarms:

$F_Q(Z)$ Warning Setpoint = 2%

$F_Q(Z)$ Alarm Setpoint = 0%

Figure 2.5.1: K(Z) - Normalized Fq(Z) as a Function of Core Height



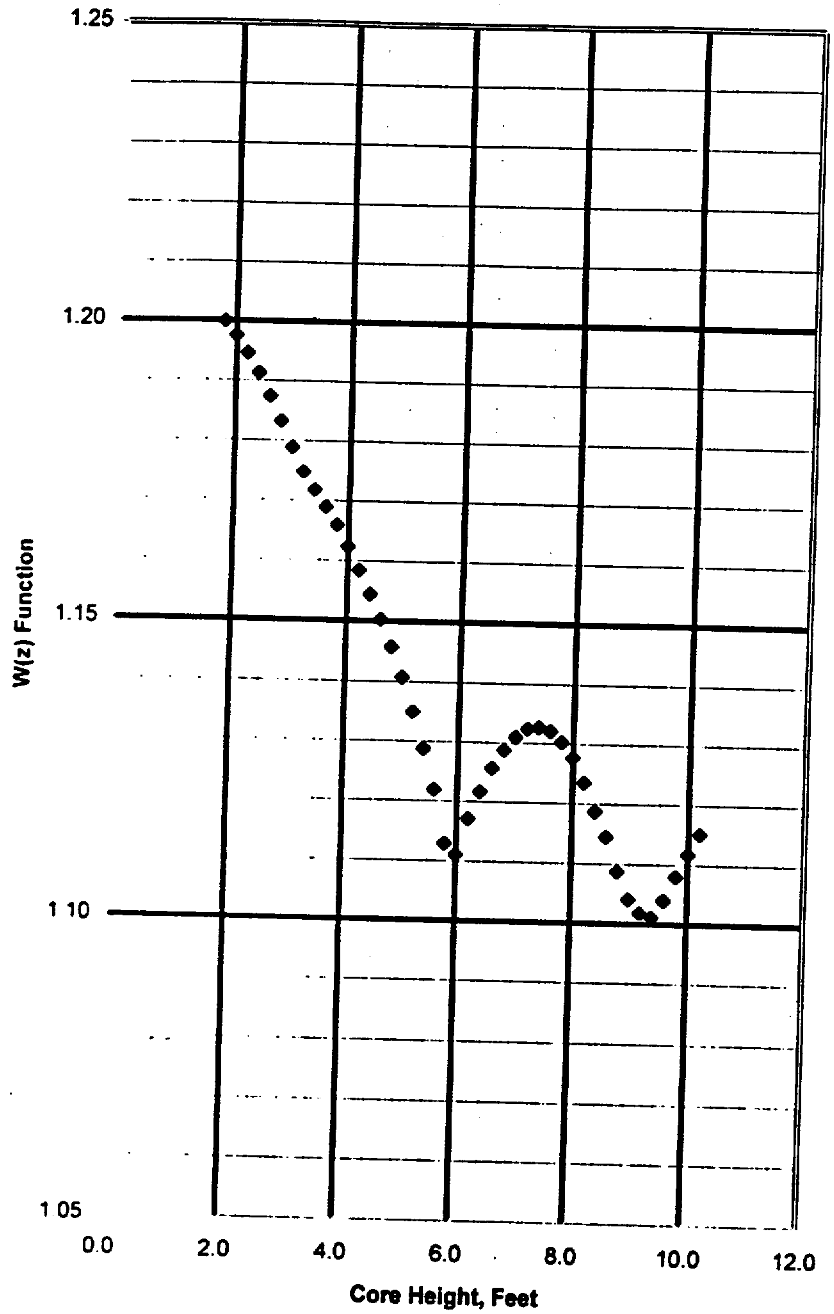
CORE OPERATING LIMITS REPORT (COLR) for BRAIDWOOD UNIT X CYCLE X

Height Feet	Max W(z)
0.0	1.0000
0.2	1.0000
0.4	1.0000
0.6	1.0000
0.8	1.0000
1.0	1.0000
1.2	1.0000
1.4	1.0000
1.6	1.0000
1.8	1.1999
2.0	1.1975
2.2	1.1946
2.4	1.1912
2.6	1.1874
2.8	1.1832
3.0	1.1788
3.2	1.1747
3.4	1.1717
3.6	1.1689
3.8	1.1659
4.0	1.1623
4.2	1.1584
4.4	1.1544
4.6	1.1503
4.8	1.1457
5.0	1.1407
5.2	1.1349
5.4	1.1287
5.6	1.1219
5.8	1.1129
6.0	1.1110
6.2	1.1171
6.4	1.1216
6.6	1.1256
6.8	1.1287
7.0	1.1309
7.2	1.1323
7.4	1.1326
7.6	1.1320
7.8	1.1302
8.0	1.1276
8.2	1.1235
8.4	1.1187
8.6	1.1145
8.8	1.1087
9.0	1.1041
9.2	1.1018
9.4	1.1011
9.6	1.1040
9.8	1.1080
10.0	1.1117
10.2	1.1151
10.4	1.0000
10.6	1.0000
10.8	1.0000
11.0	1.0000
11.2	1.0000
11.4	1.0000
11.6	1.0000
11.8	1.0000
12.0	1.0000

Braidwood Unit X Cycle X

Figure 2.5.3.a

Summary of W(z) Function at 150 MWD/MTU
(Top and Bottom 15% Excluded per WCAP-10216)



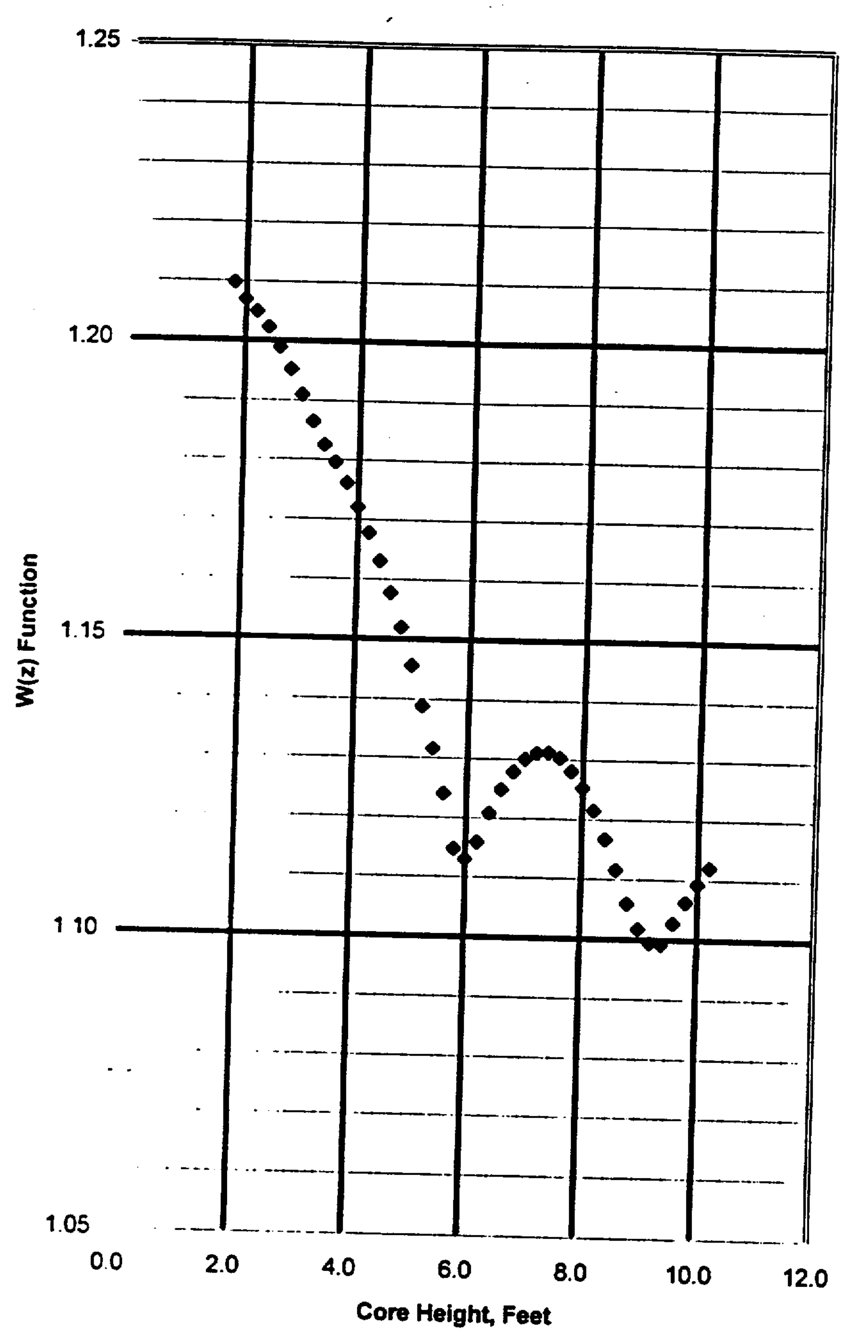
CORE OPERATING LIMITS REPORT (COLR) for BRAIDWOOD UNIT X CYCLE X

Height Feet	Max W(z)
0.0	1.0000
0.2	1.0000
0.4	1.0000
0.6	1.0000
0.8	1.0000
1.0	1.0000
1.2	1.0000
1.4	1.0000
1.6	1.0000
1.8	1.2097
2.0	1.2069
2.2	1.2049
2.4	1.2022
2.6	1.1989
2.8	1.1952
3.0	1.1909
3.2	1.1864
3.4	1.1825
3.6	1.1795
3.8	1.1761
4.0	1.1720
4.2	1.1677
4.4	1.1630
4.6	1.1577
4.8	1.1519
5.0	1.1456
5.2	1.1388
5.4	1.1316
5.6	1.1240
5.8	1.1147
6.0	1.1129
6.2	1.1158
6.4	1.1208
6.6	1.1248
6.8	1.1279
7.0	1.1300
7.2	1.1311
7.4	1.1312
7.6	1.1303
7.8	1.1281
8.0	1.1253
8.2	1.1215
8.4	1.1166
8.6	1.1115
8.8	1.1059
9.0	1.1017
9.2	1.0994
9.4	1.0992
9.6	1.1027
9.8	1.1062
10.0	1.1092
10.2	1.1120
10.4	1.0000
10.6	1.0000
10.8	1.0000
11.0	1.0000
11.2	1.0000
11.4	1.0000
11.6	1.0000
11.8	1.0000
12.0	1.0000

Braidwood Unit X Cycle X

Figure 2.5.3.b

Summary of W(z) Function at 8000 MWD/MTU
(Top and Bottom 15% Excluded per WCAP-10216)



CORE OPERATING LIMITS REPORT (COLR) for BRAIDWOOD UNIT X CYCLE X

Height Feet	Max W(z)
0.0	1.0000
0.2	1.0000
0.4	1.0000
0.6	1.0000
0.8	1.0000
1.0	1.0000
1.2	1.0000
1.4	1.0000
1.6	1.0000
1.8	1.2188
2.0	1.2148
2.2	1.2099
2.4	1.2044
2.6	1.1982
2.8	1.1914
3.0	1.1846
3.2	1.1773
3.4	1.1705
3.6	1.1670
3.8	1.1642
4.0	1.1611
4.2	1.1577
4.4	1.1538
4.6	1.1495
4.8	1.1449
5.0	1.1396
5.2	1.1341
5.4	1.1276
5.6	1.1203
5.8	1.1210
6.0	1.1267
6.2	1.1319
6.4	1.1363
6.6	1.1397
6.8	1.1421
7.0	1.1435
7.2	1.1438
7.4	1.1430
7.6	1.1410
7.8	1.1386
8.0	1.1352
8.2	1.1304
8.4	1.1245
8.6	1.1178
8.8	1.1099
9.0	1.1034
9.2	1.1011
9.4	1.1015
9.6	1.1053
9.8	1.1088
10.0	1.1120
10.2	1.1149
10.4	1.0000
10.6	1.0000
10.8	1.0000
11.0	1.0000
11.2	1.0000
11.4	1.0000
11.6	1.0000
11.8	1.0000
12.0	1.0000

Braidwood Unit X Cycle X

Figure 2.5.3.c

Summary of W(z) Function at 18800 MWD/MTU
(Top and Bottom 15% Excluded per WCAP-10216)

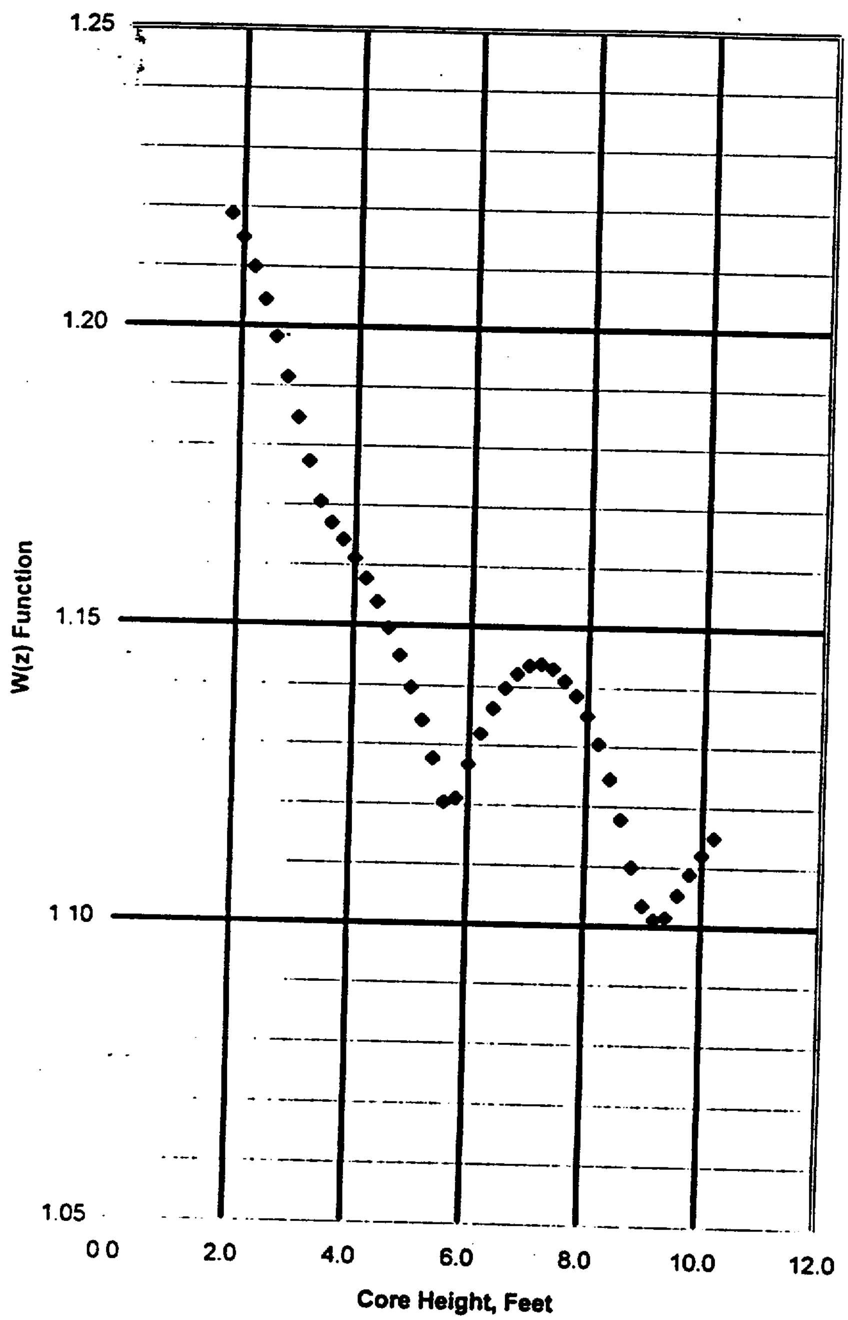


Table 2.5.3

Fq Margin Decreases in Excess of 2% per 31 EFPD

Cycle Burnup (MWD/MTU)	Max % Decrease in Fq Margin
150	3.62
800	4.21
1200	2.74
1800	2.21
2000	2.00

Note: All cycle burnups outside the range of the table shall use a 2% decrease in Fq margin for compliance with the 3.2.1.2 Surveillance Requirements.

2.6 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) (LCO 3.2.2)

2.6.1 $F^N_{\Delta H} \leq F^{RTP}_{\Delta H}[1.0 + PF_{\Delta H}(1.0 - P)]$

where: P = the ratio of THERMAL POWER to RATED THERMAL POWER
$F^{RTP}_{\Delta H} = 1.70$
$PF_{\Delta H} = 0.3$

2.6.2 Uncertainty when PDMS is inoperable

The uncertainty, $U_{F\Delta H}$, to be applied to the Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}{}^N$ shall be calculated by the following formula:

$U_{F\Delta H} = U_{F\Delta Hm}$

where:

$U_{F\Delta Hm}$ = Base FDH measurement uncertainty = 1.04

2.6.3 PDMS Alarms:

$F^N_{\Delta H}$ Warning Setpoint = 2%

$F^N_{\Delta H}$ Alarm Setpoint = 0%

2.7 Axial Flux Difference (AFD) (LCO 3.2.3)

2.7.1 When PDMS is OPERABLE, no AFD Acceptable Operation Limits are applicable.

2.7.2 When PDMS is inoperable, the AFD Acceptable Operation Limits are provided in Figure 2.7.2 or the latest valid PDMS Surveillance Report, whichever is more conservative.

2.8 Departure from Nucleate Boiling Ratio (DNBR) (LCO 3.2.5)

2.8.1 $DNBR_{APSL} = 1.4$

The $DNBR_{APSL}$ limit is applicable with THERMAL POWER ≥ 50% RTP when PDMS is OPERABLE.

2.8.2 PDMS Alarms:

DNBR Warning Setpoint = 2%

DNBR Alarm Setpoint = 0%

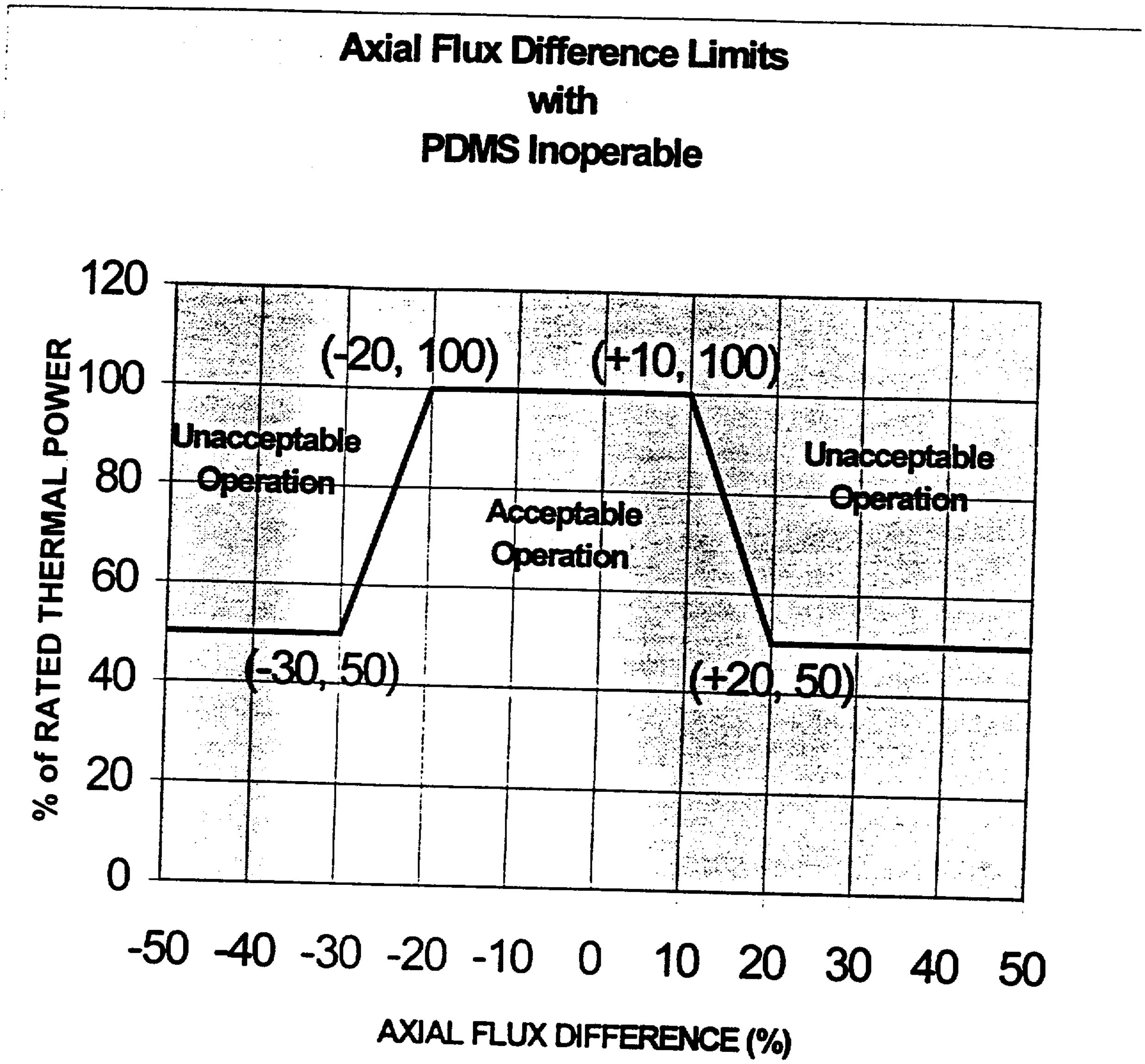
2.9 Boron Concentration (LCO 3.9.1)

2.9.1 The refueling boron concentration shall be greater than or equal to 2000 ppm.

2.9.2 The Reactor Coolant System boron concentration shall be greater than or equal to 1919 ppm to maintain adequate shutdown margin for Rod Drop Time Measurements. (TLCO 3.1.k)

CORE OPERATING LIMITS REPORT (COLR) for BRAIDWOOD UNIT X CYCLE X

Figure 2.7.2 Axial Flux Difference Limits as a Function of Rated Thermal Power



ATTACHMENT C

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison (ComEd) has evaluated the proposed changes and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

- Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated; or

- Create the possibility of a new or different kind of accident from any previously analyzed; or

- Involve a significant reduction in a margin of safety.

ComEd proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-37, NPF-66, NPF-72, and NPF-77 for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, respectively. The proposed changes to the TS involve allowing the use of the Best Estimate Analyzer for Core Operations Nuclear (BEACON) Power Distribution Monitoring System (PDMS) to perform core power distribution surveillances. The proposed changes allow for the power distribution surveillances to be performed by PDMS rather than using the Movable Incore Detector (MID) System. In addition, a Relaxed Axial Offset Control (RAOC)-type axial flux distribution methodology is planned to be implemented along with the proposed implementation of PDMS. The proposed TS changes for PDMS and RAOC are supported by the NRC approved methodologies. All reload specific input will be confirmed via approved reload methodology employed by ComEd and Westinghouse.

In addition, sections have been added to the COLR to define the equations and constants to be used to determine the applicable measurement uncertainties to be applied to the core peaking factors depending upon the Operability status of PDMS. The constants found in these sections of the COLR are used as coefficients in the PDMS uncertainty calculations and are determined using NRC approved methodology.

In support of this determination, an evaluation of the three criteria set forth in 10 CFR 50.92 is provided below.

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

Power Distribution Monitoring System (PDMS) performs continuous core power distribution monitoring. It in no way provides any protection or control system functionality. Fission product barriers are not impacted by these proposed changes. The proposed changes occurring with PDMS will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident. The changes associated with the PDMS do not affect plant systems such that their function in the control of radiological consequences is adversely affected. These proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

Continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This results in more time (i.e., earlier determination of an adverse condition developing) for operator action prior to having any adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an accident.

Each accident analysis addressed in the Byron and Braidwood Stations' UFSAR will be examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be performed in accordance with the requirements set forth in 10 CFR 50.59, "Changes, tests and experiments," will ensure that future reloads will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

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

As stated previously, the implementation of the PDMS system has no influence or impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with PDMS does not result in a change to the design basis of any plant component or system. The evaluation of the effects of the PDMS changes shows that all design standards and applicable safety criteria limits are met. These changes, therefore, do not cause the initiation of any accident nor create any new failure mechanisms. All equipment important to safety will operate as designed. Component integrity is not challenged. The proposed changes do not result in any event previously deemed incredible being made credible. The PDMS changes will not result in more adverse conditions and will not result in any increase in the challenges to safety systems. The cycle specific variables required by the PDMS are calculated using NRC approved methods. The Technical Specifications (TS) will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded.

The proposed change, therefore. does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is not affected by the implementation of PDMS. The margin of safety presently provided by current TS remains unchanged. Appropriate measures exist to control the values of these cycle-specific limits. The proposed changes continue to require operation within the core limits that are based on NRC approved reload design methodologies. The proposed changes continue to ensure that appropriate actions will be taken if limits are violated. These actions remain unchanged. The development of the reload specific limits, including Relaxed Axial Offset Control (RAOC) bands, for future reloads will continue to conform to those methods described in NRC approved documentation. In addition, each future reload involves a 10 CFR 50.59, "Changes, tests and experiments," safety review to assure that operation of the units, within the cycle-specific limits, will not involve a reduction in margin of safety.

The proposed changes, therefore, do not impact the operation of the Byron and Braidwood Stations in any manner that involves a reduction in the margin of safety.

ATTACHMENT D

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Commonwealth Edison (ComEd) has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that these proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment C, these proposed changes do not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

Implementation of the Best Estimate Analyzer for Core Operations Nuclear (BEACON) Power Distribution Monitoring System (PDMS) does not result in an increase in power level, does not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore the proposed changes will not affect the types or increase the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the configuration of the facility. The proposed changes only affect operation of the plant in that core power distribution monitoring can be performed on a continuous basis. There will be no increase in individual or cumulative occupational radiation exposure resulting from this change. There will be no change in the level of controls or methodology used for processing radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant Therefore, there will be no increase in individual or cumulative occupational radiation exposure from these proposed changes.