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Norman W. Curtis
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May 31, 1983

Mr. J. M. Allan
Acting Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

SUSQUEHANNA STEAM ELECTRIC STATION-UNIT 2
INTERIM REPORT OF A DEFICIENCY INVOLVING FEEDWATER
BYPASS LEAKAGE - POTENTIAL EXCESSIVE OFF-SITE DOSE
ER 100508 FILE 821-10
PLA-1687

Docket No. 50-388

Dear Mr. Allan:

This letter serves to provide the Commission with an interim report on a deficiency involving potential excessive off-site dose related to the feedwater containment penetrations not maintaining a water seal for 30-days post-LOCA. This deficiency was originally reported by telephone to Mr. D. Johnson of NRC Region I on April 27, 1983 by Mr. J. Saranga of PP&L as potentially reportable under the requirements of 10CFR50.55(e) for SSES Unit 2. Further evaluation of the deficiency by PP&L resulted in the conclusion that the condition should now be classified as reportable as a significant deficiency in final design, as defined in 10CFR50.55(e).

The attachment to this letter contains a description of the deficiency, its cause and extent, and the safety implications. Alternative plans for corrective action are currently being assessed. PP&L anticipates providing the Commission with a final report in July, 1983, including the corrective action taken or planned. This information is furnished for Unit 2 pursuant to the provisions of 10CFR50.55(e).

Since the details of this report provide information relevant to the reporting requirements of 10CFR21 for Unit 2, this correspondence is considered to also discharge any formal responsibility PP&L may have in compliance thereto.

This information was reported to the Commission for Unit 1 in a letter (PLA-1662) dated 5/12/83 which transmitted Licensee Event Report No. 83-057.

We trust the Commission will find this report to be satisfactory.

Very truly yours,

RJ Shovlin for

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

Attachment

IE27 1/1

cc: Mr. Richard C. DeYoung (15)
Director-Office of Inspection & Enforcement
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Washington, D.C. 20555

Mr. G. McDonald, Director
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Interim Report on Feedwater Bypass Leakage (8.1.2A #83-12)

PROBLEM:

The feedwater containment penetrations at Susquehanna were designed to remain water filled for 30 days post-LOCA. Review of the design has yielded the conclusion that the water seal may not exist under certain design basis conditions.

CAUSE:

The Susquehanna FSAR states that the feedwater containment penetrations are designed to remain water sealed for 30-days post-LOCA. The water seal is relied upon to prevent the primary containment atmosphere from bypassing secondary containment and the Standby Gas Treatment System through the feedwater lines during a LOCA. During the Technical Specification review, NRC questioned whether the penetrations were truly water sealed and suggested that local leak rate testing be performed using air as the working fluid. PP&L agreed to use air, but retained the contention that the lines will remain water filled. Bechtel Power Corporation provided an analysis of post-accident flashing in feedwater lines and concluded that enough water remained above the top of the valve seats to seal the penetration for 30-days. This analysis assumed valve leakage would be less than or equal to the leak test criteria in Manufacturer's Standard MSS-SP-61. The calculated allowable leak rate was transmitted to the Susquehanna Integrated Startup Group for use as a test acceptance criteria.

It was later discovered that leak rate testing was never performed to confirm that the volume of feedwater remaining in the lines would last for 30-days. During the investigation of this potential problem, the feedwater flashing analysis was reviewed. Three important deficiencies were found in the calculations. First, the calculation considered only the volume of piping from the RPV nozzles out to the inboard isolation valves, not the outboard valves. This is a nonconservative assumption. Second, the sensible heat of the pipe walls and the valves was neglected. More seal water would be lost by boiling while cooling the hot metal. Third, flashing was assumed to occur at the free surface of the water in the pipe. Due to the large volume of steam relative to water, any flashing that occurred in portions of the pipe upstream of the free surface would entrain some of the liquid in the downstream piping and carry it out into the RPV.

Our evaluation of these three deficiencies caused us to conclude that there is insufficient water in the feedwater lines immediately following rapid depressurization of the RPV to cover the seating surface of the isolation valves for 30 days, regardless of the leak tightness of the valves.

EXTENT:

FSAR Table 6.3-15 lists all potential bypass paths for secondary containment. Several other penetrations use water seals to prevent bypass leakage: Reactor Water Clean-up (RWCU) from Reactor Vessel, Suppression Pool Purification Line, Liquid Radwaste Collection, and Reactor Building Closed Cooling Water (RBCCW). The justification of water seals for these penetrations was reviewed to determine whether the feedwater problem bore any implications in these cases.

RWCU and the Suppression Pool drain draw their seal water from relatively inexhaustible sources and therefore were not considered further. Liquid Radwaste and RBCCW, however, are sealed by water in piping at a higher elevation than their respective isolation valves. For these two penetrations, a maximum allowable leak rate was calculated such that the inventory remains for 30-days post-LOCA. This allowable value was compared with expected leak rates for the valves used and in each case, the expected leak rate is much lower than the allowable. A program of testing and maintenance will verify that this is true throughout the life of the plant. The major difference between the seal water analysis for these lines and the feedwater lines is flashing. In all the lines where we now take credit for seal water, the water is relatively cold throughout the postulated accident.

SAFETY IMPLICATIONS:

The analysis of flashing in the feedwater lines following rapid depressurization of the reactor pressure vessel has been shown to be invalid. This determination has no bearing on any system other than feedwater. The implication of this determination is that the design of the feedwater system is insufficient to perform as described in the FSAR.

Since the feedwater isolation valves will not remain water sealed post-LOCA, the safety implication depends on how well the valves perform when tested with air. The measure of success of an air test depends upon the amount of bypass air leakage assumed in the analysis of the radiological consequences of a large break LOCA. This analysis, presented in Chapter 15 of the FSAR, used 5.0 scf per hour of bypass leakage. The Technical Specifications allotted 1.2 scf per hour of leakage to the Main Steam Drain lines which are the only other viable bypass paths (per FSAR Section 6.2.3.2.3). Therefore, as long as the feedwater lines leak less than or equal to 3.8 scf per hour, the accident analysis is still valid. Though the feedwater design does not support a water seal, there is no adverse safety effect if the demonstrable leak rate is less than 3.8 scf per hour.

However, the 3.8 scf per hour leakage criterion was not specified in the Technical Specifications; it was not deemed necessary because the water seal should have prevented the leakage entirely. The current leakage criteria for the feedwater valves is based only on the requirement that all Type C tested (per 10CFR50 Appendix J) penetrations leak less than 0.6 La. During the life of the plant, the valves may have been allowed to leak so much that off-site doses would have been excessive. This could constitute an unsafe condition if a LOCA occurred while the valves were in this state.

Since this deficiency could have adversely affected the safety of operations of the plant, and it represents a significant deficiency in final design, we conclude that it is reportable as defined in 10CFR50.55(e)(1)(ii).

CORRECTIVE ACTION:

Alternative plans for corrective action are currently being assessed. Finalized corrective action will be specified in the final report.