UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO UTILIZATION OF LEAK-BEFORE-BREAK METHODOLOGY FOR REACTOR COOLANT SYSTEM PIPING

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By a letter dated February 2, 1994, Florida Power and Light Company requested to eliminate from the design basis the dynamic effects of postulated pipe ruptures in the reactor coolant loop piping for Turkey Point Units 3 & 4. The request was based on a plant-specific leak-before-break (LBB) analysis as permitted by General Design Criteria 4 (GDC-4) of Appendix A to 10 CFR 50. The analysis is documented in a proprietary Westinghouse report, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for The Turkey Point Units 3 & 4 Nuclear Power Plants," WCAP-14237, December 1994.

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, as defined in NUREG-1061 and draft Standard Review Plan (SRP) 3.6.3, 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 postulated flaw(s) should be calculated based on a leak rate that is 10 times the capability of the leak detection system of the reactor pressure boundary.

The analysis should demonstrate that the postulated leakage flaw is stable under 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, a margin of two should be maintained between the leakage crack size and the critical crack size 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 modifications; 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 postulated flaws will not grow significantly during 40 years of service.

3.0 EVALUATION

The reactor coolant system (RCS) piping at Turkey Point Units 3 & 4 have various diameters and wall thicknesses. The outside diameter of the hot leg varies from 34.00 to 37.75 inches; its minimum wall thickness varies from 2.395 to 3.270 inches. The crossover leg has an outside diameter of 36.25 inches with a minimum wall thickness of 2.520 inches. The outside diameter of the cold leg varies from 32.25 to 33.56 inches; its wall thickness varies from 2.270 to 2.930 inches. The pipe is made of austenitic wrought stainless steel SA376 TP316 and pipe elbows are made of cast stainless steel SA351 CF8M.

Based on applied load and material toughness, the licensee selected the following critical pipe locations in the crack calculations: (1) the weld between the reactor vessel outlet nozzle and the hot leg, (2) the weld between the hot leg and the elbow that is connected to the steam generator inlet nozzle, and (3) the weld between the cold leg and the elbow that is connected to the reactor vessel inlet nozzle. 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 the leakage flaw size and critical flaw size. The staff finds that the selection of the critical locations and loads are acceptable.

To determine the stability of the leakage flaws, the licensee used the modified limit load method as specified in draft SRP 3.6.3 to qualify for the austenitic stainless steel piping and the J-integral method to qualify for the cast stainless steel elbows. The staff determined that the licensee's limit load analysis of the austenitic stainless steel piping followed the NRC accepted procedure and, therefore, is acceptable.

In the J-integral analysis, material toughness parameters, J values (i.e., J_{IC} and $J_{maximum}$) and tearing modulus $T_{material}$, are compared to the applied tearing modulus, $T_{applied}$, and $J_{applied}$, at the crack. A crack is stable (i.e., not predicted to grow) when $J_{applied}$ is less than J_{IC}. For the case when $J_{applied}$ is greater than J_{IC}, the crack will grow in a stable manner if $T_{applied}$ is less than $T_{material}$ and $J_{applied}$ is less than $J_{maximum}$. The crack propagation will cease when $J_{applied}$ equals $J_{material}$.

Using chemical contents of the cast stainless steel material, the licensee derived values of J_{IC}, $T_{material}$, and $J_{maximum}$ at the end of license based on a staff-approved Westinghouse report (Ref. 1). The thermal aging of cast stainless steel was considered. For the limiting cast stainless steel elbows, the licensee showed that the $J_{applied}$ was less than J_{IC} under the absolute sum of normal plus SSE loads. Therefore, the postulated cracks in the elbows were shown to be stable.

The licensee demonstrated that the margin between the leakage flaw size and the critical flaw size satisfies the staff recommended value (two or greater) for the above three critical locations.

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 0.4435 inch, propagated from a postulated 0.425 inch deep crack. The staff finds the fatigue analysis results acceptable.

The licensee showed that, for Westinghouse plants, there is no history of stress corrosion cracking in the reactor coolant system piping because of controls in the water chemistry and there is a low probability for water hammer because the reactor coolant system 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 calculations to evaluate the licensee's LBB analysis of the large diameter reactor coolant piping stated above for the Turkey Point Units 3 & 4 Nuclear Power Plants. 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. Hence, 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 considered in the design basis.

5.0 REFERENCE

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

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