UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO A LEAK-BEFORE-BREAK ANALYSIS

FOR

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

On March 16, 1992, Pacific Gas & Electric Company requested to eliminate from the design basis the dynamic effects of postulated pipe ruptures in the reactor coolant loop piping for the Diablo Canyon Power Plant, Units 1 and 2. The request was based on a leak-before-break (LBB) analysis performed by Westinghouse (Ref. 1) as permitted by General Design Criteria 4 (GDC-4) of Appendix A to 10 CFR 50.

2.0 DISCUSSION

The design basis for the Class 1 piping requires that the dynamic effects of pipe breaks be evaluated and that pipe whip restraints and barriers be installed to protect safety systems from steam and water jet impingement. Since the mid-1980s, the NRC has determined that such breaks are unlikely and may be eliminated from the design basis if the piping system can be shown to qualify for leak-before-break.

GDC-4 allows the use of the plant-specific LBB analysis to eliminate the dynamic effects of postulated pipe ruptures in high energy piping from the design basis. Licensees with NRC approved LBB analysis may remove pipe whip restraints and jet impingement barriers. The acceptance criteria for the LBB analysis are defined in NUREG-1061 and draft Standard Review Plan (SRP) 3.6.3. They are summarized as follows:

The LBB analysis should provide data on materials specifications and limitations, and age-related degradations such as thermal aging of cast stainless steel. The piping materials must be free from brittle cleavage-type failure over the full range of the system operating temperature.

The analysis should consider the forces and moments due to pressure, deadweight, thermal expansion, operating basis earthquake, and safe shutdown

earthquake (SSE). The analysis should identify location(s) at which the highest stresses are coincident with the poorest material properties for base metals, weldments, and safe ends.

The analysis should postulate a through-wall flaw at the highest stressed locations. The flaw size should be large enough so that any leakage is assured of being detected with at least a margin of 10 on leakage using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

The analysis should show that the postulated leakage flaw is stable under faulted conditions (normal plus SSE loads). The leakage flaw should also be stable under larger loads at least 1.4 times the normal plus SSE loads. However, the margin of 1.4 may be reduced to 1.0 if the individual normal and SSE loads are summed absolutely.

Under normal plus SSE loads, the safety margin should be at least a factor of 2 between the leakage-size flaw and the critical-size flaw to account for the uncertainties inherent in the analyses and leakage detection capability.

The analysis should provide operating experience to show that the pipe will not experience stress corrosion cracking, fatigue, or water hammer. The operating history should include system operational procedures; system or component modification; water chemistry parameters, limits, and controls; resistance of piping material to various forms of stress corrosion; and performance of the pipe under cyclic loadings.

For Class 1 piping, a fatigue crack growth analysis should be performed to show that the postulate flaw(s) at highest stress location(s) will not grow significantly during 40 years of service.

3.0 EVALUATION

The reactor coolant system (RCS) pipe at Diablo Canyon Units 1 and 2 consists of various diameter and wall thickness. The outside diameter of the hot leg varies from 33.90 to 37.19 inches; its wall thickness varies from 2.395 to 2.99 inches. The crossover leg has an outside diameter of 37.19 inches with a wall thickness of 2.99 inches. The outside diameter of the cold leg varies from 32.26 to 33.06 inches; its wall thickness varies from 2.275 to 2.625 inches. The pipe is made of austenitic wrought stainless steel SA376 TP316 and the elbow fittings are cast stainless steel SA351 CF8M.

The licensee selected following critical pipe locations for crack stability analysis based on applied load and material toughness: (1) the weld connecting the reactor vessel outlet nozzle and the hot leg, (2) the elbow to elbow welds (2 locations) connecting the steam generator outlet and the crossover leg, and (3) the weld connecting the crossover leg and reactor coolant pump. The licensee applied loads from effects of pressure, deadweight, thermal expansion, and safe shutdown earthquake to the postulated crack at the above critical locations to determine crack stability. The staff finds that the selection of the critical locations and loads are acceptable.

To determine the crack stability, the licensee used the modified limit load method as specified in draft SRP 3.6.3 for the austenitic stainless steel pipe and the J-integral method for the cast stainless steel elbow. The licensee's limit load analysis followed the NRC accepted procedure and the associated Z factor applied to welds followed Article C-3320 of Section XI of the ASME Code. The staff finds that the licensee's limit load analysis is acceptable.

The J-integral method considers the thermal aging of cast stainless steel. Using chemistry of the cast material, the licensee derived the J_{IC}, tearing modulus (T-mat), and J-max at end-of-life based on the Westinghouse report (Ref. 2) which the NRC has approved. These material toughness parameters were then compared to the applied tearing moduli (T-applied) and J values (J-applied) at the critical cast stainless elbows.

In order for the crack to be stable, J-applied and T-applied should be less than J-max and T-mat. For the critical cast stainless elbows, the licensee showed that the T-applied was less than the T-mat and the J-applied was less than the J-max except at the elbow weld connecting the crossover leg and reactor coolant pump. At this location, J-applied equals J-max. The staff judges that fracture toughness of the cast material at this location will not be compromised so long as the J-applied is not greater than the J-max.

The licensee showed that the postulated leakage flaw is stable under normal plus SSE loads. The loads were combined absolutely and the safety margin on loads was shown to comply with recommended value of one in NUREG-1061. The licensee showed that the margin between the leakage-size flaw and the critical-size flaw satisfies the recommended value of 2 for all critical locations except at the elbow weld connecting the crossover leg and reactor coolant pump. The licensee calculated a margin of 1.95 at this location. Considering the overall crack size calculation, the staff believes that the margin of 1.95 is within the uncertainty bounds of 2.0 and is acceptable. The structural integrity of the pipe during a leak-before-break event will not be compromised.

The licensee stated that the leak detection system for the reactor coolant pressure boundary meets the intent of Regulatory Guide 1.45 which recommends that a leakage of one gallon per minute in one hour be detected. The licensee used a margin of 10 on leakage in calculating the leakage crack size. This is consistent with the LBB criteria in NUREG-1061.

To determine crack growth under thermal fatigue, the licensee calculated the growth in 40 years of postulated cracks using equations in Appendix A to Section XI of the ASME Code. Thermal transients, including number of cycles and temperature differentials, were used. The licensee performed a parametric study using crack depth of 0.29, 0.3, 0.375, and 0.425 inch. The maximum crack size at end 40 years was calculated to be 0.474 inch, propagated from a postulated 0.425 inch deep crack. The staff finds fatigue analysis results acceptable.

The licensee showed that for Westinghouse plants there is no history of stress corrosion cracking in the RCS piping because of controls in the water

chemistry and there is a low probability for water hammer because the RCS is designed and operated to preclude the voiding condition necessary to generate severe water hammer transients. The staff finds that the licensee has addressed stress corrosion cracking and water hammer satisfactorily.

4.0 CONCLUSION

The NRC staff has performed independent flaw stability calculations to evaluate the licensee's LBB analysis of the large diameter reactor coolant piping stated above for the Diablo Canyon Power Plant, Units 1 and 2. The staff concludes that the licensee's LBB analysis is consistent with the criteria in NUREG-1061, Volume 3, and draft SRP 3.6.3.; therefore, the analysis complies with GDC-4. Thus, the probability of large pipe breaks occurring in the RCS line is sufficiently low that the dynamic effects associated with postulated pipe breaks need not be a design basis.

5.0 REFERENCES

1.0 WCAP-13039, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for The Diablo Canyon Units 1 & 2 Nuclear Power Plants," Westinghouse Electric Corporation, November 1991 (proprietary).

2.0 WCAP-10931, "Toughness Criteria for Thermally Aged Cast Stainless Steel," Westinghouse Electric Corporation, May 1986.

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