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Docket No. 50-313

NOV 15 1972

Mr. J. D. Phillips
Vice President & Chief Engineer
Arkansas Power & Light Company
Sixth and Pine Streets
Pine Bluff, Arkansas 71601

Dear Mr. Phillips:

On the basis of our continuing review of the Final Safety Analysis Report for the Arkansas Nuclear One - Unit No. 1, we find that we need additional information to complete our evaluation. The specific information required is listed in the enclosure.

In order to maintain our licensing review schedule we will need a completely adequate response by December 8, 1972. Please inform us within seven (7) days after receipt of this letter of your confirmation of the schedule or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our requests it is highly likely that the overall schedule for completing the licensing review for this project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the extent of extension will most likely be greater than the extent of delay in your response.

Sincerely,

Original Signed by
Irving A. Peltier for

A. Schwencer, Chief
Pressurized Water Reactors Branch No. 4
Directorate of Licensing

Enclosure:
Request for Additional Information

cc w/encl:
Horace Jewell, Esquire
House, Holms & Jewell
1550 Tower Building
Little Rock, Arkansas 72201

POOR ORIGINAL

OFFICE ▶	PWR-4	PWR-4				
SURNAME ▶	RMBernero	ASchwencer		8004210	557	
DATE ▶	11/14/72	11/15/72				

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REQUEST FOR ADDITIONAL INFORMATION
ARKANSAS NUCLEAR ONE - UNIT 1
DOCKET NO. 50-313

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.47 Your response to Request for Information 9.47 should include a discussion of the results of an evaluation of the effect on the reactor vessel supports and the reactor cavity liquid seal that could result from the postulated dropping of the heavy components indicated.

14.0 SAFETY ANALYSIS

14.11 Your analysis of the steam line break accident assumes that only one of the steam generators blows down through the break. Your analysis indicates that the cooling effect of this blowdown can cause the plant to return to power momentarily 44.5 seconds after the break occurs. In your design the main steam block valves, just outside the reactor building, are the boundary between seismic category I and seismic category II steam piping; these valves are closed by a manual switch in the control room. Thus, if the category II piping of both main steam lines fails in a major seismic event, as we would assume, then both steam generators will blow down through the breaks until the main steam block valves are closed. Manual actuation is not acceptable to assure timely closure of these valves. Therefore, you should present the results of your analysis for the simultaneous rupture of both steam lines and attendant blowdown of both steam generators, or you should explain what changes to your system you will make to prevent this occurrence.

14.12 In your analysis of recovery from a steam line break accident you indicate that at least one steam generator must be intact for use with the emergency feedwater system to remove reactor decay heat. Our evaluation indicates that delivery of this emergency feedwater may be frustrated by a number of single failures under steam line break accident conditions, for example:

a. Emergency feedwater inlet valve failure. If the steam line break occurs within the isolation boundary of a steam generator, that generator may not be usable for decay heat removal. The supply of emergency feedwater to the intact steam generator requires that the electric-motor-operated inlet valve open on signal. If the motor operator of this valve jams, preventing opening by either electric signal or manually, the supply of emergency feedwater is blocked from the intact steam generator.

b. Emergency feed pump failure. To assure an emergency feedwater supply you have provided both a turbine-driven and an electric-motor-driven emergency feed pump. If a seismic event causes failure of Category II sections of the main steam piping, these emergency feed pumps are needed for decay heat removal. Figure 1-6 of the FSAR indicates that these two pumps lie side-by-side in the same compartment of the auxiliary building. Thus, it appears that a single failure of one of these pumps can cause the failure of the other by flooding or mechanical damage.

c. Control system failure. It is not clear to us from your system description how the Integrated Control System (ICS) controls the normal and emergency feedwater valves and pumps. It appears that failure of the ICS, say in a seismic event, could leave the emergency feedwater system inoperative.

Provide the results of your analysis to show that no such single failure will prevent the supply of adequate emergency feedwater in the event of a steam system failure, or show that emergency feedwater is not needed to shut the plant down safely. In your analysis it is appropriate to assume that offsite power is lost at the time the accident occurs.