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DEC 13, 1971

Docket No. 50-313

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

Dear Mr. Phillips:

I indicated in my letter of November 1, 1971, that further requests for additional information would be transmitted to you as our review of your application for an operating license for the Arkansas Nuclear One - Unit No. 1 facility progressed. Additional information required before we can complete our review is described in the enclosure to this letter.

Please contact us if you have any questions regarding the additional information required.

Sincerely,

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Division of Reactor Licensing

Enclosure:
Additional Information Required

cc: Mr. Harlan T. Holmes
Nuclear Project Manager
Arkansas Power & Light Company
Sixth & Pine Streets
Pine Bluff, Arkansas 71601

Mr. Horace Jewell
House, Holms, & Jewell
1550 Tower Building
Little Rock, Arkansas 72201

Mr. Roy B. Snapp
1725 K Street, N. W.
Washington, D. C. 20006

OFFICE ▶	DRL:PWR-3 x7415	DRL:PWR-3	DRL:AD/PWRs			
SURNAME ▶	JMMcGough:esp	KRG KRGoller	RCDeYoung			LB
DATE ▶	12/10/71	12/13/71	12/ /71			

ADDITIONAL INFORMATION REQUIRED

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

1.0 GENERAL

1.8 Discuss the considerations given to accidents involving commercial traffic on nearby roads, railroads and navigable waterways in the design and in the preparation of operating procedures for the plant. Consider the potential for explosions, fires, and release of toxic gases due to such accidents and their potential effect on the safety of the nuclear plant.

2.0 SITE AND ENVIRONMENT

2.6 Provide maps of the plant site and surrounding areas of suitable scale, that clearly indicate the following:

2.6.1 That portion of the plant site to be established as a "restricted area", as defined in 10 CFR Part 20.3.

2.6.2 The boundary that you propose be used to establish technical specification limits for radioactive gaseous effluents.

2.6.3 Each point within the facility from which gaseous effluents containing or potentially containing radioactivity may be released and the distance of each from the nearest boundary line in 2.6.2 above.

2.6.4 The location and nature of any non-plant related activities (e.g., farming, picnic areas, camps) that will be conducted on the plant site.

2.6.5 The Low Population Zone and location of all schools, hospitals, institutions and other facilities that may require special consideration in the implementation of the emergency plan.

5.0 STRUCTURES

5.61 State your criteria for selection of protective coatings and paints for use within the containment to withstand accident conditions, including consideration of radiation, boric acid spray washdown, steam environment, and jet impingement effects. Include an evaluation of the potential impairment of the performance capabilities of engineered safety features due to flow blockage, fouling of heat transfer surfaces, or other events that might result from failure of the protective coatings and paints.

5.62 Identify any aspects of your program for testing reinforcing bars used in Class I concrete structures that do not conform to Safety Guide 15 and indicate your bases for believing that these are acceptable.

5.63 Identify any aspects of your containment structural acceptance test that will not conform to Safety Guide 18 and indicate your bases for believing that these are acceptable.

5.64 Provide the missing information in Table 5.1 pertaining to containment penetrations 25 and 45.

5.65 Does the isolation valving for the containment penetrations satisfy General Design Criteria Nos. 54, 55, 56 and 57? For each penetration for which the valving does not satisfy these criteria, indicate your bases for believing that the design is acceptable.

7.0 INSTRUMENTATION AND CONTROL

7.21 We note that all the nuclear power range channels required for reactor protection are also used for control functions. Discuss how this design meets the requirements of Section 4.7 of IEEE-279 (1971) with particular emphasis on (1) results of tests and analyses that verify the isolation devices can withstand all credible faults without preventing the protection system from meeting its minimum performance requirements, (2) measures proposed to verify that degradation of the isolation devices during the plant operating life will not violate the requirements of IEEE-279 and (3) the methods used to isolate the averaging networks from the protection system.

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.4 Identify any aspects of the fuel storage facility that do not conform to the provisions of Safety Guide 13 and provide your basis for believing that these are acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEMS

10.1 The turbine stop valves serve as containment isolation valves and for isolation of the unaffected steam generator in the event of a steamline rupture accident. Provide information on the design, operation, inspection and testing of these valves relative to these functions. Indicate whether the control system that closes these valves meets IEEE-279 criteria.

10.2 Provide information describing the locations of both emergency feedwater pumps, the location of the atmospheric exhaust associated with the steam turbine driven pump and a description of how the electrically driven pump can be manually connected to the diesel generator busses. Estimate the time associated with such a transfer of power sources.

11.0 RADIOACTIVE WASTE AND RADIATION PROTECTION

11.6 The Commission, on June 9, 1971, published for comment a proposed Appendix I "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Practicable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents" to 10 CFR Part 50.

Review your waste-processing system design in light of the new Appendix I and evaluate whether each of the radioactive waste processes meet the "As Low As Practicable" criterion. Identify each effluent stream from the plant and justify why further processing is not practicable. Calculate the total radiological impact of operation of the ARK-1 plant on the environs in view of the proposed numerical guides stated in Appendix I. Consider (1) operation with expected levels of radioactivity in the primary and secondary systems, and (2) operation with technical specification limits of radioactivity and primary to secondary system leakage.

12.0 CONDUCT OF OPERATIONS

12.10 Provide information to demonstrate compliance with the provisions contained in Safety Guide 17, Protection Against Industrial Sabotage.

14.0 SAFETY ANALYSIS

14.1 Discuss the probable consequences associated with the following types of fuel loading errors. Indicate by what means and with what degree of confidence these loading errors could be detected after the fuel assembly was placed in the reactor.

14.1.1 Erroneous enrichment loading of fuel pellets or fuel pins during the fabrication process.

14.1.2 Erroneous location or orientation of fuel assemblies during initial core loading or during subsequent refueling operations.

14.2 Relative to the steam line failure accident analysis:

14.2.1 Justify the use of a moderator temperature coefficient of -3.0×10^{-4} $\Delta k/k/^{o}F$ for this analysis.

14.2.2 Provide the results of an analysis indicating the effects on the primary system during this accident. Include the time-history traces for all significant parameters such as DNBR, pressure, temperature, pressurizer level, and reactivity. Indicate the time sequences for significant events such as reactor trip, turbine valve closure, feedwater valve closure, steam dump and bypass valve operation, relief valve operation, and high pressure injection actuation.

14.2.3 We note that you have analyzed this accident assuming certain operator actions in one case and no operator action in another case. The results obtained appear to be identical. Provide a clarification of what actions the operator is expected to perform during this accident and discuss whether these actions can significantly effect the potential consequences of the accident.

14.3 Submit a summary of a detailed analysis to show that the plant can be shutdown safely in the event of a loss of all AC power (offsite and onsite), including:

14.3.1 A list of all necessary components, control systems, instrumentation and other necessary electrical systems that must be supplied from the station batteries and discuss how power will be provided from the batteries. Estimate the power consumption of each component and indicate the maximum time the plant could maintain this condition safely.

14.3.2 Section 14.1.2.8.4 of the FSAR indicates that in the event the condensate storage tank is unavailable as a source of emergency feedwater, an alternate supply is available from the service water system. Provide an analysis showing that NPSH requirements for both the steam driven and motor driven emergency feed pumps can be met in the event suction must be taken through the idle service water pumps.

14.4 Relative to your analysis of the potential consequences of a loss of load incident:

14.4.1 Justify the use of 10 CFR Part 100 limits as acceptable criteria for radiological doses associated with this event.

14.4.2 Describe the testing that will be performed during the startup test program to verify the assumptions and results of your analyses of the loss of load incident.

15.0 TECHNICAL SPECIFICATIONS

15.1 The proposed technical specifications submitted in Amendment 20 are not complete. Submit a complete set of proposed technical specifications, preferably based on the set recently developed for Unit 1 of the Oconee Station and identify any specifications that are different and the bases for these differences. Note that the reporting requirements of your proposed technical specifications should be based on Safety Guide 16, Reporting of Operating Information.