February 18, 1986 DMB 016

Dockets Nos. 50-269, 50-270
and 50-287

Mr. Hal B. Tucker
Vice President - Nuclear Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: SAFETY EVALUATION OF B&W OWNERS GROUP REPORTS DEALING WITH ELIMINATION OF POSTULATED PIPE BREAKS IN PWR PRIMARY MAIN LOOPS

Re: Oconee Nuclear Station, Units Nos. 1, 2 and 3

The NRC staff has reviewed the B&W Owners Group reports BAW-1847, Rev. 1, and BAW-1889P which apply "leak-before-break" technology as an alternative to designing against dynamic loads associated with postulated ruptures of primary coolant loop piping. As discussed in the enclosed letter to the B&W Owners Group, we have concluded that an acceptable technical basis has been provided to eliminate, as a design basis, the dynamic effects of large ruptures in the main loop piping of those B&W Owners Group facilities listed in the enclosure. Authorization by the NRC to not provide protection against the dynamic loads resulting from postulated breaks of primary main loop piping will require an exemption from General Design Criterion 4 (GDC4). Such exemptions must be justified on a facility specific basis. Each request for an exemption should include a safety balance in accordance with the guidance provided in NRC Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," February 1, 1984. In addition, information for each facility should be submitted to demonstrate that leakage detection systems installed at the facility comply with Regulatory Guide 1.45.

Generic Letter 84-04 informed all operating PWR licensees, construction permit holders and applicants for construction permits of the staff's intent to proceed with rulemaking changes to GDC-4 to permit the use of analyses that demonstrate the probability of rupturing piping is extremely low under design basis conditions. On July 1, 1985, the Commission published a proposed modification to GDC-4 which would permit the use of such analyses for PWR primary coolant loop piping. The NRC staff is currently in the

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process of final rulemaking. Promulgation of the final rule will eliminate the need for exemption requests and performance of safety balances; however, the requested information on leakage detection systems should be submitted.

Sincerely,

ORIGINAL SIGNED BY
JOHN F. STOLZ

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosure:
As Stated

cc w/enclosure:
See next page

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Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

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

December 12, 1985

Mr. L. C. Oakes, Chairman
B&W Owners Group Leak-Before-Break Task Force
Washington Public Power Authority
P.O. Box 460
3000 George Washington Way
Richland, Washington 99352

SUBJECT: SAFETY EVALUATION OF B&W OWNERS GROUP REPORTS DEALING WITH ELIMINATION OF POSTULATED PIPE BREAKS IN PWR PRIMARY MAIN LOOPS

Reference:
1. B&W Owners Group report BAW-1847, Rev. 1, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," September 1985.
2. B&W Owners Group Report BAW-1889P, "Piping Material Properties for Leak-Before-Break Analysis," A. L. Lowe, Jr., K. K. Yoon, and R. H. Emanuelson, October 1985, proprietary.
3. NRC Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Breaks in PWR Primary Main LOOPS," February 1, 1984.

The NRC staff has completed its review of the referenced B&W Owners Group reports which apply "leak-before-break" technology as an alternative to designing against dynamic loads associated with postulated ruptures of primary coolant loop piping.

The staff evaluation concludes that an acceptable technical basis has been provided to eliminate, as a design basis, the dynamic effects of large ruptures in the main loop primary piping of the B&W Owners Group facilities.* Authorization by the NRC to not provide protection against the dynamic loads resulting from postulated breaks of primary main loop piping will require an exemption from General Design Criterion 4 (GDC4). Such exemptions must

*1. ANO-1
2. Midland-2
3. Oconee 1,2,3
4. Crystal River 3
5. Rancho Seco
6. WNP-1
7. Bellefonte 1,2
8. Davis-Besse 1

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be justified on a facility specific basis. Each request for an exemption should include a safety balance in accordance with the guidance provided in NRC Generic Letter 84-04 (Reference 3). In addition, information for each facility should be submitted to demonstrate that leakage detection systems installed at the facility comply with Regulatory Guide 1.45.

Reference 3 informed all operating PWR licensees, construction permit holders and applicants for construction permits of the staff's intent to proceed with rulemaking changes to GDC-4 to permit the use of analyses that demonstrate the probability of rupturing piping is extremely low under design basis conditions. On July 1, 1985, the Commission published a proposed modification to GDC-4 which would permit the use of such analyses for PWR primary coolant loop piping. The NRC staff is currently in the process of final rulemaking. Promulgation of the final rule will eliminate the need for exemption requests and performance of safety balances; however, the requested information on leakage detection systems should be submitted.

By copy of this letter with enclosed safety evaluation report, Mr. J. F. Walters of Babcock & Wilcox is being informed of this action. This information is also being transmitted to participating licensees and applicants of the B&W Owners Group.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Assistant Director
for Technical Support
Division of PWR Licensing-B
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: J. F. Walters, B&W

ATTACHMENT

THE B&W OWNERS GROUP
DOCKET NOS. 50-269, 270, 287, 302, 312, 313, 329,
346, 438, 439, & 460

SAFETY EVALUATION REPORT ON THE ELIMINATION OF LARGE PRIMARY LOOP RUPTURES AS A DESIGN BASIS

Section A
Engineering Branch
Division of PWR Licensing-B

INTRODUCTION

By letter dated September 7, 1984, the B&W Owners Group (B&WOG), on behalf of participating utilities with B&W designed facilities, submitted a generic report (Reference 1) on the technical bases for eliminating large primary loop piping ruptures as a design basis. Reference 1 presented the results of a bounding evaluation for the following B&WOG members:

Licensee or Applicant	Facility
Arkansas Power & Light Co.	ANO-1
Consumers Power Co.	Midland-2
Duke Power Co.	Oconee 1, 2, 3
Florida Power Corp.	Crystal River 3
Sacramento Municipal Utility District	Rancho Seco
Supply System	WNP-1
Tennessee Valley Authority	Bellefonte 1, 2
Toledo Edison Co.	Davis-Besse 1

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The Reference 1 submittal was made to provide technical justification for the preceding licensees and applicants of the B&WOG in regard to a request for an exemption to General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50 in regard to the need for protection against dynamic effects from postulated pipe breaks. After meeting with the B&WOG, the staff formally responded by letter (Reference 2) dated March 12, 1985, to transmit the staff's comments and questions on the submittal. The response to the staff's concerns resulted in a revision to the submittal, Reference 3, and an additional report (Reference 4) on material properties data, both of which were transmitted to the NRC on October 22, 1985.

By means of deterministic fracture mechanics analyses, the B&WOG contends that postulated double-ended guillotine breaks (DEGBs) of the primary loop reactor coolant piping will not occur in the facilities addressed in References 3 and 4 and therefore need not be considered as a design basis for installing protective devices such as pipe whip restraints to guard against the dynamic effects associated with such postulated breaks. No other changes in design requirements are addressed within the scope of the referenced reports; e.g., no changes to the definition of a LOCA nor its relationship to the regulations addressing design requirements for ECCS (10 CFR 50.46), containment (GDC 16, 50), other engineered safety features and the conditions for environmental qualification of equipment (10 CFR 50.49).

The Commission's regulations require provision of protective measures against the dynamic effects of postulated pipe breaks in high energy fluid system piping. Protective measures include physical isolation from postulated pipe rupture locations if feasible or the installation of pipe whip restraints, jet impingement shields or compartments. In 1975, concerns arose as to the asymmetric loads on pressurized water reactor (PWR) vessels and their internals which could result from these large postulated breaks at discrete locations in the main primary coolant loop piping. This led to the establishment of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on PWR Primary Systems."

The NRC staff, after several review meetings with the Advisory Committee on Reactor Safeguards (ACRS) and a meeting with the NRC Committee to Review Generic Requirements (CRGR), concluded that an exemption from the regulations would be acceptable as an alternative for resolution of USI A-2 for 16 facilities owned by 11 licensees in the Westinghouse Owner's Group (one of these facilities, Fort Calhoun has a Combustion Engineering nuclear steam supply system). This NRC staff position was stated in Generic Letter 84-04, published on February 1, 1984 (Reference 5). The generic letter states that the affected licensees must justify an exemption to GDC 4 on a plant-specific basis. Other PWR applicants or licensees may request similar exemptions from the requirements of GDC 4 provided that they submit an acceptable technical basis for eliminating the need to postulate pipe breaks.

The acceptance of an exemption was made possible by the development of advanced fracture mechanics technology. These advanced fracture mechanics techniques deal with relatively small flaws in piping components (either postulated or real) and examine their behavior under various pipe loads. The objective is to demonstrate by deterministic analyses that the detection of small flaws by either inservice inspection or leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as the DEGB or its equivalent. The concept underlying such analyses is referred to as "leak-before-break" (LBB). There is no implication that piping failures cannot occur, but rather that improved knowledge of the failure modes of piping systems and the application of appropriate remedial measures, if indicated, can reduce the probability of catastrophic failure to insignificant values.

Advanced fracture mechanics technology was applied in topical reports (References 6, 7, and 8) submitted to the staff by Westinghouse on behalf of the licensees belonging to the USI A-2 Owners Group. Although the topical reports were intended to resolve the issue of asymmetric blowdown loads that resulted from a limited number of discrete break locations, the technology advanced in these topical reports demonstrated that the probability of breaks occurring in the primary coolant system main loop piping is sufficiently low such that these breaks need not be considered as a design basis for requiring installation of pipe whip restraints or jet impingement shields. The staff's Topical Report Evaluation is attached as Enclosure 1 to Reference 5.

Probabilistic fracture mechanics studies conducted by the Lawrence Livermore National Laboratories (LLNL) on both Westinghouse and Combustion Engineering nuclear steam supply system main loop piping (Reference 9) confirm that both the probability of leakage (e.g., undetected flaw growth through the pipe wall by fatigue) and the probability of a DEGB are very low. The results given in Reference 9 are that the best-estimate leak probabilities for Westinghouse nuclear steam supply system main loop piping range from 1.2×10^{-8} to 1.5×10^{-7} per plant year and the best-estimate DEGB probabilities range from 1×10^{-12} to 7×10^{-12} per plant year. Similarly, the best-estimate leak probabilities for Combustion Engineering nuclear steam supply system main loop piping range from 1×10^{-8} per plant year to 3×10^{-8} per plant year, and the best-estimate DEGB probabilities range from 5×10^{-14} to 5×10^{-13} per plant year. In addition, LLNL recently conducted an evaluation of B&W nuclear steam supply main loop piping with the result that the best-estimate leak and DEGB probabilities are nominally identical to those calculated for the Westinghouse and Combustion Engineering studies. These results do not affect core melt probabilities in any significant way.

During the past few years it has also become apparent that the requirement for installation of large, massive pipe whip restraints and jet impingement shields is not necessarily the most cost effective way to achieve the desired level of safety, as indicated in Enclosure 2, Regulatory Analysis, to Reference 5. Even for new plants, these devices tend to restrict access for future inservice inspection of piping; or if they are removed and reinstalled for inspection, there is a potential risk of damaging the piping and other safety-related components in this process. If installed in operating plants, high occupational radiation exposure (ORE) would be incurred while public risk reduction would be very low. Removal and reinstallation for inservice inspection also entail significant ORE over the life of a plant.

PARAMETERS EVALUATED BY THE STAFF

The B&WOG facilities evaluated in Reference 3 include both 177-FA and 205-FA plants and configurations of the lowered-and-raised-loop designs. The primary coolant loop piping of these facilities are comprised of straight sections and elbows in each of four pipe sizes - 28, 32, 36 and 38 inch diameters. The piping materials in the primary main loops are low alloy ferritic steels (SA-106 GrC, SA-508 Cl 1, and SA-516 Gr 70) and wrought stainless steel safe ends (SA-376 TP 316). In its review of References 3 and 4, the staff evaluated the B&WOG analyses and materials data with regard to:

- the location of maximum stresses in the piping, associated with the combined loads from normal operation and the Safe Shutdown Earthquake (SSE);
- potential cracking mechanisms;
- size of postulated through-wall cracks that would leak a detectable amount under normal loads and pressure;
- stability of a "leakage-size crack" under normal plus SSE loads and the expected margin in terms of load;
- margin based on crack size; and
- the fracture toughness properties of low alloy, ferritic steel piping, wrought stainless steel safe ends and associated weld material.

STAFF CRITERIA USED IN THE EVALUATION

The NRC staff's criteria for evaluation of the above parameters are delineated in the Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, NUREG-1061, Volume 3, "Evaluation of Potential for Pipe Breaks." These criteria are enumerated in Chapter 5.0 of Volume 3 of the NUREG 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 is 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; resistance of material to various forms of stress corrosion, and performance under cyclic loadings.

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

(4) It should be demonstrated that the postulated leakage crack 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 the $\sqrt{2}$ and should be determined by a crack stability analysis, i.e., that the leakage-size crack will not experience unstable crack growth even if larger loads (larger than design loads) are applied. This analysis should demonstrate that crack growth is stable and the final crack size is limited, such that a double-ended pipe break will not occur.

(5) The crack size should be determined by comparing leakage-size crack to critical-size cracks. Under normal plus SSE loads, it should be demonstrated that there is a margin of at least 2 between the leakage-size crack and the critical-size crack 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.

(6) 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 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, maximum crack growth).

STAFF EVALUATION AND CONCLUSIONS

Based on its evaluation of the analysis contained in BAW-1847, Rev. 1 (Reference 3) and the materials data presented in BAW-1889P (Reference 4), the staff finds that the B&WOG has presented acceptable technical justification, addressing the preceding criteria, to eliminate, as a design basis, the dynamic effects of large ruptures in the main loop primary coolant piping of the B&WOG facilities evaluated. Specifically:

(1) The loads associated with the highest stressed location in the main loop primary system piping are 1,685.7 kips (axial), 37,171 in-kips (bending moment) and result in maximum stresses of about 51% of Service Level D limits specified in Section III of the ASME Code.

(2) For the B&WOG facilities, there is no history of cracking failure in reactor primary coolant system main loop piping. The reactor coolant system 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 53 reactor-years spanning 13 years of operation.

(3) The leak rate calculations performed for the B&WOG facilities used initial postulated throughwall flaws larger in size than those of Enclosure 1 to Reference 5. B&WOG facilities have an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide 1.45 such that leakage of one (1) 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 plant leak detection systems; the margin is at least a factor of ten (10) on leakage.

(4) The margin in terms of load based on fracture mechanics analyses for the leakage-size crack under normal plus SSE loads (Service Level D loads) meets NUREG-1061, Volume 3, guidance on margins. Based on a limit-load analysis, the load margin is at least $\sqrt{2}$. Similarly, based on the J limit, the margin is at least $\sqrt{2}$.

(5) The margin between the leakage-size crack and the critical-size crack was calculated by a limit load analysis. Again, the results demonstrated that a margin of at least 2.0 exists and is within the guidelines of NUREG-1061, Volume 3.

(6) In their review of the reactor coolant piping, the B&WOG first listed all the base metals and weld metals represented. From a review of published test data -- J-R curves and tensile properties -- the materials from the list that were most likely to be limiting were identified. A test program was then conducted to obtain the toughness and tensile data required. From these data, a limiting J-R curve and the associated tensile stress-strain curve was selected for the fracture analyses of the base metal and weld metal in the straight sections and elbows of the piping identified for evaluation. The staff concludes that the choice of limiting materials is satisfactory.

In view of the analytical results presented in Reference 3, the materials data contained in Reference 4, and the staff's evaluation findings related above, the staff concludes that the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of the B&WOG facilities is sufficiently low such that dynamic effects associated with postulated pipe breaks in these facilities need not be a design basis.

1. B&W Owners Group Report BAW-1847, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," September 1984.

2. Letter to L. C. Oakes of the B&W Owners Group," B&WOG Leak-Before-Break Report, BAW-1847," dated March 12, 1985.

3. B&W Owners Group Report BAW-1847, Rev. 1, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," September 1985.

4. B&W Owners Group Report BAW-1889P, "Piping Material Properties for Leak-Before-Break Analysis," A. L. Lowe, Jr., K. K. Yoon and R. H. Emanuelson, October 1985, proprietary.

5. NRC Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Breaks in PWR Primary Main Loops," February 1, 1984.

6. Westinghouse Report WCAP-9558, Rev. 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-wall Crack," May 1981, Class 2 proprietary.

7. Westinghouse Report WCAP-9687, "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation," May 1981, Class 2 proprietary.

8. Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981 Letter Report NS-EPR-2519, E. P. Rahe to Darrell G. Eisenhut, November 10, 1981, Westinghouse Class 2 proprietary.

9. T. Lo, H. H. Woo, G. S. Holman and C. K. Chou, "Failure Probability of PWR Reactor Coolant Loop Piping," presented at the ASME PVP Conference and Exhibition, June 17-21, 1984, San Antonio, Texas.