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January 5, 2012

PG&E Letter DCL-12-002

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

10 CFR 50.90

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
License Amendment Request 12-01,
Revision to the Updated Final Safety Analysis Report Section 4.3.2.2, "Power Distribution;" and Technical Specification Bases 3.1.7, "Rod Position Indication;" 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor" ($F^N_{\Delta H}$); 3.2.4, "Quadrant Power Tilt Ratio" (QPTR); and 3.3.1, "Reactor Trip System Instrumentation" (RTS).

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby requests approval of the enclosed proposed amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP) respectively. The enclosed license amendment request (LAR) proposes to revise the DCPP Units 1 and 2 Updated Final Safety Analysis Report (UFSAR) Section 4.3.2.2, "Power Distribution;" and Technical Specification (TS) Bases Sections 3.1.7, "Rod Position Indication;" 3.2.1, "Heat Flux Hot Channel Factor" ($F_Q(Z)$), 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor" ($F^N_{\Delta H}$); 3.2.4, "Quadrant Power Tilt Ratio" (QPTR); and 3.3.1, "Reactor Trip System Instrumentation" (RTS). These revisions are to allow use of the BEACON Power Distribution Monitoring System (PDMS) methodology described in WCAP-12472-P-A, Addendum 1-A, "BEACON Core Monitoring and Operation Support System," dated January 2000.

The Enclosure provides a detailed description and technical evaluation of the proposed changes for the PDMS, including PG&E's determination that the proposed changes involve no significant hazards.

Attachment 1 provides marked up pages to the DCPP UFSAR.

Attachment 2 provides marked-up TS Bases page changes for information only.

PG&E requests approval of this LAR no later than January 16, 2013. PG&E requests the license amendments be made effective upon NRC issuance, to be implemented within 120 days from the date of issuance.

PG&E makes no regulatory commitments (as defined by NEI 99-04) in this letter. This letter includes no revisions to existing regulatory commitments.

In accordance with site administrative procedures and the Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91, PG&E is sending a copy of this proposed amendment to the California Department of Public Health.

If you have any questions or require additional information, please contact Tom Baldwin at 805-545-4720.

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

Executed on January 5, 2012.

Sincerely,

James R. Becker
Site Vice President

kjse/4328 50386872
Enclosure
cc: Diablo Distribution
cc/enc: Gary W. Butner, Branch Chief, California Dept of Public Health
Elmo E. Collins, NRC Region IV
Michael S. Peck, NRC, Senior Resident Inspector
Alan B. Wang, NRR Project Manager

Evaluation of the Proposed Change

License Amendment Request 12-01
Revision to the Updated Final Safety Analysis Report Section 4.3.2.2, "Power Distribution," and Technical Specification Bases Sections 3.1.7, "Rod Position Indication," 3.2.1, "Heat Flux Hot Channel Factor" ($F_Q(Z)$), 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor" ($F^N_{\Delta H}$), 3.2.4, "Quadrant Power Tilt Ratio" (QPTR), and 3.3.1, "Reactor Trip System Instrumentation" (RTS)

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1. UFSAR Pages Markups
2. TS Bases Pages Markups (For Information Only)

EVALUATION

1. SUMMARY DESCRIPTION

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

The proposed changes would revise the Updated Final Safety Analysis Report (UFSAR), Section 4.3.2.2, "Power Distribution;" and Technical Specification (TS) Bases Sections 3.1.7, "Rod Position Indication;" 3.2.1, "Heat Flux Hot Channel Factor" ($F_Q(Z)$), 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor" ($F^N_{\Delta H}$); 3.2.4, "Quadrant Power Tilt Ratio" (QPTR); and 3.3.1, "Reactor Trip System Instrumentation" (RTS) to add a new reference for the Westinghouse BEACON Core Monitoring and Operation Support System methodology contained in WCAP-12472-P-A, Addendum 1-A.

2. DETAILED DESCRIPTION

UFSAR Section 4.3.2.2, "Power Distribution"

> Add WCAP-12472-P-A, Addendum 1-A, as reference in the text and list of references.

TS Basis 3.1.7, "Rod Position Indication":

> Add WCAP-12472-P-A, Addendum 1-A, as reference in the text and list of references.

TS Basis 3.2.1, "Heat Flux Hot Channel Factor":

> Add WCAP-12472-P-A, Addendum 1-A, as reference in the text and list of references.

TS Basis 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor":

> Add WCAP-12472-P-A, Addendum 1-A, as reference in the text and list of references.

TS Basis 3.2.4, "Quadrant Power Tilt Ratio (QPTR)":

> Add WCAP-12472-P-A, Addendum 1-A, as reference in the text and list of references.

TS Basis 3.3.1, "Reactor Trip System (RTS) Instrumentation":

Add WCAP-12472-P-A, Addendum 1-A, as reference in the text and list of references.

The BEACON Power Distribution Monitoring System (PDMS) currently is declared nonfunctional at DCPP since changes have been made by Westinghouse to the BEACON software that are not reflected in the BEACON WCAP-12472-P-A contained in the DCPP UFSAR. The BEACON PDMS serves as a three dimensional (3-D) core monitor, operational analysis tool, and operational support package. Currently, DCPP Units 1 and 2 only use BEACON to process reactor core flux maps performed using the movable incore detector system (MIDS). It is desired to make BEACON PDMS functional so that BEACON may be used for TS Monitoring (TSM). This will allow MIDS use for surveillances to be less frequent and will increase the usable life of the MIDS components.

In June 1991, the NRC approved Westinghouse neutronics SPNOVA code for use in licensing (Reference 1). The original version used mathematical formulae and methods (e.g. Green's Functions) to speed operation on early computers with limited capabilities.

In August 1994, the NRC approved the topical report for BEACON, WCAP-12472-P-A, for use in licensing (Reference 2). The BEACON code described in WCAP-12472-P-A used the neutronics code SPNOVA.

In September 1999, the NRC approved BEACON WCAP-12472-P-A, Addendum 1, and the NRC approved version was subsequently issued in WCAP-12472-P-A, Addendum 1-A, dated January 2000. This addendum added in the SPNOVA code the optional "Use of 3-D ANC Code" that includes the Nodal Expansion Method. The current version of BEACON software uses the ANC code and Nodal Expansion Method. The addendum also added optional use of fixed incore self-powered detectors (SPDs) to BEACON (Reference 5). DCPP does not use fixed incore SPDs.

In April 2004, the NRC approved the License Amendments for DCPP Units 1 and 2 to use BEACON PDMS (Reference 6). These amendments included revision of applicable TS and included the original BEACON WCAP-12472-P-A as a methodology referenced in the TS Bases.

PG&E proposes to use the version of BEACON described in WCAP-12472-P-A, Addendum 1-A, for PDMS and TSM. This change requires addition of WCAP-12472-P-A, Addendum 1-A to the UFSAR and TS Bases.

The proposed UFSAR changes are noted on the marked-up UFSAR pages provided in Attachment 1.

The proposed TS Bases changes are noted on the marked-up TS Bases pages provided in Attachment 2 for information only.

3. TECHNICAL EVALUATION

System Description

DCPP employs two methods for performing core power distribution calculations. Either BEACON or the MIDS can be used to satisfy core power peaking TS Surveillance Requirements. The BEACON system also provides an on-line monitoring of the reactor core using current plant instrumentation. Excore neutron detectors and core exit thermocouples are used with a 3-D calculated power distribution on a nearly continuous basis. The generation of 3-D power distribution involves periodic calibration of BEACON using the MIDS, a 3-D nodal simulation of the core, and frequent processing of excore neutron detector and core exit thermocouple readings. Power distributions can be compared to TS limits for surveillance purposes.

The BEACON system also allows processing of MIDS flux map data. Results can be compared to TS limits for surveillance purposes. BEACON can provide core depletion and fuel isotopic distribution information by calculating 3-D power distributions and associated neutron flux distributions. BEACON can be used to provide core reactivity calculations such as estimated critical conditions and shutdown margin calculations in order to meet TS requirements and can provide load follow simulations.

Current Licensing Basis and Current Licensing Basis Acceptance Criteria

The current licensing basis for BEACON methodology at DCPP is WCAP-12472-P-A. In the NRC's Safety Evaluation Report (SER) acceptance of the WCAP, three conditions were specified that are addressed below:

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

Power peaking uncertainties are not changed in WCAP-12472-P-A, Addendum 1-A. Westinghouse continues to check the uncertainties in each DCPP unit/cycle core design to ensure they remain valid.

2) In order to ensure that the assumptions made in the BEACON uncertainty analysis remain valid, the generic uncertainty components may require reevaluation when BEACON is applied to plant or core designs that differ sufficiently to have a significant impact on the WCAP-12472-P-A data-base (Section-3.4).

Westinghouse continues to check each unit/cycle core design to ensure that it conforms to expected results. PG&E informs Westinghouse of plant changes that can impact BEACON each cycle as part of the core reload design process. Should any plant instrumentation changes occur during an operating cycle, PG&E compares these to the minimum required channels in TS to ensure that the minimum requirements are met. This verifies that uncertainty requirements are met.

3) The BEACON TS 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 (Section 3.6).

The TS were previously revised as part of implementation of the DCPP license amendments that allowed use of the BEACON system described in WCAP-12472-P-A. No additional changes are required to implement the methodology contained in WCAP-12472-P-A, Addendum 1-A, for PDMS and TSM.

This license amendment request does not propose any changes that could impact the technical basis for the methodology described in WCAP-12472-P-A, such as rod insertion limits, thermocouple calibration uncertainty, or the criteria for updating the reference power distribution.

Changes to plant/cycle-specific component performance continue to be confirmed each cycle before or during startup. Westinghouse continues to monitor core designs that may differ significantly from the BEACON WCAP database for changes in generic uncertainty components.

WCAP-12472-P-A, Addendum 1-A

The following information is provided to address the Safety Evaluation Report Acceptance Criteria contained in WCAP-12472-P-A, Addendum 1-A.

"PHOENIX/ANC is a proven and licensed methodology that is supported by many critical experiments and plant data." The ANC code will be used as part of the methodology in WCAP-12472-P-A, Addendum 1-A, and is expected to provide increased accuracy in 3-D power distribution calculations.

The NRC comments regarding fixed, incore SPDs do not apply to DCPP. DCPP has only movable incore detectors.

The NRC SERs for WCAP-12472-P-A and WCAP-12472-P-A, Addendum 1-A, contain a condition for acceptance regarding plant-cycle-specific uncertainty components. The staff required that "plant-cycle-specific components will be determined on a plant-specific basis and confirmed each cycle."

Westinghouse continues to evaluate whether DCPP unit/cycle core designs conform to the BEACON database so as not to impact generic uncertainty components. PG&E informs Westinghouse of plant changes that can impact BEACON each cycle as part of the core reload design process.

Based on a review of the criteria contained in the SER for WCAP-12472-P-A and WCAP-12472-P-A, Addendum 1-A, it is concluded that the criteria are met and therefore the proposed change is acceptable.

System Safety Analysis Basis

The BEACON PDMS methodology contained in Westinghouse Topical Report WCAP-12472-P-A, Addendum 1-A, that is proposed to be added as a reference to the UFSAR and TS bases, has already been approved by the NRC.

No changes are being made to the UFSAR accident analyses, and no TS limits or surveillance frequencies are being changed. The newer methodology for calculating 3-D power distributions in WCAP-12472-P-A, Addendum 1-A, is more accurate than the methodology in the original WCAP-12472-P-A.

The proposed change is limited to adding reference to the BEACON methodology, WCAP-12472-P-A, Addendum 1-A, to the UFSAR and TS Bases.

System Summary/Conclusion

Since all applicable criteria in the NRC SER for WCAP-12472-P-A, Addendum-1-A, are met, the addition of NRC-approved WCAP-12472-P-A, Addendum 1-A, to the UFSAR and TS Bases is judged to be acceptable.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses include TS as part of the facility operating license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. Pursuant to 10 CFR 50.36, TS are required to include items in the five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS. Under 10 CFR 50.36(c)(2)(ii), a limiting condition for operation must be included in TS for any item meeting one of the following four criteria:

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

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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

Those items that do not fall within or satisfy any of the above criteria do not need to be included in Section 3 of TS. The PDMS instrumentation does not meet any of the criteria of 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. PG&E has included the PDMS instrumentation requirements in equipment control guidelines (ECGs) as part of implementation of the amendments for WCAP-12472-P-A. The ECGs are plant-specific administrative controls, similar to TS controls, but ECGs are controlled by PG&E in accordance with 10 CFR 50.59.

4.2 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 proposed change is to revise the Updated Final Safety Analysis Report to allow the use of the BEACON code methodology contained in WCAP-12472-P-A, Addendum 1-A. The BEACON code will be used to perform core flux mapping to support the performance of Technical Specification surveillances for power distribution limits and the use of the BEACON code will not cause an accident.

No physical changes are being made to the plant. With the change, Diablo Canyon Power Plant will continue to operate within the power distribution limits contained in the plant Technical Specifications.

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

2. Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?

Response: No.

The proposed change does not involve any physical changes to the plant. The BEACON code performs flux mapping of the core and is not used to control the performance of any plant equipment. Therefore, use of the BEACON code cannot cause an accident. If it is determined that the plant is not operating within the power distribution limits during the performance of a Technical Specification Surveillance using BEACON, then the applicable Technical Specification Condition and Required Action(s) will be entered.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

With the use of the BEACON code methodology contained in WCAP-12472-P-A, Addendum 1-A, the plant will continue to operate within the power distribution limits contained in the plant Technical Specifications. The use of the BEACON code does not involve any changes to the fuel, reactor vessel, or containment fission product barriers. The use of the BEACON code methodology includes requirements for control of uncertainties associated with use of the methodology and therefore there will be no impact on the accident analyses that are contained in the Updated Final Safety Analysis Report.

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

Based on the above evaluation, PG&E concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

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

5. ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. Chao, Y.A., et. al., “SPNOVA – A Multidimensional Static and Transient Computer Program for PWR Core Analysis,” WCAP-12983-A, June 1991.
2. Beard, C.L., et. al., “BEACON, Core Monitoring and Operations Support System,” WCAP-12472-P-A, August 1994.
3. Chao, Y.A., et. al., “Westinghouse Dynamic Rod Worth Measurement Technique,” WCAP-13360-P-A, January 1996.
4. Westinghouse letter Liparulo, N.J. to Jones, R.C. (NRC), “Process Improvement to the Westinghouse Neutronics Code System,” March 29, 1996.
5. Morita, T., “BEACON, Core Monitoring and Operations Support System,” WCAP-12472-P-A, Addendum 1-A, January 2000.
6. NRC Issuance of License Amendment Nos. 164 and 166 to Facility Operating License Numbers 80 and 82 for Diablo Canyon Power Plant, Units 1 and 2, respectively, March 31, 2004.

Updated Final Safety Analysis Report Markups material in the form of boron dissolved in the primary coolant and burnable absorber rods or boron coated fuel pellets.

Boric acid concentration in the primary coolant is varied to control and to compensate for long-term reactivity requirements, such as those due to fuel burnup, fission product poisoning, including xenon and samarium, burnable absorber material depletion, and the cold-to-operating moderator temperature change. Using its normal makeup path, the chemical and volume control system (CVCS) is capable of inserting negative reactivity at a rate of approximately 30 pcm/min when the reactor coolant boron concentration is 1000 ppm, and approximately 35 pcm/min when the reactor coolant boron concentration is 100 ppm. In an emergency, the CVCS can insert negative reactivity at approximately 65 pcm/min when the reactor coolant concentration is 1000 ppm, and 75 pcm/min when the reactor coolant boron concentration is 100 ppm. The peak xenon burnout rate is 25 pcm/min (Section 9.3.4 discusses the capability of the CVCS to counteract xenon decay). Rapid transient reactivity requirements and safe shutdown requirements are met with control rods.

As the boron concentration increases, the MTC becomes less negative. Using soluble poison alone would result in a positive MTC at beginning of life (BOL) at full power operating conditions. Therefore, burnable absorber rods are used to reduce the soluble boron concentration sufficiently to ensure that the MTC is not positive for full power operating conditions. During operation, the absorber content in these rods is depleted, thus adding positive reactivity to offset some of the negative reactivity from fuel depletion and fission product buildup. The depletion rate of the burnable absorber material is not critical since chemical shim is always available and flexible enough to cover any possible deviations in the expected burnable absorber depletion rate. Figure 4.3-3 shows typical core depletion curves with burnable absorbers.

In addition to reactivity control, the burnable absorbers are strategically located to provide a favorable radial power distribution. Figures 4.3-4 and 4.3-5 show the typical burnable absorber distribution within a fuel assembly for the several burnable absorber patterns used for both discrete and integral fuel burnable absorbers. The burnable absorber loading pattern for a typical equilibrium cycle reload core is shown in Figure 4.3-6 using the integral fuel burnable absorber.

Tables 4.1-1, and 4.3-1 through 4.3-3, summarize the reactor core design parameters for a typical reload fuel cycle, including reactivity coefficients, delayed neutron fraction, and neutron lifetimes.

4.3.2.2 Power Distribution

DCPP employs two methods for performing core power distribution calculations. The Power Distribution Monitoring System (PDMS) generates a continuous measurement of the core power distribution using the methodology documented in ~~Reference 32~~. The measured core power distribution is used to determine the most

References 32 & 33

limiting core peaking factors, which are used to verify that the reactor is operating within the design limits.

The PDMS requires information on current plant and core conditions in order to determine the core power distribution using the core peaking factor measurement and measurement uncertainty methodology described in ~~Reference 32~~ References 32 & 33. The core and plant condition information is used as input to the continuous core power distribution measurement software that continuously and automatically determines the current core peaking factor values. The core power distribution calculation software provides the measured peaking factor values at nominal one-minute intervals to allow operators to confirm that the core peaking factors are within design limits. In order for the PDMS to accurately determine the peaking factor values, the core power distribution measurement software requires accurate information about the current reactor power level average reactor vessel inlet temperature, control bank positions, the power range detector currents, and the core exit thermocouples.

Data obtained from the movable neutron flux detectors, described in Section 7.7.1, are used to calibrate the PDMS, and may also be used independent of the PDMS to generate a flux map of the core power distribution. The accuracy of these power distribution calculations has been confirmed through more than 1,000 flux maps during some 20 years of operation, under conditions very similar to those expected for DCPP. Details of this confirmation are given in References 1 and 3 and in Section 4.3.2.2.7.

4.3.2.2.1 Definitions

Power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core and the total energy produced in a coolant channel and are expressed in terms of quantities related to the nuclear or thermal design; namely:

Power density is the thermal power produced per unit volume of the core (kW/liter).

Linear power density is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, this is the unit of power density most commonly used. For all practical purposes, it differs from kW/liter by a constant factor that includes geometry effects and the fraction of the total thermal power which is generated in the fuel rods.

Average linear power density is the total thermal power produced in the fuel rods divided by the total active fuel length of all rods in the core.

Local heat flux is the heat flux at the surface of the cladding (Btu $ft^{-2}hr^{-1}$). For nominal fuel rod parameters, this differs from linear power density by a constant factor.

Rod power or *rod integral power* is the linear power density in one rod integrated over its length (kW).

24. Deleted in Revision 12.

25. T. M. Camden. et al., PALADON - Westinghouse Nodal Computer Code, WCAP-9485-P-A. December 1979 and Supplement 1, September 1981.

26. S. L. Davidson, (Ed), et al., ANC: Westinghouse Advanced Nodal Computer Code, WCAP-10965-P-A, September 1986.

27. T. Q. Nguyen, et al, Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, WCAP-11596-P-A, June 1988.

28. C. M. Mildrum, L. T. Mayhue, M. M. Baker and P. G. Issac, Qualification of the PHOENIX/POLCA Nuclear Design and Analysis Program for Boiling Water Reactors, WCAP-10841 Proprietary and WCAP-10842 (Non-Proprietary), June 1985.

29. WCAP-10216-P-A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, June 1983 (Westinghouse Proprietary).

30. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (Westinghouse Proprietary).

31. Kersting, P. J., et al., Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel, WCAP-13589-A (Proprietary), March 1995, and WCAP-14297-A (Non-Proprietary), March 1995.

32. WCAP-12472-P-A, BEACON Core Monitoring and Operations Support System, August 1994 (Westinghouse Proprietary).

33. WCAP-12472-P-A, Addendum 1-A, BEACON Core Monitoring and Operations Support System, January 2000 (Westinghouse Proprietary).

Enclosure
Attachment 2
PG&E Letter DCL-12-002

Technical Specification Bases Markups

(for information only)

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.

<u>A.1</u>

When one DRPI per group fails, the position of the rod may still be determined indirectly by use of the 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. References 4 & 5

(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. 5. WCAP-12472-P-A, Addendum 1-A, "BEACON Core Monitoring and Operations Support System," January 2000.

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 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 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)

References 3 & 4

BASES

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 is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
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. 4. WCAP-12472-P-A, Addendum 1-A, "BEACON Core Monitoring and Operations Support System," January 2000.

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 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~~).

References 4 & 5

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.2.2.1 (continued) After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle. 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 $F^N_{\Delta H}$ limit, provides assurance that the $F^N_{\Delta H}$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
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. 5. WCAP-12472-P-A, Addendum 1-A, "BEACON Core Monitoring and Operations Support System," January 2000.

BASES

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 $F_{\Delta H}^{N}$ 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 $F_{\Delta H}^{N}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support 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 $F_{\Delta H}^{N}$ 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.

References 4 & 5

A.4

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

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. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

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 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 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.
5. WCAP-12472-P-A,Addendum 1-A, "BEACON Core Monitoring and Operations Support System," January 2000.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT. 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested 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 Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

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 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 26~~ References 26 & 33). 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

REFERENCES (continued)

17. WCAP-11082, "Westinghouse Setpoint Methodology for Protection Systems, Diablo Canyon Units 1 and 2, 24 Month Fuel Cycle Evaluation and Replacement Steam Generator," September 2007.
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. License Amendments 157/157, June 2, 2003.
26. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.
27. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.
28. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.
29. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.
30. WCAP-11394-P-A, "Methodology For The Analysis of the Dropped Rod Event," January, 1990
31. License Amendments 205/206, April 29, 2009
32. WCAP-16769-P Revision 1, "Westinghouse SSPS Universal Logic Board Replacement Summary Report 6D30225G01/G02/G03/G04," July 2008.
33. WCAP-12472-P-A, Addendum 1-A,"BEACON Core Monitoring and Operations Support System," January 2000.