10/6/75

SUPPLEMENTAL TESTIMONY

OF

NRC STAFF

ON

EMERGENCY CORE COOLING SYSTEM

WASHINGTON NUCLEAR PROJECTS NOS. 1 AND 4
DOCKET NOS. 50-460 AND 50-513

BY

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1.0 INTRODUCTION

The NRC staff completed its evaluation of Applicant's preliminary design criteria and designs with respect to compliance with 10 CFR §50.46. The staff conclusion was that, with certain required modifications, Applicant's design will be in conformance with regulatory requirements. Applicant has conditionally committed to staffs position in these matters, and staff concludes that Applicant has met all regulatory requirements for preliminary design of the ECCS, at the construction permit stage of review.

The following sections will summarize the development of the staff review and conclusions. Attachment A to this testimony is a numbered list of references. The following sections contain references to individual numbers on the Attachment A listing.

ECCS EVALUATION

2.0 By letter of June 19, 1975 (reference 1), Applicant incorporated the Babcock & Wilcox (B&W) topical report No. BAW-10102 (reference 2) into their application to construct the WNP-1,4 facility. Pursuant to the requirements of the Commission's regulations, 10 CFR §50.46, the B&W topical report was submitted to demonstrate compliance with the ECCS Final Acceptance Criteria for the WNP-1,4 facility. The Applicant has submitted additional information in references 3, 4 and 5.

Compliance with the NRC Interim Acceptance Criteria was previously demonstrated, as reported in staff's SER on this application.

In addition to a revised LOCA analysis, the staff's recent review has addressed the specific areas of minimum containment pressure, single failure criterion, effects of boron precipitation on long term cooling capability, operability of valves that might be submerged within containment following a LOCA, and partial loop operation.

These individual review areas are discussed in this testimony, and detailed technical conclusions are presented supporting the staff conclusions stated in Section 1.0 above.

2.1 <u>Revised LOCA Analyses for Final Acceptance Criteria Evaluation</u>

Applicant's submittals to the staff addressed the loss of coolant from postulated small pipe ruptures of 0.5 ft^2 and smaller (reference 6), and postulated major pipe ruptures of the reactor coolant system (reference 2). Analyses submitted were performed with an evaluation model (reference 7) which is considered by the staff to conform to Appendix K to 10 CFR §50.

A selected number of break sizes, configurations and locations were analyzed in accordance with staff requirements. The analyses identified the worst-case break as the 8.55 ft^2 double-ended break at the pump discharge.

The table below summarizes Applicant's results of the LOCA analyses which determine the allowable linear heat rate limits as a function of elevation in the core.

Elevation, ft.	LOCA limit, kW/ft.	Peak Clad Temp., ^oF		Maximum Local Oxidation, %	Time to Rupture, Sec. after LOCA
		Ruptured	Unruptured		
2	14.9	2097	1931	4.1	25.90
4	16.2	2002	2156	5.9	26.19
6	16.8	2017	2126	5.3	25.40
8	15.3	1763	2177	6.7	32.30
10	14.2	1880	2171	6.3	26.75

As shown in the above table, the calculated values for the peak clad temperature, and local total clad oxidation were below the allowable limits of 2200^{o}F and 17%, respectively, which are specified in 10 CFR §50.46. Reference 2 also demonstrates from results of analyses that the core geometry remains amenable to cooling and that long term cooling can be established.

The maximum core-wide metal/water reaction was calculated to be 0.62% which is below the allowable limit of 1%.

The analyses of postulated breaks during partial loop operation (shutdown of one or more reactor coolant pumps) has not been provided by Applicant. Applicant has indicated that these analyses will be provided during the first quarter of 1976.

The staff position on partial loop operation is that such operation must be specifically prohibited by preliminary design Technical Specifications until staff review and approval of Applicant's analyses is complete. Inclusion of such prohibition in the preliminary design Technical Specifications is considered acceptable at the CP stage of the licensing process.

2.2 Potential Boron Precipitation During Long Term Cooling After LOCA

Applicant has described procedures and equipment that would be used to assure reactor coolant circulation through the reactor core after a LOCA. This circulation would prevent excessive boron concentration and subsequent precipitation from the coolant within the reactor vessel. The staff has reviewed Applicant's procedure, designated Mode 1 cooling, and the applicable referenced analyses. Staff has concluded that the Mode 1 design and procedure are adequate to provide long term cooling and to prevent excessive boron concentration, provided that this mode of operation is achieved from the control room. Staff also finds that the Mode 1 cooling design and operating procedure will also meet single failure criteria, provided that certain modifications are incorporated as discussed in paragraph 2.3 in this testimony.

The Mode 1 cooling procedure consists of establishing suction from the decay heat drop (let down) line and the reactor building sump with one of the low pressure injection trains. In order to establish suction from the drop line, an operator must operate a throttling valve in the bypass line. The staff's position is that each of the throttling valves in both trains must be motor-operated valves with control and position indication in the control room. In addition, the staff requires that flow indication in the bypass lines shall also be provided in the control room.

Applicant has indicated in a letter to the staff dated September 26, 1975 (Ref. 9) that their design will incorporate the operating controls and flow and position indicators in the control room as required by the staff, if a planned applicant submittal to the staff fails to convince the staff that local controls and indicators (at or near the valve locations) are adequate. This submittal, to be presented on or before December 31, 1975, will be based on the Applicant position that sufficient time is available after any design basis LOCA to permit operator access to the plant location necessary to place the Mode 1 cooling procedure into effect.

The staff finds Applicant's commitment acceptable at the CP stage of review.

2.3 Single Failure Criterion

Appendix K to 10 CFR §50 of the Commission's regulations requires that the combinations of ECCS subsystems to be assumed operative shall be those available after the most damaging single failure of ECCS equipment has occurred.

The analyses submitted by the Applicant conservatively assumed all containment cooling systems operating for the independent containment calculation, and assumed the diesel failure for the ECCS calculation.

Applicant responded (Ref. 4) to staff requirements for a single failure analysis of ECCS valves. Eight valves were identified as subject to single failures for which the consequences to ECCS function could be unacceptable. Applicant initially committed to administrative controls (plant Technical Specifications) which would assure that electrical power to six of the valves was locked out during normal operation, preventing spurious operation of these electric motor operated valves due to an electrical failure. In the same letter, Applicant committed to unspecified design changes to provide single failure protection for the remaining two air-operated valves.

In response to a staff letter of September 19, 1975 (Ref. 8), Applicant has further committed, in a letter to the staff on September 26, 1975 (Ref. 9), to positively lock out motive power to the air-operated valves in a manner that assures that the valves will not close by inadvertant admission of air to the valves. We find Applicant's commitments regarding protection against single failure of ECCS valves acceptable.

Applicant identified seven valves that could be submerged following a LOCA. Of these, only the submergence of the two Decay Heat Removal System (DHRS) letdown isolation valves could result in unacceptable consequences. Applicant relocated the DHR valves to an elevation of 412 feet, positioning the valve motor operator at an elevation of 417 feet. Applicant indicated that the maximum sump flood level is 410 feet. Staff finds this design acceptable.

2.4 <u>ECCS Containment Pressure Evaluation</u>

The ECCS containment pressure calculations for WNP-1,4 were completed using the Babcock & Wilcox (B&W) ECCS generic containment pressure evaluation model. The staff required justification of the plant-dependent input parameters used in the generic analysis. Applicant provided, in reference 5, its analysis using plant-dependent parameters specific to the WNP-1,4 facility design. The staff judged that Applicant's values for containment net free volume, and assumptions concerning operation of containment heat removal systems, were conservatively selected for the ECCS analysis.

Passive heat sink data were determined using guidelines provided by the staff.

We have concluded that the plant-dependent information provided by Applicant for WNP-1,4 is conservative for ECCS analysis Therefore, Applicant's calculated containment pressures are in accordance with Appendix K to 10 CFR §50.46 of the Commission's regulations.

The plant dependent data inputs specific to WNP-1,4 resulted in a containment pressure slightly lower (much less than 1 psi) than that obtained using the B&W generic model for the first 60 seconds after the postulated LOCA, and higher thereafter. Applicant concluded, and the staff concurs, that little or no difference in peak cladding temperature would result from the two different pressure calculations.

A B&W sensitivity study showing peak clad temperature variation with containment pressure showed that a containment pressure decrease of 1 psi would cause a peak clad temperature increase of 17 °F

2.5 Conclusions

The staff's completed review of the WNP-1,4 facility preliminary design compliance with the ECCS final acceptance criteria has resulted in the following conclusions:

1) Compliance with the acceptance criteria of 10 CFR §50, Appendix K has been demonstrated.
2) The single failure criterion (Appendix K, Section I.D.1) is satisfied with Applicant's commitment to modify the WNP-1,4 Technical Specifications to require the positive locking out of motive power to specified valves.
3) The procedure to control boron concentration during the post-LOCA, long term cooling period is acceptable with Applicant's commitment concerning the provisions for operation of the system from the control room.
4) ECCS minimum containment pressure calculations were performed in accordance with 10 CFR §50, Appendix K.
5) Partial loop operation of the WNP-1,4 units shall be prohibited by Technical Specifications until Applicant's analyses of the consequences of postulated breaks during partial loop operation is reviewed and approved by the staff.
6) Applicant's analysis of submerged valves and commitment to relocate the two DHRS letdown line isolation valves is acceptable.

ATTACHMENT A

REFERENCES

1. Letter from Applicant, J. J. Stein to A. Giambusso - G01-75-122, "WPPSS Nuclear Projects Nos. 1 & 4 Submittal of Information Demonstrating Compliance with the ECCS Final Acceptance Criteria," June 19, 1975.

2. Lowe, R. J., et al., "ECCS Evaluation of B&W's 205-FA NSS Revision 1," Babcock and Wilcox Company, BAW-10102, July 1975.

3. Letter from Applicant, N. O. Strand to A. Giambusso - G01-75-147, "WPPSS Nuclear Projects Nos. 1 & 4 Schedule for Submittal of Additional Information for ECCS Analysis," July 11, 1975.

4. Letter from Applicant, N. O. Strand to A. Giambusso - G01-75-150, "WPPSS Nuclear Projects Nos. 1 & 4 Submittal of Additional ECCS Information," July 18, 1975.

5. Letter from Applicant, N. O. Strand to A. Giambusso - G01-75-153, "WPPSS Nuclear Projects Nos. 1 & 4 - Submittal of Additional ECCS Information," July 25, 1975.

6. Jones, R. C., et al., "Multinode Analysis of Small Breaks for B&W's 205 - Fuel Assembly Nuclear Plants with Internals Vent Valves," BAW-10074, Babcock & Wilcox Company, November 1973.

7. Dunn, B. M., et al., "B&W's ECCS Evaluation Model," BAW-10104, Babcock & Wilcox Company, May 1975.

8. Letter to Applicant, A. Schwencer to J. Stein, Docket No. 50-460 and 50-513, September 19, 1975.

9. Letter from Applicant, N. O. Strand to R. Boyd - "WPPSS Nuclear Projects Nos. 1 and 4 Additional ECCS Information," September 26, 1975.