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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ELIMINATION OF POSTULATED PRIMARY LOOP PIPE RUPTURES

AS A DESIGN BASIS

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-390 & 50-391

1.0 INTRODUCTION

By letter dated April 17, 1989, Tennessee Valley Authority (the applicant) requested the elimination of the dynamic effects of postulated primary loop pipe ruptures from the design basis of Watts Bar Nuclear Plant Units 1 and 2 using "leak-before-break" (LBB) technology as permitted by the revised General Design Criterion 4 (GDC-4) of Appendix A to 10 CFR Part 50.

The applicant submitted the technical basis for the elimination of primary loop pipe ruptures for Watts Bar Nuclear Plant Units 1 and 2 in Westinghouse report WCAP-11985 (Reference 1). The applicant also referenced Westinghouse reports WCAP-10456 (Reference 2) and WCAP-10931, Revision 1 (Reference 3), which have been reviewed previously by the staff as discussed in References 4 and 5, respectively. By letter dated February 14, 1990, the applicant submitted additional information in Westinghouse Report WCAP-12500 (Reference 6).

The revised GDC-4 is based on the development of advanced fracture mechanics technology using the LBB concept. On October 27, 1987, a final rule was published (52 FR 41288), effective November 27, 1987, amending GDC-4 of Appendix A to 10 CFR Part 50. The revised GDC-4 allows the use of analyses to eliminate from the design basis the dynamic effects of postulated pipe ruptures in high energy piping in nuclear power units. The new technology reflects an engineering advance which allows simultaneously an increase in safety, reduced worker radiation exposures, and lower construction and maintenance costs. Implementation permits the removal of pipe whip restraints and jet impingement barriers as well as other related changes in operating plants, plants under construction, and future plant designs. Although functional and performance requirements for containments, emergency core cooling systems, and environmental qualification of equipment remain unchanged, local dynamic effects uniquely associated with postulated ruptures in piping which qualified for LBB may be excluded from the design basis (53 FR 11311). The acceptable technical procedures and criteria are defined in NUREG-1061, Volume 3 (Reference 7).

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Using the criteria in Reference 7, the staff has reviewed and evaluated the applicant's submittal for compliance with the revised GDC-4. This Safety Evaluation Report provides the staff's findings.

2.0 EVALUATION

2.1 Watts Bar Nuclear Plant Units 1 and 2 Primary Loop Piping

The Watts Bar Nuclear Plant Units 1 and 2 primary loop piping consists of 34-inch, 36-inch, and 32-inch nominal diameter hot leg, cross-over leg, and cold leg, respectively. The piping material in the primary loops is austenitic cast stainless steel (SA-351 CF8A). The piping is centrifugally cast and the fittings are statically cast.

2.2 Staff Evaluation Criteria

The staff's criteria for evaluation of compliance with the revised GDC-4 are discussed in Chapter 5.0 of Reference 7 and are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight, and thermal expansion) due to normal operation, and the forces and moments associated with the safe shutdown earthquake (SSE). These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments, and safe ends.

(2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue, or water hammer are not likely, should be provided. Relevant operating history should be cited, which includes system operational procedures; system or component modification; water chemistry parameters, limits, and controls; and resistance of material to various forms of stress corrosion and performance under cyclic loadings.

(3) The materials data provided should include types of materials and materials specifications used for base metal, weldments, and safe ends; the materials properties including the fracture mechanics parameter "J-integral" (J) resistance (J-R) curve used in the analyses; and long-term effects such as thermal aging and other limitations to valid data (e.g., J maximum, and maximum crack growth).

(4) A through-wall flaw should be postulated at the highest stressed locations determined from criterion (1) above. The size of the flaw should be large enough so that the leakage is assured of detection with at least a factor of 10 using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

(5) It should be demonstrated that the postulated leakage flaw is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be at least 1.4 and should be determined by a flaw stability analysis, i.e., that the leakage-size flaw will not experience unstable crack growth even if larger loads (larger than design loads) are applied. However, the final rule permits a reduction of the margin of 1.4 to 1.0 if the individual normal and seismic (pressure, deadweight, thermal expansion, SSE, and seismic anchor motion) loads are summed absolutely. This analysis should demonstrate that crack growth is stable and the final flaw size is limited, such that a double-ended pipe break will not occur.

(6) The flaw size should be determined by comparing the leakage-size flaw to the critical-size flaw. Under normal plus SSE loads, it should be demonstrated that there is a margin of at least 2 between the leakage-size flaw and the critical-size flaw to account for the uncertainties inherent in the analyses and leakage detection capability. A limit-load analysis may suffice for this purpose; however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.

2.3 Staff Evaluation of GDC-4 Compliance

The staff has evaluated the information presented in References 1 and 6 for compliance with the revised GDC-4. Furthermore, the staff performed independent flaw stability computations using an elastic-plastic fracture mechanics procedure developed by the staff (Reference 8).

On the basis of its review, the staff finds the Watts Bar Nuclear Plant Units 1 and 2 primary loop piping in compliance with the revised GDC-4. The following paragraphs in this section present the staff's evaluation.

(1) Normal operating loads, including pressure, deadweight, and thermal expansion, were used to determine leak rate and leakage-size flaws. The flaw stability analyses performed to assess margins against pipe rupture at postulated faulted load conditions were based on normal plus SSE loads. In the stability analysis, the individual normal and seismic loads were summed absolutely. In the leak rate analysis, the individual normal load components were summed algebraically. Leak-before-break evaluations were performed for the limiting location in the piping.

(2) For Westinghouse facilities, there is no history of cracking failure in reactor coolant system (RCS) primary loop piping. The RCS primary loop has an operating history which demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history totals over 450 reactor-years, including 5 plants each having over 17 years of operation and 15 other plants each with over 12 years of operation.

(3) The material tensile and fracture toughness properties were provided in Reference 1. Because the Watts Bar Nuclear Plant Units 1 and 2 primary loop piping consists of cast stainless steel, the thermal aging toughness properties of cast stainless steel materials were estimated according to procedures in References 2 and 3. The material tensile properties were estimated using plant specific material certifications and generic procedures. For flaw stability evaluations, the lower-bound stress-strain properties were used. For leakage rate evaluations, the average stress-strain properties were used.

(4) Watts Bar Nuclear Plant Units 1 and 2 have RCS pressure boundary leak detection systems which are consistent with the guidelines of Regulatory Guide 1.45 such that a leakage of one gallon per minute (gpm) in one hour can be detected. The calculated leak rate through the postulated flaw is large relative to the staff's required sensitivity of the plant's leak detection systems; the margin is a factor of 10 on leakage and is consistent with the guidelines of Reference 7.

(5) In the flaw stability analyses, the staff evaluated the margin in terms of load for the leakage-size flaw under normal plus SSE loads. The staff's calculations indicated the margin exceeded 1.0 when the individual normal and seismic loads were summed absolutely. The margin is consistent with the guidelines of the final rule.

(6) Similar to item (5) above, the margin between the leakage-size flaw and the critical-size flaw was also evaluated in the flaw stability analyses. The staff's calculations indicated the margin in terms of flaw size exceeded 2 for the load combination method considered. The margin is consistent with the guidelines of Reference 7.

3.0 CONCLUSION

The staff has reviewed the information submitted by the applicant and has performed independent flaw stability computations. On the basis of its review, the staff concludes that the Watts Bar Nuclear Plant Units 1 and 2 primary loop piping complies with the revised GDC-4 according to the criteria in NUREG-1061, Volume 3 (Reference 7). Thus, the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of Watts Bar Nuclear Plant Units 1 and 2 is sufficiently low such that dynamic effects associated with postulated pipe breaks need not be a design basis.

4.0 REFERENCES

(1) Westinghouse Report WCAP-11985, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Watts Bar Units 1 & 2", November 1988, Westinghouse Proprietary Class 2.

(2) Westinghouse Report WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems", November 1983, Westinghouse Proprietary Class 2.

(3) Westinghouse Report WCAP-10931, Revision 1, "Toughness Criteria for Thermally Aged Cast Stainless Steel", July 1986, Westinghouse Proprietary Class 2.

(4) Letter from B. J. Youngblood of NRC to M. D. Spence of Texas Utilities Generating Company dated August 28, 1984.

(5) Letter from D. C. DiIanni of NRC to D. M. Musolf of Northern States Power Company dated December 22, 1986.

(6) Westinghouse Report WCAP-12500, "Additional Information in Support of Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Watts Bar Units 1 and 2", January 1990, Westinghouse Proprietary Class 3.

(7) NUREG-1061, Volume 3, "Report of the U. S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks", November 1984.

(8) NUREG/CR-4572, "NRC Leak-Before-Break (LBB.NRC) Analysis Method for Circumferentially Through-Wall Cracked Pipes Under Axial Plus Bending Loads", May 1986.

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