July 9, 2009

TVA-WBN-TS-09-07 10 CFR 50.90

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

Gentlemen:

In the Matter of	)	Docket No. 50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT UNIT 1 - TECHNICAL SPECIFICATIONS CHANGE - "CORE POWER DISTRIBUTION MONITORING SYSTEM"

Pursuant to 10 CFR 50.90, Tennessee Valley Authority (TVA) requests a Technical Specifications (TS) change, WBN-TS-09-07, for Watts Bar Nuclear Plant Unit 1 Operating License NPF-90. Specifically, TVA proposes to revise TS 1.1, "Definitions," TS 3.1.8, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor," TS 3.2.4, "Quantrant Power Tilt Ratio (QPTR)", and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation."

The proposed TS change will allow use of a dedicated on-line core power distribution monitoring system (PDMS) to enhance surveillance of core thermal limits. The PDMS to be used at Watts Bar Nuclear Plant is the NRC-approved Westinghouse proprietary computer code, Best Estimate Analyzer for Core Operations - Nuclear (BEACON™).

TVA intends to use the BEACON PDMS as the primary method for performing core power distribution measurements and surveillances when thermal power is greater than 25 percent rated thermal power (% RTP). The movable incore detector system will be used for periodic calibration of the PDMS when thermal power is greater than 25% RTP. Additionally, the movable incore detector system will be used whenever the PDMS is inoperable or whenever a power distribution measurement is obtained with thermal power less than or equal to 25% RTP.

Enclosure 1 to this letter provides a detailed description and technical evaluation of the proposed TS changes for the PDMS, including TVA's determination that the proposed changes involve no significant hazards consideration. Enclosure 2 provides the affected, existing TS pages marked-up to show the proposed changes. Enclosure 3 contains annotated versions of the appropriate TS Bases pages for information only. Enclosure 4 documents TVA's conclusion that the PDMS does not meet the selection criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. Enclosure 5 and 6 are provided for information only and contain annotated versions of the affected pages for the Technical Requirements Manual (TRM), and the TRM Bases, respectively.

In accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and attachments to the Tennessee State Department of Public Health.

TVA requests approval of this TS change by July 2010 and that implementation of the revised TS be within 90 days of NRC approval.

There are no regulatory commitments associated with this submittal. If you have any questions about this change, please contact Robert Clark at (423) 365-1818 or Mike Brandon, Site Licensing and Industry Affairs Manager, at (423) 365-1824.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 9th day of July, 2009.

Sincerely,

Mike Skaggs
Site Vice President
Watts Bar Nuclear Plant

Enclosures:

1. TVA Evaluation of Proposed Technical Specifications Change
2. Proposed Technical Specifications Change (Mark-Up)
3. Proposed Technical Specifications Bases Change (Mark-Up)
4. Evaluation for Excluding PDMS Requirements from the Technical Specifications
5. Proposed Technical Requirements Change (Mark-Up)
6. Proposed Technical Requirements Bases Change (Mark-Up)

cc: See page 3

Enclosure
cc (Enclosures):

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Sequoyah Licensing Files, OPS 4C-SQN
EDMS, WT 3B-K

ENCLOSURE 1

TVA EVALUATION OF PROPOSED TECHNICAL SPECIFICATIONS CHANGE WBN-TS-09-07

1.0 SUMMARY DESCRIPTION

The proposed Technical Specifications (TS) change would allow use of a dedicated on-line core power distribution monitoring system (PDMS) to enhance surveillance of core thermal limits. The PDMS to be used at Watts Bar Nuclear Plant is the NRC-approved Westinghouse proprietary computer code, Best Estimate Analyzer for Core Operations - Nuclear (BEACON^{TM}).

The BEACON PDMS is an advanced core monitoring and support software package that utilizes existing plant instrumentation to generate detailed three dimensional (3D) core power distributions for comparison with core thermal limits.

Core thermal limits or peaking factors are dependent upon fuel manufacturing tolerance, accuracies of the core physics/thermo hydraulic models, and core instrumentation. Fuel manufacturing tolerance is vendor specific and is controlled by the vendor's quality assurance program. The accuracy of the core physics/thermo hydraulic models is determined by benchmarking with other qualified codes and comparing calculated core parameters with actual plant data. Plant instrumentation accuracies are based on well established calibration methodologies using vendor supplied data (reference accuracies) and instrument drift based on operating experience. Total uncertainty is quantified by propagating the core model and instrumentation uncertainties through the BEACON system using a Monte Carlo statistical simulation. The results are used to establish the appropriate tolerance for the measured hot channel factors. Details of the methodology used to determine core thermal limit uncertainties are described in Ref. 1.

The BEACON PDMS is also capable of providing fuel burnup distribution, burnable absorber depletion distribution, xenon and samarium concentration, fission product concentration, estimated critical condition for return to criticality, minimum soluble boron concentration to meet shutdown margin requirements, and load follow simulation. This in-depth look at the core provides additional insights into tritium production core behavior and enhanced reactor operations.

TVA intends to use the BEACON PDMS as the primary method for performing core power distribution measurements and surveillances when thermal power is greater than 25 percent rated thermal power (% RTP). The movable incore detector system will be used for periodic calibration of the PDMS when thermal power is greater than 25% RTP. Additionally, the movable incore detector system will be used whenever the PDMS is inoperable or whenever a power distribution measurement is obtained with thermal power less than or equal to 25% RTP.

The proposed TS changes will revise the limiting condition for operations (LCOs) for the following categories; (1) reactivity control systems, (2) power distribution limits, and (3) instrumentation.

2.0 DETAILED DESCRIPTION

2.1 Background

Westinghouse reactors have two neutron flux measuring systems for power operations, incore and excore. These, however, do not by themselves provide direct, continuous determination of core power distribution or provide direct relationship to fuel safety limits, i.e., peak power density or departure from nucleate boiling (DNB).

Power distribution in Westinghouse reactors is currently determined and controlled by a combination of:

(1) Continuous monitoring with four quadrant-base, split excore neutron detectors,

(2) Periodic (monthly) incore neutron flux mapping using incore detectors traveling through thimbles within the fuel assemblies,

(3) Limits on control rod alignment, sequence, overlap, and insertion as a function of power level,

(4) Controlling the axial flux difference as measured by the excore detectors,

(5) Maintaining quadrant symmetry as measured by excore readings,

(6) Determining an upper limit on axial power peaking (as a function of core height) via preoperational calculations of power distribution,

(7) Maintaining operation within the power peaking limits by using, for example, constant axial offset control or relaxed axial offset control.

The current TS accommodate and tie together this rather indirect system of pre-calculations, measurements, and controls to set acceptable limits on power density and heat flux in order to meet core safety limits. As a consequence, the controls on power operations are generally more restrictive than would be required if a more direct power distribution measurement process was used.

As described in WCAP-12472- P-A, the Westinghouse BEACON 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 to generate detailed 3D core power distribution for comparison with core thermal limits. The data provided by existing instruments (i.e., core exit thermocouple, core inlet temperature, rod position, control bank position, excore power range detector, and reactor power) are processed by the plant integrated computer system in the form of a file that BEACON can interpret.

By coupling the calculated 3D power distribution with an on-line evaluation, actual core power distribution margins can be better understood. This methodology improves the quality of the surveillance process since it uses a depleted model to match the actual operational profile. BEACON is calibrated periodically using the movable incore detectors to force the calculated power distribution to match the measured power distribution and to obtain calibration data for the core exit thermocouples and excore detectors.

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

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

The BEACON-TSM (Tech Spec Monitor) system level was developed to provide licensees with the functionality needed to integrate BEACON into the plant Technical Specifications (Tech Specs) for monitoring of current Tech Spec thermal power limits such as peak linear power density (F_Q) and peak enthalpy rise ($F_{\Delta H}$). BEACON-TSM includes all of the base functionality in the BEACON-OLM level. Added to this are the procedures, system operational status information and on-line calculations needed to provide the core monitoring capability for Tech Specs compliance. The licensing of BEACON for core monitoring allows the BEACON on-line monitoring functions to potentially eliminate most flux maps for normal and off-normal Tech Spec thermal power limit verification. Once integrated into the plant TS and procedures, the BEACON-TSM system has the potential to provide the following features:

- Essentially continuous monitoring of the core power distribution

- Increased interval for flux maps (using incore detectors) from 31 EFPDs to 180 EFPDs (flux maps are only required for BEACON calibration, when thermal power is less than 25% RTP, or when PDMS is inoperable)

- Verification of the position of a rod with an inoperable rod position indicator (RPI) using the BEACON core monitor function

- Reduced incore detector instrumentation requirements to 50% after initial calibration for a fuel cycle. The BEACON system uses surface spline fitting to compensate for sparse instrumentation and automatically adjusts the applied thermal limit uncertainties allowing for operation with reduced instrumentation

The BEACON-DMM (Direct Margin Monitor) system level was developed to provide licensees with the full functionality and benefits of the BEACON license granted by the NRC. BEACON-DMM includes all of the functionality of BEACON-TSM and also provides for direct monitoring and use of DNBR as a thermal limit in the plant TS. The DMM level of BEACON will also perform continuous on-line margin trade-off between thermal parameters such as $F_{\Delta H}$ and F_Q while providing continuous verification of the TS DNBR limit. This allows nearly all available margins to be used for core operations or for fuel cycle cost benefits on a cycle specific basis. The licensee can then choose how and when to use the available margin. The DMM level requires that the BEACON system be used by the reactor operators to monitor reactor core conditions and determine responses to TS limit alarms. BEACON-DMM has the potential to provide the following additional features over the TSM level:

- Continuous verification of the DNBR limit
- Continued operation with a misaligned rod without a reduction to 75% RTP with the BEACON system operational
- On-line trade-off of margin between core thermal limits such as peak linear power density (F_Q) and peak rod power ($F_{\Delta H}$) when DNB margin is available
- Elimination of the axial flux difference (AFD) TS requirements - The continuous verification of the DNBR limit will make the AFD TS requirement redundant and unnecessary.
- Relaxed Quadrant Power Tilt Ratio (QPTR) requirements - The continuous verification of the DNBR limit will allow the QPTR requirement to be eliminated in most cases or relaxed to support the worst case accident analysis.

WBN is seeking the “BEACON-TSM” application of the PDMS as opposed to the “BEACON-DMM” application. The “BEACON-TSM” application does not provide continuous monitoring of DNBR limits or control room annunciators as described above. Instead, the PDMS will be used as a Tech Spec monitor for present peaking factor limits. Consequently, this license amendment request does not seek to relax the requirement to limit the transient initial conditions via power distribution control.

2.2 Technical Specification Changes

TVA intends to use the BEACON PDMS as the primary method for performing core power distribution measurements and surveillances when thermal power is greater than 25 percent rated thermal power (% RTP). The movable incore detector system will be used for periodic calibration of the PDMS when thermal power is greater than 25% RTP. The PDMS must be calibrated to a flux map obtained above 25% RTP to ensure the accuracy of the calibration data set which is generated from the incore flux map, core exit thermocouples, excore detectors and other input channels.

Additionally, the movable incore detector system will be used whenever the PDMS is inoperable or whenever a power distribution measurement is obtained with thermal power less than or equal to 25% RTP.

The Technical Specifications that are to be revised for implementation of the PDMS are:

1.1 Definitions
3.1.8 Rod Position Indication
3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)
3.2.4 Quadrant Power Tilt Ratio (QPTR)
3.3.1 Reactor Trip System (RTS) Instrumentation

The PDMS software and hardware changes have already been implemented at WBN. Movable incore detectors will continue to be used for the monthly flux map surveillances prior to issuance and implementation of the license amendment.

The following is a detailed description of the proposed TS changes given in Enclosure 2:

2.2.1. <u>TS 1.1, Definition</u>

The term "PDMS" with the following definition will be added to TS 1.1 to read:

"The Power Distribution Monitoring System (PDMS) is a real-time three dimensional core monitoring system. The system utilizes existing core instrumentation data and an on-line neutronics code to provide surveillance of core thermal limits."

This change is necessary to avoid using a commercial product name or trademark in the TS.

2.2.2. <u>TS 3.1.8, Rod Position Indication</u>

For conditions involving inoperable Analog Rod Position Indication (ARPI), Required Actions A.1, A.2.1, A.2.3, and B.1 of TS 3.1.8 require plant operators to:

"Verify the position of the rod(s) with (an) inoperable position indicator(s) by using movable incore detectors."

The Required Actions will be revised to state:

"Verify the position of the rod(s) with (an) inoperable position indicator(s) by using either the movable incore detectors or the PDMS."

2.2.3. <u>TS 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$)</u>

Surveillance Requirement (SR) 3.2.1.2, for verifying that $F^W_Q(Z)$ is within its limit, is modified by a Note. This Note currently states that if $F^C_Q(Z)$ is within limits and measurements indicate that

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

$F^C_Q(Z)$, plant operators are required (per part "b" of this Note) to do the following:

b. “Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate

maximum over z $\left[\frac{F_Q^C(Z)}{K(z)}\right]$ has not increased.”

This requirement will be revised so that it is worded as follows:

b. “Repeat SR 3.2.1.2 once per 7 EFPD using either the movable incore detectors or the PDMS until two successive power distribution measurements indicate

maximum over z $\left[\frac{F_Q^C(Z)}{K(z)}\right]$ has not increased.”

This change would allow the surveillance to be performed using either the movable incore detectors or an operable PDMS.

2.2.4. 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 either the movable incore detectors or the PDMS.”

This change would allow the surveillance to be performed using either the movable incore detectors or an operable PDMS.

2.2.5. TS 3.3.1, Reactor Trip System (RTS) Instrumentation

SR 3.3.1.3 requires comparing results from the incore detector system to the nuclear instrumentation system (NIS) channel output with respect to the indicated axial flux difference (AFD).

a. Specifically, SR 3.3.1.3 currently states:

“Compare results of the incore detector measurements to NIS AFD.”

SR 3.3.1.3 will be revised to state:

“Compare results of the incore detector or PDMS measurements to NIS AFD.”

b. SR 3.3.1.6 requires periodically calibrating the excore channels to the incore detector measurements. 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 incore detector or PDMS measurements."

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

Markup of the corresponding TS Bases pages reflecting the above TS changes is provided in Enclosure 3 for information only. In addition to these Bases changes, the Bases for TS 3.1.5, "Rod Group Alignment Limits," TS 3.1.9, "PHYSICS TESTS Exceptions - MODE 1," and TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," were also revised due to references to "incore flux mapping." The TS Bases will be implemented in accordance with TS 5.6, "Technical Specifications (TS) Bases Control Program," as part of implementing the requested amendment following NRC approval and issuance of the requested amendment.

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 BEACON-TSM version of the PDMS does not meet the selection criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the Technical Specifications. The evaluation for this determination is provided in Enclosure 4.

In lieu of TS requirements, operability requirements for the PDMS and associated instrumentation will be contained in the WBN Technical Requirements Manual (TRM). The TRM is dedicated to providing administrative controls and operational/testing requirements as appropriate for equipment not required to be included in the TS. Markup changes to the TRM and TRM Bases are provided for information only in Enclosure 5 and 6, respectively, and are controlled in accordance with 10 CFR 50.59.

In summary, the proposed license amendment would allow use of the Westinghouse PDMS (BEACON-TSM) to perform power distribution surveillances provided that the PDMS is operable. The proposed amendment would also allow (or continue to allow) the use of the movable incore detector system for meeting power distribution surveillances and TS actions, in addition to its use for calibration of the PDMS and excore channels.

3.0 TECHNICAL EVALUATION

As noted above, the requested license amendment will allow utilization of the BEACON PDMS at WBN to provide operational support for the facility. The BEACON PDMS maintains an on-line 3D nodal model that is continuously updated to reflect the current plant operating conditions. The technical evaluation of the acceptability of using the BEACON PDMS to monitor core thermal limits is based on the Technical Evaluation Report developed by Brookhaven National Laboratory for WCAP-12472-P-A.

3.1 BEACON Neutronics Code

The BEACON core monitoring system uses the NRC-approved SPNOVA nodal method to calculate the 3D core power distribution. The data libraries and core models used in the SPNOVA code are consistent with the NRC approved Phoenix/Advance Nodal Code (ANC) and have been extensively benchmarked against operating reactor measurements.

The original SPNOVA method used one radial node per assembly coupled with a Green's function solution that simplified the numerical analysis by eliminating the inner iterations (Ref.1). Due to workstation advancements, coupled with improvements in numerical solution techniques, the SPNOVA nodal method described above was replaced with the two group, 3D Nodal Expansion Method (NEM) which is the same methodology used in the Phoenix/ANC neutronics code approved by the NRC. These changes to the SPNOVA neutronics code were reviewed and approved by the NRC as documented in the SER for Addendum 1-A to WCAP-12472-P-A (Ref. 2).

Use of the NEM methodology in the SPNOVA code with four radial nodes per assembly allowed plant personnel using the BEACON system to use the same model and neutronic solution method that are used by the core design group to perform reload safety analysis work. Therefore, all predictive calculations performed by the plant reactor engineering personnel have the same pedigree and accuracy as the calculations performed by the core design personnel.

Because WBN is a tritium production core, the Phoenix-L/ANC-L code is used to account for the tritium isotopic chain produced in the Tritium Producing Burnable Absorber Rods (TPBARs). Westinghouse has confirmed that core designs that use TPBARs do not differ significantly from traditional core designs and that the Phoenix-L/ANC-L codes do not change the licensed ANC methodology as documented in Ref. 3.

The BEACON 6.2.0 and SPNOVA 7.6.0 codes were updated in 2003 to include the tritium isotopic chains consistent with those in the Phoenix-L/ANC-L code. This update allowed the BEACON system to accurately track the depletion and isotopic changes due to the TPBARs consistent with the Phoenix-L/ANC-L code for core depletion calculations, flux map analysis and on-line core monitoring (see Ref. 11).

3.2 BEACON Calibration / Monitoring Methodology

The BEACON calibration/monitoring process is carried out in three steps and is outlined below. A detailed discussion of the power distribution calibration/monitoring process is given in Ref. 1.

Step 1: Calibration Factor Update using Incore Flux Map

Incore flux maps are used approximately every three months to determine calibration factors for the SPNOVA nodal model, core exit thermocouples, and the excore detectors. The calibration factors associated with the nodal model are defined as the ratio of the measured incore detector reaction rates to the SPNOVA predicted detector reaction rates. The corrected 3D power distribution is obtained at the time of the flux map by multiplying the predicted power distribution by the nodal calibration factors.

These calibration factors are extended to the non-instrumented assemblies using the surface spline fit method outlined in Section 3.2 of Ref. 1.

The core exit thermocouple calibration factors (mixing factors) are defined as the ratio of the inferred assembly power to the relative enthalpy rise as measured by the assembly inlet and outlet temperatures. The inferred assembly power is the assembly axially integrated power derived from the corrected 3D power distribution described above. Calibration factors for the excore detectors are based on the correlation between the 3D nodal model peripheral axial offset and the axial offset measured by the excore detector signals.

Step 2: Nodal Model Update

The 3D nodal model is depleted approximately every 15 minutes by following the core operation history. The nodal calibration factors determined in Step 1 are applied to the calculated 3D power distribution to obtain a best estimate power distribution. The radial power distribution for the instrumented assemblies is then further adjusted by the core exit thermocouple measurements using appropriate mixing factors. The corrections are extended to non-instrumented assemblies using surface spline fits. The model is also adjusted to reproduce the axial offset as measured by the excore detectors (see response to RAI question 4 in Ref. 1). This revised 3D power distribution will serve as the reference power distribution for the frequent power distribution updating described in Step 3.

Step 3: Continuous Power Distribution Update

The reference power distribution in Step 2 is updated every minute to represent the latest power distribution using measurements from the core exit thermocouples and the excore detectors to adjust the radial and axial power distribution. The radial power distribution for the instrumented assemblies is adjusted as in Step 2 to reproduce the actual assembly power. Again an interpolation process is used for the non-instrumented assemblies. The axial power distribution is adjusted by adding a sinusoidal harmonic term to preserve the axial offset measured by the excore detectors.

If the core power level or axial offset change by a few percent the reference power distribution in Step 2 is automatically regenerated using the 3D nodal model and the latest core operation history.

Generation of a new reference power shape (Step 2) is performed as a parallel operation without hindering Step 3 operation. As soon as generation of the power shape of Step 2 is completed, the reference shape is replaced. The power distribution calibration/monitoring process is depicted in Figure 3-2 of Ref.1.

3.3 BEACON Core Peaking Factor Measurement Uncertainty

Core thermal limit uncertainties are dependent upon the accuracies of the core physics/thermo hydraulic models, plant instrumentation accuracies and the number and distribution of incore detectors and core exit thermocouples available. The accuracy of the core physics/thermo hydraulic models was determined by benchmarking with other qualified codes and comparing calculated core parameters with actual plant data. Plant instrumentation accuracies are based on well established methodologies using vendor generic data (reference accuracies) and instrument drift based on operating experience. Total uncertainty is quantified by propagating the model and instrumentation uncertainties through the BEACON system using a Monte Carlo statistical simulation. The results are used to define the appropriate tolerance factors for the core peaking factors, i.e. $F^M_Q(Z)$ and $F^N_{\Delta H}$.

The following summary/excerpts from the Brookhaven National Laboratory's (BNL) Technical Evaluation Report (TER) for WCAP-12472-P-A are applicable to the updated BEACON.

> *As an initial assessment of the power distribution calculation, Westinghouse has performed detailed comparisons of BEACON to the predictions of the INCORE System (References - 6 and 7) presently used at Westinghouse plants. These comparisons were made for three plants over four cycles, and included a range of fuel burnup, core loadings, power level and control rod insertion. For the high powered assemblies (P>1), BEACON reproduced a set of axially integrated measurements to within a few percent. The BEACON and INCORE axially integrated assembly powers also agreed to within a few percent for a sample of high powered assemblies.*

The following is an excerpt from Section 4.1.1 of Westinghouse's WCAP-12472-P, BEACON: Core Monitoring and Operations Support System.

> *The assemblywise (axially integrated) reaction rates were compared between prediction and measurement... The standard deviations of the percent differences for all plants analyzed are summarized in Table 4-5. In the table, two statistics are shown: one is for all measurements, the other is for assemblies with powers greater than the average (1.0) value. The averages of the standard deviation are 1.5% for high power assemblies and 2% for all measured assemblies.*

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, reaction rates, 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-A relevant to the uncertainty analysis are 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 in the BEACON system results in a relaxation of the local calibration to measurement, and an increase in the power distribution uncertainty. The effect of random failures of the incore and thermocouple detectors on the assembly power was evaluated for failure rates of up to 75%. It is noteworthy that the assembly power uncertainty was found to increase linearly with incore* [detector] *failure and quadratically with the failure of thermocouples.*

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 ~ twenty highest powered assemblies.

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

The enthalpy-rise ($F_{\Delta h}$) and power peaking factor (F_Q) uncertainties are determined by a statistical combination of the assembly peaking factor, axial peaking factor, calibration interval, inoperable detector and local power peaking component uncertainties.

Operability and calibration of the PDMS is also dependent on the number and distribution of available core exit thermocouples. The criteria for the core exit thermocouples with PDMS operable require that at least 25% of the thermocouples, with at least two per quadrant, be operable. An added requirement that the thermocouple pattern cover all internal fuel assemblies within a chess knight's move (an adjacent plus a diagonal square away) of an operable thermocouple, otherwise a more frequent calibration is required. With the minimum operable thermocouples satisfying the chess knight's move pattern, calibration with the movable incore detectors is required every 180 Effective Full Power Days (EFPD). Calibration is required every 31 EFPD, however, when the chess knight's move pattern 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 BNL TER section 2.3). The analysis concluded that the minimum available incore and core exit thermocouple detectors, when coupled with the increased uncertainty penalties, provide reasonable and acceptable power distribution information.

3.4 Acceptance Criteria/Conditions

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

WBN utilizes a Westinghouse 4-loop nuclear steam supply system (NSSS) with Westinghouse movable incore instrumentation. All fuel is presently of Westinghouse manufacture.

WBN does utilize TPBARs in the core design which behaves similarly to standard burnable poison rodlet assemblies. As a consequence, the WBN core designs use the PHOENIX-L and ANC-L codes which have been updated to add isotopic chains for tritium. Westinghouse has confirmed that core designs that use TPBARs do not differ significantly from traditional core designs and that the PHOENIX-L and ANC-L codes do not change the licensed ANC methodology as documented in Ref. 3.

During the review of the Westinghouse topical report WCAP-12472-P-A, 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 calibrations/uncertainty components are updated as necessary. In addition, the initial flux mapping at the start of the cycle ensures model calibration factors that reflect the actual fuel in the reactor before the PDMS 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.*

WCAP-12472-P-A described an application of BEACON (i.e. BEACON-DMM) where the core operating limits are changed. As noted previously, TVA is proposing only to use BEACON as a core TS monitor for conformance to WBN's existing limits (i.e. BEACON-TSM). The recommended changes to Specifications 3.1.3.1 and 3.1.3.2 and the COLR mentioned above apply to the BEACON-DMM application and not to the BEACON-TSM application of BEACON. Therefore, the issue addressed by this condition is not applicable to this license amendment requested.

3.5 Impact on UFSAR Accident Analyses

There is no impact on UFSAR Accident Analyses. The PDMS is used for periodic measurement of the core power distribution to confirm operation within design limits and periodic calibration of the excore detectors. The PDMS does not initiate any automatic protection action and is not assumed to be operable to mitigate the consequences of a design basis accident or transient.

3.6 Summary of Technical Evaluation

The detailed evaluation above concludes that the updated BEACON will be implemented in a manner consistent with the original BEACON methodologies approved by the NRC in WCAP-12472-P-A.

4.0 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 WBN does not eliminate, replace, or modify existing plant instrumentation. The PDMS software runs on a workstation connected to the plant integrated computer system. 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.

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.

4.2 Precedent

The BEACON Technical Specification monitor (BEACON-TSM) was approved by the NRC for use at the Virgil C. Summer Nuclear Station in License Amendment No. 142 (Unit 1) on April 9, 1999 and then for Salem Nuclear Generating Station in License Amendment Nos. 237 (Unit 1) and 218 (Unit 2) on November 6, 2000. The BEACON Direct Margin Monitor (BEACON-DMM) application was approved for Byron Nuclear Power Station in License Amendment No. 116 for Units 1 and 2 and for Braidwood Nuclear Power Station in License Amendment No. 110 for Units 1 and 2 on February 13, 2001. The BEACON-TSM application was again approved on 3/31/04 for the Diablo Canyon Power Plant in License Amendment Nos. 164 (Unit 1) and 166 (Unit 2), on 3/31/06 for the South Texas Project in License Amendment Nos. 175 (Unit 1) and 163 (Unit 2), and finally on 3/21/07 for Callaway in License Amendment No. 182. The license amendment requested for Watts Bar Nuclear is most similar to the amendment requested and approved for Callaway.

4.3 Significant Hazards Consideration

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The Power Distribution Monitoring System (PDMS) performs essentially 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 Final Safety Analysis Report (UFSAR).

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 result 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 consequences of an accident previously evaluated.

2. **Does the proposed amendment 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.

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

3. **Does the proposed amendment 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, TVA concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change a 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 effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCE

1. WCAP-12472-P-A, “BEACON Core Monitoring and Operations Support System,” August 1994 (NRC approved version with Safety Evaluation Report).

2. WCAP-12472-P-A Addendum 1-A, “BEACON Core Monitoring and Operations Support System,” January 2000 (NRC approved version with Safety Evaluation Report).

3. NDP-00-0344, “Implementation and Utilization of Tritium Producing Burnable Absorber Rods (TPBARS) at Watts Bar Unit 1,” Revision 1, July 2001.

4. License Amendment 142 to Facility Operating License NPF-12 Regarding Best Estimate Analyzer to Core Operations - Nuclear (BEACON), Virgil C. Summer Nuclear Stations, Unit 1.

5. License Amendments 237 and 218 to Facility Operating Licenses 50-272 and 50-311 for Salem Nuclear Generating Station, Units 1 and 2.

6. License Amendment 116 to Facility Operating Licenses NPF-37 and NPF-66 for the Byron Nuclear Power Station, Units 1 and 2.

7. License Amendment 110 to Facility Operating Licenses NPF-72 and NPF-77 for the Braidwood Nuclear Power Station, Units 1 and 2.

8. License Amendments 164 and 166 to Facility Operating Licenses DPR-80 and DPR-82 for the Diablo Canyon Power Plant, Units 1 and 2.

9. License Amendments 175 and 163 to Facility Operating Licenses NPF-76 and NPF-80 for the South Texas Project Nuclear Operating Company, Units 1 and 2.

10. License Amendment 182 to Facility Operating License NPF-30 for the Callaway plant, Unit 1.

11. Westinghouse Letter, NF-TV-09-30, “BEACON and SPNOVA Features Supporting TPBARs,” dated May 13, 2009.

Enclosure 2

Proposed Technical Specifications Change Markup

1.1 Definitions

LEAKAGE (continued)

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary-to-secondary LEAKAGE);

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. <u>Pressure Boundary LEAKAGE</u>

LEAKAGE (except primary-to-secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE-OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PDMS

The Power Distribution Monitoring System (PDMS) is a real-time three dimensional core monitoring system. The system utilizes existing core instrumentation data and an on-line neutronics code to provide surveillance of core thermal limits.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter 14, Initial Test Program of the FSAR;

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Rod Position Indication

LCO 3.1.8 The Analog Rod Position Indication (ARPI) 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 per group and each demand position indicator per bank.

CONDITION	REQUIRED ACTION	COMPLETION TIME
------------------NOTE------------------ Rod position monitoring by Required Actions A.2.1 and A.2.2 may only be applied to one inoperable ARPI and shall only be allowed: (1) until the end of the current cycle, or (2) until an entry into MODE 5 of sufficient duration, whichever occurs first, when the repair of the inoperable ARPI can safely be performed. Required Actions A.2.1, A.2.2 and A.2.3 shall not be allowed after the plant has been in MODE 5 or other plant condition, for a sufficient period of time, in which the repair of the inoperable ARPI could have safely been performed. --- A. One ARPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators by using ***either the*** movable incore detectors ***or the PDMS.*** <u>OR</u> A.2.1 Verify the position of the rod with the inoperable position indicator by using ***either the*** movable incore detectors ***or the PDMS.*** <u>AND</u>	Once per 8 hours 8 hours <u>AND</u> Once every 31 days thereafter <u>AND</u> 8 hours, if rod control system parameters indicate unintended movement (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Review the parameters of the rod control system for indications of unintended rod movement for the rod with an inoperable position indicator. AND	16 hours AND Once per 8 hours thereafter
	A.2.3 Verify the position of the rod with an inoperable position indicator by using ***either the*** movable incore detectors ***or the PDMS.***	8 hours, if the rod with an inoperable position indicator is moved greater than 12 steps. AND Prior to increasing THERMAL POWER above 50% RTP and within 8 hours of reaching 100% RTP
	OR	
	A.3 Reduce THERMAL POWER to less than or equal to 50% RTP.	8 hours
B. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1 Verify the position of the rods with inoperable position indicators by using ***either the*** movable incore detectors ***or the PDMS.*** OR	4 hours
	B.2 Reduce THERMAL POWER to less than or equal to 50% RTP.	8 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2.	--------------------------------NOTE-------------------------------- If $F^W_Q(Z)$ is within limits and measurements indicate Maximum over z $\left[\frac{F^C_Q(Z)}{K(Z)}\right]$ has increased since the previous evaluation of $F^C_Q(Z)$: a. Increase $F^W_Q(Z)$ by the appropriate factor specified in the COLR and reverify $F^W_Q(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD ***using either the movable incore detectors or the PDMS*** until two successive ***power distribution measurements*** ~~flux maps~~ indicate Maximum over z $\left[\frac{F^C_Q(Z)}{K(Z)}\right]$ has not increased --	
	Verify $F^W_Q(Z)$ is within limit.	Once after initial fuel loading and each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> (continued)

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 if adequate power range neutron flux channel inputs are not OPERABLE. ---------- Verify QPTR is within limit by calculation.	7 days <u>AND</u> Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable
SR 3.2.4.2	-----NOTE----- Only required to be performed if input from one or more power range neutron flux channels are inoperable with THERMAL POWER ≥ 75% RTP. ---------- Verify QPTR is within limit using ***either the*** movable incore detectors ***or the PDMS.***	Once within 12 hours AND 12 hours thereafter

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	-------------------------------NOTES---------------------------------- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Required to be performed within 96 hours after THERMAL POWER is ≥ 15% RTP. --- Compare results of the incore detector ***or PDMS*** measurements to NIS AFD.	31 effective full power days (EFPD)
SR 3.3.1.4	------------------------------NOTE--------------------------------- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. -- Perform TADOT.	62 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	---------------------------------NOTE---------------------------------- Required to be performed within 6 days after THERMAL POWER is ≥ 50% RTP. --- Calibrate excore channels to agree with **incore detectors or PDMS** measurements.	92 EFPD

(continued)

Enclosure 3

Proposed Technical Specifications Bases Change (Mark-Up)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip in response to a main steam pipe rupture 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. 5). The reactor is shutdown by the boric acid injection delivered by the ECCS.

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~~ ***using incore power distribution measurements***. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

(continued)

ACTIONS

B.2.1.1 and B.2.1.2 (continued)

Power operation may continue with one RCCA tripable 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, RTP 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. 6). 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~~ ***an incore*** power distribution ~~using the incore flux mapping system~~ ***measurement*** and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

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

(continued)

ACTIONS (continued)

A.1

When one ARPI channel per group fails, the position of the rod can still be determined by use of ~~the incore movable detectors~~ ***incore power distribution measurement information. Incore power distribution measurement information can be obtained from flux traces using the Movable Incore Detector System or from an OPERABLE Power Distribution Monitoring System (PDMS) (ref. 15)*** Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2.1, A.2.2

The control rod drive mechanism (a portion of the rod control system) consists of four separate subassemblies; 1) the pressure vessel, 2) the coil stack assembly, 3) the latch assembly, and 4) the drive rod assembly. The coil stack assembly contains three operating coils; 1) the stationary gripper coil, 2) the moveable gripper coil, and 3) the lift coil. In support of Actions A.2.1 and A.2.2, a Temporary Alteration (TA) to the configuration of the plant is implemented to provide instrumentation for the monitoring of the rod control system parameters in the Main Control Room. The TA creates a circuit that monitors the operation and timing of the lift coil and the stationary gripper coil. Additional details regarding the TA are provided in the FSAR (Ref. 14).

Required Actions A.2.1 and A.1 are essentially the same. Therefore, the discussion provided above for Required Action A.1 applies to Required Action A.2.1. The options provided by Required Actions A.2.1 and A.2.2 allow for continued operation in a situation where the component causing the ARPI to be inoperable is inaccessible due to operating conditions (adverse radiological or temperature environment). In this situation, repair of the ARPI cannot occur until the unit is in an operating MODE that allows access to the failed components.

In addition to the initial 8 hour verification, Required Action A.2.1 also requires the following for the rod with the failed ARPI:

1. Verification of the position of the rod every 31 days using ***either*** the incore movable detectors ***or the PDMS***.

2. Verification of the position of the rod using ***either*** the incore movable detectors ***or the PDMS*** within 8 hours of the performance of Required Action A.2.2 whenever there is an indication of unintended rod movement based on the parameters of the rod control system.

(continued)

ACTIONS

A.2.1, A.2.2 (continued)

Required Action A.2.2 is in lieu of the verification of the position of the rod using ***either*** the incore movable detectors ***or PDMS*** every 8 hours as required by Required Action A.1. This action alleviates the potential for excessive wear on the incore system due to the repeated use of the incore detectors. Once the position of the rod with the failed ARPI is confirmed through the use of ***either*** the moveable incore detectors ***or PDMS*** in accordance with Required Action A.2.1, the parameters of the rod control system must be monitored until the failed ARPI is repaired. Should the review of the rod control system parameters indicate unintended movement of the rod, the position of the rod must be verified within 8 hours in accordance with Required Action A.2.1. Should there be unintended movement of the rod with the failed ARPI, an alarm will be received. Alarms will also be received if the rod steps in a direction other than what was demanded, and if the circuitry of the TA fails. Receipt of any alarm requires the verification of the position of the rod in accordance with Required Action A.2.1.

Required Actions A.2.1, A.2.2 and A.2.3 are modified by a note. The note clarifies that rod position monitoring by Required Actions A.2.1 and A.2.2 shall only be applied to one rod with an inoperable ARPI and shall only be allowed until the end of the current cycle. Further, Required Actions A.2.1, A.2.2 and A.2.3 shall not be allowed after the plant has been in MODE 5 or other plant condition, for a sufficient period of time, in which the repair of the inoperable ARPI(s) could have safely been performed.

As indicated previously, the modifications required for the monitoring of the rod control system will be implemented as a TA. Implementation of the TA includes a review for the impact on plant procedures and training. This ensures that changes are initiated for key issues like the monitoring requirements in the control room, and operator training on the temporary equipment.

A.2.3

Required Action A.2.3 addresses two contingency measures when the TA is utilized:

1. Verification of the position of the rod with the inoperable ARPI by use of ***either*** the moveable incore detectors ***or PDMS***, whenever the rod is moved greater than 12 steps in one direction.

2. Operation of the unit when THERMAL POWER is less than or equal to 50% RTP.

For the first contingency, the rod group alignment limits of LCO 3.1.5 require that all shutdown and control rods be within 12 steps of their group step counter demand position. The limits on shutdown or control rod alignments ensure that

(continued)

ACTIONS

A.2.3 (continued)

the assumptions in the safety analysis will remain valid and that the assumed reactivity will be available to be inserted for a unit shutdown. Therefore, this conservative measure ensures LCO 3.1.5 is met whenever the rod with the inoperable ARPI is moved greater than 12 steps. For the second contingency, the reduction of THERMAL POWER to less than or equal to 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 13). Consistent with LCO 3.0.4 and this action, unit startup and operation to less than or equal to 50% RTP may occur with one ARPI per group inoperable. However, prior to escalating THERMAL POWER above 50% RTP, the position of the rod with an inoperable ARPI must be verified by use of ***either*** the moveable incore detectors ***or PDMS***. Once 100% RTP is achieved, the position of the rod must be reverified within 8 hours by use of ***either*** the moveable incore detectors ***or PDMS***. Monitoring of the rod control system parameters in accordance with Required Action A.2.2 for the rod with an inoperable ARPI may resume upon completion of the verification at 100% RTP.

A.3

Required Action A.3 applies whenever the TA is not utilized. The discussion for Required Action A.2.3 (above) clarified that a reduction of THERMAL POWER to less than or equal to 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 13). The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to less than or equal to 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above. Consistent with LCO 3.0.4 and this action, unit startup and operation to less than or equal to 50% RTP may occur with one ARPI per group inoperable. Thermal Power may be escalated to 100% RTP as long as Required Action A.1 is satisfied.

B.1 and B.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to less than or equal to 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at greater than 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."

2. Watts Bar FSAR, Section 15.2.1, “Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition."

3. Watts Bar FSAR, Section 15.2.2, “Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power."

4. Watts Bar FSAR, Section 15.2.3, “Rod Cluster Control Assembly Misalignment."

5. Watts Bar FSAR, Section 15.2.4, “Uncontrolled Boron Dilution."

6. Watts Bar FSAR, Section 15.2.5, “Partial Loss of Forced Reactor Coolant Flow."

7. Watts Bar FSAR, Section 15.2.13, “Accidental Depressurization of the Main Steam System."

8. Watts Bar FSAR, Section 15.3.4, “Complete Loss of Forced Reactor Coolant Flow."

9. Watts Bar FSAR, Section 15.3.6, “Single Rod Cluster Control Assembly Withdrawal At Full Power."

10. Watts Bar FSAR, Section 15.4.2.1, “Major Rupture of Main Steam Line."

11. Watts Bar FSAR, Section 15.4.4, “Single Reactor Coolant Pump Locked Rotor."

12. Watts Bar FSAR, Section 15.4.6, “Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)."

13. Watts Bar FSAR, Section 4.3, “Nuclear Design."

14. Watts Bar FSAR, Section 7.7.1.3.2, “Main Control Room Rod Position Indication.”

15. WCAP-12472-P-A, “BEACON Core Monitoring and Operations Support System,” August 1994.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the THERMAL POWER level is $\leq$ 85% RTP will ensure that the required core protection is provided during the performance of PHYSICS TESTS. Control of the reactor power level is a vital parameter and is closely monitored during the performance of PHYSICS TESTS. A Frequency of 1 hour is sufficient for ensuring that the power level does not exceed the limit.

SR 3.1.9.2

Verification of the Power Range Neutron Flux - High trip setpoints within 8 hours prior to initiation of the PHYSICS TESTS will ensure that the RTS is properly set to perform PHYSICS TESTS.

SR 3.1.9.3

The performance of SR 3.2.1.1 and SR 3.2.2.1 measures the core $F_Q(Z)$ and the $F^N_{\Delta H}$, respectively. If the requirements of these LCOs are met, the core has adequate protection from exceeding its design limits, while other LCO requirements are suspended. The Frequency of 12 hours is based on operating experience and the practical amount of time that it may take to ~~run an incore flux map~~ ***obtain an incore power distribution measurement*** and calculate the hot channel factors.

SR 3.1.9.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.9.4 (continued)

f. Samarium concentration; and

~~g. Design isothermal temperature coefficient (ITC)~~.

~~Using the ITC accounts for Doppler reactivity in the calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.~~

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident without the required SDM.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

2. Title 10, Code of Federal Regulations, Part 50.59, "Changes, Tests, and Experiments."

3. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978.

4. ANSI/ANS-19.6.1, "Reload Startup PHYSICS TESTS for Pressurized Water Reactors," American National Standards Institute.

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

6. Watts Bar FSAR, Section 14.2, "Test Program."

7. WCAP-11618, "MERITS Program - Phase II, Task 5, Criteria Application," dated November 1987, including Addendum 1, April 1989.

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

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

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

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

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.7, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

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

$F_Q(Z)$ is measured periodically using ~~the~~ ***either the Movable Incore Detector System*** ~~incore detector system~~ ***or the Power Distribution Monitoring System (PDMS) (ref.5)***. These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents a steady state 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.

(continued)

BASES (continued)

LCO

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

$$F_Q(Z) \leq \frac{CFQ}{P} K(Z) \quad \text{for} \quad P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} K(Z) \quad \text{for} \quad P \leq 0.5$$

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

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

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

The actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of 2.4, and K(Z) is a function that looks like the one provided in Figure B 3.2.1-1.

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

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

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

~~where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.~~

The measured value, $F^M_Q(Z)$, of $F_Q(Z)$ is obtained from the incore power distribution measurement and then corrected for fuel manufacturing tolerances and measurement uncertainty.

If the Moveable Incore Detector System is used to obtain the incore power distribution measurement, then:

$$F^C_Q(Z) = 1.03\, F^M_Q(Z)\, F^{MU}_Q$$

where 1.03 is the factor that accounts for the fuel manufacturing tolerances and F^{MU}_Q, which accounts for flux map measurement uncertainty, is 1.05 (Ref. 4).

(continued)

When the PDMS is used to obtain the incore power distribution measurement, then:

$$\boldsymbol{F^C_Q(Z) = 1.03\ F^M_Q(Z)\ (1+U_Q/100)}$$

where 1.03 is the factor that accounts for the fuel manufacturing tolerances and the factor (1+U_Q/100)), which accounts for PDMS measurement uncertainty, is calculated and applied automatically by the BEACON software (Ref.5).

$F^C_Q(Z)$ is an approximation of the steady state $F_Q(Z)$.

(continued)

BASES

LCO (continued)

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

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

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

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1).

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

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

APPLICABILITY

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

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^C_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F^C_Q(Z)$ is $F^M_Q(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F^M_Q(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F^C_Q(Z)$ and $F^W_Q(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

SR 3.2.1.1

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

If the Movable Incore Detector System is used to obtain the incore power distribution measurement, then:

$$F^C_Q(Z) = 1.03\ F^M_Q(Z)\ F^{MU}_Q$$

where 1.03 is the factor that accounts for the fuel manufacturing tolerances and F^{MU}_Q, which accounts for flux map measurement uncertainty, is 1.05 (Ref. 4).

When the PDMS is used to obtain the incore power distribution measurement, then:

$$F^C_Q(Z) = 1.03\ F^M_Q(Z)\ (1+U_Q/100)$$

where 1.03 is the factor that accounts for the fuel manufacturing tolerances and the factor $(1+U_Q/100)$), which accounts for PDMS measurement uncertainty, is calculated and applied automatically by the BEACON software (Ref.5)

The limit with which $F^C_Q(Z)$ 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 $F^C_Q(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by $\geq$ 10% RTP since the last determination of $F^C_Q(Z)$, another evaluation of this factor is required

(continued)

12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F^C_Q(Z)$ 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).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because ~~flux maps~~ ***incore power distribution measurements*** are taken ~~in~~ ***at or near*** steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the ~~flux map~~ ***measured*** 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, $F^C_Q(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F^W_Q(Z)$.

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

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

a. Lower core region, from 0 to 10% inclusive; and

b. Upper core region, from 90 to 100% inclusive.

The top and bottom 10% of the core are excluded from the evaluation because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F^W_Q(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to $F^M_Q(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

(continued)

BASES

REFERENCES (continued)

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

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

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

(continued)

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.

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Application of these criteria provides assurance that the hottest fuel rod in the core does not experience a DNB.

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

The LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3) model $F^N_{\Delta H}$ as well as the Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$).

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 3.1.7, "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 ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using ***either*** the Movable Incore Detector System ***or the PDMS (Ref.5).*** Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

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

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

(continued)

BASES

ACTIONS (continued)

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore ~~flux map~~ ***power distribution measurement*** (SR 3.2.2.1) must be obtained and the measured value of $F^{N}_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore ~~flux map~~ ***power distribution measurement***, perform the required calculations, and evaluate $F^{N}_{\Delta H}$.

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 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.

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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using ***either*** the Movable Incore Detector System ***or the PDMS to obtain an incore power distribution measurement.*** ~~to obtain a flux distribution map~~. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by ~~1.04~~ ***a factor*** to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

If the Moveable Incore Detector System is used to obtain the incore power distribution measurement, then the factor ($F^N_{\Delta H}$) is 1.04 (Ref. 4). When the PDMS is used to obtain the incore power distribution measurement, the factor ($1+U_{\Delta H}/100$) is calculated and applied automatically by the BEACON software (Ref. 5).

After the initial fuel loading and 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. Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

3. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1994.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Channel Factor ($F^N_{\Delta}H$), ***rod group alignment***, ***sequence, overlap, and control*** bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

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

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

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

LCO

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

APPLICABILITY

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

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta}H$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient

(continued)

BASES

ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

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 ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform ~~a flux map~~ ***an incore power distribution measurement***. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta}H$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta}H$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those changes of QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is required only when the input from one or more power range neutron flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

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.

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

~~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,~~

For the purpose of monitoring the QPTR when the input from one or more power range neutron flux channels are inoperable, incore power distribution measurement information is used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and the reference normalized symmetric power distribution. The incore power distribution measurement information can be used to generate an incore "tilt." This tilt can be compared to the reference incore

(continued)

tilt to generate an incore QPTR. Therefore, incore QPTR can be used to confirm that excore QPTR is within limits.

The incore power distribution measurement information can be obtained from either the Movable Incore Detector System or from an OPERABLE Power Distribution Monitoring System (PDMS) (Ref. 4). If the Movable Incore Detector System is used to obtain the incore power distribution measurement information, then the incore detector monitoring is performed with a full core 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 reference normalized symmetric power distribution is available from the last incore power distribution measurement information used to calibrate the excore axial offset. The reference incore power distribution measurement information may have been obtained from either a full core flux map using the Movable Incore Detector System or from an OPERABLE PDMS. The full core flux map information may be reduced to the information from only the two sets of four symmetric thimbles with quarter core symmetry for like comparisons, if practical.

With the input from one or more power range neutron flux channels inoperable, the indicated QPTR 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 normalized quadrant tilt is compared against the reference normalized quadrant tilt. Nominally, quadrant tilt from the surveillance should be within 2% of the tilt shown by the reference incore power distribution measurement information.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

~~the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data~~.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

2. Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

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

)

BASES

ACTIONS

D.1.1, D.1.2, D.2.1, D.2.2, and D (continued)

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 14.

Required Action D.2.2 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 ***or the PDMS*** once per 12 hours may not be necessary.

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux—Low; and
- Power Range Neutron Flux—High Positive Rate

A known inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 14.

If the inoperable channel cannot be placed in the trip condition within the specified Completion Time, the plant must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the plant in MODE 3. Six hours is a reasonable time, based on operating experience, to place the plant in MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 14.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore ~~system~~ ***power distribution measurement*** to the NIS channel ***AFD*** output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function. ***The incore power distribution measurement may be obtained using the Movable Incore Detector System or an OPERABLE Power Distribution Monitoring System (PDMS) (Ref. 16).***

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%. Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 96 hours is allowed for performing the first Surveillance after reaching 15% RTP. This surveillance is typically performed at ***greater than or equal to*** 50% RTP to ensure the results of the evaluation are more accurate and the adjustments more reliable. Ninety-six (96) hours are allowed to ensure Xenon stability and allow for instrumentation alignments.

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

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 62 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.4 (continued)

The Frequency of every 62 days on a STAGGERED TEST BASIS is justified in Reference 15.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 92 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection Function. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 15.

SR 3.3.1.6

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 measurements~~ ***power distribution measurement(s).*** 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. ***The incore power distribution measurement(s) may be obtained using the Moveable Incore Detector System or an OPERABLE PDMS (Ref. 16).***

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 6 days is allowed for performing the first surveillance after reaching 50% RTP.

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

(continued)

BASES

REFERENCES

1. Watts Bar FSAR, Section 6.0, "Engineered Safety Features"

2. Watts Bar FSAR, Section 7.0, "Instrumentation and Controls"

3. Watts Bar FSAR, Section 15.0, "Accident Analysis."

4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.

5. 10 CFR Part 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."

6. WCAP-12096, Rev. 7, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," March 1997.

7. WCAP-10271-P-A, Supplement 1, and Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," May 1986 and June 1990.

8. Watts Bar Technical Requirements Manual, Section 3.3.1, "Reactor Trip System Response Times."

9. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar, Westinghouse Letter WAT-D-10128.

10. ISA-DS-67.04, 1982, "Setpoint for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants."

11. WCAP-13632-P-A Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements,” January 1996

12. WCAP-14036-P-A, Revision 1, “Elimination of Periodic Protection Channel Response Time Tests,” October 1998.

13. WCAP-16067-P, Rev. 0, “RCS Flow Measurement Using Elbow Tap Methodology at Watts Bar Unit 1,” April 2003.

14. WCAP-14333 P-A, Revision 1, “Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,” October 1998.

15. WCAP-15376-P-A, Revision 1, “Risk Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,” March 2003.

16. WCAP-12472-P-A, “BEACON Core Monitoring and Operations Support System,” August 1994.

ENCLOSURE 4

Evaluation for Excluding PDMS Requirements from the Technical Specifications

The justification for not including requirements for the PDMS and associated instrumentation in the Technical Specifications 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 Technical Specifications. This evaluation is done in accordance with the requirements contained in 10 CFR 50.36(c)(2)(ii).

Per 10 CFR 50.36(c)(2)(ii), a Technical Specification 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.

PDMS is not associated with the monitoring of any aspect of the reactor coolant system 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 i.e., $F_Q(Z)$, $F^N_{\Delta H}$, QPTR, and rod alignments are operating restrictions which ensure that the accident analyses and assumptions for all applicable, analyzed DBAs remain valid. These limits are included in the Technical Specifications. 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 affect any condition assumed in the accident/transient analyses. The PDMS provides the capability to monitor power distribution parameters at more frequent intervals than is currently required by the Technical Specifications. Additionally, these parameters or limits can be determined independent of the operability of PDMS via the Movable Incore Detectors. 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. While the PDMS can provide a measurement of the core average axial offset used in the calibration of the OTΔT Reactor Trip System function, this parameter can be determined independent of the operability of PDMS via the Movable Incore Detector System (MIDS). Measurement of the core average axial offset by either the PDMS or the MIDS are equivalent. Neither the PDMS nor the MIDS have active/control functions or actuation capability and as such are not part of any primary success path for mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) 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 Technical Specifications.

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 detector system) 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 the PDMS does not meet any of the criteria for inclusion in the Technical Specifications. The PDMS requirements and controls to be incorporated into the Technical Requirements Manual 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 Technical Specifications.

Enclosure 5

Proposed Technical Requirements Manual Change (Mark-Up)

TR 3.3 INSTRUMENTATION

TR 3.3.3 Movable Incore Detectors

TR 3.3.3 The Movable Incore Detection System shall be OPERABLE with ≥ 75% of the detector thimbles, ≥ 2 detector thimbles per core quadrant, and sufficient movable detectors, drive, and readout equipment to map these thimbles.

--------------------------------------NOTES--

1. ***Either a full core flux map or quarter-core flux maps may be used in calibrations of the Excore Neutron Flux Detection System.***

2. ***Either a full core flux map or two sets of four thimble locations with quarter core symmetry may be used for monitoring the QUADRANT POWER TILT RATIO.***

3. ***Only ≥ 50% of the detector thimbles and ≥ 2 detector thimbles per core quadrant are required for Power Distribution Monitoring System (PDMS) calibration after the initial PDMS calibration after each refueling.***

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

a. Calibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ ***and*** $F_Q(Z)$ or

d. ***Calibration of the Power Distribution Monitoring System (PDMS).***

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Movable Incore Detection System inoperable.	A.1 ---------------NOTE------------------- TR 3.0.3 is not applicable. -- Restore the inoperable system to OPERABLE status.	Prior to using the system for monitoring or calibration functions.

TR 3.3 INSTRUMENTATION

TR 3.3.8 Power Distribution Monitoring System (PDMS)

TR 3.3.8 ***The PDMS shall be OPERABLE with:***

a. THERMAL POWER $\geq$ 25% RTP, and

b. The required channel inputs from the plant computer for each function defined in Table 3.3.8-1.

APPLICABILITY: ***When the PDMS is used for:***

a. Calibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$, or

d. Verifying the position of a rod with inoperable position indicators.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. PDMS inoperable.	A.1	-----------NOTE------------- TR 3.0.3 is not applicable. ------------------------------- Restore the inoperable system to OPERABLE status.	Prior to using the system for incore power distribution measurement purposes.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		*FREQUENCY*
TSR 3.3.8.1	*Perform CHANNEL CHECK for each required instrumentation channel specified in Table 3.3.8-1.*	*24 hours*
TSR 3.3.8.2	*Verify by administrative means that the surveillance requirements for each required channel specified in Table 3.3.8-1 are satisfied.*	*24 hours*
TSR 3.3.8.3	*Perform PDMS calibration.*	*Once after each refueling* ***AND*** *---------NOTE---------* *Not required to be performed until 31 Effective Full Power Days (EFPD) after the Core Exit Thermocouple (CETC) chess knight move pattern not satisfied.* *----------------------* *31 EFPD thereafter with the CETC chess knight move pattern not satisfied* ***AND*** *180 EFPD thereafter with the CETC chess knight move pattern satisfied*

Table 3.3.8-1 (Page 1 of 1)

Power Distribution Monitoring System (PDMS) Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	SURVEILLANCE TYPE
1. Power Range Neutron Flux Monitors	3	SR 3.3.1.6 SR 3.3.1.11(3)	Excore Calibration CHANNEL CALIBRATION
2. RCS Cold Leg Temperature	2	SR 3.3.3.3	CHANNEL CALIBRATION
3. Reactor Power	1(1)	SR 3.3.1.2(4) SR 3.3.1.10(5) TSR 3.3.7.1(4)	Calorimetric Heat Balance CHANNEL CALIBRATION LEFM Availability
4. Control Bank Position (per bank)	1(2)	SR 3.1.8.1	CHANNEL CHECK
5. Core Exit Thermocouple Temperature	17 with $\geq$ 2 per core quadrant	SR 3.3.3.3	CHANNEL CALIBRATION

(1) *Either secondary calorimetric power, average power range neutron flux power, or average RCS Loop ΔT power*

(2) *Either the Demand Position Indication or the average of the individual Rod Position Indications*

(3) *Neutron detectors are excluded from CHANNEL CALIBRATION*

(4) *Not applicable to average RCS Loop ΔT power*

(5) *Applies to average RCS Loop ΔT power only*

Enclosure 6

Proposed Technical Requirements Manual Bases Change (Mark-Up)

B 3.3 INSTRUMENTATION

B 3.3.3 Movable Incore Detectors

BASES

BACKGROUND

The Movable Incore Detection System uses six miniature fission chamber neutron detectors to measure fuel assembly ~~power level~~ ***reaction rates***. The miniature fission chambers are positioned by a Detector Drive System which pushes and pulls the detectors in and out of the reactor core through one of 58 thimbles, which are open at one end. The thimbles have Reactor Coolant System (RCS) pressure on the outside and atmospheric pressure inside. Each thimble is inserted into the center position of the fuel assembly, all the way to the top of the fuel assembly. The drive system slowly pushes the detector up through the fuel assembly, inside the thimble, to the top of the core. An x-y plot (position verses flux level) is initiated with the slow withdrawal of the detectors through the core from top to a point below the bottom. As the detector traverses the thimble tube it obtains the raw currents for 61 axial levels. At each level, the computer takes three rapid looks, averages the readings and uses this as the base reading for that axial point. In a similar manner, other core locations are selected and plotted. Each detector provides axial flux distribution data along the center of a fuel assembly.

Each of the six miniature neutron detectors has its own drive unit. A series of five- and ten-path transfer devices are then used to direct a detector into one of several possible fuel assemblies. In this manner, the six detectors and drive units are used to monitor 58 fuel assemblies in the core.

The operability of the Movable Incore Detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of the data obtained. The detectors are normalized to one another through cross-calibration comparison (looking for relative readings) of each detector's output. This effectively makes the data from all the detectors approximately the same as the reference detector.

Either the Movable Incore Detector System or the Power Distribution Monitoring System (PDMS) may be used for calibration of the Excore Neutron Flux Detection System, monitoring the QUADRANT POWER TILT RATIO, or measurement of $F^N_{\Delta H}$ and $F_Q(z)$. Since the Movable Incore Detector System is utilized by the PDMS, it is required to be OPERABLE before it is used to calibrate the PDMS. The initial calibration of PDMS after a refueling requires a full incore flux map ($\geq$ 75% of the detector thimbles) but subsequent calibrations require only $\geq$ 50% of the detector thimbles.

(continued)

When the Movable Incore Detectors are used directly for measuring $F^N_{\Delta H}$, the Nuclear Enthalpy Rise Hot Channel Factor (Technical Specification 3.2.2) and $F_Q(z)$, the Heat Flux Hot Channel Factor (Technical Specification 3.2.1), a full incore flux map is required. Quarter-core flux maps, as designed in Reference 1, may be used to calibrate the excore axial offset. Either full incore flux maps or symmetric incore thimbles may be used for monitoring the QPTR.

~~The OPERABILITY of the Movable Incore Detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of the data obtained. For the purpose of measuring~~ $\cancel{F_Q(Z)}$~~, the Heat Flux Hot Channel Factor (Technical Specification 3.2.1), or~~ $\cancel{F^N_{\Delta H}}$~~, the Nuclear Enthalpy Rise Hot Channel Factor (Technical Specification 3.2.2), a full incore flux map is used. Quarter-core flux maps, as designed in Reference 1, may be used in~~

(continued)

BACKGROUND (continued)

~~recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QPTR.~~

~~The detectors are normalized to one another through cross-calibration comparison (looking for relative readings) of each detector's output. This effectively makes the data from all the detectors approximately the same as the reference detector.~~

APPLICABLE SAFETY ANALYSES

The Movable Incore Detector System is used for periodic surveillance of the power distribution and calibration of the excore detectors. Surveillance of the power distribution verifies that the peaking factors are within the design envelope. The system is not used continuously and does not initiate any automatic protection action. The Movable Incore Detector System is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient (Ref. 2).

TR

TR 3.3.3 specifies that the Movable Incore Detection System shall be OPERABLE with at least 75% of the detector thimbles and a minimum of two detector thimbles per core quadrant. Also, sufficient movable detectors, drive, and readout equipment to map these thimbles.

This TR ensures the OPERABILITY of the Movable Incore Detector Instrumentation when required to monitor the flux distribution within the core. The Movable Incore Detector System is used for periodic surveillance of the power distribution, and calibration of the excore detectors. The surveillance of power distribution verifies that the peaking factors are within their design envelope (Ref. 2).

Three notes modify TR 3.3.3. Note 1 clarifies that quarter-core flux maps may be used to calibrate the Excore Neutron Flux Detection System in lieu of a full core flux map with $\geq$ 75% of the detector thimbles in accordance with Reference 1. Note 2 clarifies that symmetric incore thimbles may be used for monitoring the QPTR in lieu of a full incore flux map with $\geq$ 75% of the detector thimbles in accordance with Technical Specification 3.2.4. Note 3 clarifies that a flux map with $\geq$ 50% of the detector thimbles and $\geq$ 2 detector thimbles per core quadrant may be used for subsequent calibration of the PDMS after the initial calibration after a refueling in lieu of a full incore flux map with $\geq$ 75% of the detector thimbles and $\geq$ 2 detector thimbles per core quadrant in accordance with Reference 3.

APPLICABILITY

The Movable Incore Detection System must be OPERABLE when it is used for ~~re~~calibration of the Excore Neutron Flux Detection System, ~~or~~ monitoring the QPTR, ~~or~~ measurement of $F^N_{\Delta H}$ ***and*** $F_Q(z)$ ~~F_{XY}~~ ***or calibration of the PDMS***.

(continued)

ACTIONS

A.1

The Required Action A.1 has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

An inoperable Movable Incore Detection System~~***s***~~ cannot be used for ~~***re***~~calibration of the Excore Neutron Flux Detection System, ~~***or***~~ monitoring the QPTR***,*** ~~***or***~~ measurement of $F^{N}_{\Delta H}$ ***and*** $F_{Q}(z)$ ~~and F_{XY}~~ ***or calibration of the PDMS***. Therefore, the Required action A.1 prohibits the use of the inoperable system for the above applicable monitoring or calibration functions.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.3.1

The Movable Incore Detector System must be demonstrated OPERABLE at least once per 24 hours by the setting of each detector's operating voltage. The operating voltage is set by determining the operating region for each detector after inserting it into a high flux region of the core. The acceptability of each detector is verified by the performance of a detector drift check. The operating voltage must be determined prior to using the Movable Incore Detector System for ~~***re***~~calibration of the Excore Neutron Flux Detection System, ~~or~~ monitoring the QPTR, ~~or~~ measurement of $F^{N}_{\Delta H}$ ***and*** $F_{Q}(z)$, ~~and F_{XY}~~ ***or calibration of the PDMS.*** This surveillance ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The Frequency of 24 hours has been established, based on engineering judgment, and has been shown to be acceptable through operating experience.

REFERENCES

1. WCAP-8648, "Excore Detector Recalibration Using Quarter-core Flux Maps," June 1976.

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

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

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8 Power Distribution Monitoring System (PDMS)

BASES

BACKGROUND

The Power Distribution Monitoring System (PDMS) generates a continuous measurement of the incore power distribution using the methodology documented in Reference 1. The PDMS employs an advanced three-dimensional nodal code to calculate the incore power distribution. The reference incore power distribution is periodically normalized to the incore flux measurements from the movable incore detectors. On a nominal once-per-minute basis, the incore power distribution is updated with plant instrumentation, most notably from the Core Exit Thermocouples (CETCs). In this way, the information from the up-to-the-minute PDMS incore power distribution is equivalent to a full incore flux map using the Movable Incore Detector System (Technical Requirement 3.3.3).

The PDMS incore power distribution measurement can be used to determine the most limiting core peaking factors, $F^N_{\Delta H}$, the Nuclear Enthalpy Rise Hot Channel Factor (Technical Specification 3.2.2) and $F_Q(z)$, the Heat Flux Hot Channel Factor (Technical Specification 3.2.1). The incore power distribution measurement can also be used in the ~~r~~ecalibration of the excore neutron flux detection system (Technical Specification 3.3.1), monitoring the QUADRANT POWER TILT RATIO (QPTR) (Technical Specification 3.2.4), and verifying the position of a rod with inoperable position indicators (Technical Specification 3.1.8).

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 OPERABILITY of the PDMS with the specified minimum complement of instrumentation channel inputs ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The PDMS requires input for power range detector calibrated voltage values, average reactor vessel inlet temperature, reactor power level, control bank positions, and temperatures from the Core Exit Thermocouples (CETCs).

Either the PDMS or the Movable Incore Detector System may be used for ~~r~~ecalibration of the Excore Neutron Flux Detection System, monitoring the QUADRANT POWER TILT RATIO, or measurement of $F_Q(z)$ or $F^N_{\Delta H}$. Similarly, either the PDMS or the Movable Incore Detector System may be used for verifying the position of a rod with inoperable position indicators, but only the PDMS must satisfy OPERABILITY requirements prior to this function.

APPLICABLE SAFETY ANALYSES

The PDMS is used for periodic measurement of the core power distribution to confirm operation within design limits 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 (References 2 and 3).

TR	***TR 3.3.8 requires the PDMS to be OPERABLE with the specified number of instrument channel inputs from the plant computer for each function listed in Table 3.3.8-1. The PDMS is OPERABLE when the required channel inputs are available, the calibration data set is valid, and reactor power is $\geq$ 25% RTP.*** ***This TR ensures the OPERABILITY of the PDMS when required to monitor the power distribution within the core. The PDMS is used for periodic surveillance of the incore power distribution and calibration of the excore detectors. The surveillance of incore power distribution verifies that the peaking factors are within their design envelope (Reference 3). The peaking factor limits include measurement uncertainty which bounds the actual measurement uncertainty of an OPERABLE PDMS (Reference 1).*** ***Maintaining the minimum number of instrumentation channel inputs ensures the uncertainty is bounded by the uncertainty methodology. Similarly, when THERMAL POWER is less than 25% RTP, then the accuracy of the adjustment provided by the CETCs to the measured PDMS power distribution may not be bounded by the uncertainties documented in Reference 1.***
APPLICABILITY	***The PDMS must be OPERABLE when it is used for calibration of the Excore Neutron Flux Detection System, monitoring the QPTR, measurement of $F^N_{\Delta H}$ and $F_Q(z)$, or verifying the position of a rod with inoperable position indicators.***
ACTIONS	***<u>A.1</u>*** ***The Required Action A.1 has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.*** ***With THERMAL Power less than 25% RTP or with one or more required channel inputs inoperable or unavailable to the PDMS, the PDMS must not be used to obtain an incore power distribution measurement. Therefore, the Required Action A.1 prohibits the use of the inoperable system for the applicable monitoring or calibration functions.***
TECHNICAL SURVEILLANCE REQUIREMENTS	***<u>TSR 3.3.8.1</u>*** ***Performance of the CHANNEL CHECK 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.***

TECHNICAL SURVEILLANCE REQUIREMENTS

<u>TSR 3.3.8.1 (continued)</u>

A CHANNEL CHECK will detect gross channel failure, thus it is a key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency of 24 hours is sufficient considering the PDMS provides automatic validation of the channel inputs and either discards the inoperable channel input or declares itself inoperable, but at the same time ensures that the required channel inputs to the PDMS are manually verified to be valid within a reasonable time frame prior to using the PDMS to obtain an incore power distribution measurement.

<u>TSR 3.3.8.2</u>

Verification by administrative means of the surveillance requirements required elsewhere ensures the instrumentation channels satisfy nominal accuracy and reliability for power operation. Many of these surveillance requirements are CHANNEL CALIBRATIONS.

CHANNEL CALIBRATIONS are typically performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATION must be performed consistent with the assumptions of the Watts Bar setpoint methodology. The difference between the current "as found" values and the previous test "as Left" values must be consistent with the drift allowance used in the setpoint methodology.

Five notes modify the instrumentation channels specified in Table 3.3.8-1.

Note 1 allows up to three parameters to be used for reactor power input into PDMS, but BEACON will only accept two options at any one time.

Note 2 allows the control bank position input to come from either the Demand Position Indication or the average of the individual Pod Position Indications.

Note 3 clarifies 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 performed above 15% RTP in accordance with SR 3.3.1.2.

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TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

Note 4 clarifies that the calorimetric heat balance adjustment is not applicable to the average RCS Loop ΔT power input.

Note 5 clarifies that the CHANNEL CALIBRATION requirements do not apply to the secondary calorimetric power or average power range neutron flux power inputs.

<u>TSR 3.3.8.3</u>

The PDMS must be calibrated to a flux map obtained above 25% RTP to ensure the accuracy of the calibration data set which is generated from the incore flux map, CETCs, and other input channels. The initial calibration in each fuel cycle must utilize incore flux measurements from at least 75% of the detector thimbles, with at least two incore thimbles in each core quadrant. The incore flux measurements in combination with at least the minimum channel inputs from Table 3.3.8-1 are used to generate the calibration data set, including nodal calibration factors and the thermocouple mixing factors. Subsequent PDMS calibrations require only ≥ 50% of the detector thimbles, with at least two incore thimbles in each core quadrant.

The subsequent PDMS calibration frequency is 31 Effective Full Power Days (EFPD) when the CETC chess knight move pattern is not satisfied. The CETC chess knight move pattern is satisfied when every interior core location (fuel assemblies not face adjacent to the core baffle) is no further than a chess knight's move from an OPERABLE CETC. The 31 EFPD frequency calibration requirement is modified by a note that clarifies that subsequent PDMS calibration is not required to be performed until 31 EFPD after the CETC chess knight move pattern is not satisfied.

The subsequent PDMS calibration frequency is 180 EFPD when the CETC chess knight move pattern is satisfied. The CETC chess knight move pattern provides coverage of all interior fuel assemblies (coverage of fuel assemblies with a face along the baffle is not required). Fuel assemblies which are within a chess knight's move of an OPERABLE CETC are covered.

REFERENCES

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

2. 10 CFR 50.46

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.