

The probability and consequences of accidents previously evaluated in the Final Safety Analysis Report Update (FSARU) are not adversely affected because changes to the RPS and ESFAS trip setpoints assure a conservative response of the affected trip functions, consistent with the safety analyses and licensing basis.

The proposed changes will not affect the probability of any accident initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSARU.

Therefore the proposed changes do not involve a significant increase in the probability or consequences 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 proposed changes do not change any hardware or the design functions of any structures, systems or components involved, other than to revise the SG narrow range (NR) water level-low setpoints; changes that have already been implemented. The proposed changes will not affect the normal method of plant operation or change any operating parameters. No new accidents, accident initiators, or failure mechanisms are created by the proposed changes.

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

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

The SG NR water level-low setpoints specified in the Technical Specifications have already been increased in the conservative direction. The safety analysis limits assumed in the transient and accident analyses remain unchanged. None of the acceptance criteria for any accident analysis are changed.

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.

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*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:* October 22, 2003.

*Description of amendment requests:*

The proposed license amendments would revise Surveillance Requirement 3.6.3.7 of Technical Specification (TS) 3.6.3, "Containment Isolation Valves," by extending the leakage rate testing frequency of the containment purge supply and exhaust and vacuum/pressure relief valves, all with resilient seals, from 184 days to 24 months. The amendments would also delete the requirement to leakage rate test the containment vacuum/pressure relief valves within 92 days after opening.

*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.

Operability and leakage control effectiveness of the containment purge supply and exhaust and containment vacuum/pressure relief isolation valves have no effect on whether an accident occurs. Consequently, increasing the interval between surveillances of isolation valve leak rate does not involve any significant increase in the probability of an accident previously evaluated. The consequences of a unisolated reactor containment building at the time of a fuel-handling accident or loss of coolant accident (LOCA) are the release of radionuclides to the environment. Offsite exposures due to containment leakage during a LOCA and fuel-handling accident have been evaluated in Final Safety Analysis Report Update (FSARU) sections 15.5.17.3 and 15.5.22, respectively. For a LOCA, the Diablo Canyon Power Plant (DCPP) analyses assume containment leakage of 0.1 percent of the containment volume per day for the first 24 hours and 0.05 percent per day for the rest of the duration of the accident. Calculated radiological exposures from the LOCA are listed in FSARU Chapter 15, Table 15.5-75 and are within the 10 CFR part 100 limits. The good performance history of these valves, along with the very low total containment leakage rate, are reasonable bases that there should not be any significant increase in the consequences of [an] accident previously evaluated. For the fuel-handling accident inside containment, DCPP analyses do not credit these valves to provide a containment isolation function. It was assumed that activity released from the containment refueling pool is transported to the environment over a short time period through the open equipment hatch. Calculated radiological exposures from the fuel-handling accident inside containment are listed in FSARU Chapter 15, Table 15.5-50 and are also within the 10 CFR part 100

limits. In summary, increasing the interval between leakage rate surveillances of these isolation valves will not involve any significant increase in the probability or consequences 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 proposed changes do not involve a modification to the physical configuration of the plant (*i.e.*, no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. The functions of the containment purge and containment vacuum/pressure relief systems are not altered by this change. Therefore, the proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

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

This proposed change only increases the interval between surveillance tests of the containment purge supply and exhaust, and containment vacuum/pressure relief valves. These valves have a good performance history and should be able to perform their intended containment isolation function reliably when called upon. In FSARU Chapter 15, two offsite exposure scenarios are applicable to the containment isolation function. These scenarios are LOCA containment leakage and fuel-handling accident inside containment. For LOCA containment leakage, the DCPP analyses assume containment leakage of 0.1 percent of the containment volume per day for the first 24 hours and 0.05 percent per day for the remainder of the accident. Calculated radiological exposures from a LOCA are listed in FSARU Chapter 15, Table 15.5-75 and meet the 10 CFR part 100 limits. For the fuel-handling accident inside containment, the DCPP analyses do not credit these valves to provide a containment isolation function. The analyses assume that activity released from the containment refueling pool is transported to the environment over a short time period through the open equipment hatch. Calculated radiological exposures from the fuel-handling accident inside containment are listed in FSARU Chapter 15, Table 15.5-50 and also meet the 10 CFR part 100 limits. If in the unlikely event that these valves exceed their leakage rate limits due to the extension of the surveillance interval, the consequences will be consistent with the containment leakage assumed in the accident analyses. Therefore, the extension of leakage rate test interval will have an insignificant radiological consequence, and the proposed change will not involve any significant reduction in the 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