

amendment requests involve no significant hazards consideration.

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*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:* October 22, 2003.

*Description of amendment requests:* The proposed license amendments would revise Technical Specifications (TS) Section 5.5.9, "Steam Generator (SG) Tube Surveillance Program," and TS Section 5.6.10, "Steam Generator (SG) Tube Inspection Report," to allow use of leak limiting Alloy 800 sleeves to repair degraded SG tubes as an alternative to plugging the SG tubes. The proposed amendments would also remove an unnecessary reporting requirement contained in TS Table 5.5.9-2, "Steam Generator (SG) Tube Inspection."

*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 leak limiting Alloy 800 sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator (SG) tubing. The applied stresses and fatigue usage for the sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of sleeves under normal, upset, emergency, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Burst testing of sleeve-tube assemblies has confirmed the analytical results and demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

The leak limiting Alloy 800 sleeve depth-based structural limit is determined using NRC guidance and the pressure stress equation of ASME Code, Section III with additional margin added to account for the configuration of long axial cracks. A sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly.

Evaluation of the repaired SG tube testing and analysis indicates no detrimental effects on the leak limiting Alloy 800 sleeve or sleeved tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Diablo Canyon Power Plant (DCPP) Units 1 and 2. Corrosion testing and historical performance of sleeve-tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The implementation of the proposed change has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the leak limiting Alloy 800 sleeve-tube assembly is bounded by the current SG tube rupture (SGTR) analysis described in the DCPP Final Safety Analysis Report Update. Due to the slight reduction in the inside diameter caused by the sleeve wall thickness, primary coolant release rates through the parent tube would be slightly less than assumed for the SGTR analysis and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feedwater line break will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the DCPP safety analysis. The sleeve-tube assembly leakage during plant operation would be minimal and is well within the Technical Specification (TS) leakage limits.

The proposed change to TS 5.5.9 Table 5.5.9-2, "Steam Generator (SG) Tube Inspection," to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) if the first sample inspection or the second sample inspection results in a C-3 classification, is an administrative change only and does not affect plant equipment or accident analyses.

Therefore, the proposed change does 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 leak limiting Alloy 800 sleeves are designed using the applicable ASME Code as guidance, and therefore meet the objectives of the original SG tubing. As a result, the functions of the SG will not be significantly affected by the installation of the proposed sleeve. The proposed sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing SGTR accident analysis. The continued integrity of the installed sleeve-tube assembly is periodically verified by the TS requirements and a sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly.

Implementation of the proposed change has no significant effect on either the configuration of the plant, or the manner in

which it is operated. The proposed change to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) from TS 5.5.9 Table 5.5.9-2 is an administrative change only and does not affect plant equipment or accident analyses.

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.

The repair of degraded SG tubes with leak limiting Alloy 800 sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions and thereby maintains current core cooling margin as opposed to plugging the tube and taking it out of service. The design safety factors utilized for the sleeves are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in the original SG design. The sleeve and portions of the installed sleeve-tube assembly that represent the reactor coolant pressure boundary will be monitored and a sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly. Use of the previously identified design criteria and design verification testing assures that the margin to safety is not significantly different from the original SG tubes.

The proposed change to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) from TS 5.5.9 Table 5.5.9-2 is an administrative change only, does not affect plant equipment or accident analyses, does not relax any safety system settings, and does not relax the bases for any limiting conditions for operations.

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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*STP Nuclear Operating Company, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas.*

*Date of amendment request:* November 4, 2003.

*Description of amendment request:* The proposed amendments would revise the South Texas Project, Units 1 and 2 Technical Specifications for the Remote Shutdown System to reflect requirements consistent with those in NUREG-1431, "Standard Technical Specifications—Westinghouse Plants."