

correlation limit, the design DNBR limits, or the safety analysis DNBR limits.

The basis for the accumulator LCO, as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP-15049-A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming accumulators are needed to mitigate the design basis event, has a very small impact on plant risk. Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP-15049-A evaluation, the impact of increasing the accumulator CT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP-15049-A evaluation.

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

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

**Pacific Gas and Electric Company,
Docket Nos. 50-275 and 50-323, Diablo
Canyon Nuclear Power Plant, Unit Nos.
1 and 2, San Luis Obispo County,
California**

Date of amendment requests: June 11, 2003.

Description of amendment requests: The license amendment request proposes to revise Technical Specification (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.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

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

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. The proposed change does not involve a significant reduction in a margin of safety.

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.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

**PSEG Nuclear, LLC, Docket No. 50-354,
Hope Creek Generating Station, Salem
County, New Jersey**

Date of amendment request: June 17, 2003.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) Surveillance Requirement 4.6.2.1.b.2.b. This change would remove the requirement to verify that the reactor thermal power output is less than, or equal to, 1% of rated thermal power when the suppression chamber average water temperature is above 95 °F. Additionally, the amendment would correct two typographical errors on TS index page "x."

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR) Section