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U.S. Nuclear Regulatory Commission
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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

License Amendment Request to Use BEACON Power Distribution Monitoring System

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the operating license for Prairie Island Nuclear Generating Plant (PINGP). Specifically, NSPM proposes to revise Technical Specification (TS) 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation."

The proposed TS changes will allow the use of a dedicated on-line core power distribution monitoring system (PDMS) to enhance surveillance of core thermal limits. The PDMS to be used at PINGP is the NRC-approved Westinghouse proprietary core analysis system called Best Estimate Analyzer for Core Operations – Nuclear (BEACON™).

Enclosure 1 to this letter provides the evaluation of the proposed TS changes for the PDMS. Enclosure 2 provides an evaluation entitled, "Evaluation for Excluding Power Distribution Monitoring System Requirements from the Technical Specifications." Enclosure 3 provides the existing TS pages marked-up to show the proposed changes. Enclosure 4 provides, for information only, the existing TS Bases pages marked-up to show the associated proposed Bases changes. Enclosure 5 provides marked-up pages from the PINGP Technical Requirements Manual (TRM) indicating the changes to incorporate appropriate administrative controls and surveillance requirements for the PDMS. The proposed TRM changes are provided for information only and will be implemented at the time the amendment is implemented. Final TS Bases changes will be implemented pursuant to TS 5.5.12, "Technical Specifications (TS) Bases Control Program," at the time the amendment is implemented. Enclosure 6 provides a list of regulatory commitments made by NSPM in this submittal.

NSPM has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment meets the categorical exclusion requirements of 10 CFR 51.22(c)(9) and an environmental impact assessment need not be prepared.

A copy of this submittal, including the Determination of No Significant Hazards Consideration, without enclosures 2 through 6, is being forwarded to the designated State of Minnesota official pursuant to 10 CFR 50.91(b)(1).

NSPM requests approval of this proposed amendment by July 1, 2011. Once approved, the amendment will be implemented within 120 days or prior to December 31, 2011.

If there are any questions or if additional information is needed, please contact Glenn Adams at 612-330-6777.

Summary of Commitments

This letter makes one new commitment as described in Enclosure 6, and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on JUN 14 2010

Mark A. Schimmel
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosures (6)

cc: Regional Administrator, Region III, USNRC
Project Manager, Prairie Island Nuclear Generating Plant, USNRC
Resident Inspector, Prairie Island Nuclear Generating Plant, USNRC
State of Minnesota (without enclosures 2 through 6)

ENCLOSURE 1

Evaluation of the Proposed Change

License Amendment Request to Use BEACON Power Distribution Monitoring System

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the operating license for Prairie Island Nuclear Generating Plant (PINGP). Specifically, NSPM proposes to revise Technical Specification (TS) 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," to incorporate use of the Best Estimate Analyzer for Core Operations – Nuclear (BEACON™) Power Distribution Monitoring System (PDMS) described in WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System." The purpose of this system is to perform core power distribution surveillances.

2.0 DETAILED DESCRIPTION

The proposed changes to the TS are as follows:

1. **Proposed Change to TS 3.1.7, "Rod Position Indication"**

   Currently, for conditions involving inoperable rod position indicators, Required Actions A.1 and B.3 of Limiting Condition for Operation (LCO) 3.1.7 require plant operators to verify the position of the rod(s) with inoperable position indicators by using movable incore detectors.

   The proposed change will modify Required Action A.1 to state: *"Verify the position of the rod(s) with inoperable position indicators by using core power distribution measurement information."* The proposed change will modify Required Action B.3 to state: *"Verify, using core power distribution measurement information, position of rods with inoperable RPIs which have been moved in excess of 24 steps in one direction since last determination of their position."*

   The generic phrase "core power distribution measurement information" is substituted for "movable incore detectors," as this would allow the use of either an OPERABLE PDMS or the movable incore detectors for verifying the position of the rod(s) with an inoperable rod position indicator.

2. **Proposed Change to TS 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)"**

   Currently, the Note to the Surveillance Requirements Table states: *"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."*

   The proposed change will modify the Note to state: *"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 measurement is obtained."*

In addition, another Note associated with Surveillance Requirement (SR) 3.2.1.2 (for verifying that $F^W_Q(Z)$ is within limits) will be modified by the proposed change. Part b. of this Note currently states: *"Repeat SR 3.2.1.2 once per 7 EFPD [Effective Full Power Days] until either a. above is met or two successive flux maps indicate that the maximum over z* $\left[\frac{F^C_Q(Z)}{K(Z)}\right]$ *has not increased."*

The proposed change will modify the SR so that part b. of the Note is worded as follows: *"Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive power distribution measurements indicate that the maximum over z* $\left[\frac{F^C_Q(Z)}{K(Z)}\right]$ *has not increased."*

These changes would allow the Surveillance to be performed using the movable incore detectors or an OPERABLE PDMS.

3. Proposed Change to TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)"

Currently, SR 3.2.4.2 states, *"Verify QPTR is within limit using the movable incore detectors or thermocouples."*

The proposed change will modify SR 3.2.4.2 to state, *"Verify QPTR is within limit using core power distribution measurement information."*

This change would allow the Surveillance to be performed using either the movable incore detectors, the thermocouples, or an OPERABLE PDMS.

4. Proposed Change to TS 3.3.1, "Reactor Trip System (RTS) Instrumentation"

SR 3.3.1.3 requires comparing results from the incore detector system to the Nuclear Instrumentation System (NIS) channel output with respect to the indicated axial flux difference (AFD). Currently, SR 3.3.1.3 states, *"Compare results of the incore detector measurements to NIS AFD."*

The proposed change will modify SR 3.3.1.3 to state, *"Compare results of the core power distribution measurements to NIS AFD."*

In addition, SR 3.3.1.6 requires periodically calibrating the excore channels against the incore channels. Currently, SR 3.3.1.6 states, *"Calibrate excore channels to agree with incore detector measurements."*

The proposed change will modify SR 3.3.1.6 to state, *"Calibrate excore channels to agree with core power distribution measurements."*

These changes would allow SR 3.3.1.3 and SR 3.3.1.6 to be performed using either the movable incore detectors or an OPERABLE PDMS.

The proposed TS changes described above are shown as mark-ups to the current TS on the pages provided in Enclosure 3.

The TS Bases will also be revised for consistency with the proposed TS changes. A markup of the TS Bases pages reflecting these changes is provided in Enclosure 4 for information only. In addition to the Bases for TS 3.1.7, TS 3.2.1, TS 3.2.4, and TS 3.3.1, the Bases for TS 3.1.4, "Rod Group Alignment Limits" and TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)" are also marked for revision due to references to "incore flux mapping," and related information in those Bases sections. The proposed TS Bases changes will be implemented in accordance with TS 5.5.12, "Technical Specifications (TS) Bases Control Program," at the same time that the proposed TS changes of this LAR are implemented.

With regard to the OPERABILITY and control requirements of the PDMS and its associated instrumentation, NSPM has determined that no TS changes are needed for this purpose because the PDMS does not meet the selection criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. The evaluation for this determination is provided in Enclosure 2. Further, precedent has been set by similar facilities with respect to use of the PDMS in the application as described herein. See Section 4.2 of this enclosure for details.

In lieu of TS requirements, requirements for the PDMS and associated instrumentation will be placed in the PINGP Technical Requirements Manual. The changes for incorporating PDMS instrumentation requirements and controls into the Technical Requirements Manual are indicated in Enclosure 5. The indicated changes are provided for information only.

3.0 TECHNICAL EVALUATION

The PDMS to be used at the PINGP utilizes the NRC-approved Westinghouse proprietary core analysis system called the Best Estimate Analyzer for Core Operations - Nuclear (BEACON), together with continuous information from plant instrumentation. Incore detector measurements are used to periodically calibrate the BEACON PDMS. The BEACON PDMS serves as a three dimensional (3-D) core monitor, operational analysis tool, and operational support package.

Westinghouse submitted topical report WCAP-12472-P, "BEACON Core Monitoring and Operations Support System," to the NRC on May 21, 1990. The NRC issued a Safety Evaluation Report (SER) approving the topical report on February 16, 1994. In its SER, the NRC concluded that BEACON provides a greatly improved continuous online power distribution measurement and display, limit surveillance, and operation prediction information system for Westinghouse reactors.

PINGP currently uses a module of BEACON called INCORE. BEACON INCORE is a stand-alone system that provides a modern user interface to interpret the measurements from the Unit's movable incore detector system.

BEACON has three additional operational levels that interface with plant instrumentation; BEACON-OLM (On-Line Monitor), BEACON-TSM (Technical Specification Monitor), and BEACON-DMM (Direct Margin Monitor).

The BEACON-OLM system level was developed to provide licensees with the same level of functionality and application that was being used before the licensing of BEACON. This system level provides the base functionality of the BEACON system which includes continuous core monitoring, core predictive capability and operational history analysis. This system level is used for information and analysis purposes and does not require operational action based on results from the core monitor displays. This level of the BEACON system is purely an information and analysis tool that plant operational personnel can use at their option. The use of the BEACON-OLM level can be integrated into the plant procedures. If this is done, then the flux map analysis and estimated critical condition (ECC) functions from BEACON can be used to replace other off-line codes and procedures used for these calculations.

The BEACON-TSM system level was developed to provide licensees with the functionality needed to integrate BEACON into the plant TS for monitoring of current TS thermal power limits such as peak linear power density (F_Q - TS 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$)) and peak enthalpy rise ($F^N_{\Delta H}$ - TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)). BEACON-TSM includes all of the base functionality in the BEACON-OLM level. Added to this are the procedures, system operational status information and on-line calculations needed to provide the core monitoring capability for TS compliance. The licensing of BEACON for core monitoring allows the BEACON on-line monitoring functions to potentially eliminate most flux maps for normal and off-normal TS thermal power limit verification. Once integrated into the plant TS and procedures, the BEACON-TSM system has the potential to provide the following features:

- Essentially continuous monitoring of the core power distribution
- Increased interval for flux maps (using incore detectors) from 31 EFPD to 180 EFPD (flux maps are only required for BEACON calibration, when thermal power is less than 25% Rated Thermal Power (RTP), or when PDMS is inoperable)
- Verification of the position of a rod with an inoperable rod position indicator (RPI) using the BEACON core monitor function
- Reduced incore detector instrumentation requirements to 50% after initial calibration for a fuel cycle. The BEACON system uses surface spline fitting to compensate for sparse instrumentation and automatically adjusts the applied thermal limit uncertainties allowing for operation with reduced instrumentation.

The BEACON-DMM level was developed to provide licensees with the full functionality and benefits of the BEACON license granted by the NRC. BEACON-DMM includes all of the functionality of BEACON-TSM and also provides for direct monitoring and use of Departure from Nucleate Boiling Ratio (DNBR) as a thermal limit in the plant TS. NSPM does not propose to license the BEACON-DMM application of the PDMS for the PINGP.

NSPM proposes to license the BEACON-TSM application of the PDMS as the primary method for performing power distribution measurements and surveillances when thermal power is greater than or equal to 25 percent RTP. The Technical Requirements Manual (TRM) will implement the associated PDMS operability requirements. At thermal power levels less than 25 percent RTP, or when the PDMS is inoperable or as an alternative for performing power distribution measurements, the movable incore detector system will be used.

3.1 Background

As described in WCAP-12472-P-A, the Westinghouse BEACON PDMS was developed to improve the operational support for pressurized water reactors (PWRs). BEACON is an advanced core monitoring and support package that utilizes existing plant instrumentation such as core exit thermocouple temperatures, reactor coolant system (RCS) cold leg temperatures, control bank positions, power range detector output, and reactor power level. These data are sent by the plant computer in the form of a file that BEACON can interpret to perform nodal power distribution prediction calculations. The PDMS does not provide any protection or control system function.

The PDMS includes an on-line 3-D nodal model (SPNOVA) that is continuously updated to reflect the current plant operating conditions. The nodal solution method used by the PDMS is consistent with the NRC-approved Westinghouse Advanced Nodal Code (ANC) core design code. The core-exit thermocouple and excore neutron flux detector readings 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 can be better understood. The PDMS provides an understanding of operating and design margins to address strategic fuel cycle changes. The BEACON methodology improves the quality of the surveillance process because it uses a frequently calibrated model to match the actual operational profile. The PDMS continuously monitors the limiting $F_Q(Z)$ and $F^N_{\Delta H}$.

As previously noted, the movable in-core detector system will remain available for use. The movable in-core detector system will also be used to calibrate BEACON.

NSPM intends to utilize the BEACON PDMS to take advantage of its capability for nearly continuous monitoring of the limiting core thermal peaking factors, $F_Q(Z)$ and $F^N_{\Delta H}$, without the need to obtain a full-core flux map. The BEACON PDMS will provide operational support for TS compliance, and its nearly continuous monitoring feature will permit instantaneous identification of core anomalies, as well as providing predictive capabilities for both operators and reactor engineers.

3.2 BEACON Core Monitoring Methodology

The following is a summary description of BEACON, taken primarily from Brookhaven National Laboratory's (BNL) Technical Evaluation Report (TER) for WCAP-12472-P.

The primary function of the BEACON core monitoring system is the determination of the three-dimensional core power distribution. In BEACON, this calculation is performed with the NRC-approved Westinghouse SPNOVA nodal method. The SPNOVA data libraries and core models are consistent with the NRC-approved Westinghouse PHOENIX/ANC design models and have been benchmarked against operating reactor measurements.

The BEACON core monitoring process is carried out in three steps. In the first step, the SPNOVA model, individual thermocouples, and the excore axial offset are calibrated to the full-core incore flux measurement. In the second step, the SPNOVA model is updated based on the most recent operating history, and adjusted using the thermocouple and excore measurements. The continuous monitoring is performed in step-3, using the thermocouples and excore detectors to update the BEACON model.

The BEACON power distribution calculation is updated using the thermocouple and excore detector measurements. The thermocouple measurements are interpolated / extrapolated radially using the spline fit. The BEACON system provides both a full three-dimensional nodal power distribution calculation as well as a simplified, more approximate one-dimensional calculation. The BEACON on-line limits evaluation will be performed in three dimensions, and the one-dimensional calculation will only be used as a scoping tool in predictive analysis.

The continuous core monitoring of the current reactor statepoint (fuel burnup, xenon distribution, soluble boron concentration, etc.) provided by BEACON allows a more precise determination of the parameters used in the transient analyses, and therefore relaxes the requirement to limit the transient initial conditions via power distribution control. As part of the continuous monitoring, the fuel limits and DNBR limits are calculated using the standard Westinghouse methods.

For the application of BEACON at PINGP, credit will not be taken for the calculation and direct monitoring of DNBR as a thermal limit (as described above). Instead, BEACON will be used as a TS monitor for present peaking factor limits, and the transient initial condition limits for the PINGP will not be relaxed.

OPERABILITY and calibration of the BEACON PDMS is dependent on the number and distribution of available core exit thermocouples. The criteria for the core exit thermocouples, with BEACON OPERABLE, require at least 25% of the thermocouples to be OPERABLE, including at least two per quadrant, with the added requirement that the OPERABLE pattern normally covers all internal fuel assemblies within a chess "knight move" (an adjacent plus a diagonal square away); otherwise a more frequent calibration is required. With optimum thermocouple coverage, calibration with the movable incore detectors is required every 180 effective full power days (EFPD). However, calibration is required every 31 EFPD when the "knight move" requirement is not satisfied. The accuracy of the power distribution information with decreased incore detector or thermocouple OPERABILITY has been analyzed by Westinghouse, and penalties are applied to the calculated peaking factors (refer to TER Section 2.3). The analysis concluded that the minimum available incore detectors and thermocouples, when coupled with the increased uncertainty penalties, provide reasonable and acceptable power distribution information.

3.3 Model Calibration and Uncertainty

BEACON uses the incore flux detector measurements, core-exit thermocouples, and excore detectors to perform the local calibration of the SPNOVA three-dimensional power distribution. The SPNOVA-predicted detector reaction rates are normalized to the incore measurements at the incore radial locations and over an axial mesh. The thermocouple adjustment is two-dimensional and is made by normalizing the SPNOVA radial power distribution to the assembly power inferred from the core-exit thermocouples. The thermocouple assembly power measurement is periodically calibrated to the incore-measured assembly power.

Since the incore detectors and core-exit thermocouples do not provide complete coverage of the core, BEACON employs a two-dimensional spline fit to interpolate/extrapolate these measurements to the unmonitored assemblies. The spline fit includes a tolerance factor which controls the degree to which the fit is forced to match the individual measurements. If, for example, the measurements are believed to be extremely accurate (inaccurate), a low (high) tolerance factor is used and the SPNOVA solution is (is not) forced to be in exact agreement with the measurements.

The BEACON axial power shape is adjusted to ensure agreement with the axial offset measured by the excore detectors. This adjustment is made by adding a sinusoidal component to the SPNOVA-calculated axial power shape. The SPNOVA excore axial offset is determined by an appropriate weighting of the peripheral assembly powers. The excore detector axial offset is periodically calibrated to the incore power distribution measurements.

As an initial assessment of the power distribution calculation, Westinghouse performed detailed comparisons of BEACON to the predictions of the INCORE system used at Westinghouse plants. INCORE is a data analysis code written to process information obtained by the movable incore detector system in Westinghouse pressurized water reactors. INCORE is presently used at PINGP for processing information obtained by the movable incore detector system and verifying TS Surveillance Requirements. These comparisons were made for three plants over four cycles, and included a range of fuel burnup, core loadings, power levels and control rod insertion levels. The averages of the standard deviation between the BEACON results and the actual measured reaction rates were 1.5% for assemblies with power greater than the average (1.0) value and 2% for all measured assemblies (Reference 6.1, Section 4.1.1). The averages of the standard deviation of the inferred assembly power between BEACON and INCORE were 1.10% for assemblies with power greater than the average (1.0) value and 1.37% for all assemblies (Reference 6.1, Table 4-6). From the results of this study, Westinghouse concluded that the BEACON processing of the incore flux map and the inferred assembly power distribution accuracy is statistically consistent with the INCORE computer code.

The uncertainties applied to the BEACON power distribution measurements are different than those applied to the traditional flux map systems because BEACON uses a more comprehensive scope of instrumentation. An uncertainty analysis of the BEACON power distribution measurement is reported in Reference 6.1. Portions of the BNL TER for Reference 6.1 relevant to the uncertainty analysis are summarized in this section below:

Due to the change in reactor statepoint, SPNOVA modeling approximations, and instrumentation error, a model calibration uncertainty is introduced into the BEACON predictions. Westinghouse has evaluated this uncertainty by comparing BEACON predicted and measured incore reaction rates over four cycles and a range of operating conditions, and has found that the model calibration uncertainty was very small and varied only slightly for these comparisons.

The thermocouple calibration uncertainty is due to the change in reactor statepoint and to instrument error. Westinghouse has evaluated this uncertainty by comparing the assembly powers inferred from the thermocouples to SPNOVA incore-corrected assembly powers. Comparisons for three plants and a range of operating conditions indicate a difference of less than a few percent at full power. The observed calibration uncertainty increased at lower powers due to the reduced enthalpy rise and changes in cross-flow.

In order to determine the axial power distribution uncertainty, Westinghouse has compared SPNOVA incore-updated and SPNOVA excore-updated predictions of the axial power shape. These comparisons included a range of fuel burnups and rod insertions, and indicated a 95/95 upper tolerance limit of less than a few percent with a slight dependence on rod movement since calibration.

Based on an extensive set of calibration data, the model calibration uncertainty is observed to increase as the calibration interval (in units of fuel burnup) increases. Using the observed fuel burnup dependence, an additional assembly power uncertainty is determined to account for the effects of increased calibration intervals.

The failure of detectors in the BEACON system results in a relaxation of the local calibration to measurement, and an increase in the power distribution uncertainty. The effect of random failures of the incore and thermocouple detectors on the assembly power was evaluated for failure rates of up to 75%. The assembly power uncertainty was found to increase linearly with incore detector failure and quadratically with the failure of thermocouples.

The BEACON calculation requires local power distribution factors for (1) the ratio of assembly power-to-detector response, (2) assembly local peaking factor, and (3) the grid power-depression factor, which is a correction factor to the assembly axial power distribution to account for the power depression due to the grid of the assembly. The BEACON uncertainty analysis employs previously approved upper tolerance values for the assembly power-to-detector response ratio and the local peaking factor. The grid factor uncertainty was determined by comparison to measured flux traces and is found to be relatively small.

The uncertainty in the BEACON power peaking resulting from errors in the SPNOVA model calibration and thermocouple calibration is determined using an analog 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 thermocouple 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 approximate 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 thermocouple uncertainties is used together with a relatively large tolerance factor which results in substantial smoothing of the thermocouple measurements. The upper tolerance limit on the assembly power peaking factor is calculated and found to increase as the square-root of the thermocouple uncertainty.

The enthalpy-rise ($F_{\Delta H}$) and power peaking factor (F_Q) 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.

3.4 Acceptance Criteria/Conditions

In the NRC Safety Evaluation Report for WCAP-12472-P, the NRC staff evaluated the BEACON methodology, calibration process, the uncertainty analysis, and the operation of the overall system and concluded that the BEACON PDMS is acceptable for performing core monitoring and operations support functions for Westinghouse PWRs but subject to certain conditions as specified in the BNL TER. These conditions are listed below. After each condition listed, a description of how the condition will be met at the PINGP is provided.

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

   Cycle-specific BEACON calibrations performed before startup and at beginning-of-cycle conditions will ensure that power peaking uncertainties provide 95% probability upper tolerance limits at the 95% confidence level. These calibrations are to be performed using the Westinghouse methodology. Until these calibrations are complete, more conservative default uncertainties will be applied. The calibrations will be documented and retained as records. See the commitment made in Enclosure 6.

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 database.*

   The PINGP utilizes a Westinghouse 2-loop pressurized water reactor (PWR) nuclear steam supply system (NSSS) with movable incore instrumentation and other core power distribution monitoring instrumentation described by Section 1.0 of the Safety Evaluation Report (SER) for WCAP-12472-P-A. That SER states the general applicability of the WCAP to Westinghouse PWRs. Therefore, PINGP does not differ significantly from the plants that form the WCAP database.

   During the review of Reference 6.1, the NRC requested additional information on how the BEACON methodology treats core loadings with different fuel designs and the impact to the BEACON uncertainty analysis. Westinghouse responded that for all BEACON applications, the previous operating cycle is examined to establish reference uncertainties. This examination accounts for loading of fuel of different designs by comparing a BEACON model to actual operating data over the cycle. At the beginning of cycle, thermocouple data is verified and calibration/uncertainty components are updated as necessary. In addition, the initial flux mapping at the start of the cycle insures model calibration factors that reflect the actual fuel in the reactor before the BEACON system is declared OPERABLE.

3. *The BEACON Technical Specifications should be revised to include the changes described in Section 3* [of the BNL TER] *concerning Specifications 3.1.3.1 and 3.1.3.2 and the Core Operating Limits Report.*

   The WCAP describes an application of BEACON where the core operating limits are changed. As noted previously, NSPM is proposing only to use BEACON as a core TS monitor for conformance to the existing PINGP TS limits. The TS changes pertaining to this question are not applicable or of concern to the more limited changes being proposed by NSPM for the intended use of BEACON. Therefore, this condition does not apply to the amendment requested for PINGP.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The US Atomic Energy Commission (AEC) issued their Safety Evaluation (SE) of the PINGP on September 28, 1972. The SE, Section 3.1, "Conformance with AEC General Design Criteria," described the conclusions the AEC reached associated with the General Design Criteria in effect at the time. The AEC stated:

The Prairie Island plant was designed and constructed to meet the intent of the AEC's General Design Criteria, as originally proposed in July 1967. Construction of the plant was about 50% complete and the Final Safety Analysis Report (Amendment No. 7) had been filed with the Commission before publication of the revised General Design Criteria in February 1971 and the present version of the criteria in July 1971. We did not require the applicant to reanalyze the plant or resubmit the FSAR. However, our technical review did assess the plant against the General Design Criteria now in effect and we are satisfied that the plant design generally conforms to the intent of these criteria.

Based on the above, the most applicable PINGP GDC states:

CRITERION: Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core. PINGP GDC-13

Implementation of the PDMS at the PINGP does not replace, eliminate, or modify existing plant instrumentation. The PDMS operates on a workstation that obtains data from a separate data file generated by the plant process computer. The PDMS combines inputs from currently installed plant instrumentation and design data generated for each fuel cycle. Together, this provides a means to continuously monitor the power distribution limits including limiting peaking factors and quadrant power tilt ratio.

4.2 Precedent

BEACON, at the Technical Specification Monitor (BEACON-TSM) operational level, has been approved by the NRC for use at the following stations:

- Diablo Canyon Power Plant in License Amendment Nos. 164 (Unit 1) and 166 (Unit 2) on March 31, 2004 (ADAMS Accession Number ML040920245). In this application, a PDMS LCO was not added to the TS, but placed in a document under licensee control.
- South Texas Project in License Amendment Nos. 175 (Unit 1) and 163 (Unit 2) on March 31, 2006 (ADAMS Accession Number ML060760501). In this application, a PDMS LCO was not added to the TS, but placed in a document under licensee control.
- Callaway in License Amendment No. 182 on March 21, 2007 (ADAMS Accession Number ML070460584). In this application, a PDMS LCO was not added to the TS, but placed in a document under licensee control.

More recently in Amendment No. 144 issued on April 2, 2008, for the Comanche Peak Steam Electric Station, Units 1 and 2, the NRC approved a more extensive use of the BEACON system. This amendment approved (1) implementation of Westinghouse methodologies for determining selected core operating parameter values; (2) implementation of relaxed axial offset control (RAOC) of the reactor core; and (3) implementation of the BEACON method for determining the core power distribution. Section 3.2.1.5 of the NRC Safety Evaluation provides the specific evaluation for the use of a BEACON system for TS monitoring (ADAMS Accession Number ML080500627). In this application, a PDMS LCO was not added to the TS, but placed in a document under licensee control.

4.3 Significant Hazards Consideration

Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, proposes to amend the facility operating licenses of Prairie Island Nuclear Generating Plants (PINGP) Units 1 and 2. The purpose of this amendment is to modify the PINGP Technical Specifications (TS) to support the use of the Best Estimate Analyzer for Core Operations - Nuclear (BEACON) Power Distribution Monitoring System (PDMS).

NSPM has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

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

Response: No

The PDMS performs continuous core power distribution monitoring with data input from existing plant instrumentation. The system passively supports Technical Specification (TS) surveillances which ensure that core power distribution is within the same limits that are currently prescribed. Further, the proposed TS Actions are comparable to existing operator actions such that no new plant configurations are prompted by the proposed change. The system's physical interface with plant equipment is limited to an electronic link from a new workstation to the plant process computer. The system is passive in that it provides no control or alarm functions, and does not promote any new plant configuration which would affect the initiation, probability, or consequences of a previously-evaluated accident.

Continuous on-line core monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This system performance may result in an earlier determination of an adverse core condition and more time for operator action, thus reducing the probability of an accident occurrence and reduced consequences should a previously-evaluated accident occur.

By virtue of its inherently passive surveillance function and limited interface with plant systems, structures, or components, the proposed changes will not result in any additional challenges to plant equipment that could increase the probability or occurrence of any previously-evaluated accident. Further, the proposed changes will ensure conformance to the same core power distribution limits that form the basis for initial conditions of previously-evaluated accidents. Thereby, the proposed changes will not affect the consequences of any previously-evaluated accident.

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

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No

The system's physical interface with plant equipment is limited to an electronic link from a new workstation to the plant process computer. The system is passive in that it provides no control or alarm functions, and the proposed changes (including operator actions prescribed by the proposed

TS) do not promote any new plant configuration which would create the possibility for an accident of a new or different type.

The NRC previously evaluated the effects of using the PDMS to monitor core power distribution parameters and determined that all design standards and applicable safety criteria limits are met. The Technical Specifications will continue to require operation within the required core operating limits, and appropriate actions will continue to be taken when or if limits are exceeded. Thus, the reactor core will continue to be operated within its reference bounds of design such that an accident of a new or different type is not credible.

The proposed change, therefore, does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

Response: No

No margin of safety is adversely affected by the implementation of the PDMS. The margins of safety provided by current TS requirements and limits remain unchanged, as the TS will continue to require operation within the core limits that are based on NRC-approved reload design methodologies. The proposed change does not result in changes to the core operating limits. Appropriate measures exist to control the values of these cycle-specific limits, and appropriate actions will continue to be specified and taken when limits are violated. Such actions remain unchanged.

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

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

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATIONS

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

6.0 REFERENCES

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

Enclosure 2

Evaluation for Excluding Power Distribution Monitoring System Requirements from the Technical Specifications

The purpose for this enclosure is to demonstrate that Limiting Conditions for Operation (LCOs) for the Best Estimate Analyzer for Core Operations – Nuclear (BEACON^{TM}) Power Distribution Monitoring System (PDMS) and associated instrumentation are not required to be included in the Prairie Island Nuclear Generating Plant (PINGP) Technical Specifications (TS). The justification for this statement is explained in the evaluation provided below. The evaluation demonstrates that the structures, systems, or components (i.e., instrumentation) that constitute the PDMS are not required to be contained in the TS in accordance with the requirements contained in 10 CFR 50.36(c)(2)(ii).

10 CFR 50.36(c)(2)(ii) requires that a TS LCO must be established for each item meeting one or more of the following criteria:

(A) Criterion 1. Installed Instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The PDMS instrumentation is not associated with monitoring of any aspect of the reactor coolant pressure boundary. Therefore, the PDMS cannot be used to detect or indicate any degradation of the reactor coolant pressure boundary.

(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident 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 $F_Q(Z)$ and $F_{\Delta H}$ are operating restrictions which ensure that the accident analyses and assumptions for all applicable, analyzed Design Basis Accidents (DBAs) remain valid. These limits are included in the TS and are not changed through implementation of the PDMS. The PDMS supports the capability to monitor core power distribution for verifying conformance to such limits, but it does not control core power distribution. In addition the PDMS cannot by itself cause or affect any condition assumed in the accident/transient analyses.

The PDMS provides the capability to monitor power distribution parameters at more frequent intervals than is currently required by the TS. These parameters can be determined independent of the OPERABILITY of PDMS. Therefore, the PDMS does not constitute 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) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The PDMS performs only a monitoring function and does not affect any of the key safety parameter limits or levels of margin considered in the DBA evaluations. The PDMS performs no active/control functions, nor does the PDMS have an actuation capability. Therefore, the PDMS is not part of any primary success path for mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The PDMS and its associated instrumentation provide the capability to monitor power distribution parameters at more frequent intervals than is currently required by the TS, but the PDMS has no active safety functions and its use has no impact on the results or consequences of any DBA or transient analysis. Further, the PDMS is only an alternative means for performing core power distribution measurements and related surveillances, because the current means of performing such activities (by use of the movable incore detectors) is still available. PDMS unavailability, therefore, is not significant relative to plant risk. Based on these considerations and facts, the PDMS is not a feature that is significant to public health and safety.

The evaluation completed above indicates that the BEACON PDMS does not meet any of the criteria for inclusion in the TS. The PDMS requirements and controls to be incorporated into the Technical Requirements Manual (TRM) are consistent with the recommendations in WCAP-12472-P-A and will suffice to provide the necessary OPERABILITY and test requirements for the PDMS apart from the TS. Enclosure 5 provides (for information only) the proposed new Technical Requirement (TR) 3.2.4, "Power Distribution Monitoring System (PDMS)."

Enclosure 3
Marked-Up Technical Specification Pages

7 pages follow

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Rod Position Indication (RPI) System and demand position indication shall be OPERABLE.

-----NOTE-----
Individual RPIs may be outside their limits for ≤ 1 hour following substantial rod movement.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RPI per group inoperable for one or more groups.	A.1 Verify the position of the rod(s) with inoperable position indicators by using core power distribution measurement information ~~movable incore detectors~~.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. More than one RPI per group inoperable for one or more groups.	B.1 Monitor and record demand position indication for rods with inoperable RPI.	Once per hour
	AND	
	B.2 Monitor and record reactor coolant system average temperature.	Once per hour
	AND	
	B.3 Verify, using core power distribution measurement information~~movable incore detectors~~, position of rods with inoperable RPIs which have been moved in excess of 24 steps in one direction since last determination of their position.	Once per 4 hours
	AND	
	B.4 Restore inoperable RPIs to OPERABLE status such that a maximum of one RPI per group is inoperable.	24 hours

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified AND 31 effective full power days (EFPD) thereafter

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	---------------------------NOTE--------------------------- If 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 an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive power distribution measurements~~flux maps~~ indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ has not increased. --- Verify $F_Q^W(Z)$ is within limit.	Once within 12 hours after achieving equilibrium conditions after each refueling after THERMAL POWER exceeds 75% RTP AND

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------------------------NOTES------------------------ 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 85% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	------------------------NOTE------------------------ Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 85% RTP. --- Verify QPTR is within limit using core power distribution measurement information~~the movable incore detectors or thermocouples~~.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	-----NOTES----- 1. Adjust NIS channel if absolute difference is ≥ 2%. 2. Not required to be performed until 72 hours after THERMAL POWER is ≥ 15% RTP. ---------- Compare results of the core power distribution~~incore detector~~ measurements to NIS AFD.	31 effective full power days (EFPD)
SR 3.3.1.4	-----NOTE----- This Surveillance must be performed on the reactor trip bypass breaker 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

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
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 core power distribution~~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

Enclosure 4

Marked-Up Technical Specification Bases Pages

25 pages follow

BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by core power distribution measurements~~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 control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) 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 a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The rod alignment requirements are satisfied when individual actual rod positions are within 24 steps of their group step counter demand position when the demand position is between 30 and 215 steps, or within 36 steps of their group step counter demand position when the demand position is $\leq$ 30 steps, or $\geq$ 215 steps.

BASES

ACTIONS

B.2.1.1, B.2.1.2, B.2.2, B.3, and B4 (continued)

Verifying that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within the required limits (i.e., SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1) ensures that current operation at RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 8 hours allows sufficient time to obtain ~~flux maps of the~~ core power distribution measurements using either the incore flux mapping system or the Power Distribution Monitoring System and to calculate $F_Q(Z)$ and $F_{\Delta H}^N$.

In lieu of determining hot channel factors ($F_Q(Z)$ and $F_{\Delta H}^N$) within the Completion Time of 8 hours, reducing the high neutron flux trip setpoint to 85% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. The Completion Time of 8 hours gives the operator sufficient time to accomplish an orderly power reduction and setpoint change without challenging the Reactor Protection System.

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.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analyses to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in Ref. 3 that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions.

If the analyses do not support continued operation at RTP, then the power must be reduced to a level consistent with the safety analyses.

BASES (continued)

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 limits, ejected rod worth, and with minimum SDM (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 RPI System and bank demand position indication be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires the following:

a. The RPI System indicates within 12 steps of the group step counter demand position when the demand position is between 30 and 215 steps, or within 24 steps of their group step counter demand position when the demand position is greater than or equal to 215 steps, or less than or equal to 30 steps, or individual rod position indication has been verified to be in agreement with actual rod position through independent means such as moveable incore detectors <u>or the Power Distribution Monitoring System</u> in response to Required Actions; and

BASES (continued)

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable RPI and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one RPI channel per group fails, the position of the rod may still be determined indirectly by core power distribution measurement using either~~use of~~ the moveable incore detectors or the Power Distribution Monitoring System. Based on experience, normal power operation does not require excessive movement of banks. 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. Verification may determine that the RPI is OPERABLE and the rod is misaligned, then the Conditions of LCO 3.1.4, "Rod Group Alignment Limits" must be entered.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors.

BASES

ACTIONS

A.2 (continued)

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

When more than one RPI channel per group fails, additional monitoring shall be performed to assure that the reactor remains in a safe condition. The demand position from the group step counters associated with the rods with inoperable position indicators shall be monitored and recorded on an hourly basis. This ensures a periodic assessment of rod position to determine if rod movement in excess of 24 steps has occurred since the last determination of rod position. If rod movement in excess of 24 steps has occurred since the last determination of rod position, the Required Action of B.3 is required.

The reactor coolant system average temperature shall be monitored and recorded on an hourly basis. Monitoring and recording of the reactor coolant system average temperature may provide early detection of mispositioned or dropped rods.

If THERMAL POWER has not been reduced ≤ 50% RTP in accordance with Required Action A.2 and one or more rods have been moved in excess of 24 steps in one direction, since the position was last determined via Required Action A.1, then action is initiated sooner in accordance with Required Action B.3 to begin verifying that these rods are still properly positioned relative to their group positions. The 4 hour allowance for completion of this action allows adequate time to complete the rod position verification using either the moveable incore detectors or the Power Distribution Monitoring System.

BASES

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

The position of rods with inoperable RPIs will also continue to be verified indirectly using either the moveable incore detectors or the Power Distribution Monitoring System (PDMS) every 8 hours in accordance with Required Action A.1 if THERMAL POWER has not been reduced ≤ 50% RTP in accordance with Required Action A.2. Using the moveable incore detectors or the PDMS provides further assurance that the rods have not moved.

Based on experience, normal power operation does not require excessive movement of banks. Therefore, the actions specified in this condition are adequate for continued full plant operation for up to 24 hours since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour allowed out of service time also provides sufficient time to troubleshoot and restore the RPI System to operation following a component failure in the system, while avoiding the challenges associated with a plant shutdown.

C.1.1 and C.1.2

Demand position indication is provided by any of the following means: step counters; ERCS; calculations using rod drive cabinet counters and Pulse to Analog counters. With all indication for one demand position per bank inoperable, the rod positions can be determined by the RPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the rod position indication of the most withdrawn rod and the rod position indication of the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. This ensures that the most withdrawn and least withdrawn rod are no more than 24 steps apart (including instrument

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

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

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

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

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

BASES

LCO (continued)

An $F^C_Q(Z)$ evaluation requires obtaining a power distribution measurement~~an incore flux map~~ in MODE 1~~.~~ ~~F~~from which~~the incore flux map results~~ a measured value ($F^M_Q(Z)$) of $F_Q(Z)$ is obtained. If the power distribution measurement is obtained with the movable incore detector system ~~Then~~,

$$F^C_Q(Z) = F^M_Q(Z)*(1.0815)$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances (1.03) multiplied by a factor associated with the flux map measurement uncertainty (1.05) (Ref. 3 and 5).

If the power distribution measurement is obtained with the Power Distribution Monitoring System,

$$F^C_Q(Z) = F^M_Q(Z)*(1.03)\left(1.0 + \frac{U_Q}{100}\right)$$

where 1.03 is a factor that accounts for fuel manufacturing tolerances and U_Q is a factor that accounts for Power Distribution Monitoring System measurement uncertainty (%), determined as described in Reference 5.

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

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

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

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

BASES

LCO (continued)

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 precludes core power distributions that could violate the assumptions in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F^C_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required, and if $F^W_Q(Z)$ cannot be maintained within the LCO limits, reduction of the AFD limits is required. Note that sufficient reduction of the AFD limits will also result in a reduction of the core power.

Violating the LCO limits for $F_Q(Z)$ may result in 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

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^C_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F^C_Q(Z)$ is $F^M_Q(Z)$ multiplied by factors accounting for manufacturing tolerances and measurement uncertainties. $F^M_Q(Z)$ is

BASES

ACTIONS

A.1 (continued)

the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

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

BASES

ACTIONS (continued)

C.1

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

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

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution measurement~~map~~ can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F^C_Q(Z)$ and $F^W_Q(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F^C_Q(Z)$ 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 $F^C_Q(Z)$ before exceeding 75% RTP. This ensures that some determination of $F_Q(Z)$ is 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 $F^C_Q(Z)$ and $F^W_Q(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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.

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 $F_Q^C(Z)$ as described in the preceeding LCO section. ~~Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z)$ = $F_Q^M(Z)$*(1.0815) (Ref. 4).~~ $F_Q^C(Z)$ is then compared to its specified limits. The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

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

If THERMAL POWER has been increased by $\geq$ 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 the power increase to stay within the LCO limits).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

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 plant is operated in accordance with the Technical Specifications (TS).

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 power distribution measurements~~flux maps~~ are taken at or near~~in~~ steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the measurements~~flux map data~~. These variations are, however, conservatively calculated during the nuclear design process 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 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

SR 3.2.1.2 (continued)

During the power ascension following a refueling outage, startup physics testing program controls ensure that the $F_Q(Z)$ will not exceed the values assumed in the safety analysis. These controls include power distribution measurement~~flux mapping~~, ramp rate restrictions, and restrictions on RCCA motion. They provide the necessary controls to precondition the fuel and ensure that the reactor power may be safely increased to equilibrium conditions at or near RTP, at which time $F_Q^W(Z)$ and AFD target band are determined. Performing the Surveillance within 12 hours after achieving equilibrium conditions after each refueling after THERMAL POWER exceeds 75% RTP, ensures that the $F_Q(Z)$ limit is met when the unit is released for normal operations.

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

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

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

B 3.2 POWER DISTRIBUTION LIMITS

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

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 during either normal operation or a postulated accident analyzed in the safety analyses.

$F_{\Delta H}^{N}$ 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, $F_{\Delta H}^{N}$ is a measure of the maximum total power produced in a fuel rod.

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

$F_{\Delta H}^{N}$ is not directly measurable but is inferred from a power distribution measurement~~map~~ obtained with either the movable incore detector system or the Power Distribution Monitoring System. Specifically, the results of the three dimensional power distribution measurement~~map~~ are analyzed by a computer to determine $F_{\Delta H}^{N}$. This factor is calculated at least every 31 effective full power days (EFPD). However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

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

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

LCO

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

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise 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 unit safety analyses as described in the Applicable Safety Analyses section above.

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

BASES

ACTIONS (continued)

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1, a~~n~~ power distribution measurement~~incore flux map~~ (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 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 power distribution measurement~~incore flux map~~, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.4

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

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

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.

BASES

ACTIONS (continued)

B.1

When Required Actions A.1 through A.4 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using either the movable incore detector system or the Power Distribution Monitoring System to obtain a power distribution measurement~~flux distribution map~~. A computer calculation determines ~~data reduction computer program then calculates~~ the maximum value of $F^N_{\Delta H}$ from the measured power~~flux~~ distributions~~s~~. The measured value of $F^N_{\Delta H}$ must be increased~~multiplied~~ by 4%~~1.04~~ (if using the movable incore detector system) or increased by $U_{\Delta H}$% (if using the Power Distribution Monitoring System, where $U_{\Delta H}$ is determined as described in Reference 3) 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.

BASES (continued)

REFERENCES

1. USAR Section 14.

2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

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

BASES

ACTIONS (continued)

A.3

The peaking factor $F_Q(Z)$ (as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$) and $F_{\Delta H}^N$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 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 a power distribution measurement using either the movable incore detector system or the Power Distribution Monitoring System~~flux mapping~~. A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a power distribution measurement~~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 $F_{\Delta H}^N$ and $F_Q(Z)$ for 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 $F_{\Delta H}^N$ 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

BASES

ACTIONS

A.4 (continued)

power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

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 limits 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 QPTR is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). 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,

BASES

ACTIONS

A.5 (continued)

which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing a power distribution measurement, using either the movable incore detector system or the Power Distribution Monitoring System~~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 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at 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.

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.

BASES

ACTIONS (continued)

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

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 85% 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.

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.

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

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channel inputs are inoperable and the THERMAL POWER is > 85% RTP.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

With 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. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that the QPTR remains within its limits.

For purposes of monitoring changes in radial core power distribution when one power range channel is inoperable, the Power Distribution Monitoring System, or at least 2 moveable incore detectors, or 4 thermocouples per quadrant may be used to calculate an incore core power tilt. This incore core power tilt may be used, instead of the excore detectors, to confirm that the QPTR is within the limits by comparing it to previous power distribution measurements~~flux maps~~.

REFERENCES

1. USAR, Section 14.
2. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criterion 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

BASES

ACTIONS

D.1.1, D.1.2, and D.2 (continued)

inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the core power distribution measurement information~~movable incore detectors~~ once per 12 hours may not be necessary.

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Overtemperature ΔT;
- Overpower ΔT;
- 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-two logic for actuation of the two-out-of-three trips and 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 6.

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 unit systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up

Enclosure 5

Proposed Marked-Up Technical Requirements Manual

6 pages follow

TECHNICAL REQUIREMENTS MANUAL

TABLE OF CONTENTS

3.2 POWER DISTRIBUTION LIMITS

TRM 3.2.4 Power Distribution Monitoring System (PDMS)

TLCO 3.2.4 The PDMS shall be OPERABLE with the minimum required channels shown in Table 3.2.4-1.

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

APPLICABILITY: MODE 1 with THERMAL POWER $\geq$ 25% RTP when the PDMS is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or
b. Monitoring the QUADRANT POWER TILT RATIO, or
c. Measurement of $F^N_{\Delta H}$, or
d. Measurement of $F_Q(Z)$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER less than 25% RTP.	A.1 Suspend use of the PDMS for calibration and core power distribution measurement functions.	Immediately
B. One or more functions from Table 3.2.4-1 with less than the required channel inputs OPERABLE.	B.1 Suspend use of the PDMS for calibration and core power distribution measurement functions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TSR 3.2.4.1 Perform CHANNEL CHECK of the PDMS instrumentation input specified in Table 3.2.4-1.	Prior to use for core power distribution measurement
TSR 3.2.4.2 Perform calibration of the PDMS using the movable incore detector system with at least 75% of the detector thimbles and at least 2 detector thimbles per core quadrant, using a minimum of 10 OPERABLE core exit thermocouples with at least 2 thermocouples per quadrant, with THERMAL POWER ≥ 25% RTP.	Initial calibration after each refueling outage

TSR 3.2.4.3	------------------------NOTE------------------------ For PDMS calibration, the quantity and the coverage distribution of core exit thermocouples used as data input must meet certain criteria. With respect to thermocouple coverage, the available core exit thermocouple distribution can be "optimum" or "minimum" as measured by the chess "knight move". This criterion affects the TSR Frequency: Optimum thermocouple coverage satisfies the thermocouple minimum OPERABILITY requirements of Table 3.2.4-1 with the added requirement that the OPERABLE pattern covers all internal fuel assemblies within a chess "knight move" (an adjacent plus a diagonal square away). Minimum thermocouple coverage satisfies the thermocouple minimum OPERABILITY requirements of Table 3.2.4-1, but does not meet the "knight move" pattern discussed above. --	--------NOTE----- Frequency applies after TSR 3.2.4.2 is performed. --------------------
	Perform calibration of the PDMS using the movable incore detector system with at least 50% of the detector thimbles and at least 2 detector thimbles per core quadrant, using a minimum of 10 OPERABLE core exit thermocouples with at least 2 thermocouples per quadrant, with THERMAL POWER $\geq$ 25% RTP.	31 EFPD with minimum thermocouple coverage <u>OR</u> 180 EFPD with optimum thermocouple coverage

TSR 3.2.4.4	--------------------------NOTE-------------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION of the PDMS instrumentation input specified in Table 3.2.4-1.	Prior to use for core power distribution measurement <u>AND</u> 24 months thereafter

Table 3.2.4-1
Power Distribution Monitoring System Instrumentation

	Function	Required Channel Inputs
1.	Control Bank Position	4[(a)]
2.	RCS Cold Leg Temperature, Tcold	1
3.	Reactor Power Level	1[(b)]
4.	NIS Power Range Excore Detector Section Signals	6[(c)]
5.	Core Exit Thermocouple Temperatures	10 with $\geq$ 2 per quadrant

(a) Control bank position inputs may be bank positions from either valid Demand Position indications or the average of all valid individual RCCA positions in the bank determined from Rod Position Indication (RPI) System values for each Control Bank. A maximum of 1 rod position indicator may be inoperable when RCCA position indications are being used as input to the PDMS.

(b) Reactor Power Level inputs may be reactor thermal power derived from either a valid secondary calorimetric measurement, the average Power Range Neutron Flux Power, or the average RCS Loop ΔT.

(c) The total must consist of 3 pairs of corresponding upper and lower detector section signals.

Enclosure 6

List of Regulatory Commitments

1. Cycle-specific BEACON calibrations performed before startup and at beginning-of-cycle conditions will ensure that power peaking uncertainties provide 95% probability upper tolerance limits at the 95% confidence level. These calibrations are to be performed using the Westinghouse methodology. Until these calibrations are complete, more conservative default uncertainties will be applied. The calibrations will be documented and retained as records.