WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President Engineering

October 10, 2009

ET 09-0019

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket No. 50-482: Proposed Revision to Technical Specifications for Use of BEACON Power Distribution Monitoring System

Gentlemen:

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests amendment of the Renewed Facility Operating License (No. NPF-42) for the Wolf Creek Generating Station (WCGS) in order to incorporate proposed changes to the WCGS Technical Specifications. Specifically, WCNOC proposes to revise Technical Specification (TS) 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor," TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," for use of the BEACON Power Distribution Monitoring System (PDMS) described in WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," to perform power distribution surveillances.

Attachment I provides the evaluation of the proposed TS changes for the PDMS. Attachment II provides a special evaluation for the PDMS as described in Attachment I. Attachment III provides the existing TS pages marked-up to show the proposed changes. Attachment IV provides a copy of the revised TS pages retyped with the proposed changes incorporated. Attachment V provides for information only the existing TS Bases pages marked-up to show the associated proposed Bases changes. Attachment VI provides marked-up pages from the WCGS Technical Requirements Manual indicating the changes to incorporate appropriate administrative controls and surveillance requirements for the PDMS. The proposed Technical Requirements Manual 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 5.5.14, "Technical Specification (TS) Bases Control Program," at the time the amendment is implemented. Attachment VII provides a list of regulatory commitments made by WCNOC in this submittal.

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It has been determined that his amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. The amendment application was reviewed by the WCNOC Plant Safety Review Committee. In accordance with 10 CFR 50.91, a copy of this application is being provided to the designated Kansas State official.

WCNOC requests approval of this proposed amendment by September 30, 2010. Once approved, the amendment will be implemented within 90 days.

If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Richard D. Flannigan at (620) 364-4117.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachments I - Evaluation of Proposed Change
II - Evaluation for Excluding Power Distribution Monitoring System Instrumentation Requirements from Technical Specifications
III - Markup of Technical Specification Pages
IV - Retyped Technical Specification Pages
V - Markup of Technical Specification Bases Pages (for information only)
VI - Markup of Technical Requirement Manual Pages (for information only)
VII - List of Regulatory Commitments

cc: E. E. Collins (NRC), w/a
T. A. Conley (KDHE), w/a
V. G. Gaddy (NRC), w/a
B. K. Singal (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By________________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 10th day of October, 2009.

NOTARY PUBLIC OFFICIAL SEAL STATE OF KANSAS
RHONDA L. TIEMEYER
MY COMMISSION EXPIRES
January 11, 2010

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2010

EVALUATION OF PROPOSED CHANGE

Subject: Proposed Revision to Technical Specifications for Use of BEACON Power Distribution Monitoring System

1. SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Wolf Creek Nuclear Operating Corporation (WCNOC) requests a change to Wolf Creek Generating Station (WCGS) Technical Specifications (TS). This evaluation supports a request to amend Renewed Facility Operating License(s) NPF-42.

The proposed change would revise TS 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor," TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," for use of the BEACON Power Distribution Monitoring System (PDMS) described in WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," to perform power distribution surveillances.

2. DETAILED DESCRIPTION

Proposed changes to the TSs are as follows:

- TS 3.1.7, "Rod Position Indication"

  For conditions involving inoperable digital control rod position indicators, Required Actions A.1, B.3 and C.1 of Limiting Condition for Operation (LCO) 3.1.7 require plant operators to "verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors." The Required Actions will be revised to state, "Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information." The generic phrase "core power distribution measurement information" would allow the use of an OPERABLE PDMS or the movable incore detectors for verifying the position of the rod with an inoperable digital rod position indicator.

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

  LCO 3.2.1 is revised from $F_Q^C(Z)$ to $F_Q(Z)$. WCNOC letter WO 04-0031, dated October 7, 2004, proposed an editorial change to TS 3.2.1, unrelated to LCO 3.2.1. With the issuance of Amendment No. 159, $F_Q(Z)$ in LCO 3.2.1 had been inadvertently changed to $F_Q^C(Z)$. This change corrects the equation terminology for heat flux hot channel factor.

  The Note to the Surveillance Requirements Table currently states: "During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained." The Note is revised to state: "During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution measurement is obtained."

Surveillance Requirement (SR) 3.2.1.2, for verifying that $F_Q^W(Z)$ is within its limit, is modified by a Note. This Note currently states that if $F_Q^C(Z)$ measurements indicate

maximum over z $\left[\dfrac{F_Q^C(Z)}{K(Z)}\right]$ has increased since the previous evaluation of $F_Q^C(Z)$:

b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate maximum over z $\left[\dfrac{F_Q^C(Z)}{K(Z)}\right]$ has not increased.

The Note in the SR will be revised so that it is worded as follows:

b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive power distribution measurements indicate maximum over z $\left[\dfrac{F_Q^C(Z)}{K(Z)}\right]$ has not increased.

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

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

  The Note to the Surveillance Requirements Table currently states: "During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained." The Note is revised to state: "During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution measurement is obtained." This change would allow the Surveillance to be performed using either the movable incore detectors or an OPERABLE PDMS.

- TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)"

  SR 3.2.4.2 currently states, "Verify QPTR is within limit using the movable incore detectors." It will be revised 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 or an OPERABLE PDMS.

- TS 3.3.1, "Reactor Trip System (RTS) Instrumentation"

  SR 3.3.1.3 requires comparing results from the incore system to the Nuclear Instrument System (NIS) channel output with respect to the indicated axial flux difference (AFD). Specifically, SR 3.3.1.3 currently states, "Compare results of the incore detector measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%." SR 3.3.1.3 will be revised to state, "Compare results of core power distribution measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%."

SR 3.3.1.6 requires periodically calibrating the excore channels against the incore channels. Specifically, SR 3.3.1.6 currently states, "Calibrate excore channels to agree with incore detector measurements." It will be revised 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 above-described TS changes are shown as mark-ups to the current TSs on the pages provided in Attachment III.

The TS Bases will also be revised for consistency with the proposed TS changes. A markup of the TS Bases pages reflecting the needed changes is provided in Attachment IV for information only. In addition to the Bases for TS 3.1.7, TS 3.2.1, TS 3.2.2, TS 3.2.4, and TS 3.3.1, the Bases for TS 3.1.4, "Rod Group Alignment Limits," is also to be revised due to references to "incore flux mapping," etc. in those Bases sections. The TS Bases changes will be implemented in accordance with TS 5.5.14, "Technical Specification (TS) Bases Control Program," at the time the amendment is implemented.

With regard to maintenance, OPERABILITY and control of the PDMS and its associated instrumentation, it has been determined that no TS changes are needed for this purpose since the PDMS does not meet the selection criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs. The evaluation for this determination is provided in Attachment II.

In lieu of TS requirements, requirements for the PDMS and associated instrumentation will be contained in the WCGS Technical Requirements Manual. The changes for incorporating PDMS instrumentation requirements and controls into the Technical Requirements Manual are indicated in Attachment VI. The indicated changes are provided for information only.

3. TECHNICAL EVALUATION

The PDMS to be used at the WCGS utilizes the NRC-approved Westinghouse proprietary computer code, 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 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) thereby 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 operation prediction information system for Westinghouse reactors.

WCNOC proposes to use BEACON to augment the functional capability of the flux mapping system for the purpose of power distribution Surveillances. WCAP-12472-P-A discusses an application of BEACON in which the TSs and core power distribution limits are revised to take credit for continuous monitoring by plant operators. However, WCNOC proposes to use a more conservative application of BEACON in which the core power distribution limits themselves remain unchanged. WCNOC intends to use the BEACON PDMS as the primary method for performing power distribution measurements and Surveillances, and to use the flux mapping system as an alternative for such purposes, when thermal power is greater than 25 percent RATED THERMAL POWER (RTP). At thermal power levels less than or equal to 25 percent RTP, or when the PDMS is inoperable, 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 provide operational support for pressurized water reactors (PWRs). BEACON is an advanced core monitoring and support software package that utilizes existing plant instrumentation for providing incore thermocouple temperatures, reactor coolant system (RCS) cold leg temperatures, control bank positions, power range detector output, and reactor power measurement data to the PDMS. 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 includes an on-line 3-D nodal model 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 since it uses a depleted 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.

WCNOC personnel intend to utilize the BEACON PDMS to take advantage of its capability for 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 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 of the Brookhaven National Laboratory's (BNL) Technical Evaluation Report (TER) for WCAP-12472-P.

> The BEACON core monitoring system uses the NRC-approved Westinghouse SPNOVA nodal method for core power distribution measurements. 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. Continuous monitoring is performed for the third step, using the thermocouples and excores 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 are calculated using the standard Westinghouse methods.

For the application of BEACON at WCGS, credit will not be taken for the continuous monitoring of the power distribution (as described above). Instead, BEACON will be used as a TS monitor for present peaking factor limits, and the transient initial condition limits for WCGS 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 17 of the thermocouples, with 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) or 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's move requirement is not satisfied. The accuracy of the power distribution information with decreased incore or thermocouple detector 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 and thermocouple detectors, 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 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 (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 the 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 presently 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 WCGS 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 (WCAP-12472-P-A 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. (See WCAP-12472-P-A 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 WCAP-12472-P-A. Portions of the BNL TER for WCAP-12472-P relevant to the uncertainty analysis are summarized/excerpted as follows:

> 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 interval.

The failure of (incore and thermocouple) detectors (used by) 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 (correction factor to the assembly axial power distribution to take the power depression due to the grid of the assembly into account). 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 (power depression) 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 (FΔH) and power peaking factor (FQ) 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, 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 pressurized water reactors (PWR) 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 WCGS 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.*

Although not specifically described in this submittal, 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.

2. *In order to ensure 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.*

WCGS utilizes a Westinghouse 4-loop nuclear steam supply system (NSSS) with Westinghouse movable incore instrumentation. All fuel is presently of Westinghouse manufacture. Therefore, WCGS does not differ significantly from the plants that form the WCAP database, and no additional review of WCAP applicability to WCGS is necessary.

During the review of the Westinghouse topical report WCAP-12472-P, the NRC requested additional information on how BEACON treats core loadings with fuel designs from multiple fuel vendors, 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 supplied by multiple vendors 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, WCNOC is proposing only to use BEACON as a core TS monitor for conformance to WCGS's existing limits. The TS changes of concern per this question or condition are not applicable or of concern to the more limited changes being proposed by WCNOC for the intended use of BEACON. Therefore, this condition does not apply to the amendment requested for WCGS.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, General Design Criterion 13 states:

Criterion 13 -- Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Implementation of the PDMS at WCGS does not replace, eliminate, or modify existing plant instrumentation. The PDMS software runs on a workstation connected to the plant process computer. The PDMS combines 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

- Amendment No. 144 was issued on April 2, 2008, for the Commanche Peak Steam Electric Station, Units 1 and 2. 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 of use of a BEACON system for TS monitoring. (ADAMS Accession Number ML080500627)

- Amendment No. 182 was issued on March 21, 2007, for the Callaway Plant. This amendment approved the use of the BEACON Power Distribution Monitoring System (PDMS) for augmenting the functional capability of the neutron flux mapping system for the purposes of power distribution Surveillances. (ADAMS Accession Number ML070460584)

4.3 Significant Hazards Consideration

Wolf Creek Nuclear Operating Corporation (WCNOC) has evaluated whether or not a significant hazards consideration is involved with the proposed Technical Specification changes for supporting use of the BEACON Power Distribution Monitoring System (PDMS), 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. This system utilizes an NRC approved Westinghouse proprietary computer code, i.e., Best Estimate Analyzer for Core Operations - Nuclear (BEACON), to provide data reduction for incore flux maps, core parameter analysis, load follow operation simulation, and core prediction. The PDMS does not provide any protection or control system function. Fission product barriers are not impacted by these proposed changes. The proposed changes occurring with PDMS will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident. The changes associated with the PDMS do not affect plant systems such that their function in the control of radiological consequences is adversely affected. These proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Safety Analysis Report (USAR).

Use of the PDMS supports maintaining the core power distribution within required limits. Further continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This results in more time (i.e., earlier determination of an adverse condition developing) for operator action prior to having an adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an 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

Other than use of the PDMS to monitor core power distribution, implementation of the PDMS and associated Technical Specification changes has no impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with implementation of the PDMS do not result in a change to the design basis of any plant component or system. The evaluation of the effects of using the PDMS to monitor core power distribution parameters shows that all design standards and applicable safety criteria limits are met.

The proposed changes do not result in any event previously deemed incredible being made credible. Implementation of the PDMS will not result in any additional adverse condition and will not result in any increase in the challenges to safety systems. The cycle-specific variables required by the PDMS are calculated using NRC-approved methods. The Technical Specifications 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.

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 Technical Specification requirements and limits remain unchanged, as the Technical Specifications will continue to require operation within the core limits that are based on NRC-approved reload design methodologies. Appropriate measures exist to control the values of these cycle-specific limits, and appropriate actions will continue to be specified and taken for when limits are violated. Such actions remain unchanged.

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

Based on the above evaluations, WCNOC concludes that the activities associated with the proposed amendment present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC 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 CONSIDERATION

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

6.0 REFERENCES

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

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

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

The justification for not including requirements for the Power Distribution Monitoring System (PDMS) and associated instrumentation in the Technical Specifications (TSs) is explained per the evaluation provided below. The purpose of this evaluation is to demonstrate that the structures, systems, or components (i.e., instrumentation) that constitute the PDMS are not required to be contained in the TSs. This evaluation is performed in accordance with the requirements contained in 10 CFR 50.36(c)(2)(ii).

Per 10 CFR 50.36(c)(2)(ii) a TS Limiting Condition for Operation must be established for each item meeting one or more of the following criteria:

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

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

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

The limits for the power distribution parameters $F_Q(Z)$ and $F^N_{\Delta H}$ are operating restrictions which ensure that the accident analyses and assumptions for all applicable, analyzed DBAs remain valid. These limits are included in the TSs. The PDMS supports the capability to monitor core power distribution for verifying conformance to such limits, but it does not control core power distribution and cannot itself cause or effect 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 TSs. Additionally, these parameters or limits 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) A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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 design-basis evaluations. The PDMS has no active/control functions or actuation capability and as such is not part of any primary success path for mitigation of a DBA or transient that either assumes the failure of, or presents the challenge to the integrity of a fission product barrier.

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

The PDMS and its associated instrumentation provide the capability to monitor power distribution parameters at more frequent intervals than is currently required by the TSs, 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 an alternative means for performing core power distribution measurements and related surveillances, as the current means of performing such activities (by use of the movable incore detectors) will still be 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 PDMS instrumentation does not meet any of the criteria for inclusion in the TSs. 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 TSs. Attachment VI provides for information the proposed new TR 3.3.19, "Power Distribution Monitoring System (PDMS)," and associated TRM Bases.

Markup of Technical Specification Pages

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one DRPI per group inoperable for one or more groups.	B.1	Place the control rods under manual control.	Immediately
	<u>AND</u>		
	B.2	Monitor and record RCS T_{avg}.	Once per 1 hour
	<u>AND</u>		
	B.3	Verify the position of the rods with inoperable position indicators indirectly by using ~~movable incore detectors.~~	Once per 8 hours
	<u>AND</u>		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable position indicators indirectly by using ~~movable incore detectors.~~	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

core power distribution measurement information.

(continued)

3.2 POWER DISTRIBUTION LIMITS

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

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

APPLICABILITY: MODE 1.

ACTIONS

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

(continued)

SURVEILLANCE REQUIREMENTS

measurement

-----NOTE-----
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution ~~map~~ is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	----------NOTE---------- If $F_Q^C(Z)$ measurements indicate $$\text{maximum over z} \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the 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 two successive ~~flux maps~~ power distribution measurements indicate $$\text{maximum over z} \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ has not increased. ------------------------------	
	Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> (continued)

SURVEILLANCE REQUIREMENTS

measurement

-----NOTE-----
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution ~~map~~ is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	---NOTES--- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range 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 Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. ------ Verify QPTR is within limit using ~~the movable incore detectors~~ core power distribution measurement information.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	------------------------------NOTES------------------------------ Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculation results exceed power range channel output by more than + 2% RTP.	24 hours
SR 3.3.1.3	------------------------------NOTES------------------------------ Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. -- core power distribution Compare results of the ~~incore detector~~ measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%.	31 effective full power days (EFPD)
SR 3.3.1.4	------------------------------NOTE------------------------------ This Surveillance must be performed on the reactor trip bypass breaker for the local manual shunt trip only prior to placing the bypass breaker in service. -- Perform TADOT.	62 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	------------------------------NOTE------------------------------ Not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 75 % RTP. -- Calibrate excore channels to agree with ~~incore detector~~ core power distribution measurements.	92 EFPD
SR 3.3.1.7	------------------------------NOTES------------------------------ 1. 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. 2. Source range instrumentation shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -- Perform COT.	184 days

(continued)

Retyped Technical Specification Pages

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	Once per 8 hours
	<u>OR</u>		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one DRPI per group inoperable for one or more groups.	B.1	Place the control rods under manual control.	Immediately
	AND		
	B.2	Monitor and record RCS T_{avg}.	Once per 1 hour
	AND		
	B.3	Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	Once per 8 hours
	AND		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	4 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

3.2 POWER DISTRIBUTION LIMITS

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

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

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q^C(Z)$ not within limit.	A.1	Reduce THERMAL POWER $\geq$ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	AND		
	A.2	Reduce Power Range Neutron Flux - High trip setpoints $\geq$ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND		
	A.3	Reduce Overpower ΔT trip setpoints $\geq$ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND		
	A.4	Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

SURVEILLANCE REQUIREMENTS

-----NOTE-----
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution measurement 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 <u>AND</u> Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified <u>AND</u> 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-------------------------------NOTE------------------------------- If $F_Q^C(Z)$ measurements indicate $$\text{maximum over z} \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the 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 two successive power distribution measurements indicate $$\text{maximum over z} \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$ has not increased. ---	
	Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> (continued)

SURVEILLANCE REQUIREMENTS

---NOTE---
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution measurement is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-----------------------------NOTES------------------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range 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 Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. --- Verify QPTR is within limit using core power distribution measurement information.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	------------------------------NOTES------------------------------ Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculation results exceed power range channel output by more than + 2% RTP.	24 hours
SR 3.3.1.3	------------------------------NOTES------------------------------ Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. -- Compare results of the core power distribution measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%.	31 effective full power days (EFPD)
SR 3.3.1.4	------------------------------NOTE------------------------------ This Surveillance must be performed on the reactor trip bypass breaker for the local manual shunt trip only prior to placing the bypass breaker in service. -- Perform TADOT.	62 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	------------------------------NOTE------------------------------ Not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 75 % RTP. -- Calibrate excore channels to agree with core power distribution measurements.	92 EFPD
SR 3.3.1.7	------------------------------NOTES------------------------------ 1. 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. 2. Source range instrumentation shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -- Perform COT.	184 days

(continued)

Markup of Technical Specification Bases Pages (for information only)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

b. The core remains subcritical after accident transients.

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

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from bank D inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 3).

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

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

core power distribution measurement.

BASES

ACTIONS (continued)

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be inserted to approximately 100 steps in order to maintain proper overlap.

Power operation may continue with one RCCA OPERABLE but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 4). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

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

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

BASES

LCO (continued)

These requirements ensure that rod position indication during power operation and startup are accurate, and that, design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

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

or the Power Distribution Monitoring System

A.1

core power distribution measurement using either

When one DRPI per group fails, the position of the rod may still be determined indirectly by ~~use of~~ the movable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 2).

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

Placing the Rod Control System in manual assures unplanned rod motion will not occur. The Immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition. Monitoring and recording Reactor Coolant System T_{avg} help to assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

When more than one DRPI per group fails, the position of the rod(s) can still be determined ~~by use of~~ indirectly by core power distribution measurement using either the moveable incore detectors or the Power Distribution Monitoring System. Based on experience, normal power operation does not require excessive movement of banks. If one or more banks has been significantly moved, the Required Action of C.1 or C.2 is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation for up to 24 hours since the probability of simultaneously having a rod significantly out of position and an event sensitive to that position is small.

C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and B.1, as applicable, are still appropriate but must be initiated promptly under Required Action C.1 to begin indirectly verifying that these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps.

BASES

ACTIONS

C.1 and C.2 (continued)

The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions using either the movable incore detectors or the Power Distribution Monitoring System.

D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod within each affected bank are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

D.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power to ≤ 50% RTP.

E.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Verification at 24, 48, 120, and 228 steps withdrawn for the control banks and at 18, 210, and 228 steps withdrawn for the shutdown banks provides assurance that the DRPI is operating correctly over the full range of indication. Since the DRPI does not display the actual shutdown rod positions between 18 and

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

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

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. 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 TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," and 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 not directly measurable but is inferred from a power distribution map measurement obtained with either the movable incore detector system or the Power Distribution Monitoring System. The results of the three-dimensional power distribution map are analyzed to derive a measured value for $F_Q(Z)$. These measurements are generally taken with the core at or near equilibrium conditions. However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations, such as load following.

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

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

BASES

LCO (continued)

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

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

The actual values of CFQ and K(Z) are given in the COLR.

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

a power distribution measurement

, from which

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

If the power distribution measurement is obtained with the movable incore detector system,

$$F_Q^C(Z) = F_Q^M(Z)\ (1.03)\ (1.05) = F_Q^M(Z)\ (1.0815)$$

s

where 1.03 is a factor that account for fuel manufacturing tolerances and 1.05 is a factor that accounts for flux map measurement uncertainty. (Ref. 4)

INSERT B 3.2.1-3

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

power distribution measurement

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

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

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) information is included in the COLR.

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

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

INSERT B 3.2.1-3

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

$$F_Q^C(Z) = F_Q^M(Z)\ (1.03)\ (1.00 + U_Q/100)$$

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 6.

measurement

BASES

SURVEILLANCE REQUIREMENTS (continued)

power distribution ~~map~~ can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in a reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified within 24 hours from when equilibrium conditions are achieved at RTP (or any other level for extended operation). Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to perform ~~flux mapping.~~ 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 was last measured.

a power distribution measurement.

, as described in the preceeding LCO section

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. ~~Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)(1.0815)$ (Ref. 4). $F_Q^C(Z)$ is then compared to its specified limits.~~

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

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

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required within 24 hours after achieving equilibrium conditions at this higher power level

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

(to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

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

power distribution measurements

SR 3.2.1.2

at or near

measurements.

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

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

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

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

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

The amount of the axial core region that can be excluded during the performance of SR 3.2.1.2 shall not exceed 15% of the upper and lower core regions, and may be reduced on a cycle-specific basis as determined during the core reload design process. The amount of the axial core region that can be excluded during the performance of SR 3.2.1.2 is identified in the COLR. The axial core regions are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making

BASES

REFERENCES

1. 10 CFR 50.46, 1974.

2. USAR, Section 15.4.8.

3. 10 CFR 50, Appendix A, GDC 26.

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

5. Performance Improvement Request 2005-3311.

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

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

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

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

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

either

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD.

measurement

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. Compliance with these LCOs, along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

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

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible

BASES

APPLICABLE SAFETY ANALYSES (continued)

acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank 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 ($F_Q(Z)$)."

either

or the Power Distribution Monitoring System

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 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 is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

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

A power multiplication factor in this equation includes an additional allowance 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 cycle-dependent factor, $PF_{\Delta H}$, specified in the COLR for each 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power.

BASES

ACTIONS

A.1.2.1 and A.1.2.2 (continued)

A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints.

A.2

a power distribution measurement

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, ~~an 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 68 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 72 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 72 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the ~~incore flux map,~~ perform the required calculations, and evaluate $F^N_{\Delta H}$.

power distribution measurement,

A.3

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 identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

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

BASES

ACTIONS (continued)

B.1

When Required Actions A.1.1 through A.3 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

SR 3.2.2.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving Mode 1). The Note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution ~~map~~ measurement can be obtained. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to perform ~~flux mapping.~~ the measurement.

INSERT B 3.2.2-6

~~The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A calculation determines the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ does not require a correction for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit since a measurement uncertainty of 4% has been included in 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.

REFERENCES

1. USAR, Section 15.4.8.

2. 10 CFR 50, Appendix A, GDC 26.

3. 10 CFR 50.46.

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

INSERT B 3.2.2-6

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. A calculation determines the maximum value of $F^N_{\Delta H}$ from the measured power distribution. The measured value of $F^N_{\Delta H}$ must be increased by 4% (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 4) to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

BASES

ACTIONS

A.1 (continued)

within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable THERMAL POWER level. Decreases in QPTR would allow raising the maximum allowable THERMAL POWER level and increasing THERMAL POWER up to this revised limit.

A.2

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

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required ~~Actn~~ 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 the intended operating conditions to support ~~flux mapping~~ a power distribution measurement using either the movable incore detector system or the Power Distribution Monitoring System. 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 ~~flux map~~ power distribution measurement. If these peaking factors are not within their limits, the Required Actions associated with these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

core distribution measurement, using either the movable incore detector system or the Power Distribution Monitoring System,

BASES

ACTIONS (continued)

A.6

If the QPTR remains above the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limit prior to increasing THERMAL POWER to above the limit of Required Action A.1. The process of normalization is accomplished by measuring currents for each detector during ~~flux mapping~~ and using this information to normalize the output from each detector (either through calibration of the NIs or through the use of constants in calculations) 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.6 is modified by two Notes. Note 1 states that excore detectors are not normalized to restore QPTR to within limit 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.5). Note 2 states that if Required Action A.6 is performed, then Required Action A.7 shall be performed. Required Action A.6 normalizes the excore detectors to restore QPTR to within limit, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing ~~flux mapping~~ to verify peaking factors per Required Action A.7. These Notes are intended to prevent any ambiguity about the required sequence of actions.

a power distribution measurement

A.7

Once the excore detectors are normalized to restore QPTR to within limit (i.e., Required Action A.6 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.7 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of achieving equilibrium conditions. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support ~~flux mapping.~~ As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

power distribution measurement, using either the movable incore detector system or the Power Distribution Monitoring System.

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 any tilt remains within its limits.

system or the Power Distribution Monitoring System is

either

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detector~~s are~~ used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. If one of the symmetric thimbles is not available, then other pairs (triples) of symmetric thimbles can be monitored to gain information about the quadrant with the out-of-service thimble, provided the reference case is set up with the same thimble groupings. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

power distribution

core power distribution measurements

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the ~~incore result~~ may be compared against previous ~~flux maps~~ either using the symmetric thimbles as described above or a complete ~~flux map~~.

core power distribution measurement.

REFERENCES

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

BASES

ACTIONS

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

continued unit operation at power levels > 75% RTP. At power levels ≤ 75% RTP, operation of the core with radial power distributions beyond the design limits, at a power level where DNB conditions may exist, is prevented. The 12 hour Frequency is consistent with the Surveillance Requirement Frequency in LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." Required Action D.1.1 has been modified by a Note which only requires SR 3.2.4.2 to be performed if the Power Range Neutron Flux input to QPTR becomes inoperable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using ~~the movable incore detectors~~ may not be necessary.

core power distribution measurement information

The NIS power range detectors provide input to the Rod Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 12.

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Seventy-eight (78) hours are allowed to place the plant in MODE 3. The 78-hour Completion Time includes 72 hours for channel corrective maintenance, and an additional 6 hours for the MODE reduction as required by Required Action D.2. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 12 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 12 hour time limit is justified in Reference 12.

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux - Low;

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued

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

core power distribution measurement, obtained using either the movable incore detector system or the Power Distribution Monitoring System,

SR 3.3.1.3

SR 3.3.1.3 compares the ~~incore system~~ to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. The excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

The Note to SR 3.3.1.3 clarifies that the Surveillance is required only if reactor power is ≥ 50 % RTP, and that 24 hours is allowed for performing the first Surveillance after reaching 50% RTP. This Note allows power ascensions and associated testing to be conducted in a controlled and orderly manner, at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use. Due to such effects as shadowing from the relatively deep control rod insertion and, to a lesser extent, the axially-dependent radial leakage which varies with power level, the relationship between the incore and excore indications of axial flux difference (AFD) at lower power levels is variable. Thus, it is acceptable to defer the calibration of the excore AFD against the incore AFD until more stable conditions are attained (i.e., withdrawn control rods and a higher power level). The AFD is used as an input to the Overtemperature ΔT reactor trip function and for assessing compliance with LCO 3.2.3., "AXIAL FLUX DIFFERENCE (AFD)." Due to the DNB benefits gained by administratively restricting power level to 50% RTP, no limits on AFD are imposed below 50% RTP by LCO 3.2.3; thus, the proposed change is consistent with the LCO 3.2.3 requirements below 50% RTP. Similarly, sufficient DNB margins are realized through operation below 50% RTP that the intended function of the Overtemperature ΔT reactor trip function is maintained, even though the excore AFD indication may not exactly match the incore AFD indication. Based on plant operating experience, 24 hours is a

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

core power distribution, measured using either the movable incore detector system or the Power Distribution Monitoring System.

SR 3.3.1.6 is a calibration of the excore channels to the ~~incore channels.~~ If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the ~~incore detector~~ core power distribution measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER ≥ 75% RTP. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to ~~perform flux mapping.~~ obtain a core power distribution measurement. The SR is deferred until a scheduled testing plateau above 75% RTP is attained during a power ascension. During a typical power ascension, it is usually necessary to control the axial flux difference at lower power levels through control rod insertion. After equilibrium conditions are achieved at the specified power plateau, a ~~flux map~~ core power distribution measurement must be taken and the required data collected. The data is typically analyzed and the appropriate excore calibrations completed within 48 hours after achieving equilibrium conditions. An additional time allowance of 24 hours is provided during which the effects of equipment failures may be remedied and any required re-testing may be performed.

The allowance of 72 hours after equilibrium conditions are attained at the testing plateau provides sufficient time to allow power ascensions and associated testing to be conducted in a controlled and orderly manner at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be OPERABLE for subsequent use.

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

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 184 days.

A COT is performed on each required channel to ensure the channel will perform the intended Function.

Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

Markup of Technical Requirement Manual Pages (for information only)

3.3 INSTRUMENTATION

3.3.10 Movable Incore Detectors

TR 3.3.10 The Moveable Incore Detection System shall be OPERABLE.

APPLICABILITY: When the Movable Incore Detection System 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 $F_Q(Z)$, or

e. Calibrating the Power Distribution Monitoring System.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Movable Incore Detection System inoperable.	A.1 Suspend use of the system for monitoring, calibration, and measurement functions.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.10.1	Normalize each detector output when required for monitoring, calibration and measurement functions.	24 hours

3.3 INSTRUMENTATION

3.3.19 Power Distribution Monitoring System (PDMS)

TR 3.3.19 The PDMS shall be OPERABLE with the minimum required channels shown in Table TR 3.3.19-1.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 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. PDMS inoperable.	A.1	Suspend use of the PDMS for monitoring, calibration, and measurement functions.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.19.1	Perform a CHANNEL CHECK of the PDMS instrumentation input specified in Table TR 3.3.19-1.	24 hours

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.3.19.2	Perform a calibration of the PDMS.	Once after each refueling AND 31 EFPD thereafter with minimum thermocouple coverage OR 180 EFPD thereafter with optimum thermocouple coverage
TSR 3.3.19.3	-----------------------------NOTE------------------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform a CHANNEL CALIBRATION of the PDMS instrumentation input specified in Table TR 3.3.19-1.	18 months
TSR 3.3.19.4	Technical Specification (TS) SR 3.1.7.1 is applicable for the Control Bank Position instrumentation to be OPERABLE.	In accordance with applicable TS SRs

Table TR 3.3.19-1
Power Distribution Monitoring System Instrumentation

FUNCTION/INPUT	REQUIRED CHANNELS	TECHNICAL SURVEILLANCE REQUIREMENTS
1. Control Bank Position	4[(a)]	TSR 3.3.19.1 TSR 3.3.19.4
2. RCS Cold Leg Temperature, T_{cold}	2	TSR 3.3.19.1 TSR 3.3.19.3
3. Reactor Power Level	1[(b)]	TSR 3.3.19.1 TSR 3.3.19.3
4. NIS Power Range Excore Detector Section Signal	6[(c)]	TSR 3.3.19.1 TSR 3.3.19.3
5. Core Exit Thermocouple Temperatures	17 with $\geq$ 2 per core quadrant	TSR 3.3.19.1 TSR 3.3.19.3

(a) Control Bank Position inputs may be bank positions from either valid demand position indications or the average of all valid individual rod cluster control assembly (RCCA) positions in the bank determined from Digital Rod Position Indication System values for each control bank. A maximum of one rod position indicator per group 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 secondary calorimetric measurement, the average Power Range Neutron Flux power, or the average RCS loop ΔT.

(c) The total must consist of three pairs of corresponding upper and lower detector sections.

B 3.3 INSTRUMENTATION

TR B 3.3.19 Power Distribution Monitoring System (PDMS)

BASES

BACKGROUND

The Power Distribution Monitoring System (PDMS) generates a continuous measurement of the core power distribution using the methodology documented in Reference 1. The measured core power distribution is used to determine the most limiting core peaking factors, $F^N_{\Delta H}$ and $F_Q(Z)$. The most limiting measured core peaking factor values are used to verify that the reactor is operating within the design limits.

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

In order for the PDMS to accurately determine the peaking factor values, the core power distribution measurement software requires accurate information about the current reactor power level, average reactor vessel inlet temperature, control bank positions, the Power Range Detector calibrated voltage values, and measured temperatures from Core Exit Thermocouples (T/C).

APPLICABLE SAFETY ANALYSES

The PDMS is used for periodic measurement of the core power distribution to confirm operation within design limit, and periodic calibration of the excore detectors. This system does not initiate any automatic protection action. The PDMS is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient (Reference 2).

TR

The TR requires the PDMS to be OPERABLE with the minimum required channels shown in Table TR 3.3.19-1. The PDMS is OPERABLE when the required channel inputs are available, the calibration data set is valid, and the reactor power level is at least 25% of RTP. The PDMS must be calibrated above 25% RTP to assure the accuracy of the calibration data set which can be generated from the incore flux map, core exit thermocouples, and the other input channels. Below 25% RTP, the PDMS is inoperable, since the calculated power distribution is of reduced accuracy, and may not be used to demonstrate compliance with the LCO limits.

BASES

TR (continued)

This ensures that the measured plant and core condition information input to the core power distribution measurement software with the PDMS OPERABLE is adequate to accurately calculate the core peaking factors. The peaking factor calculations include measurement uncertainty that bounds the actual measurement uncertainty of an OPERABLE PDMS.

APPLICABILITY

The PDMS must be OPERABLE in MODE 1 with THERMAL POWER $\geq$ 25% when it is used for recalibration of the Excore Neutron Flux Detection System, or monitoring of QPTR, or measurement of $F^N_{\Delta H}$ or $F_Q(Z)$.

ACTIONS

A.1

An inoperable PDMS cannot be used for recalibration of the Excore Neutron Flux Detection System, or monitoring of QPTR, or measurement of $F^N_{\Delta H}$ or $F_Q(Z)$. Required Action A.1 requires the immediate suspension of the use of the PDMS for these activities if the system is inoperable.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.19.1

Performance of a CHANNEL CHECK at a frequency of 24 hours ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels. A CHANNEL CHECK will detect gross channel failure; thus it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

The Frequency is based on the need to establish that the required inputs to the PDMS are valid when using the PDMS to obtain a core power distribution measurement to be used to confirm that the reactor is operating within design limits.

TSR 3.3.19.2

Upon initial plant startup following refueling, the PDMS uses a calibration data set calculated by the core designer for the new core.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.19.2 (continued)

An incore flux map for PDMS calibration may be obtained above 25% RTP. The initial calibration data set generated in each operating cycle must utilize incore flux measurements from at least 75% of the incore thimbles, with at least two incore thimbles in each core quadrant. Subsequent to the initial calibration, the calibration data set can be updated using incore flux measurements from at least 50% of the incore thimbles, with at least two incore thimbles in each core quadrant. The incore flux measurements in combination with inputs from the Table TR 3.3.19-1 channels are used to generate the updated calibration data set, including the nodal calibration factors and the thermocouple mixing factors.

Following the initial calibration, the Frequency is 31 EFPD when the minimum core exit thermocouple coverage is available for the PDMS calibration. The minimum thermocouple coverage consists of at least 17 thermocouples with a minimum of two per core quadrant.

Following the initial calibration, the Frequency is 180 EFPD when the optimum core exit thermocouple coverage is available for the PDMS calibration. The optimum thermocouple coverage consists of at least 17 thermocouples with a minimum of two per core quadrant, distributed such all interior fuel assemblies (coverage of fuel assemblies with a face along the baffle is not required) are within a chess knight's move of a thermocouple.

TSR 3.3.19.3

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter with the necessary range and accuracy. This SR is modified with a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)	<u>TSR 3.3.19.4</u> Verfication that the Digital Rod Position Indication (DRPI) agrees with the demand position ensures that the DRPI and demand position indication is operating correctly. Refer to the corresponding Bases for Technical Specification 3.1.7 for a discussion of SR 3.1.7.1.	
REFERENCES	1. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994. 2. Amendment No. XXX, [date].	

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by WCNOC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Richard Flannigan at (620) 364-4117.

REGULATORY COMMITMENT	DUE DATE/EVENT
Once approved, the amendment will be implemented within 90 days.	Within 90 days of approval
Although not specifically described in this submittal, 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 approved methodology. Until these calibrations are complete, more conservative default uncertainties will be applied. The calibrations will be documented and retained as records.	Implementation of Amendment