UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
GEORGIA POWER COMPANY, ET. AL	)	Docket Nos. 50-424
	)	and 50-425
(Vogtle Electric Generating Plant, Units 1 and 2)	)	

EXEMPTION

I.

On August 1, 1972, the Georgia Power Company, the Municipal Electric Authority of Georgia, the Oglethorpe Power Corporation, and the City of Dalton, Georgia (the licensees) tendered an application for licenses to construct Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle or the facility) with the Atomic Energy Commission (currently the Nuclear Regulatory Commission or the Commission). Following a public hearing before the Atomic Safety and Licensing Board, the Commission issued Construction Permit Nos. CPPR-108 and CPPR-109 permitting the construction of Units 1 and 2, respectively, on June 28, 1974. Each unit of the facility is a pressurized water reactor, containing a Westinghouse Electric Company nuclear steam supply system, located at the licensee's site in Burke County, Georgia.

On June 30, 1983, the licensees tendered an application for Operating Licenses for each unit of the facility, currently in the licensing review process.

II.

The Construction Permits issued for constructing the facility provide, in pertinent part, that the facility units are subject to all rules, regulations and Orders of the Commission. This includes General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50. GDC 4 requires that structures, systems and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, discharging fluids that may result from equipment failures, and from events and conditions outside the nuclear power unit.

In a submittal dated October 25, 1983, the applicants enclosed Westinghouse Report MT-SME-3082 (Reference 1) containing the technical basis for their request to: (1) eliminate the need to postulate circumferential and longitudinal pipe breaks in the Reactor Coolant System (RCS) primary loop (hot leg, cold leg and cross-over leg piping); (2) eliminate the need to install pipe whip restraints and jet impingement shields associated with previously postulated breaks in the RCS primary loops and; (3) eliminate the need to consider dynamic effects and loading conditions specifically associated with previously postulated pipe breaks in the RCS primary loop, including jet impingement loads, cavity pressure loads, blowdown loads in the RCS and attached piping, and subcompartment pressure loads. By a subsequent submittal dated April 2, 1984, the applicants requested an exemption from a portion of the requirements of GDC 4 related to the above, in support of the prior request. The applicants also stated in their submittals that the exemption request does not apply to the design bases for the containment including the design basis for structural loading of subcompartment walls and floors, the emergency core cooling system, or environmental qualification. The applicants also stated that the design of their reactor coolant system supports would remain unchanged.

Based on its review of the applicants' October 25th submittal, the NRC staff requested additional information and provided comments on the reports (References 1 and 9) which were transmitted to the applicant in the form of questions by NRC letter dated March 19, 1984 (Reference 2).

By a submittal dated May 17, 1984, the applicants responded to the staff's questions, providing a new report identified as Westinghouse Report WCAP-10551 (Reference 3). In a separate submittal, dated April 2, 1984, the applicants provided a value-impact analysis which, together with the technical information contained in the Reference 3 report, provided a comprehensive justification for requesting a partial exemption from the requirements of GDC 4.

By letter dated December 21, 1984, the applicants described their present installation status of the pipe whip restraints and jet impingement shields for both Unit 1 and Unit 2. For Unit 1, of the twenty-four (24) pipe whip restraints per unit, only support structures for sixteen (16) are installed. No bearing bars or attachments have been installed and no shimming has begun. For Unit 2, eight (8) are similarly partially installed. Installation has not begun for the remainder of the Unit 1 and Unit 2 restraints. Additionally, none of the jet impingement shields for either unit have been installed.

From the deterministic fracture mechanics analysis contained in the technical information furnished, the applicants concluded that the postulated double-ended guillotine breaks (DEGB) of the primary loop coolant piping in Vogtle, Units 1 and 2, need not be considered as a design basis for installing protective structures, such as pipe whip restraints and jet impingement shields, to guard against the dynamic effects associated with such postulated breaks. However, the applicant proposes to continue to postulate the DEGB as the design basis for the containment subcompartments, for the ECCS and for environmental qualification.

III.

The Commission's regulations require that applicants provide 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 sixteen facilities owned by eleven 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 4). 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 5, 6 and 7) 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 included as part of Reference 4.

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 8) 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 8 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. 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 4. 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.

IV.

The primary coolant system of Vogtle, Units 1 and 2, described in Reference 3, has four main loops each comprising a 33.9 inch diameter hot leg, a 36.2 inch diameter crossover leg and 32.14 inch diameter cold leg piping. The material in the primary loop piping is cast stainless steel (SA 351 CF8A). In its review of Reference 3, the staff evaluated the Westinghouse analyses with regard to:

- the location of maximum stresses in the piping, associated with the combined loads from normal operation and the SSE;
- potential cracking mechanisms;
- size of 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 thermally-aged cast stainless steel piping and weld material.

The NRC staff's criteria for evaluation of the above parameters are delineated in its Topical Report Evaluation, Enclosure 1 to Reference 4, Section 4.1, "NRC Evaluation Criteria", 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 and the lowest material toughness are coincident 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 adequate margin 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 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 the leakage-size crack to critical-size cracks. Under normal plus SSE loads, it should be demonstrated that there is adequate margin 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).

V.

Based on its evaluation of the analysis contained in Westinghouse Report WCAP-10551 (Reference 3), the staff finds that the applicants have presented an acceptable technical justification, addressing the above criteria, for not installing protective devices to deal with the dynamic effects of large pipe ruptures in the main loop primary coolant system piping of Vogtle, Units 1 and 2. This finding is predicated on the fact that each of the parameters evaluated for Vogtle is enveloped by the generic analysis performed by Westinghouse in Reference 5, and accepted by the staff in Enclosure 1 to Reference 4. Specifically:

(1) The loads associated with the highest stressed location in the main loop primary system piping are 1,962 kips (axial), 28,810 in-kips (bending moment) and result in maximum stresses of about 75% of the bounding stress used by Westinghouse in Reference 5. Further, these loads are approximately 70% of those established by the staff as limits (e.g. a moment of 42,000 in-kips in Enclosure 1 to Reference 4).

(2) For Westinghouse plants, there is no history of cracking failure in reactor primary coolant system loop piping. The Westinghouse 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 400 reactor-years, including five (5) plants each having 15 years of operation and 15 other plants with over 10 years of operation.

(3) The results of the leak rate calculations performed for Vogtle, using an initial through-wall crack of 7.5 inches, are identical to those of Enclosure 1 to Reference 4. The Vogtle plant has an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide 1.45, and it can detect leakage of one (1) gpm in one hour. The calculated leak rate through the postulated flaw results in a factor of at least 10 relative to the sensitivity of the Vogtle plant leak detection system.

(4) The margin in terms of load based on fracture mechanics analyses for the leakage-size crack under normal plus SSE loads is within the bounds calculated by the staff in Section 4.2.3 of Enclosure 1 to Reference 4. Based on a limit-load analysis, the load margin is about 2.9 and based on the J limit discussed in (6) below, the margin is at least 1.5.

(5) The margin between the leakage-size crack and the critical-size crack was calculated by a limit load analysis. Again, the results demon-

strated that a margin of at least 3 on crack size exists and is within the bounds of Section 4.2.3 of Enclosure 1 to Reference 4.

(6) As an integral part of its review, the staff's evaluation of the material properties data of Reference 9 is enclosed as Appendix 1 to this exemption. In Reference 9, data for ten (10) plants, including the Vogtle units, are presented, and lower bound or "worst case" materials properties were identified and used in the analysis performed in the Reference 3 report by Westinghouse. The applied J for Vogtle in Reference 3 was substantially less than 3000 in-lb/in^2. Hence, the staff's upper bound of 3000 in-lb/in^2 on the applied J (refer to Appendix 1, page 6) was not exceeded.

In view of the analytical results presented in the Westinghouse Report for Vogtle (Reference 3) and the staff's evaluation findings related above, the staff concludes that the probability or likelihood of large pipe breaks occurring at the eight (8) locations in each primary coolant system loop of Vogtle, Units 1 and 2 is sufficiently low such that such pipe breaks and their associated dynamic loads as indicated in the applicants' October 25 letter need not be considered as a design basis for requiring pipe whip restraints and jet impingement shields. Eliminating the need to consider these dynamic loads for this particular application does not in any way affect the design bases for the containment, the emergency core cooling system, or the environmental qualification for Vogtle.

The staff also reviewed the value-impact analysis provided by the applicant in their April 2, 1984, submittal for not providing protective structures against postulated reactor coolant system loop pipe breaks to assure as low as reasonably achievable (ALARA) exposure to plant personnel. Consideration was given to design features for reducing doses to personnel who must operate, service and maintain the Vogtle instrumentation, controls, equipment, etc. The Vogtle value-impact analysis shows that the elimination of protective devices for RCS pipe breaks will save an occupational dose for plant personnel of approximately 700 person-rem for both units over their operating lifetime. The staff review of the analysis shows it to be a reasonable estimate of dose savings. Therefore, with respect to occupational exposure, the staff finds that there is a radiological benefit to be gained by eliminating the need for the protective structures.

VI.

In view of the staff's evaluation findings, conclusions, and recommendations above, the Commission has determined that, pursuant to 10 CFR 50.12(a), this exemption is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest. The Commission hereby approves the limited exemption from GDC 4 of Appendix A to 10 CFR Part 50, to permit the applicants not to further install pipe whip restraints and jet impingement shields and not to consider dynamic effects and loading conditions as detailed in Part II of this exemption associated with postulated pipe breaks of the eight (8) locations per loop in the Vogtle, Units 1 and 2 primary coolant system, as specified in Enclosure D of the applicants' letter dated October 25, 1983.

Pursuant to 10 CFR 51.32, the Commission has determined that the issuance of the exemption will have no significant impact on the environment (50 FR 4605).

The exemption will become effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 5th day of February 1985

REFERENCES

(1) Westinghouse Report MT-SME-3082, "Technical Bases for Eliminating Large Primary Loop Pipe Ruptures as the Structural Design Basis for Vogtle, Units 1 and 2," September 1983, Westinghouse Class 2 proprietary.

(2) Letter to D. O. Foster of Georgia Power Company, "Request for Additional Information Concerning Leak-Before-Break Analysis for Vogtle Electric Generating Plants, (Units 1 and 2)," dated March 19, 1984.

(3) Westinghouse Report WCAP-10551, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Vogtle Units 1 and 2," May 1984, Westinghouse Class 2 proprietary.

(4) 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.

(5) Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack, WCAP-9558, Rev. 2, May 1981, Westinghouse Class 2 proprietary.

(6) Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation, WCAP-9787, May 1981, Westinghouse Class 2 proprietary.

(7) 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.

(8) Lawrence Livermore National Laboratory Report, UCRL-86249, "Failure Probability of PWR Reactor Coolant Loop Piping," by T. Lo, H. H. Woo, G. S. Holman and C. K. Chou, February 1984 (Preprint of a paper intended for publication).

(9) 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 Class 2 proprietary.

(10) Georgia Power Company letter, "Alternative Pipe Break Design Considerations" (D. O. Foster to H. R. Denton) dated April 2, 1984.

Notes: See next page

REFERENCES

NOTE: Non-proprietary versions of References 1, 3, 5, 6, 7 and 9 are available in the NRC Public Document Room as follows:

(1) MT-SME-3082, non-proprietary
(3) WCAP 10552
(5) WCAP 9570
(6) WCAP 9788
(7) Non-proprietary version attached to the Letter Report
(9) WCAP 10457