UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION REPORT

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE WASHINGTON NUCLEAR PROJECT NO. 2 FIRE PROTECTION

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

1.0 INTRODUCTION

In a memorandum dated March 12, 1986, Region V requested NRR to evaluate three areas of concern with respect to WNP-2 fire protection. These are the acceptability of using the automatic depressurization system (ADS) in conjunction with low pressure coolant injection (LPCI); the need for the licensee to perform a "hot short to overvoltage" associated circuits common enclosure analysis; and the determination as to whether the WNP-2 control or cable spreading room fire analysis is adequate and does not represent any unreviewed or unanalyzed safety concern. This evaluation presents the results of our review of the WNP-2 facility design bases as they relate to these three concerns.

2.0 EVALUATION

2.1 With respect to the licensee's changes in safe shutdown methodology, Region V indicated the following concern:

> The licensee, apparently without NRC concurrence, changed safe shutdown methodology from using the RCIC/RHR systems to cool the plant down to using the ADS/LPCI methodology. The original method was identified in the program submitted for review prior to licensing and is the method identified by their license condition on fire protection. In addition, on March 21, 1983, the licensee submitted to NRR a letter informing NRR that this action was planned. The analysis in the March 21, 1983 letter is based on a generic BWR-4. Another plant specific analysis which reflects the real cooldown method, not submitted to NRR, results in a greater amount of core uncovering. In January 1986, the NRR staff reviewed a similar issue with Grand Gulf. The staff required that GGNGS use 6 ADS valves, instead of the three that they proposed, to minimize the amount of core uncovering. WNP-2 also uses 3 ADS valves. Appendix R guidelines say that the core should have no uncovering.
>
> Please determine if the licensee's analysis supports safe shutdown of the facility.

The licensee's change of the safe shutdown methodology was necessitated by the fact that special thermal protection (thermolagging) has not been applied to the cable trays of the HPCS and the RCIC systems. The cable trays for the ADS valves and for RHR loop B and SSW loop B are thermally protected (thermolagged). Region V staff raised the issue of the adequacy of using the ADS/LPCI methodology for safe shutdown. The use of ADS/LPCI results in uncovery of the core and produces a transient on the suppression pool and, therefore, would not be the preferred means of maintaining reactor core cooling. Nevertheless, the use of ADS and LPCI is an approved and accepted means of achieving and maintaining a safe shutdown condition. This methodology is used by other licensees and the use of ADS and LPCI is considered to be an acceptable alternative shutdown capability.

The licensee initially had controls for only three ADS valves in the remote shutdown panel. During the first refueling outage at WNP-2, the licensee installed controls for three additional ADS valves in the remote shutdown panel. The licensee performed a plant-specific analysis using six ADS valves for safe shutdown. The analysis was based on the licensee's own models, which are not approved by the staff. Hence, the licensee contracted with GE to perform an independent verification of the analysis. However, it should be noted that three of the ADS valves are controlled from the remote shutdown panel and three are controlled from the alternate remote shutdown panel. The remote and alternate remote shutdown panels are located in different rooms and different fire areas. The alternate remote shutdown system has not been reviewed by NRR with respect to fire protection considerations. It is NRR's intent to review this subject with respect to the assessment of Amendment 37 to the FSAR. The alternate remote shutdown system is under review by NRR with respect to GDC 19.

The initial conditions for the postulated fire event used in the GE analysis are described as follows. At the start of the event, the reactor is assumed to be operating at full power, normal water level, and steady state conditions. The event is assumed to occur simultaneously with the instantaneous loss of all unprotected safe shutdown systems. It is also conservatively assumed that the loss of offsite power occurs at the same time, and that this leads to events such as reactor scram, turbine trip, loss of feedwater and isolation.

The GE analysis identifies that immediately after scram and isolation, the reactor pressure increase is controlled by operation of the SRVs. Normally, upon a sufficient drop in reactor water level, the high pressure makeup systems (i.e., High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC)) will supply coolant inventory. For this event, however, the RCIC and the HPCS systems are assumed to be unavailable for the reasons stated before.

The GE analysis assumes that after 10 minutes, six relief valves will be manually opened to depressurize the vessel. The capability to manually open the three ADS valves in the alternate remote shutdown panel and the three valves on the remote shutdown panel within the specified 10 minutes has not been reviewed. Generally,

no credit is given for operator action for the first 20 minutes if the required action is to be taken within the control room, or for the first 30 minutes if the required action is to be taken outside of the control room. Furthermore, the operator may not be able to open the six ADS valves within 10 minutes, since three of these valves are controlled from the remote shutdown panel, and three are controlled from the alternate remote shutdown panel which is located in a different fire area. The acceptability of the short period of time should be evaluated by the Region.

According to the GE analysis, as the reactor depressurizes, only one LPCI pump is assumed to be available. The LPCI pump rapidly restores the water level to near normal, after the vessel pressure has been reduced to below the LPCI shutoff head. The minimum water level inside the shroud occurs about 15 minutes into the event and would be approximately 4.1 feet below the top of the active fuel. As soon as LPCI flow enters the vessel, the water level begins to rise. The total fuel uncovery time is estimated to be about 5.4 minutes. The fuel node having the highest calculated Peak Cladding Temperature (PCT)) is uncovered for 4.5 minutes. The PCT is calculated to be 762°F and occurs approximately 19 minutes after initiation of the event.

No fuel cladding damage is expected to occur at this low temperature. No fuel rod perforations are expected to occur below 1700°F on the basis of General Electric's cladding swelling and rupture model (References 1, 2, 3). Cladding expansion begins at a temperature which is about 200°F lower than the perforation temperature (Reference 4). Hence, fuel cladding damage should not occur, since the peak cladding temperature remains below 1500°F.

GE has used SAFE and CHASTE which were previously reviewed by the staff for similar analyses. The assumptions used by GE are considered to be conservative.

BTP CMEB 9.5-1, Paragraph C.5.c guideline indicates that the core should be covered. However, short-term uncovery of the upper third of the core predicted by the GE analysis is an acceptable variance from the BTP, since fuel cladding integrity is maintained. This is supported by previous analyses for other BWR plants, wherein the same exemption was granted. Hence, we find this variance to be acceptable.

2.2 Region V had the following concern with respect the need for a "hot short to overvoltage" analysis:

> The team found that the licensee did not perform the "hot short to overvoltage" portion of the associated circuits common enclosure analysis. The licensee stated that NRR knew and approved (apparently informally) of their not doing this analysis and that it was too difficult for a plant of the WNP-2 vintage to perform.
>
> Please determine the necessity of performing this analysis and the safety significance of not performing it.

Specifically, Region V postulated that, with high voltage impressed on low voltage circuits, the lower voltage circuit loads would fail. This would lead to high currents in the lower voltage circuit within the fire area, as well as in other areas, outside the fire area. Because the high voltage circuit protective device may not clear the fault, the high current is postulated to produce cable insulation failures resulting in localized electrically generated fires and subsequent hot shorts along the lower voltage circuits.

The Region V's postulated event that high voltage applied to the low voltage circuit includes the low voltage equipment is improbable. Voltage which is sufficiently high, so that the circuit insulation strength (level) is exceeded, causes a short circuit. The short circuit, in turn, draws the fault current. This instantaneous fault current is sufficiently high to actuate the interrupting devices which are located in the local fire area, and upstream of the high voltage and low voltage circuits. Also, the short circuiting of cables routed in grounded steel trays or enclosures will result in short circuits to ground. The ground fault actuates protective devices to clear immediately the fault, before there is a sufficient buildup of thermal energy to cause a fire. On the basis of the above considerations, we conclude that the licensee's analysis regarding the postulated event is acceptable.

2.3 Region V also had a concern with respect to the licensee's analysis of control room and cable spreading room fires, as follows:

> Appendix R to 10 CFR 50, Section III.L.1 and NUREG-0800, Section 9.5.1, BTP CMEB 9.5.1, paragraph C.5.c, specify, that during postfire reactor shutdown, "The Reactor Coolant System process variables shall be maintained within those predicted for a loss of normal AC power."
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> During the Appendix R Audit at WNP-2, on March 3-7, 1986, the team identified, in a licensee draft FSAR fire protection program review that "The DBF (Design Basis Fire) for the main control room and cable spreading room...can result in generating transients more severe than presently analyzed in the FSAR Chapter 15 if worst case conditions are applied. These conditions are not analyzed."
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> You are requested to evaluate the licensee's analysis for the control room fire and cable spreading room fire to ensure that the analysis are adequate and that no unreviewed or unanalyzed safety questions exists.

With respect to a postulated control room fire, the reactor coolant system process variables are not assumed to be maintained within those values which are predicted for a loss of normal AC power, as specified in BTP CMEB 9.5-1, paragraph C.5.c(1). However, since the fission product boundary integrity will not be affected adversely, deviation from this guideline is acceptable (refer to Section 2.1 of this SER). The use of ADS and LPCI for a control room fire does not represent a new, unreviewed, or unanalyzed safety question, except as discussed in Section 2.1 of this SER.

The licensee maintains that alternate shutdown capability is not required with respect to the cable spreading room on the basis of cable separation and the use of fire barriers around the intervening combustible cables. Since alternate shutdown capability is not required for the cable spreading room, the above guidelines for reactor coolant system process variables do not apply. The adequacy of the fire barriers is a key factor in this determination and should be evaluated by the Region.

3.0 CONCLUSION

Since the GE analysis results are more conservative than the licensee's analysis, the staff's conclusions are based on the results of the GE analysis. The results of the analysis show that the upper portion of the core would be uncovered for a short period of time. This short term uncovery of the upper portion of the core in a BWR is an acceptable deviation from the BTP CMEB 9.5-1, paragraph C.5.c(1) guideline, which states that reactor water level should be maintained above the top of the core. The basis for acceptance stems from the determination that even with core uncovery, fuel cladding integrity is maintained. This same exception has been granted for other BWR plants that rely on the ADS/LPCI method of shutdown in the event of a fire (Reference 5).

Since short circuits caused by a high voltage cable faulting onto a low voltage cable draw sufficient current to trip the protective device, we conclude that not performing a "hot short to overvoltage" analysis is acceptable. Furthermore, on the basis of the low probability of an "Appendix R" fire, we conclude that there is minimal safety significance in not performing this analysis.

Since the use of ADS and LPCI for alternate shutdown does not lead to the loss of fuel clad integrity, we conclude that not maintaining the reactor coolant system process variables within those predicted for a normal normal loss of offsite power is acceptable.

4.0 REFERENCE

1. Letter, R. H. Buchholz (GE) to D. D. Eisenhut (NRC), "GE Cladding Hoop Stress at Perforation, "MEN 278-79, November 16, 1979.

2. "Cladding, Swelling and Rupture Models for LOCA Analysis", NUREG-0630, April, 1980.

3. Letter, Harold Bernard (NRC) to G. G. Sherwood (GE), "Supplementary Acceptance of Licensing Topical Report NEDE 20566A(P)," May 11, 1982.

4. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K, Vol. I", NEDO-20556, January, 1976.

5. NRR memorandum from L. Rubenstein to R. Mattson, dated December 3, 1982.