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Docket No. 50-313

APR 3 1968

Arkansas Power and Light Company
Pine Bluff, Arkansas, 71601

Attention: Mr. J. D. Phillips
Vice President and Chief Engineer

Gentlemen:

We have completed our review of the part of your application for a construction permit for the Russellville Nuclear Unit related to the reactor, engineered safety features, instrumentation, and safety analysis. The material that you have submitted does not meet our requirements for the contents of applications, as specified in 10 CFR Part 50 and elsewhere. We will not be able to continue our review of these matters until your application is complete in this regard.

Specifically, the proposed Part 50 requires coverage as fully as available information permits on the preliminary design of the facility, including the principal design criteria, the design bases and the relation of the design bases to the principal design criteria and information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

Our Guide for the Organization and Contents of Safety Analysis Reports states that information submitted should show how these principal design criteria are met by:

(1) Identifying the design bases and explaining the reasons therefor.

(2) Describing the reactor to show how the design bases have been satisfied.

(3) Showing through evaluations that design bases have been met with a reasonable margin for contingencies.

(4) Providing a basis for such limits upon operation that might be appropriate in the interest of safety.

Other guidance is given in this document with respect to tests, inspection and surveillance.

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Your information on research and development programs is incomplete; in particular the time schedule for completing the programs is not provided.

The site and environment section does not sufficiently describe and evaluate design compatibility with such conditions as dam failure, site flooding and failure of the natural gas pipe line. The atmospheric diffusion model used to evaluate the off-site consequences of postulated accidents has not been justified in sufficient detail to warrant its use.

The safety analysis section is considered incomplete in the areas of loss-of-flow, small size breaks, thermal shock and steam generator failures. Chemical changes in emergency core coolant, which may affect accident severity, have not been evaluated.

Your description of spatially dependent kinetics is incomplete. The detection system for xenon oscillations is not given in sufficient detail, nor is the proposed method for stabilizing and/or controlling potential oscillations adequately described.

The instrumentation and control section does not adequately describe the proposed systems. In particular, diversification of engineered safety feature actuation signals and separation of control and safety are not adequately treated in the report.

In the electrical systems section, the engineered safety features loads are not adequately related to diesel generator capacity, the description of the emergency condition off-site power connections to the engineered safeguard busses is incomplete, and justification for and restraints on engineered safeguard bus ties is not provided.

In the initial tests and operations section you have not indicated how test procedures and test results will be documented and evaluated. There are no criteria stated for objective acceptance to be based on meeting definitive requirements established prior to test performance.

In the area of quality assurance and quality control, you have not described the ability of the design to meet the tentative supplementary criteria for nuclear pressure vessels issued last August. Further, relationships between your company and the various concerns contributing to the construction of this unit have not been sufficiently described.

The section on conduct of operations is deficient. It does not cover emergency plans, training schedules, or provision for nondestructive examination of the reactor pressure vessel.

The information we need is discussed in some detail in the attached "Request for Information." We urge that you provide full and complete answers to the enclosed request in order to minimize interruptions in the processing of your application.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Information

cc: Messrs Harlan T. Holmes
Horace Jewell
Roy B. Snapp

SEE ATTACHED FOR PREVIOUS CONCURRENCES

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ASchwencer:pt	CGLong	SLevine	RSBoyd	PAMorris
				4-3-68

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Peter A. Morris, Director
Division of Reactor Licensing

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Request for Information

cc: Messrs. Harlan T. Holmes
Horace Jewell
Roy B. Snapp

POOR ORIGINAL

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REQUEST FOR INFORMATION

ARKANSAS POWER & LIGHT COMPANY
DOCKET 50-313

1. GENERAL

1.1 Identify those items that will eventually be classified as technical specifications that now affect plant design. For a typical list of technical specifications, refer to the San Onofre Technical Specifications (Docket No. 50-206).

1.2 Describe how your design complies with General Design Criterion No. 11. In particular, describe those design features which will allow the reactor to be brought to a hot standby condition upon loss of the control room. Analyze and evaluate what additional design provisions are required to provide the capability of bringing the reactor to cold shutdown following loss of the control room during an extended reactor shutdown.

1.3 Update the description of your research and development program status. Provide a schedule for all items, including those incorporated by reference from other applications. Identify all Russellville design features dependent on these r & d programs. Compare construction schedule dates for these design features with the applicable r & d program schedule dates to show compatibility. Discuss design alternatives in the event that r & d schedule dates are not met.

1.4 Discuss the principal design decisions yet to be made that require nuclear and steam plant knowledge and which affect nuclear power plant safety. Indicate the approximate dates by which these decisions must be made and to what extent reliance will be placed upon contractors for making decisions. Indicate how the training plans for personnel are orientated toward these requirements.

1.5 By letter dated January 17, 1968, the ACRS reported, with recommendations, on the review of the nearly identical Three Mile Island plant (Docket No. 50-289). Discuss applicability to Russellville and indicate what if any design changes will be made as a result of that letter.

2. SITE AND ENVIRONMENT

2.1 The atmospheric diffusion model developed in Appendix 2A is based on the use of a building wake dilution factor equal to the cross-sectional area of all power plant buildings. Validity of the diffusion model used has not been sufficiently established to warrant use of a dilution factor that large, particularly at distances of the order of 1000 meters.

Since the off-site consequences of your postulated accidents are strongly dependent upon the atmospheric diffusion model used, that model must be based on reasonably conservative considerations. The model developed in Appendix 2A and its effect on the dose valves given in Section 14, should be re-evaluated in this regard or supplemented with more detailed information to establish it as a reasonably valid and conservative model.

2.2 To evaluate the consequences of accidents in which postulated release of activity occurs from fuel rod gaps; gas decay tanks, and liquid waste tanks, we shall need tables of isotopes and maximum quantities of each which are contained in these locations. The method used to determine these activities should be included. If this information has been provided in prior applications, it may be incorporated in this application by reference provided it is updated to the extent necessary to make it directly applicable.

2.3 It is understood that a new low population zone (LPZ) had been defined. Revise those portions of the PSAR discussing accident consequences at the LPZ to show the changes brought about by this redefinition. Show that the new LPZ is consistent with a feasible evacuation plan.

2.4 An analysis should be presented which relates primary coolant activity, assumed leakage rate from the primary to secondary system, removal and cleanup mechanisms for the secondary coolant, and the derived activity contained in the secondary system.

2.5 The PSAR descriptions of the steam generator tube and fuel handling accidents assume iodine water-to-air partition factors of 10,000 and 100 respectively. References 1 through 4, listed on Page 14-57 of the PSAR, are cited as supporting data for those assumptions. Supplement your description of those accidents by a more detailed analysis which relates conditions existing throughout the course of the accident to the specific supporting data contained in those references. In particular compare the pH, temperatures, pressure, iodine concentrations and air/water volume ratios in the references with those in this situation. Demonstrate that the effects of any differences do not invalidate use of the assumed partition factors.

2.6 Supplement the steam generator tube failure analysis by showing the effects of concurrent loss of off-site power. Also, indicate the maximum number of tube failures that can be sustained and remain within the 10 CFR 100 guidelines under the above conditions.

2.7 Analyze the consequences of failure of the Dardenelle dam and lock structure for ability to supply shutdown cooling water needs. Evaluate the need for and seismic classification of a secondary source of cooling water.

2.8 Provide the preliminary design analysis to show the ability of Class I structures, piping, and components to safely withstand the maximum probable flood and perform their intended functions throughout the flood's duration. Describe the means which will be available to allow transport of personnel and heavy equipment to the facility during this period.

2.9 Describe your criteria and programs for off-site environmental monitoring and your plans, pre- and post-operational, for survey of marine ecology above and below the plant site.

2.10 Analyze and evaluate the consequences of failures outside containment in the ECCS recirculation system. The analysis should allow for radioactive halogens being spray-washed to containment sump.

2.11 Describe the natural gas transmission pipeline shown crossing the intake structure. Provide information on physical size, date of installation, usage and corrosion protection and testing. Evaluate the likelihood and consequences of failure of this pipeline as related to safe plant operation and emergency shutdown conditions. If rerouting is necessary perform a similar analysis for the new location showing plant safety will not be impaired.

3. REACTOR

3.1 Submit additional available results of those xenon stability analyses discussed in the PSAR and specify the dates when the remaining analyses will be completed.

3.2 Assuming that control rods are used to stabilize xenon oscillations, give the maximum values anticipated for the transient and steady-state errors in local power density at the hot spots.

4. INITIAL TESTS AND OPERATIONS

4.1 Discuss the extent to which test results will be documented.

4.2 Discuss your plans for measuring and/or verifying the threshold conditions for xenon oscillations. Include in your discussion the extent to which data from earlier plants will be used.

4.3 Provide a detailed outline of the test program for each engineered safety system. The outline should provide a set of test objectives for each system, a brief description of the proposed test, and a brief discussion on how achievement of design objectives can be assured.

5. ENGINEERED SAFETY FEATURES

5.1 Identify the engineered safety feature instrumentation and electrical equipment which must function in an accident environment. Discuss your intentions with respect to qualification testing to insure that these items will function in the combined accident environment of temperature, pressure and humidity. Discuss allowances made for reduced performance towards end-of-life.

5.2 Provide an analysis of the composition and pH of the emergency core cooling solution as a function of time following the design basis loss-of-coolant accident. Consider spray system additives, soluble neutron poisons, fission and corrosion products, elements leached from concrete, etc.

5.3 Provide a discussion of the extent to which exposure to the solution discussed in item 5.2 above will be factored into the procedure for selection of materials for the engineered safety features for the facility. Discuss the systems that will be affected and the nature of the considerations that will be taken into account.

5.4 Discuss the time, temperature, and radiation dependent stability of the spray solution under both storage and post-accident recirculating conditions and indicate the possibility of forming solid decomposition products or precipitates which could potentially interfere with system performance.

5.5 Discuss both the time-dependent radiolytic and chemical hydrogen formation under post-accident conditions for the solution given in item 5.2 above. Include an estimate of total γ and β activity in both the core and in the liquid, and of the total expected irradiation dose characteristics. Indicate the extent of hydrogen formation by chemical reaction (corrosion) with exposed reactor materials.

5.6 Expand your spectrum-of-breaks analysis, for response of ECCS to loss-of-coolant accidents, for break sizes less than 0.4 feet^2. Include a core heat transfer analysis for these small breaks (down to 3.25 inch diameter) to show extent of clad damage (if any). Include plots of peak clad temperature vs. time after such breaks and describe in detail the core heating and cooling analysis.

6. INSTRUMENTATION AND CONTROL

6.1 Describe and discuss the differences between Babcock and Wilcox designed Russellville Station and Three Mile Island Station (Docket No. 50-289) (1) reactor protection systems, (2) engineered safety feature action, and (3) control systems. For (1) and (2) above, consider the complete circuit from sensors to actuation logic. For (3) above, evaluate the safety significance of each system.

6.2 For reactor protection and engineered safety feature actuation circuits to be designed by other than Babcock and Wilcox, identify and justify any design features which conflict with the proposed IEEE Standard for Nuclear Power Plant Protection Systems.

6.3 Describe and evaluate your criteria for the physical identification of the reactor protection and engineered safety feature equipment including panels, components, and cables.

6.4 Describe and evaluate changes which will be made in the design of the instrumentation and control systems as a result of the ACRS recommendations contained in the Three Mile Island letter. Include in the discussion:

a. Diversity

b. Separation of control and protection systems

6.5 Describe and evaluate the design criterion to be used to assure circuit isolation where reactor protection and engineered safety feature signals feed annunciators and/or a data logging computer.

6.6 Discuss the instrumentation systems for detecting and controlling xenon oscillations. Indicate expected minimum sensitivity of the detection system during power operation. Indicate how the control is related to reactor protection. If the existing in-core monitoring system is required to provide reactor protection capability, indicate whether revisions would be necessary to meet the proposed IEEE Standard for Reactor Protection Systems.

6.7 Fully describe all loss of flow analyses which have been performed to date. Based on this description, evaluate the feasibility of deriving flow protection inputs from signals which are direct measures of reactor coolant flow. (Refer to paragraph 4.8 of the proposed IEEE Standard for Reactor Protection Systems.) If it is concluded that special cases exist where adequate protection cannot be provided by direct flow measurement, the technical basis for such conclusion(s) should be provided.

6.8 The PSAR shows all control rod clutches on a single 125 volt d.c. scram bus. This does not satisfy the para 4.2 criterion of the IEEE Proposed Standard. Re-evaluate this design feature. If retained, it must be clearly shown by a rigorous failure analysis that a single malfunction cannot prevent a reactor scram.

6.9 Provide further details on your radiation monitoring system. For guidance as to appropriate details refer, for example, to the Metropolitan Edison PSAR (Docket 50-289), Supplement 1, Questions 10.1 through 10.8.

6.10 Describe the control room ventilation system and evaluate the need for automatically placing the system in a recirculation mode based on airborne activity level in the intake duct.

7. EMERGENCY ELECTRICAL POWER

7.1 Evaluate the ability to provide off-site power for engineered safety features with any single failure in the system. Included should be the effect of tripping of the main unit generator, a substation failure (including a control failure) concurrent with tripping of the main unit generator, failure of the startup transformer or its circuit breaker and failure of the bus tie autotransformer.

7.2 Identify all loads (HP) requiring power in the interest of safety. Evaluate the relationship between the maximum emergency load that can be placed on each diesel generator and the ability (KW rating) of the generator to accept that added load.

7.3 Describe and evaluate the provisions to prevent the two emergency diesel generators from being tied together and from being asynchronously connected to another energized bus.

7.4 Describe and evaluate the diesel fuel supply system including location, arrangement and classification of fuel tanks and fuel delivery components (piping, pumps, etc.) and the consequences of a single system failure. Identify any shared components and indicate how long the diesel generators can supply the maximum engineered safety features from this fuel supply system, without resupply, with a single system failure.

7.5 Describe and evaluate how the design ensures that batteries, reactor protection, and safeguard instrumentation systems and equipments will function properly to ensure safe shutdown under seismic loading. Discuss maximum amplification of seismic shocks as they are transmitted through the plant structure to the floors on which these items are mounted.

8. QUALITY ASSURANCE

8.1 Submit Certified Code Design Specifications for component parts of the Class I systems as required by the ASME Code Section III, paragraph N-141 (passed 6-23-67).

8.2 Provide a tabulation of every nuclear pressure vessel in the Class I (seismic design) systems in the facility. The tabulation should indicate for each vessel whether design is complete, its stage of fabrication and the extent to which it will comply with each of the 34 supplementary criteria in "Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Nuclear Pressure Vessels," issued by AEC Press Release No. IN-817, dated August 25, 1967. For each criterion not met in its entirety provide a discussion that represents the reason why total compliance is not feasible for that vessel.

8.3 Indicate AP&L's degree of participation in quality assurance (Q.A.) planning, for instance, does AP&L approve the various specifications and procedures?

8.4 Indicate AP&L's degree of participation in quality control (Q.C.) efforts by the A&E and in inspection of component fabricators' shop efforts, including B&W supplied components.

8.5 Provide a functional organization chart for Bechtel Corporation detailed to show Russelville project responsibility channels for Q.A. (design) and Q.C. (inspection) efforts, including safety related electrical, instrumentation and control systems.

8.6 Indicate planned availability of complete, up-to-date facility and system drawings and associated records and information regarding arrangements, systems diagrams, major structural plans and technical manuals of all major installed equipment. Also, indicate planned availability of all quality assurance, quality control, specification, test, inspection and examination documents related to the fabrication and erection of essential components of the plant including ASME coded pressure vessels.

8.7 Indicate whether Electro-Slag welding will be used for Class I pressure vessels and components.

8.8 Clarify description on ultrasonics examination to detect laminations "normally parallel" to the plate surface by shear wave technique, item b on page 4-17.

8.9 Extend the Q.A. planning to show responsibilities for shipping and erection phases. Explain measures to be taken to assure that components are not degraded beyond established limits during these phases.

9 CONDUCT OF OPERATIONS

9.1 Summarize emergency procedures planned for the first hour after a major loss-of-coolant accident (LOCA).

9.2 Provide a summary of emergency plans, to include:

(a) shift responsibilities
(b) alarm systems
(c) communication systems
(d) environmental monitoring equipment (portable)
(e) notification of and liaison with authorities
(f) medical facilities
(g) critical actions to be performed prior to evacuation
(h) initial assessment of damage plans
(i) evacuation plans

9.3 Provide an estimate of accumulated radiation to the operating staff during and after a major LOCA. Include radiation while in the control room, ingress and egress, and possible missions to the turbine building, auxiliary building, and borated water storage tank.

9.4 Describe the design provisions incorporated to permit nondestructive examination of the installed reactor pressure vessel during its lifetime. Indicate your willingness to demonstrate this ability by performance of an accessibility test, including removal and reinstallation of reactor vessel internals if necessary, prior to plant operation.

9.5 Referring to PSAR Section 12.5, discuss the scope, function and composition of a plant safety review committee and a general office audit committee as well as the method for the review, approval and authorization for procedural and facility changes.

9.6 Incorporate the training program schedule.

9.7 Provide and discuss an additional engineering organization chart showing the various relationships among and between: Arkansas Power & Light; Middle South Services, Incorporated; Babcock & Wilcox; Bechtel Corporation; and consultants (if appropriate). This chart and the accompanying discussion should clearly present the lines of authority and responsiblity for nuclear component fabrication and installation and facility construction regarding inspection, surveillance, quality control, testing and acceptance. Authority to "stop or resume" an operation when irregularities are revealed should be clearly stated.

10. <u>REACTOR COOLANT SYSTEM AND OTHER CLASS I SYSTEMS</u>

10.1 Discuss the ability to promptly detect fuel failures. Include detection time as a function of fuel failure severity.

10.2 Provide further loss of flow analysis. Using the 1.38 DNBR criterion cited on page 3-56 of the PSAR, show the maximum abrupt reduction in reactor coolant flow cross-sectional area that can be tolerated at three locations

(a) at the inlet tube sheet of a steam generator
(b) at the exit of a reactor inlet nozzle
(c) on the undersurface of the reactor inlet flow baffle.

10.3 Thermal Shock - With regard to thermal shock on reactor components, induced by operation of the emergency core cooling system (ECCS), provide details of an analysis which indicates that the reactor vessel and reactor internals can withstand the rapid temperature change at the end of their design life. The analysis should include both the ductile yielding and the brittle fracture modes of failure. Data in prior applications may be incorporated by reference provided it is made current and directly applicable to this plant.

10.3.1 The ductile yielding analysis for the vessel should state the geometry of the plate and the assumed cooling method. It should include also the following information:

(a) The heat transfer coefficient used, its experimental basis, and the degree of conservatism involved,

(b) The temperature profiles and the calculated thermal stress profiles through the thickness of the plate for several times during the cold water injection transient, and

(c) The magnitude of stresses due to dead load, pressure load and a potential concurrent seismic loading.

10.3.2 The brittle fracture analysis for the vessel should assume an initial crack size just below the critical crack size corresponding to the stresses present during normal operation and transients. Since the initial crack is most likely to exist in a weld or a heat affected zone, the analysis should consider two cases: a circumferential crack, and a crack parallel to the axis of the reactor vessel. The analysis should provide information on:

(a) The critical stress intensity factor (K_{IC}) assumed, and the basis for its selection,

(b) The assumed time-integrated neutron flux (nvt) at the reactor vessel inner diameter,

(c) The value of residual stresses assumed in the base metal and the weld areas,

(d) The initial crack geometry and size assumed in the analysis, and

(e) Equations used to correlate crack size with the calculated stress intensity factor (K_I).

10.3.3 Based on the analyses for the vessel provide an estimate of the maximum neutron flux exposure (nvt) of the vessel, and the maximum allowable pressure stress, which could be tolerated without failure.

10.3.4 Evaluate the capability of the piping, safety injection nozzles, and vessel nozzles to withstand the transient.

10.3.5 Evaluate the effects of this transient on the core barrel and other internals with regard to assuring that distortion would not restrict the flow path of the emergency core coolant.

10.4 Seismic Design

10.4.1 For all Class I systems and components provide the design basis load combinations and the proposed stress, strain, and deformation limits for each combination.

10.4.2 Supply criteria or specific information on the interaction forces, deformation and stresses connected with the relative motions between the reactor vessel, steam generators or other large components. Indicate how these relative motions will be controlled by snubbers or other means, and what reaction forces (and corresponding stresses) will be transmitted to the pipes.

10.4.3 Identify specific reactor internals which must maintain their functional performance capabilities to assure safe shutdown of the reactor. Provide calculated (or estimated) maximum limits of deformation or stress, at which inability to function occurs, for each component identified. Also, supply the calculated (or estimated) maximum design limit value, and the expected deformation or stress. In all cases identify the applicable loading combination and state the proposed margin of safety relative to both stress and strain.

10.4.4 For reactor internals provide information that will permit evaluation of the effect of irradiation on the material properties and on the proposed deformation limits.