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UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of ARIZONA PUBLIC SERVICE COMPANY, ET AL. (Palo Verde Nuclear Generating Station, Unit 1)	)))))))	Docket No. STN 50-528

EXEMPTION

I.

On July 11, 1974, the Arizona Public Service Company, the Salt River Project Agricultural Improvement and Power District, the El Paso Electric Company, the Public Service Company of New Mexico, and the Arizona Electric Power Cooperative, Incorporated (the applicants) tendered an application for licenses to construct the Palo Verde Nuclear Generating Station, Units 1, 2 and 3 (Palo Verde 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-141, CPPR-142 and CPPR-143 on May 25, 1976, permitting the construction of Units 1, 2 and 3, respectively. Each unit of the facility is a pressurized water reactor, containing a Combustion Engineering Company (CE) nuclear steam supply system which is a standard plant design referred to as CESSAR System 80 (CESSAR). The facility is located at the licensees' site in Maricopa County, Arizona.

On April 1978, the construction permits for Palo Verde, Units 1, 2 and 3 were amended to delete the Arizona Electric Power Cooperative, Incorporated, as a

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co-owner to the facility. On October 1, 1979, an application for operating licenses was tendered for each unit of the facility. On April 28, 1982, the construction permits for the three units were further amended to included the Southern California Public Power Authority and the Los Angeles Department of Water and Power as co-owners to the facility (the Los Angeles Department of Water and Power will actually become a co-owner after Palo Verde Unit 1 achieves commercial operation). On December 31, 1984 and June 1, 1985, Palo Verde Unit 1 was issued a low power license and a full power license, respectively. Palo Verde Units 2 and 3 are currently in the licensing review process.

II.

Facility Operating License No. NPF-41, issued for Palo Verde Unit 1 provides, in pertinent parts, that the facility is 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. The protective measures include physical isolation from postulated pipe rupture

locations, if feasible, or the installation of pipe whip restraints, jet impingement shields or compartments.

By letter dated June 7, 1984 (Reference 1) the licensees requested a partial exemption from GDC 4 for each unit of Palo Verde. By letters dated November 13 and 15, 1985 (References 2 and 3) the licensees submitted a request for a partial schedular exemption from the provisions of GDC 4 for Palo Verde Unit 1 for a period ending with the completion of the second refueling outage. Specifically, the licensees' request is to eliminate the need (1) to postulate circumferential and longitudinal pipe breaks in the RCS primary loop (hot leg, cold leg, and cross-over leg piping) specified in Section 3.6 of the Palo Verde Final Safety Analysis Report; (2) for associated pipe whip restraints in the RCS primary loop and the requirement to design for the structural effects associated with RCS primary loop pipe breaks, including jet impingement; and (3) to consider dynamic effects and loading conditions associated with these previously postulated primary loop pipe breaks. In support of the application, the licensees reference two documents: a report submitted by CE by letter dated June 14, 1983 (Reference 4) and an amendment to the CE report submitted by letter dated December 23, 1983 (Reference 5). The technical information contained in these two documents together with the value-impact analysis submitted by letter dated October 3, 1984 (Reference 6) provided a comprehensive justification for requesting a partial exemption from the requirements of GDC 4.

The CE submittals (References 4 and 5) contain the technical bases to demonstrate

that, for CESSAR plants, guillotine type failures of the RCS main loop piping need not be considered in the design basis and hence, pipe whip restraints and jet impingement shields for the RCS piping are not required. The submittals were made to support requests, by applicants with a CESSAR plant, for an exemption to GDC 4 as it relates to all postulated large pipe breaks specified in Section 3.6 of CESSAR-F, pipe whip restraints and jet impingement shields on the RCS primary piping and associated dynamic effects. 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 of 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 licensees' exemption request (References 2 and 3) also states that no other changes in design requirements are being requested.

III.

The technical bases provided by CE for the exemption request (References 4 and 5) relied on advanced fracture mechanics technology. These advanced fracture mechanics techniques, which make possible the acceptance of the technical bases, deal with relative 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 double-ended guillotine break (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 also applied to Westinghouse topical reports (References 7, 8, and 9) submitted to the staff on behalf of the licensees belonging to the Owners Group for Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on PWR Primary Systems". 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 evaluation of these Westinghouse reports is attached as Enclosure 1 to Reference 10.

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 11) 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} 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 to Reference 10. 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 CESSAR facilities, as described in References 4 and 5, has two (2) main loops each comprising a 42-inch diameter hot leg and two (2) 30-inch diameter crossover legs and cold legs. The materials in the

primary loop piping are SA 516 Gr 70 (pipes) and SA 508 CL 1, 2 or 3 (safe ends and nozzles). The piping system is cladded on the inside surface with stainless steel. In its review of References 4 and 5, the staff evaluated the CE analyses with regard to:

- the location of maximum stresses in the piping, associated with the combined loads for normal operation and the SSE;

- potential cracking mechanisms;

- size of throughwall 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 carbon steel piping and weld material.

The NRC staff's criteria for evaluation of the above parameters are delineated in Enclosure 1 to Reference 10, 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, 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 of 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 systems 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 throughwall 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-size 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 that the final crack size is limited, such that a double-ended pipe break will not occur.

(5) The crack size margin 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 in 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.

The staff's evaluation of the analysis contained in the CE submittals (References

4 and 5), is presented in Reference 12. Based on that evaluation, the staff finds that CE has 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 CESSAR facilities. As stated in Reference 12, this finding is based on the following observations:

(1) The loads associated with the highest stressed locations in the main loop primary system piping were provided and are within Code allowables.

(2) For CE plants, there is no history of cracking failure in reactor primary coolant system loop piping. CE reactor coolant system primary loops have an operating history which demonstrates their 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 includes several plants with many years of operation.

(3) The results of the leak rate calculations performed for CESSAR used initial postulated throughwall flaws that are equivalent in size to that in Enclosure 1 to Reference 10. CESSAR facilities are expected to have an RCS pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide 1.45 so that they can detect leakage of one (1)

gpm in one hour. This will be verified during the case-by-case review of each applicant's submittal. 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 expected margin in terms of load for the leakage-size crack under normal plus SSE loads is greater than a factor of three (3) when compared to the limit load. In addition, the staff found a significant margin in terms of loads larger than normal plus SSE loads.

(5) The margin between the leakage-size crack and the critical-size crack was calculated. Again, the results demonstrated that a crack size margin of at least a factor of three (3) exists.

In view of the analytical results presented in References 4 and 5 and the staff's evaluation findings related above, the staff concluded that the probability or likelihood of large pipe breaks occurring in the primary coolant system loop of a CESSAR facility is sufficiently low such that protective devices associated with postulated pipe breaks in the CESSAR primary coolant system need not be installed.

The staff evaluation (Reference 12) stated that applicants or licensees with CESSAR facilities who intend to use the "leak-before-break" approach to eliminate the need to install protective devices associated with postulated

pipe breaks in their primary coolant systems must confirm that their as-built facility design substantially agrees with the design described in References 4 and 5; specifically, the piping loads should be no greater than those cited in the references. Also, applicants or licensees must confirm that their leak detection systems meet the staff's requirements in (3) above.

Reference 6 states that the leak-before-break analysis performed by CE (References 4 and 5) was performed on the Palo Verde design (as the prototypical CESSAR plant) using pertinent Palo Verde parameters. Hence, the CE analysis envelopes the Palo Verde design with respect to such parameter as loads, material properties, postulated crack leakage and size, seismicity, and leak detection system capabilities. In addition, the leak detection system for Palo Verde is consistent with the guidelines of Regulatory Guide 1.45 so that it can detect leakage of one (1) gpm in one hour. Therefore, the Palo Verde design substantially agrees with the design described in References 4 and 5.

Based on the above evaluation, the staff concludes that the probability or likelihood of large pipe breaks occurring in the RCS main loop piping for Palo Verde, Unit 1 is sufficiently low such that pipe breaks in the RCS main loop piping and their associated dynamic loads, as indicated in the licensees' November 13 and 15, 1985, letters, need not be considered as a design basis for requiring pipe whip restraints and jet impingement shields for this piping. The Commission currently has in progress a rulemaking regarding the issue of "leak-before-break". In order to provide the Commission with an opportunity to consider the long term aspects of the NRC staff's recent acceptance criteria of the "leak-before-break" approach, this exemption is limited to a period extending until the completion of the second refueling

outage of Palo Verde Unit 1, pending the outcome of Commission rulemaking on this issue. Eliminating the need to consider these dynamic loads for this particular application does not in any way affect any other design bases for the plant and, in particular, for the containment, the emergency core cooling system, or the environmental qualification for Palo Verde.

The staff also reviewed the value-impact analysis, provided by the applicants in their October 3, 1984, submittal (Reference 6) 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. The Palo Verde value-impact analysis shows that the elimination of protective devices for RCS pipe breaks will save an occupational dose for plant personnel of approximately 560 person-rem for each unit over the operating lifetime of the facility. 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 schedular partial exemption from GDC 4 of Appendix A to 10 CFR Part 50, to permit the licensees not to consider

dynamic effects, as detailed in Part II of this exemption, and, hence, not require pipe whip restraints and jet impingement shields associated with postulated pipe breaks in the RCS main loop piping of Palo Verde, Unit 1, as specified in the licensees' letters, dated June 7, 1984 and November 13, and 15, 1985. This exemption is for a period ending with the completion of the second refueling outage, or the adoption of the proposed rulemaking for modification of GDC 4, whichever occurs first.

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 48285).

The exemption is effective upon the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Hugh L. Thompson

Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 22 day of November, 1985

References

(1) Letter E. E. Van Brunt, Jr., of Arizona Public Service Company to the Nuclear Regulatory Commission, Docket Nos. STN 50-528/529/530, June 7, 1984.

(2) Letter E. E. Van Brunt, Jr., of Arizona Public Service Company to the Director of Nuclear Reactor Regulation, Docket Nos. STN 50-528, November 13, 1985.

(3) Letter E. E. Van Brunt, Jr., of Arizona Public Service Company to the Nuclear Regulatory Commission, Docket Nos. STN 50-528, November 15, 1985

(4) Letter A. E. Scherer of Combustion Engineering, Inc., to Darrell G. Eisenhut, Docket No. STN 50-470, June 14, 1983, with enclosure, "Basis for Design of Plant Without Pipe Whip Restraints for RCS Main Loop Piping".

(5) Letter A. E. Scherer of Combustion Engineering, Inc., to Darrell G. Eisenhut, Docket No. STN 50-470F, December 23, 1983, with enclosure, "Leak Before Break Evaluation of the Main Loop Piping of a CE Reactor Coolant System," Revision 1, November 1983.

(6) Letter E. E. Van Brunt, Jr., of Arizona Public Service Company, ANPP-30736 to the Nuclear Regulatory Commission, Docket Nos. STN 50-528/529/530, October 3, 1984.

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

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

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

(10) 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.

(11) 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).

(12) NRC Letter Cecil O. Thomas to A. E. Scherer of Combustion Engineering, Inc., Docket No. STN 50-470, October 11, 1984, with enclosure, "Safety Evaluation Report on the Elimination of Large Primary Loop Ruptures as a Design Basis".

References (Cont.)

NOTE: Non-proprietary versions of References 7 and 8 are available in the NRC Public Document Room as follows:

(7) WCAP 9570
(8) WCAP 9788