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June 11, 2003

PG&E Letter DCL-03-069

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 03-09,
Revision of Technical Specifications 3.1.7, "Rod Position Indication," 3.2.1, "Heat Flux Hot Channel Factor," 3.2.4, "Quadrant Power Tilt Ratio," and 3.3.1, "Reactor Trip System (RTS) Instrumentation," for BEACON Implementation

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant, respectively. The enclosed license amendment request (LAR) proposes to revise Technical Specifications (TS) 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor," TS 3.2.4, "Quadrant Power Tilt Ratio," and TS 3.3.1, "Reactor Trip System Instrumentation," to allow use of a power distribution monitoring system as described in WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," for power distribution measurements.

Enclosure 1 contains a description of the proposed change, the supporting technical analyses, and the no significant hazards determination. Enclosures 2 and 3 contain marked-up and revised TS pages, respectively. Enclosure 4 contains marked-up TS Bases changes for information only.

PG&E has determined that this LAR does not involve a significant hazards consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

The changes in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR no later than July 2004, and requests the LAR be made effective upon NRC issuance, to be implemented within 60 days from the date of issuance.

A001

If you have any questions or require additional information, please contact Stan Ketelsen at (805) 545-4720.

Sincerely,

David H. Oatley
Vice President and General Manager – Diablo Canyon

SMG/4692
Enclosures

cc:	Edgar Bailey, DHS
	Thomas P. Gwynn
	David L. Proulx
	Diablo Distribution
cc/enc:	David H. Jaffe

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	) Docket No. 50-275) Facility Operating License) No. DPR-80
Diablo Canyon Power Plant Units 1 and 2	) Docket No. 50-323) Facility Operating License) No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath says that he is Vice President and General Manager – Diablo Canyon of Pacific Gas and Electric Company; that he has executed LAR 03-09 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

D. H. Oatley

David H. Oatley
Vice President and General Manager – Diablo Canyon

Subscribed and sworn to before me this 11th day of June, 2003.

Sandra Eatherly

Notary Public
County of San Luis Obispo
State of California

SANDRA EATHERLY
Commission # 1225422
Notary Public - California
San Luis Obispo County
My Comm. Expires Jul 18, 2003

EVALUATION

1.0 Description

This letter is a request to amend Operating Licenses DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed changes would revise the Technical Specifications (TS) to allow use of a power distribution monitoring system (PDMS). The BEACON PDMS is comprised of the NRC-approved Westinghouse proprietary computer code, the Best Estimate Analyzer for Core Operations – Nuclear (BEACON), and the plant data fed to the plant process computer from the incore thermocouples and excore nuclear instruments. BEACON serves as a 3-D core monitor, operational analysis tool, and operational support package.

Westinghouse submitted the BEACON topical report (WCAP-12472-P) to the NRC and the NRC issued a Safety Evaluation Report (SER) approving the topical report on February 16, 1994. In its SER, the NRC concluded that BEACON provides a greatly improved continuous online power distribution measurement and operation prediction information system for Westinghouse reactors.

PG&E proposes to use BEACON to augment the functional capability of the flux mapping system for the purpose of power distribution surveillances. WCAP-12472-P-A discusses an application of BEACON in which the TS and core power distribution limits are changed to take credit for continuous monitoring by plant operators. PG&E proposes to use a more conservative application of BEACON where the core power distribution limits remain unchanged. This limited application of BEACON is referred to as the BEACON Technical Specification Monitor (TSM). PG&E intends to use the BEACON PDMS as the primary method for power distribution measurements and the flux mapping system, if required, when thermal power is greater than 25 percent rated thermal power (RTP). At thermal power levels less than or equal to 25 percent RTP, or when the PDMS is inoperable, the movable incore detector system will be used.

The TS affected by the implementation of BEACON include TS 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor," TS 3.2.4, "Quadrant Power Tilt Ratio," and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation." In addition, a section is added to the Core Operating Limits Report (COLR). This section defines the equations and constants used to determine the applicable measurement uncertainties applied to the core peaking factors when determined by either the PDMS or the flux mapping system. The constants found in this section of the COLR are used as coefficients in the uncertainty calculations and are determined using the

methodology approved by the NRC in its review of the Westinghouse BEACON topical report. The constants may be revised periodically as appropriate to reflect cycle-specific variables.

PG&E requests approval of the proposed amendment by July 2004 as there are no timing constraints and it is not required to address an immediate safety concern. Movable incore detectors will continue to be used for the monthly flux map surveillances prior to issuance and implementation of the license amendment.

2.0 Proposed Change

TS 3.1.7, Rod Position Indication

TS Required Actions A.1, B.3, and C.1, which state the position of the rods with inoperable position indicators must be verified indirectly by using the movable incore detectors, will be revised to state, "Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information." The generic phrase "core power distribution measurement information" would allow the use of an operable PDMS or the movable incore detectors for verifying the position of the rod with an inoperable digital rod position indicator.

TS 3.2.1, Heat Flux Hot Channel Factor

Surveillance Requirement (SR) 3.2.1.2, for verifying the peaking factor $F_Q^W(Z)$ is within limit, is modified by a Note, which currently states if $F_Q^C(Z)$ measurements indicate that maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$ has increased since the previous evaluation of $F_Q^C(Z)$:

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

$$\text{maximum over z} \left[\frac{F_Q^C(Z)}{K(Z)}\right]$$

has not increased."

It will be revised to state:

"b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive power distribution measurements indicate

$$\text{maximum over z} \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$

has not increased."

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

TS 3.2.4, Quadrant Power Tilt Ratio

SR 3.2.4.2 currently states "Verify QPTR is within limit using the movable incore detectors." It will be revised to state "Verify QPTR is within limit using core power distribution measurement information." This change allows the surveillance to be performed using either the movable incore detectors or an operable PDMS.

TS 3.3.1, Reactor Trip System (RTS) Instrumentation

SR 3.3.1.3 currently states, "Compare results of the incore detector measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%." SR 3.3.1.3 will be revised to state "Compare results of incore power distribution measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%."

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 power distribution measurements."

These changes will allow the surveillances to be performed using either the movable incore detectors or an operable PDMS.

PDMS instrumentation does not meet the selection criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. Therefore, the PDMS instrumentation requirement will be contained in the Equipment Control Guidelines (ECG). ECGs are plant specific administrative controls (similar to those provided by the TS), over plant equipment not required to be in the TS. Changes to the ECGs are made in accordance with 10 CFR 50.59.

The justification for not including PDMS instrumentation in the TS is outlined below. The purpose of this evaluation is to demonstrate that the structures,

systems, or components associated with PDMS instrumentation are not required to be contained in the TS. This evaluation is done in accordance with the requirements contained in 10 CFR 50.36(c)(2)(ii).

A TS Limiting Condition for Operation must be established for each item meeting one or more of the following criteria:

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

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

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

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

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

PDMS instrumentation provides the capability to monitor core power distribution parameters at more frequent intervals than is currently required by TS. PDMS instrumentation does not change any of the key safety parameter limits or levels of margin as considered in the reference design basis evaluations. The PDMS instrumentation has no functions or actuations that mitigate any DBA or transient analysis that either assumes the failure of, or presents the challenge to the integrity of a fission product barrier.

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

PDMS instrumentation provides the capability to monitor power distribution parameters at more frequent intervals than is currently required by TS. PDMS instrumentation is a system utilized to monitor the core power distribution and has no impact on the results or consequences of any DBA or transient analysis. Therefore it has no impact on public health and safety.

The evaluation completed above indicates that PDMS instrumentation does not meet any of the criteria for inclusion in the TS. The ECG for PDMS operability will reflect the minimum requirements presented in Reference 1, except for changes due to PG&E's use of the BEACON TSM, according to vendor instructions.

In summary, the proposed amendment would allow the use of the Westinghouse proprietary 3-D nodal code BEACON for performing power distribution surveillances provided that the PDMS instrumentation is operable. This amendment would also allow the use of the movable incore detector system for meeting power distribution surveillances and TS actions, and for calibration of BEACON.

The COLR and TS Bases will be revised to reflect the changes to the affected TS. A markup of the changes to the TS Bases is provided in Enclosure 4 for information only. The COLR changes will be implemented in accordance with TS 5.6.5, "Core Operating Limits Report (COLR)" and the TS Bases changes will be implemented in accordance with TS 5.5.14, "Technical Specification (TS) Bases Control Program," as part of the implementation of this amendment, upon NRC approval of this amendment application.

3.0 Background

As described in Reference 1, the Westinghouse BEACON PDMS was developed to improve the operational support for pressurized water reactors (PWRs). BEACON is an advanced core monitoring and support software package that utilizes existing plant instrumentation to provide incore thermocouple temperatures, reactor coolant system cold leg temperatures, control bank demand positions, power range detector output, and reactor power measurement data. These data are sent by the plant computer in the form of a file that BEACON can interpret to perform nodal power distribution prediction calculations.

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

The Movable In-Core Detector System (MIDS) is available to use for core power distribution analysis if BEACON data becomes unavailable. The MIDS system will also be used to calibrate BEACON.

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

4.0 Technical Analysis

The proposed changes allow PG&E to use the BEACON PDMS to improve the operational support for DCPP. The PDMS maintains an on-line 3-D nodal model that is continuously updated to reflect the current plant operating conditions. The following is a summary/excerpt of Brookhaven National Laboratory's (BNL) Technical Evaluation Report (TER) for WCAP-12472-P-A.

4.1 BEACON Core Monitoring Methodology

The BEACON core monitoring system uses the NRC-approved Westinghouse SPNOVA nodal method for core power distribution measurements. The SPNOVA data libraries and core models are consistent with the NRC-approved Westinghouse PHOENIX/ANC design models and have been benchmarked against operating reactor measurements.

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

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

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

For the application of BEACON to DCPP, PG&E will not take credit for the continuous monitoring of the power distribution. Instead, PG&E will use BEACON as a Technical Specification monitor of present peaking factor limits and the transient initial condition limits will not be relaxed.

TS 3.3.3, "Post Accident Monitoring Instrumentation," requires that two incore thermocouple channels per quadrant be operable, with each channel consisting of two thermocouples. DCPP Units 1 and 2 each have 65 thermocouples, and TS 3.3.3 requires a minimum of sixteen (in the correct quadrant and train).

The criteria for the core-exit thermocouples, with BEACON operable, require at least 25% of the thermocouples, with at least two per quadrant, with the added requirement that the operable pattern normally covers all internal fuel assemblies within a chess "knight" move (an adjacent plus a diagonal square away) or more frequent calibration is required. Calibration with the movable incore detectors is required every 180 effective full-power days. However, calibration is required every 30 days when the knight's move requirement is not satisfied. The accuracy of the power distribution information with decreased incore or thermocouple detector operability has been analyzed by Westinghouse and penalties are applied to the calculated peaking factors (refer to TER section 2.3). The review has concluded that the minimum available incore and thermocouple detectors, when coupled with the increased uncertainty penalties, provide reasonable and acceptable power distribution information.

4.2 Model Calibration and Uncertainty

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

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

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

As an initial assessment of the power distribution calculation, Westinghouse performed detailed comparisons of BEACON to the predictions of the INCORE system presently used at Westinghouse plants. INCORE is a data analysis code written to process information obtained by the movable incore detector system (MIDS) in Westinghouse pressurized water reactors. PG&E is presently using INCORE for processing information obtained by the MIDS and verifying Technical Specification surveillance requirements. 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. The averages of the standard deviation between the BEACON result and the actual measured reaction rate are 1.5% for assemblies with power greater than the average (1.0) value and 2% for all measured assemblies (WCAP-12472-P-A section 4.1.1). The averages of the standard deviation of the inferred assembly power between BEACON and INCORE are 1.10% for assemblies with power greater than the average (1.0) value and 1.37% for all assemblies (WCAP-12472-P-A Table 4-6). From the result of this study,

Westinghouse concluded that the BEACON processing of the incore flux map and the inferred assembly power distribution accuracy is statistically consistent with the INCORE computer code.

The uncertainties applied to the BEACON power distribution measurements are different than those applied to the traditional flux map systems because BEACON uses a more comprehensive set 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 summarized/excerpted as follows:

> "Due to the change in reactor statepoint, SPNOVA modeling approximations and instrumentation error, a model calibration uncertainty is introduced into the BEACON predictions. Westinghouse has evaluated this uncertainty by comparing BEACON predicted and measured incore reaction rates over four cycles and a range of operating conditions, and has found that the model calibration uncertainty was very small and varied only slightly for these comparisons."
>
> "The thermocouple calibration uncertainty is due to the change in reactor statepoint and to instrument error. Westinghouse has evaluated this uncertainty by comparing the assembly powers inferred from the thermocouples to SPNOVA incore-corrected assembly powers. Comparisons for three plants and a range of operating conditions indicate a difference of less than a few percent at full power. The observed calibration uncertainty increased at lower powers due to the reduced enthalpy rise and changes in cross-flow."
>
> "In order to determine the axial power distribution uncertainty, Westinghouse has compared SPNOVA incore-updated and SPNOVA excore-updated predictions of the axial power shape. These comparisons included a range of fuel burnups and rod insertions, and indicated a 95/95 upper tolerance limit of less than a few percent with a slight dependence on rod movement since calibration."
>
> "Based on an extensive set of calibration data, the model calibration uncertainty is observed to increase as the calibration interval (in units of fuel burnup) increases. Using the observed fuel burnup dependence, an additional assembly power uncertainty is determined to account for the effects of increased calibration interval."
>
> "The failure of (incore and thermocouples) detectors (used by) the BEACON system results in a relaxation of the local calibration to measurement, and an increase in the power distribution uncertainty.

The effect of random failures of the incore and thermocouple detectors on the assembly power was evaluated for failure rates of up to 75%. The assembly power uncertainty was found to increase linearly with incore detector failure and quadratically with the failure of thermocouples. "

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

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

4.3 Acceptance Criteria

In the NRC Safety Evaluation Report on WCAP-12472-P-A, NRC staff evaluated the BEACON methodology, the uncertainty analysis, and the operation of the overall system and concluded that the BEACON system is acceptable for performing core monitoring and operations support functions for Westinghouse pressurized water reactors (PWR) but subject to certain conditions. These conditions are listed below followed by PG&E's evaluation of compliance with these conditions.

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.*

Response:

Although not specifically described in this submittal, cycle-specific BEACON calibrations performed before startup and at BOC will ensure that power peaking uncertainties provide 95% probability upper tolerance limits at the 95% confidence level. These calibrations are 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 database.*

Response:

Diablo Canyon Power Plant (DCPP) is a Westinghouse 4-loop Nuclear Steam Supply System (NSSS) with Westinghouse movable incore instrumentation. All fuel is presently of Westinghouse manufacture. Therefore, DCPP does not currently differ significantly from the plants that form the WCAP database and no additional review of WCAP applicability to DCPP is necessary.

During the review of the Westinghouse topical report WCAP-12472-P, the NRC requested additional information on how BEACON treats core loadings with fuel designs from multiple fuel vendors, and the impact to the BEACON uncertainty analysis. Westinghouse responded that for all BEACON applications, the previous operating cycle is examined to established reference uncertainties. This examination accounts for loading of fuel supplied by multiple vendors by comparing a BEACON model to actual operating data over the cycle. At the beginning of cycle, thermocouple data is verified and calibration/uncertainty components are updated as necessary. In addition, the initial flux mapping at the start of the cycle insures model calibration factors that reflect the actual fuel in the reactor before the BEACON system is declared operable.

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

Response:

Because the WCAP describes an application of BEACON where the core operating limits are changed and PG&E proposes to use BEACON as a core Technical Specification monitor of our present limits, this condition does not directly apply to this submittal.

5.0 Regulatory Analysis

5.1 No Significant Hazards Consideration

PG&E 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 change involve a significant increase in the probability or consequences of an accident previously evaluated?*

Response: No

The power distribution monitoring system (PDMS) performs continuous core power distribution monitoring. This system utilizes the NRC-approved Westinghouse proprietary computer code, the 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. It in no way provides 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 Final Safety Analysis Report Update (FSARU).

Continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This results in more time (i.e., earlier determination of an adverse condition developing) for operator action prior

to having an adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an accident.

Each accident analysis addressed in the Diablo Canyon Power Plant FSARU is examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC-approved reload design methodologies, to ensure that the transient evaluation of reload cores are bounded by previously accepted analyses. This examination, which is performed in accordance with the requirements set forth in 10 CFR 50.59, "Changes, tests and experiments," ensures that future reloads will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

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

Response: No

The implementation of the PDMS has no influence or impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with 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 more adverse conditions and will not result in any increase in the challenges to safety systems. The cycle specific variables required by the PDMS are calculated using NRC-approved methods. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded.

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

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

Response: No

The margin of safety is not affected by the implementation of the PDMS. The margin of safety provided by current TS remains unchanged. The proposed changes 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. The proposed changes continue to ensure that appropriate actions will be taken if limits are violated. These actions remain unchanged.

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

Based on the above evaluations, PG&E concludes that the activities associated with the proposed amendment present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

5.2 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 DCPP, Units 1 and 2, does not replace, eliminate, or modify existing plant instrumentation. The PDMS software runs on a workstation connected to the plant process computer. The PDMS combines inputs from currently installed plant instrumentation and design data generated for each fuel cycle. Together, this provides a means to continuously monitor the power distribution limits including limiting peaking factors and quadrant power tilt ratio.

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.

6.0 Environmental Consideration

PG&E has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7.0 References

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

2. License Amendment No. 142 to Facility Operating License No. NPF-12 Regarding Best Estimate Analyzer for Core Operations – Nuclear (BEACON), Virgil C. Summer Nuclear Station, Unit No. 1.

3. License Amendment Nos. 237 and 218 to Facility Operating License Nos. 50-272 and 50-311 for Salem Nuclear Generating Station, Unit Nos. 1 and 2.

4. License Amendment Nos. 116 to Facility Operating License Nos. NPF-37 and NPF-66 for the Byron Station, Unit Nos. 1 and 2, and License Amendment Nos. 110 for Facility Operating License Nos. NPF-72 and NPF-77 for the Braidwood Station, Unit Nos. 1 and 2.

7.1 Precedent

The BEACON-TSM was approved by the NRC for use at the Salem Nuclear Generating Station in License Amendment Nos. 237 (Unit 1) and 218 (Unit 2), on November 6, 2000, and at the Virgil C. Summer Nuclear Station in License Amendment No. 142 (Unit 1), on April 9, 1999. This LAR is consistent with the amendments issued to the Salem Nuclear Generating Station and the Virgil C. Summer Nuclear Station.

These changes were also approved for the Byron Station in License Amendment Nos. 116 (Units 1 and 2), on February 13, 2001, and for the Braidwood Station in

License Amendment Nos. 110 (Units 1 and 2), on February 13, 2001. The application of BEACON to the Byron/Braidwood stations uses BEACON to take credit for the direct and continuous monitoring of Departure from Nuclear Boiling Ratio (DNBR), whereas the application of BEACON to DCPP is for power distribution surveillances.

Proposed Technical Specification Changes

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable position indicators indirectly by using ~~movable incore detectors~~core power distribution measurement information.	Once per 8 hours
		OR	
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. More than one DRPI per group inoperable.	B.1	Place the control rods under manual control	Immediately
	AND		
	B.2	Monitor and record reactor coolant system Tavg.	Once per 1 hour
	AND		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3	Verify the position of the rods with inoperable position indicators indirectly by using ~~movable incore detectors~~core power distribution measurement information.	Once per 8 hours
	AND		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable position indicators indirectly by using ~~movable incore detectors~~core power distribution measurement information.	4 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. One demand position indicator per bank inoperable for one or more banks.	D.1.1	Verify by administrative means all DRPIs for the affected banks are OPERABLE.	Once per 8 hours
	AND		
	D.1.2	Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per 8 hours
	OR		
	D.2	Reduce THERMAL POWER to ≤ 50% RTP	8 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.2.1.2	-------------------------NOTE------------------------- If $F_Q^C(Z)$ measurements indicate $$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)}\right]$$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits: or b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive ~~flux maps~~power distribution measurements indicate $$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)}\right]$$ has not increased. --	
	Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND (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. ---------- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	-----NOTE----- Not required to be performed until 12 hours after the input from one or more Power Range Neutron Flux channels is inoperable with THERMAL POWER > 75% RTP. ---------- Verify QPTR is within limit using ~~the movable incore detectors~~core power distribution measurement information.	12 hours

SURVEILLANCE REQUIREMENTS

--NOTE--

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	-------------------------NOTE------------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP, but prior to exceeding 30% RTP. --- Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculation results exceed power range channel output by more than + 2% RTP.	24 hours
SR 3.3.1.3	-------------------------NOTE------------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. --- Compare results of incore power distribution measurements ~~the incore detector measurements~~ to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%.	31 effective full power days (EFPD)
SR 3.3.1.4	-------------------------NOTE------------------------- This Surveillance must be performed on the reactor trip bypass breaker, for the local manual shunt trip only, prior to placing the bypass breaker in service. --- Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.6	-----NOTE----- Not required to be performed until 72 hours after THERMAL POWER ≥ 75% RTP. ----- Calibrate excore channels to agree with incore power distribution ~~detector~~measurements.	92 EFPD
SR 3.3.1.7	-----NOTE----- 1. Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 2. For source range instrumentation, this Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ----- Perform COT.	92 days

(continued)

Revised Technical Specification Pages

3.1.7 Rod Position Indication

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	Once per 8 hours
		<u>OR</u>	
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. More than one DRPI per group inoperable.	B.1	Place the control rods under manual control	Immediately
	<u>AND</u>		
	B.2	Monitor and record reactor coolant system Tavg.	Once per 1 hour
	<u>AND</u>		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3	Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	Once per 8 hours
	<u>AND</u>		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable position indicators indirectly by using core power distribution measurement information.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. One demand position indicator per bank inoperable for one or more banks.	D.1.1	Verify by administrative means all DRPIs for the affected banks are OPERABLE.	Once per 8 hours
	<u>AND</u>		
	D.1.2	Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per 8 hours
	<u>OR</u>		
	D.2	Reduce THERMAL POWER to ≤ 50% RTP	8 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	----------NOTE---------- If $F_Q^C(Z)$ measurements indicate $$\text{maximum over z}\left[\frac{F_Q^C(Z)}{K(Z)}\right]$$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits: or b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive power distribution measurements indicate $$\text{maximum over z}\left[\frac{F_Q^C(Z)}{K(Z)}\right]$$ has not increased. ------------------------------	
	Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> (continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. -------------------- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	----------NOTE---------- Not required to be performed until 12 hours after the input from one or more Power Range Neutron Flux channels is inoperable with THERMAL POWER > 75% RTP. -------------------- Verify QPTR is within limit using core power distribution measurement information.	12 hours

SURVEILLANCE REQUIREMENTS

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

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	-----NOTE----- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP, but prior to exceeding 30% RTP. ----- Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculation results exceed power range channel output by more than + 2% RTP.	24 hours
SR 3.3.1.3	-----NOTE----- Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. ----- Compare results of incore power distribution measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%.	31 effective full power days (EFPD)
SR 3.3.1.4	-----NOTE----- This Surveillance must be performed on the reactor trip bypass breaker, for the local manual shunt trip only, prior to placing the bypass breaker in service. ----- Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.6	-------------------------NOTE------------------------- Not required to be performed until 72 hours after THERMAL POWER ≥ 75% RTP. --- Calibrate excore channels to agree with incore power distribution measurements.	92 EFPD
SR 3.3.1.7	-------------------------NOTE------------------------- 3. Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 4. For source range instrumentation, this Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. --- Perform COT.	92 days

(continued)

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(for information only)

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BASES

APPLICABLE ANALYSIS (continued)	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~~ core power distribution measurement information. 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 10CFR50.36(c)(2)(ii).
LCO	The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but do not impact trippability, do not necessarily result in rod inoperability. The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed. The requirement to maintain rod alignment is met by comparing individual rod DRPI indication and bank demand position indication to be within plus or minus 12 steps. If one of these position indicators become inoperable, the conditions of this LCO are still met by compliance with LCO 3.1.7. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

(continued)

BASES

ACTIONS

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 (continued)

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the ~~incore flux mapping system and~~core power distribution measurement information 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. The accident analyses of FSAR Chapter 15 are to be used to identify the appropriate design bases events requiring re-evaluation. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

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

D.1.1 and D.1.2

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

Additionally, the requirements of LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," apply if the misaligned rods are not within the required insertion limits.

(continued)

BASES

LCO (continued)

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

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

APPLICABILITY

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

ACTIONS

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

A.1

When one DRPI per group fails, the position of the rod may still be determined indirectly by use of the ~~movable incore detectors~~core power distribution measurement information. Core power distribution measurement information can be obtained from flux maps using the movable incore detectors, or from an OPERABLE Power Distribution Monitoring System (PDMS) (Reference 4). The Required Action may also be ensuring, at least once per 8 hours, that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SDM is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

(continued)

BASES

ACTIONS (continued)

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1, B.2, B.3, and B.4

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via ~~movable incore detectors~~power distribution measurement information, this will minimize the potential for rod misalignment.

The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in the Condition. Monitoring and recording reactor coolant T_{avg} help assure that significant changes in power distribution and SDM are avoided. The once per 1 hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

The position of the rods can be determined indirectly by use of the ~~movable incore detectors~~power distribution measurement information. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided that the nonindicating rods have not moved. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Allowed Outage Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication.

Based on operating experience, normal power operation does not require excessive rod movement. If one or more control rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 or B.1 are still appropriate but must be initiated promptly under Required Action C.1 to begin indirectly verifying that these rods are still properly positioned, relative to their group positions. If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions using the movable incore detectors, or other power distribution measurement methods.

D.1.1 and D.1.2

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

D.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power to ≤ 50% RTP.

E.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.1.7.1 Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Verification at 24, 48, 120, and 228 steps withdrawn for the control and shutdown banks provides assurance that the DRPI is operating correctly over the full range of indication. This surveillance is performed prior to reactor criticality after each removal of the reactor head, since there is potential for unnecessary plant transients if the SR were performed with the reactor at power.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 13. 2. FSAR, Chapter 15. 3. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control and F_Q Surveillance Technical Specification," February 1994. 4. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

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

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

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1. 6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

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

$F_Q(Z)$ is not directly measurable but is inferred from a power distribution ~~map~~measurement obtained with either the movable incore detector system or from an OPERABLE Power Distribution Monitoring System (PDMS) (Reference 3). The results of the power distribution ~~map~~measurement are analyzed to derive a measured value for $F_Q(Z)$. These measurements are generally taken with the core at or near equilibrium conditions.

However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations, such as load following.

To account for these possible variations, a transient $F_Q(Z)$ is also calculated based on the steady state value of $F_Q(Z)$. In this case, the steady state $F_Q(Z)$ is adjusted by an elevation dependent factor, W(Z), that accounts for the calculated 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.

APPLICABLE SAFETY ANALYSES

This LCO's principal effect is to preclude core power distributions that could lead to violation of the following fuel design criterion:

> During a large break loss of coolant accident (LOCA), there is a high level of probability that the peak cladding temperature will not exceed 2200° F (Ref. 1).

(continued)

BASES

LCO (continued)

An $F_Q^C(Z)$ evaluation requires obtaining ~~an incore flux map~~a power distribution measurement in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. The computed heat flux hot channel factor, $F_Q^C(Z)$ is obtained by the equation:

$$F_Q^C(Z) = F_Q^M(Z)\ \cancel{(1.03)\ (1.05)}\ \underline{U_{FQ}}$$

where ~~1.03~~$\underline{U_{FQ}}$ is a factor that accounts for fuel manufacturing tolerances and ~~1.05 is a factor that accounts for flux map~~ measurement uncertainty.

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

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

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.

Calculate the percent $F_Q(Z)$ exceeds its limit by the following expression:

$$\left\{\left(\underset{\text{over z}}{\text{maximum}}\left[\frac{F_Q^C(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)}\right]\right)-1\right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{\left(\underset{\text{over z}}{\text{maximum}}\left[\frac{F_Q^C(z) \times W(z)}{\frac{F_Q^{RTP}}{0.5} \times K(z)}\right]\right)-1\right\} \times 100 \quad \text{for } P < 0.5$$

The $F_Q(Z)$ limits define limiting values for core power peaking that, with a high level of probability, preclude peak cladding temperatures above 2200° F during either a large or small break LOCA

This LCO requires operation within the bounds assumed in the safety analyses. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produce unacceptable

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

at RTP (or any other level for extended operation). Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainties associated with the measurement are valid. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 20% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 20% higher than that power at which $F_Q(Z)$ was last measured.

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from ~~incore flux map~~core power distribution measurement results and $F_Q^C(Z) = F_Q^M(Z)$ ~~(1.03) (1.05)~~U_{FQ} (Ref. 2). The value of U_{FQ} is determined using the formulation provided in the COLR. $F_Q^C(Z)$ is then compared to its specified limits.

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

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP (and meeting the 100% RTP $F_Q(Z)$ limit) provides assurance that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 20% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 24 hours after achieving equilibrium conditions at this higher power level to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2

Because ~~flux maps~~power distribution measurements are taken either at, or ~~in~~near equilibrium conditions, the variations in power distribution resulting from normal operational maneuvers are not typically present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

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

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

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

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

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. When $F_Q^W(Z)$ is determined, an evaluation of the expression below is required to account for any increase to $F_Q^C(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over z}\left[\frac{F_Q^C(Z)}{K(Z)}\right]$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by a factor ≥ 2 percent which is specified in the COLR, or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection. Performing the Surveillance in MODE 1 prior to exceeding 75% RTP or at a reduced power at any other time, and meeting the 100% RTP $F_Q(Z)$ limit, provides assurance that the $F_Q(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels $\geq$ 20% RTP above the THERMAL POWER of its last verification, 24 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

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

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

REFERENCES

1. 10 CFR 50.46, 1974.
2. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
3. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

B 3.2 POWER DISTRIBUTION LIMITS

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

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during normal operation, operational transients, and any transient condition arising from events of moderate frequency 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.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution ~~map~~measurement obtained with either the movable incore detector system or from an OPERABLE Power Distribution Monitoring System (PDMS) (Reference 4). Specifically, the results of the power distribution ~~map~~measurement are analyzed 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. Compliance with these LCOs, along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

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

(continued)

BASES

ACTIONS (continued)

A.1.2.1 and A.1.2.2

If the value of $F_{\Delta H}^{N}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux—High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints; however, for extended operations at the reduced power level, the reduced trip setpoints are required to protect against events involving positive reactivity excursions. This is a sensitive operation that may inadvertently actuate the Reactor Protection System.

A.2

Once actions have been taken to restore $F_{\Delta H}^{N}$ to within its limits per Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, ~~an incore flux map~~a power distribution measurement (SR 3.2.2.1) must be obtained and the measured value of $F_{\Delta H}^{N}$ 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~~an incore flux map, perform the required calculations, and evaluate $F_{\Delta H}^{N}$.

(continued)

BASES

ACTIONS (continued)

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 exceeding the $F^N_{\Delta H}$ limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP. SR 3.2.2.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints. When $F^N_{\Delta H}$ is measured at reduced power levels, the allowable power level is determined by evaluating $F^N_{\Delta H}$ for higher power levels.

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.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

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

The value of $F^N_{\Delta H}$ is determined by either using the movable incore detector system to obtain a flux distribution map or from the power distribution information provided by an OPERABLE PDMS. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux ~~distributions~~ distribution map. The limit of $F^N_{\Delta H}$ in the COLR allows for 4% measurement uncertainties applicable for either flux distribution maps or PDMS-derived $F^N_{\Delta H}$ values (reference 4).

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.2.2.1 (continued) After each refueling, $F_{\Delta H}^{N}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F_{\Delta H}^{N}$ limits are met at the beginning of each fuel cycle. Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP $F_{\Delta H}^{N}$ limit, provides assurance that the $F_{\Delta H}^{N}$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased. The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F_{\Delta H}^{N}$ limit cannot be exceeded for any significant period of operation.
REFERENCES	1. Regulatory Guide 1.77, Rev. 0, May 1974. 2. 10 CFR 50, Appendix A, GDC 26. 3. 10 CFR 50.46. 4. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

ACTIONS

A.2 (continued)

continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support ~~flux mapping~~ obtaining a power distribution measurement. Power distribution information can be obtained using either the movable incore detectors or from an OPERABLE Power Distribution Monitoring System (PDMS) (Reference 4). A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map to verify peaking factors and that the incore quadrant power tilt and QPTR are consistent. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

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

(continued)

ACTIONS

A.4 (continued)

evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR remains above the 1.02 limit and a evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limit prior to increasing THERMAL POWER to above the limit of Required Action A.1. This is done to detect any subsequent significant changes in QPTR.

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

A.6

Once the excore detectors are normalized to restore QPTR to within limit (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of achieving equilibrium conditions. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support flux mapping. As an added precaution, if the peaking factor verification cannot be performed within 24 hours due to non-equilibrium core conditions, a maximum time of 48 hours is allowed for the completion of the verification.

This Completion Time is intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one or more power range channels are 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, incore 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, the incore tilt result may be compared against previous ~~flux maps~~tilt values 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~~power distribution measurement data.

REFERENCES

1. 10 CFR 50.46.
2. Regulatory Guide 1.77, Rev 0, May 1974.
3. 10 CFR 50, Appendix A, GDC 26.
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.3 (continued)

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

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. In accordance with WCAP 10271, very specific circumstances are related to the use of this bypass condition. Since the NIS channels are not designed with Bypass-capable logic that meets the requirements of IEEE 279, the provisions for bypass only apply to a specific type of channel failure. To apply, the channel must fail in such a way that it does not trip the bistables. With this type of failure, the channel may be returned to service and considered "bypassed" under this Note. Specifically, the bypass condition is the state when a failed channel is taken out of the forced "tripped" state and placed in operation. Due to the failed nature of the channel, the channel cannot be assumed to be OPERABLE, and is therefore considered to be in a state of bypass when the channel failure is such that its bistables are not tripped. The provisions of WCAP 10271 specifically prohibit the use of jumpers or lifted leads to bypass these channels. In this configuration, a second channel can be tested or setpoints adjusted with the channel in the tripped mode without completing reactor trip logic. The 4 hour time limit is justified in Reference 7.

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. The performance of SR 3.2.4.2 per ACTION D.2.2 is subject to the SR 3.2.4.2 note. 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 ~~this movable incore detectors~~ core power distribution measurement information once per 12 hours may not be necessary.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.14. The bypass breaker test shall include a local manual shunt trip only. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

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

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The seismic trip is tested every 31 days on a STAGGERED TEST BASIS. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition with the RTB bypass breaker installed, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function including operation of the P-7 permissive which is a logic function only. The P-7 alarm circuit is excluded from this testing since it only mimics the actions of the SSPS and cannot prevent the permissive from performing its function. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to 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~~power distribution measurements. The incore power distribution measurements can be obtained using the movable incore detectors or an OPERABLE Power Distribution Monitoring System (PDMS) (Reference 25). 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.6 *(continued)*

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is ≥ 75% RTP and that 72 hours after thermal power is ≥ 75% RTP is allowed for performing the first surveillance after reaching 75% RTP. The SR is deferred until a scheduled testing plateau above 75% RTP is attained during the post-outage power ascension. During a typical post-refueling power ascension, it is usually necessary to control the axial flux difference at lower power levels through control rod insertion. After equilibrium conditions are achieved at the specified power plateau, a ~~flux map~~power distribution measurement must be taken and the required data collected. The data is typically analyzed and the appropriate excore calibrations completed within 48 hours after achieving equilibrium conditions. An additional time allowance of 24 hours is provided during which the effects of equipment failures may be remedied and any required re-testing may be performed.

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

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

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days.

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

(continued)

BASES

REFERENCES (Continued)

8. WCAP 13632 - PA-1, Rev. 2 "Elimination of Pressure Sensor Response Time Testing Requirements."
9. FSAR, Chapter 9.2.7 & 9.2.2.
10. FSAR, Chapter 10.3 & 10.4
11. FSAR, Chapter 8.3.
12. DCM S-38A, "Plant Protection System"
13. WCAP-13878, "Reliability of Potter & Brumfield MDR Relays", June 1994.
14. WCAP-13900, "Extension of Slave Relay Surveillance Test intervals", April 1994.
15. WCAP-14117, "Reliability Assessment of Potter and Brumfield MDR Series Relays."
16. WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases," Revision 1, January 1978.
17. WCAP-11082, Rev. 5, "Westinghouse Setpoint Methodology for Protection Systems, Diablo Canyon Units 1 and 2, 24 Month Fuel Cycle Evaluation," January 1997.
18. NSP-1-20-13F Unit 1 "Turbine Auto Stop Low Oil Pressure."
19. NSP-2-20-13F Unit 2 "Turbine Auto Stop Low Oil Pressure."
20. J-110 "24 Month Fuel Cycle Allowable Value Determination / Documentation and ITDP Uncertainty Sensitivity."
21. IEEE Std. 338-1977.
22. License Amendment 61/60, May 23, 1991.
23. Westinghouse Technical Bulletin ESBU-TB-92-14-R1, "Decalibration Effects of Calorimetric Power Measurements on the NIS High Power Reactor Trip at Power Levels less than 70% RTP," dated February 6, 1996.
24. DCPP NSSS Calculation N-212, Revision 1.
25. <u>WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.</u>