VHWilson, DR⁻ (2)
OGC
ACRS (16)
CO (2)

DISTRIBUTION:
Docket
DRL Reading
PWR-3 Reading
CKBeck, DR
MMMann, DR
SHanauer, DR
FSchroeder, DRL
RSBoyd, DRL
RCDeYoung, DRL
DJSkovholt, DRL
HRDenton, DRL
TRWilson, DRL
EGCase, DRS
RRMaccary, DRS
RWKlecker, DRL
DRL/DRS Branch Chiefs

Docket No. 50-313

NOV 1 1971

Mr. J. D. Phillips
Vice President and Chief Engineer
Arkansas Power and 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.

We recognize that some of the information requested may be available in the public record in the context of our regulatory review of similar features of other facilities, especially the Arkansas Nuclear One - Unit No. 2 application. If such is the case, you may wish to incorporate the information by reference in your application.

The additional information requested has been categorized into groups which correspond directly to sections in your application. You may wish to amend your application by submitting revised pages for the appropriate portions of the Final Safety Analysis Report rather than by submitting separate responses to the questions. If so, please provide cross references to those pages.

As our review continues, we will be requesting additional information, particularly that pertaining to Sections 2 and 14 of the FSAR and to the Technical Specifications.

Please contact us if you desire any discussion or clarification of the material requested.

Sincerely,

Original Signed by
R. C. DeYoung

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

Enclosure:
Request for Additional Information

cc: See attached

OFFICE ▶	DRL:PWR-3	DRL:PWR-3	DRL:AD/PWRs		
SURNAME ▶	x7415 JMMc JMMcGough:esp	KRG KRGoller	RCD RCDeYoung		
DATE ▶	10/28/71	10/28/71	10/29/71		

LB

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE : 1969—O-364-598

8004180604 A

cc: Mr. Harla. 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 ▶						
SURNAME ▶						
DATE ▶						

REQUEST FOR ADDITIONAL INFORMATION

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

1.0 GENERAL

1.1 The practice of permitting small diameter piping for safety related systems to be "field run" should be limited insofar as it is practical to do so. When it is permitted (1) stringent quality assurance measures should be taken to assure that the installation has been performed in such a manner that the assumptions made for design and safety assessment purposes remain valid, and (2) tests should be performed on the completed item to provide a final indication of acceptability. In view of these requirements, provide the following information:

1.1.1 A discussion of the extent to which you permitted "field running" of piping for safety related systems, especially for engineered safety features.

1.1.2 A description of the [illegible]cial quality assurance measures and performance tests that were conducted to assure satisfactory installation.

1.1.3 Provisions for incorporating "field run" piping location and construction details on "as built" drawings.

1.2 Describe how the applicable requirements of Appendix B to 10 CFR Part 50, Quality Assurance Criteria for Nuclear Power Plants, will be implemented during the operational life of the plant.

1.3 Describe your plans with respect to assuring that design and quality assurance records (e.g., as-built drawings, equipment certifications, material test reports, deviation notices, inspection reports) will be available to your engineering and operating personnel throughout the life of the plant. Provide a list of the records that will be maintained at the site and elsewhere upon completion of construction activities, describe the method of storage of these records, and identify the individual having direct responsibility for management of these records.

1.4 Recent experience indicates that the bodies of valves and other cast components important to nuclear safety may have areas where the wall thickness is less than required. Describe the quality control procedures that you are using to verify wall measurements to demonstrate that such components meet design requirements. If such verification cannot be demonstrated, provide an engineering justification to support the acceptance of the components.

1.5 Provide a summary comparison of the Arkansas Nuclear One - Unit 1 (ARK-1) facility with Appendix A, General Design Criteria for Nuclear Power Plants published February 20, 1971, indicating any areas where the proposed design does not fully conform with these criteria.

1.6 Identify those design features parameters of ARK-1 nuclear steam supply system (NSSS) that are significantly different from those of the NSSS designed by the Babcok & Wilcox Company for the Oconee Nuclear Units. For each item, summarize the safety significance including any considerations of significant changes in safety margins. Include the technical bases that justify each change with regard to nuclear safety.

1.7 The codes and standards rule (10 CFR Part 50.55a) does not require the use of an N-symbol but does require the use of the ASME survey and inspection system, or of an AEC-approved alternate system that provides an acceptable level of quality. The rule also permits the ASME survey and inspection system to be used by an applicant or licensee in partial fulfillment of the requirements specified in 10 CFR Part 50, Appendix B (Quality Assurance Criteria for Nuclear Power Plants), provided that such use is described by the applicant or licensee, as required by 10 CFR Part 50.34 (a) (7) and is determined to be acceptable by the regulatory staff. If you have relied or intend to rely on the ASME survey and inspection system to satisfy to any extent the requirements of 10 CFR Part 50, Appendix B, provide the following information:

1.7.1 Identify those quality assurance criteria, or portions of criteria, that have been or will be complied with through the use of the ASME survey and inspection system for each applicable safety-related plant system or component.

1.7.2 Indicate how you or your designee has audited or will audit the performance of the system to determine that it has effectively accomplished or will effectively accomplish its specified objectives.

2.0 SITE AND ENVIRONMENT

2.1 Our evaluation of the seismic instrumentation system you have proposed, indicates that additional equipment and measures may be necessary to assure the integrity and performance of structures, systems, and components important to safety in the event of seismic disturbances. A seismic instrumentation system and data utilization program that we consider acceptable is set forth in Safety Guide 12, "Instrumentation for Earthquakes", published March 19, 1971. Provide the basis and justification for elements of your proposed program which differ substantially from Safety Guide 12.

2.2 In the site foundation evaluation presented in the PSAR and FSAR it was stated that all Class I (seismic) structures would be founded on shale material and therefore no foundation problems were anticipated. Provide information to demonstrate that the foundation material for these structures was found to be as anticipated during construction.

2.3 Provide a summary evaluation that demonstrates the stability of the intake and discharge canal slopes during all design basis events.

2.4 Describe how the facility was designed to protect vital equipment and assure capability for safe shutdown in the event of a design level flood (elevation 361 feet).

2.5 Provide a description of the assistance that the Arkansas Board of Health will provide in implementing your environmental monitoring program.

3.0 REACTOR

3.1 Topical reports BAW-10008, Part 1-Rev. 1 and Part 2-Rev. 1, are referenced in the FSAR as applicable to the design of the reactor internals and fuel assemblies, respectively, for blowdown and seismic loads. However, the seismic analysis presented in Part 2 of BAW-10008, utilizes the seismic ground accelerations specified in the Oconee application. Provide analyses for the fuel assemblies which consider the higher ground acceleration values used at the Arkansas site.

3.2 Evaluate and identify any single failure which could result in the continuous withdrawal of a single control rod. Estimate each failure probability. For those failure modes identified, present an evaluation of the potential consequences associated with such a rod withdrawal during any anticipated mode of operation. Indicate the initial position of the control bank and the spatial position of each rod considered.

4.0 REACTOR COOLANT SYSTEM

4.1 The list of transients that have been used in the design of components within the reactor coolant pressure boundary as specified in Table 4-8 of the FSAR appears to be incomplete. Identify all design transients and their number of cycles, such as control system or other system malfunction, component malfunctions, transients resulting from any single operator error, inservice hydrostatic tests, etc., which are specified in the ASME Code-required "Design Specifications" for the components of the reactor coolant pressure boundary. Categorize all transients or combinations of transients with respect to design Cases I through IV of paragraph 4.1.2.5.1 (comparable to the operating condition categories identified as "normal", "upset", "emergency", and "faulted" as defined in the Summer 1968 Addenda of the ASME Section III Nuclear Vessel Code). Describe the program that will be used to record and maintain an accounting of significant transients occurring during plant operation and to compare the service number of transients accumulated with those specified as the permissible number of transients for which the plant is designed.

4.2 Specify the Code Case Interpretations (Special Rulings) that have been used in implementing the component codes delineated in Appendix A of the FSAR.

4.3 Paragraph I-701.5.4 of the ANSI B31.7 Nuclear Power Piping Code requires that piping shall be supported to minimize vibration and that the designer shall make appropriate observations under startup or initial operating conditions to assure that vibration is within acceptable levels. Describe the preoperational vibration test program that will be used to verify that the piping and piping restraints within the reactor coolant pressure boundary have been designed to withstand dynamic effects resulting from valve closures, pump trips and other anticipated events. Provide a list of the transient conditions and the associated anticipated events (i.e., pump trips, valve actuations) that will be used in the vibration operational test program to verify the integrity of the system. Include those transients introduced in systems other than those within the reactor coolant pressure boundary that could result in significant vibration response of reactor coolant pressure boundary systems and components.

4.4 Specify whether the design criteria that have been used to examine the effects of pipe rupture have considered postulated pipe breaks to occur at any location within the reactor coolant pressure boundary, or only at limited areas within the system. Provide confirmation that both longitudinal and circumferential type ruptures were evaluated and describe the basis for your design approach.

4.5 Provide a more detailed description of the measures that have been used to assure that the containment liner and all essential equipment within the containment, including components of the primary and secondary coolant systems, engineered safety features, and equipment supports, have been adequately protected against blowdown, jet forces, and pipe whip. The description should include:

4.5.1 Pipe restraint design requirements to prevent plastic hinge formation.

4.5.2 The features provided to shield vital equipment from pipe whip.

4.5.3 The measures taken to physically separate piping and other components of redundant engineered safety features.

4.5.4 A description of the analyses performed to determine that the failure of lines, with diameters of 3/4 inch or less, will not cause failure of the containment liner under the most adverse design accident conditions.

4.5.6 The analytical methods that were used.

4.6 Provide a more detailed discussion of the criteria that have been applied in designing the principal reactor coolant system component supports (i.e., supports, restraints, "snubbers", guides, as applied to vessels, piping, pumps, and valves) including the stress or deformation limits and the design codes or standards applicable to each type of support.

4.7 Reported service experience of PWR steam generators has demonstrated that flow induced vibration and cavitation effects can cause tube thinning, and corrosion and erosion mechanisms both from the primary and secondary side may contribute to further structural degradation of the tube integrity during the service lifetime. The failure of a group of weakened tubes as a consequence of a design basis pipe break in the reactor coolant pressure boundary could impair the capability of emergency core cooling systems to perform their intended function. In order to evaluate the adequacy of design bases used to prevent such conditions from developing in the steam generator during service, the following additional information is required:

4.7.1 State the design conditions and transients that were specified in the design of the steam generator tubes, and the applicable design stress intensity limits associated with Cases I through IV of paragraph 4.1.2.5.1 (comparable to the "normal", "upset", "emergency", and "faulted" operating condition categories). Justify the basis for this selection.

4.7.2 Specify the margin of tube-wall thinning that could be tolerated without exceeding the allowable stress limits identified in (4.7.1) above, under the postulated condition of a design basis largest pipe break in the reactor coolant pressure boundary during reactor operation.

4.7.3 Describe the inservice inspection that will be employed to examine the integrity of steam generator tubes as a means to detect tube-wall thinning beyond acceptable limits and whether excess material will intentionally be provided in the tube wall thickness to accommodate the estimated degradation of tubes during the service lifetime.

4.8 Submit a copy of the summary technical report on overpressure protection that has been prepared in accordance with the requirements of paragraph N910.2 of the ASME Section III Nuclear Vessel Code.

4.9 Note 5 to section A.3, Appendix "A" of the FSAR, indicates that different design and nondestructive examination requirements were specified for valves within the reactor coolant pressure boundary dependent on their purchase order date (i.e., before or after January 1, 1970). For those valves ordered prior to January 1, 1970, provide the design criteria and identify the applicable design code employed. The ASME Code for Pumps and Valves for Nuclear Power (NP & V Code) is specified for valves purchased after January 1, 1970. However, this Code allows the option of selecting any of the following design procedures for Class I valves:

4.9.1 Paragraph 452.1a of the NP&V Code, Standard Pressure Rated Valve

4.9.2 Paragraph 452.1b of the NP&V Code, Non-Standard Pressure Rated Valve

4.9.3 MSS SP-66, Pressure-Temperature Ratings for Steel Butt-Welding End Valves, 1964 Edition (as referenced in the March 1970 Addenda to the NP&V Code)

Indicate which of the above design standards you have used in the design of Class I (NP&V Code) valves.

4.10 Describe the design and installation criteria for the mounting of the pressure-relieving devices (safety valves and relief valves) within the reactor coolant pressure boundary and on the main steam lines outside of containment. In particular, specify the design criteria that have been used to take into account full discharge loads (i.e., thrust, bending, torsion) imposed on valves and on connected piping in the event the valves discharge concurrently. Indicate the provisions made to accommodate these loads.

4.11 Paragraph 4.1.2.5 of the FSAR states that design cases II through IV, comparable to the "emergency" and "faulted" conditions defined in Section III of the ASME Code, have been applied to reactor coolant pressure boundary (RCPB) components. Identify any other components or systems that are not a part of the RCPB for which design Case IV stress limits will be applied. In the event that Class IV stress limits have been applied to any system or component exclusive of the RCPB, provide the bases for such application.

4.12 Recent fracture toughness test data indicate that the current ASME Code rules do not always assure adequate fracture toughness of ferritic materials. The Charpy V-notch tests are adequate to measure the upper shelf fracture energy value; however, they generally do not predict correctly the Nil Ductility Transition (NDT) temperatures or the transition temperature region in which fracture toughness increases rapidly with temperature. The NDT temperature, therefore, must be obtained from other tests, such as the drop-weight test (DWT). In addition, the transition temperature region shifts to higher temperatures when the thickness of the specimen tested is increased (size effect). In order to be able to establish appropriate heatup and cooldown limits for this plant, provide the following information:

4.12.1 For all pressure retaining ferritic components of the reactor coolant pressure boundary whose lowest pressurization temperature* will be below 250 °F, provide the material toughness test requirements and data (Charpy V-notch impact test curves and dropweight test NDT temperature, or others) that have been specified and reported for plates, forgings, piping, and weld material. Specifically, for each component, provide the following data, or your estimates based on the available data:

(a) The maximum NDT temperature as obtained from DWT,

(b) The maximum temperature corresponding to the 50 ft-lb value of the C_v fracture energy, and

(c) The minimum upper shelf C_v energy value for the weak direction (WR direction in plates) of the material.

---

* Lowest pressurization temperature of a component is the lowest temperature at which the pressure within the component exceeds 25 percent of the system normal operating pressure, or at which the rate of temperature change in the component material exceeds 50 °F/hr., under normal operation, system hydrostatic tests, or transient conditions.

4.12.2 Identify the location and type of the material (plate, forging, weld, etc.) for which the data listed above were obtained. Where these fracture toughness parameters occur in more than one plate, forging or weld, provide the information requested in 4.12.1 (a), (b), and (c) for each of them.

4.12.3 For reactor vessel beltline materials, including welds, specify:

(a) The highest predicted end-of-life transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for the weak direction of the material (WR direction in plates) and

(b) The minimum upper shelf energy value that will be acceptable for continued reactor operation toward the end-of-service life of the vessel.

4.12.4 Furnish the proposed heatup and cooldown curves that will be used to control the pressure and temperatures that the ferritic material of the reactor coolant pressure boundary will be exposed to during the first two years of operation and at the end-of-service life.

4.13 Describe the reactor vessel material surveillance program in sufficient detail to indicate conformance with ASTM E-185-70, especially in regard to Section 3.1.2 on retention of representative test stock (archive material) and Section 3.1.3 on chemical composition.

4.14 State the number of Charpy V-notch specimens oriented with respect to the weak direction (WR orientation in plates) of plates, forgings and weld materials that will be included in the reactor vessel material surveillance program.

4.15 If any component within the reactor coolant boundary has been designed or fabricated outside of the United States provide the following information:

4.15.1 Identify the manufacturer and describe his qualifications, experience in the construction of nuclear power plant components, and experience in furnishing components for nuclear power plants in the U. S.

4.15.2 Describe the steps you are planning to take to assure that the quality levels achieved in the fabrication of foreign procured components are acceptable.

4.16 Describe the process variables and the quality control procedures applied to achieve the desired material properties in the base metals, heat affected zones, and welds in connection with electroslag welding of the longitudinal seams of the shell courses of the steam generators and the pressurizer.

4.17 Indicate whether electroslag welding will be used in fabrication of any other components, particularly those made from stainless steel.

4.18 Describe the plans that were followed to avoid partial or local severe sensitization of austenitic stainless steel during heat treatments and welding operations for core structural load bearing members and component parts of the reactor coolant pressure boundary. Describe welding methods, heat input, and the quality controls that were employed in welding austenitic stainless steel components.

4.19 If nitrogen was added to stainless steel types 304 or 316 to enhance its strength (as permitted by ASME Code Case 1423 and USAS Case 71), provide justification that such material will not be susceptible to stress corrosion cracking under severely sensitized conditions.

4.20 Provide the following information regarding the primary coolant pump flywheels:

4.20.1 State the type of material used for the pump flywheel and its minimum specified yield strength. Indicate the nil-ductility transition (NDT) temperature specified for the materials, as obtained from drop-weight tests (DWT), the minimum acceptable Charpy V-notch (C_v) upper shelf energy level in the weak direction (WR orientation in plates), and the fracture toughness of the material at the normal operating temperature of the flywheel.

4.20.2 State the design stress specified for the flywheel as a percentage of the minimum specified yield strength, for the normal operating speed and the design overspeed condition.

4.20.3 State if the calculated combined primary stresses in the flywheel, at the normal operating speed will include the stresses due to the interference fit of the wheel on the shaft, and the stresses due to centrifugal forces.

4.20.4 State the highest anticipated overspeed of the flywheel and the basis for this assumption.

4.20.5 State the estimated maximum rotational speed that the flywheel attains in the event the reactor coolant piping breaks in either the suction or discharge side of the pump. In addition, describe results of any studies directed towards: (1) determining the maximum speed the pump or motors can reach due to physical limitations (e.g., the speed at which the pump impeller seizes in the wear rings due to growth from centrifugal forces or the speed at which motor parts come loose and grind or bind to prevent further increase in speed); (2) establishing speed and torque for various pipe break sizes; (3) devising means to disengage the motor from the pump in the event of pump overspeed; (4) verifying that pump fragments generated at maximum speed do not penetrate the pump casing and that any missiles leaving in the blowdown jet do not penetrate containment; (5) establishing failure speeds for motor parts and whether they will penetrate the motor frame and if so with what energy; and (6) defining a minimum rotor seizure time.

4.20.6 State the rotational speed that will be specified for the preoperational overspeed tests of the flywheel.

4.20.7 Describe the inservice inspection program proposed for the flywheels, state the areas to be inspected, access provisions, type and frequency of inspections, and the acceptance criteria.

4.21 Discuss the adequacy of the leak detection system which depends on reactor coolant activity for detection of changes in leakage during the initial period of plant operation when the coolant activity may be low.

4.22 With reference to the proposed maximum allowable leakage rate from unidentified sources in the reactor coolant pressure boundary, furnish the following information:

4.22.1 The length of a through-wall crack that would leak at the rate of the proposed limit, as a function of wall thickness.

4.22.2 The ratio of that length to the length of a critical through-wall crack, based on the application of the principles of fracture mechanics.

4.22.3 The mathematical model and data used in such analyses.

4.23 Specify the proposed maximum allowable total leakage rate for the reactor coolant pressure boundary, and the basis for the proposed limit. Furnish the ratio of the proposed limit to:

a. The normal capacity of the reactor coolant makeup system.

b. The normal capacity of the containment water removal system.

4.24 Provide the sensitivity (in gpm) and the response time of each leak detection system. For the containment air activity monitors, provide the sensitivity and the response time as a function of the percentage of failed fuel rods or of the corrosion product activity in the reactor coolant, as applicable.

4.25 Describe the proposed tests to demonstrate sensitivities and operability of the leakage detection systems.

4.26 Describe the design and arrangement provisions for access to the reactor coolant pressure boundary as required by Sections IS-141 and IS-142 of Section XI of the ASME Boiler and Pressure Vessel Code - Inservice Inspection of Nuclear Reactor Coolant System. Indicate the specific provisions made for access to the reactor vessel for examination of the welds.

4.27 Section XI of the ASME Boiler and Pressure Vessel Code recognizes the problems of examining radioactive areas where access by personnel will be impractical, and provisions are incorporated in the rules for the examination of such areas by remote means. In some cases the equipment to be used to perform such examination is under development. Provide the following information with respect to your inspection program:

4.27.1 Describe the equipment that will be used, or is under development for use, in performing the reactor vessel and nozzle inservice inspection.

4.27.2 Describe the system to be used to record and compare the data from the baseline inspection with the data that will be obtained from subsequent inservice inspections.

4.27.3 Describe the procedures to be followed to coordinate the development of the remote inservice inspection equipment with the access provisions for inservice inspection afforded by the plant design.

5.0 STRUCTURES

5.1 Identify the methods of seismic analysis (modal analysis response spectra, modal analysis time history, equivalent static load, etc.) or empirical (tests) analyses which have been employed in the design of Class I (seismic) structures, systems and components. Provide a more detailed description of all methods that were employed for seismic analysis including descriptions (sketches) of the mathematical models employed and the procedure for lumping masses.

5.2 Provide the criteria which were used to compute shears, moments, stresses, deflections and/or accelerations for each seismic-excited mode as well as for the combined total response, including the criteria for combining closely spaced modal frequencies.

5.3 Submit the basis for the methods used to determine the possible combined horizontal and vertical amplified response loadings for the seismic design of structures, systems and components including the following:

5.3.1 The possible combined horizontal and vertical amplified response loading for the seismic design of the building and floors.

5.3.2 The possible combined horizontal and vertical amplified response loading for the seismic design of equipment and components, including the effect of the seismic response of the building and floors.

5.3.3 The possible combined horizontal and vertical amplified response loading for the seismic design of piping and instrumentation, including the effect of the seismic response of the building, floors, supports, equipment, components, etc.

5.4 Provide the structure material properties and soil structure interaction which were used in seismic design analyses and the bases for the selection of these properties. Describe the measures which have been taken to assure that the calculated responses of Class I (seismic) structures will conservatively reflect the expected variations in the periods of vibration of the structures.

5.5 With reference to the seismic analysis of Class I (seismic) items by the response spectrum method using floor response spectra, the shape of these floor response spectra is dependent on the assumptions made for the structural properties, dampings, and soil structure interactions. Describe the measures which have been taken to consider the effects on floor response spectra of expected variations in structure response.

5.6 The use of both the modal analysis response spectrum and time history provides a check on the response at selected points in the station structure. List the responses obtained from both methods at selected points in the Class I (seismic) structures to provide the basis for checking the seismic system analysis.

5.7 Indicate the provisions utilized to assure that the crane located in containment is held on its rails to preclude dislodgement during seismic excitation.

5.8 With respect to Class I (seismic) piping buried or otherwise located outside of the containment structure, describe the seismic design criteria employed to assure that allowable piping and structural stresses are not exceeded due to differential movement at support points, at containment penetrations, and at entry points into other structures.

5.9 Describe the method employed to consider the torsional modes of vibration in the seismic analysis of Class I (seismic) structures.

5.10 Describe the seismic design criteria employed to assure the adequacy of Class I (seismic) mechanical components such as pumps and heat exchangers, and electrical equipment such as cable trays, battery racks, instrument racks and control consoles. Describe the measures taken for seismic restraint to meet these criteria, the analytical or testing methods employed to verify the adequacy of these restraints and the methods utilized to determine the seismic input to these components.

5.11 Describe the criteria employed to determine the field location of seismic supports and restraints for Class I (seismic) piping, piping system components, and equipment, including placement of snubbers and dampers. Describe the procedures followed to assure that the field location and characteristics of these supports and restraining devices are consistent with the assumptions made in the dynamic analyses of the system.

5.12 Describe the procedures used to account for the number of earthquake cycles during one seismic event and specify the number of loading cycles for which Class I (seismic) systems and components are designed for in this event as determined from the expected duration of the seismic motions or the number of major motion peaks.

5.13 In order to assess that the seismic design bases for this plant have been properly translated into the required specifications, drawings and procedures, systems and components, provide the following information:

5.13.1 Identify the design organizations involved in the seismic design of all safety-related items of the plant, their respective responsibilities, and the documented procedures followed to assure that these responsibilities have been met. Identify the organization that was assigned overall responsibility for the adequacy of seismic design.

5.13.2 In regard to the interchange of design information among the involved design organizations, revisions thereto, and coordination of all aspects of the seismic design, describe the documented procedures employed to assure that these interchanges and coordination among design organizations have been followed.

5.13.3 Describe the design control measures instituted to verify the adequacy of the seismic design and identify the responsible design groups or organizations who performed this function.

5.13.4 Describe the requirements that have been included in the purchase specifications for safety-related equipment to assure adequate design and functional integrity under the seismic design conditions. Describe the provisions that are included in the purchase specification to permit the purchaser to verify that these requirements are satisfied.

5.14 The Arkansas Nuclear One facility is built on semi-permeable soil. Discuss the provisions made to prevent ground water from flooding equipment that is located below the highest flood level and is necessary to shutdown the reactor and maintain it in a safe shutdown condition.

5.15 Several Class I (seismic) tunnels, underground cells, underground piping and underground cables are connected to the main structures. The seismic response of buried elements is very different from the response of the main structures. Describe the methods of design and the actual arrangement of the details of connections between these appendages and the main structures. Demonstrate that stresses due to dynamic interaction meet the criteria for allowable design stresses.

5.16 Although the foundations of the main Class I (seismic) structures are not interconnected, soil interaction will exist between them during OBE and DBE disturbances. Explain the method used to account for this condition in the design and construction of the foundation slabs and walls. Show that under the most unfavorable conditions the stresses in the foundations and soils meet the criteria for allowable design stresses.

5.17 The ACI 318-63 Code is essentially a code for framed structures where stresses are mostly uni-axial. In the containment, stress distribution is mostly tri-axial. Experimental evidence exists that when one or two of the three principal stresses are tensile stresses, the ultimate compressive strength of concrete decreases by a large amount. Therefore, the different stress limits established by the code such as: 0.45 f'_c; 0.60 f'_c; 0.85 f'_c; 6 $\sqrt{f'_c}$; 3.5 $\sqrt{f'_c}$; 2 $\sqrt{f'_c}$; 1.1 $\sqrt{f'_c}$, may not be applicable, unless some reduction of these values was considered. Demonstrate that the containment structure as designed and built, has sufficient safety margins, despite the fact that this correction has not been considered.

5.18 The definition of the yield strength of the structure as the upper limit of elastic behavior (Appendix 5A) is arbitrary and in contradiction with experimental results. When the compression stress in the extreme concrete fiber reaches f'_c the stress distribution is already far from triangular. A justification should include a discussion of shear and axial stresses which should be compatible with the other stresses.

5.19 The ACI 318-63 Code in Section 2603(a) and 2603(b) states:

"Stresses and ultimate strength shall be investigated at service conditions and at all load stages that may be critical during the life of the structure from the time prestress is first applied, and

Stress concentrations due to the prestressing or other causes shall be taken into account in the design."

In the commentary on this Code, the same Committee states, in Section 2603: ". . . the design investigation should include all load stages that may be significant. The three major stages are:

(1) initial stage--when the tensile force in the prestressed steel is transferred to the young concrete; when stress levels are high relative to cylinder strength,

(2) working load stage--after long-time volume changes have occurred, and

(3) the ultimate capacity load stage. There may be other load stages that require investigation. For example, if the cracking load is significant, this load stage may require study. From the standpoint of satisfactory behavior, the two stages of most importance are ultimate capacity and working stress (or service load). It is necessary to investigate both of these stages because of the nature of prestressed concrete."

The ACI Committee 334 in "Concrete Shell Structures Practice and Commentary" states in Section 202(d) and 202(e):

"Equilibrium checks of internal stresses and external loads are to be made to ensure consistency of results.

An ultimate strength analysis may be used only as a check on the adequacy of the design. It is not to be used as a sole criterion for design, except where it can be proven to be applicable."

In its commentary the ACI Committee 334 says in Part 4.

". . . the analysis must be made at design and at ultimate loads to ensure both proper behavior at design loads and an adequate overload capacity."

Indicate whether the design and the checking of the design have been made in accordance with the provisions indicated above.

5.20 Demonstrate that earthquake shears can safely be carried by the containment structure concrete without special shear reinforcing.

5.21 Present a study analyzing the stresses in the containment structure at end of service life. This study should include the effect of structural and thermal creep and shrinkage on stress redistribution, and should be based on a realistic evaluation of the modulus of elasticity, E, and Poisson's ratio. The use of a uniform reduced E is not considered to be satisfactory. The purpose of this study should be to demonstrate that after repeated reactor shutdowns and reactor startups no excessive tensile stresses will exist in the concrete at the liner, where there is no reinforcing to control cracking.

5.22 Justify the omission of radial reinforcing in the wall and in the dome of the containment building. Prestressing generates tensile stresses and tensile strains in the direction perpendicular to the tendon, especially where tendons are curved. Demonstrate that the absence of radial reinforcing will not jeopardize the long range stability of the containment. Indicate the ultimate tensile strength of the concrete if the design relies upon it.

5.23 Explain what provisions were made for the seismic design of the equipment, personnel and escape locks.

5.24 The 1/4 inch thick liner is attached to the concrete by means of an angle grid system with the unpainted liner face in contact with concrete. Under prestress load, concrete creep, concrete shrinkage and thermal load, the compression stress in the liner may reach the yield point during normal operating condition. In the event some plates buckle and deflect towards the inside of the containment, voids which may therefore occur between concrete and liner may be connected with the outside through cracks in concrete. Explain what provisions are made to prevent the unpainted liner from corroding under these conditions and what surveillance measures could be used to detect this condition.

5.25 The liner is not backed up by concrete at openings and behaves partly as an element of a pressure vessel. Indicate the provisions incorporated in the structure for the safe transmission of membrane and bending stresses existing in these transition pieces into the rest of the structure. Demonstrate that critical stresses or strains occurring at these locations meet the criteria for allowable design stresses and strains.

5.26 The use of the ASME Pressure Vessel Code Section III for the design of the liner is questionable since the Code applies to a structure supporting mechanical loads such as pressure. The liner is loaded by strains transmitted to it by concrete and must follow the concrete deformations. In this case the thickening of the liner at the openings does not help to reduce the stresses out only increases the local stiffness of the liner, which, in turn, may increase local stress concentrations. Demonstrate the manner by which liner failure or leakage will be precluded since the ASME code rules may not adequately cover this situation.

5.27 Explain the methods of design of the expansion joint bellows for axial and lateral displacement of the fuel transfer tube, which is a part of the containment. Discuss their accessibility for inspection and repairs.

5.28 Where Class I (seismic) structures are directly connected to Class II (seismic) elements, such as equipment and piping systems, indicate if and how the influence of seismic activity of the Class II (seismic) elements on the Class I (seismic) structures has been considered in the Class I (seismic) designs, such that damage or excessive movement of Class II (seismic) systems will not adversely affect Class I (seismic) structures.

5.29 Provide drawings of the details and final design of the anchoring and reinforcing arrangement for the Class I (seismic) auxiliary building, control room fuel handling building, pump house (portion containing the service water system), containment interior structure, and the equipment supports for the reactor, steam generator, pressurizer, and main pumps.

5.30 For the containment interior structures, describe the dynamic design used to ascertain that this structure meets seismic Class I (seismic) criteria. Explain the methods used to compute the jet forces for the structural design of the containment interior structure. Indicate the pressure and temperature safety margins resulting from the stress analyses of the interior structure.

5.31 Indicate the design criteria, materials, allowable stresses and strains, and load combinations considered for the design of the concrete and cast-in steel for the structural supports for the reactor, steam generators, pressurizer, main coolant pumps, and safety injection tanks.

5.32 Describe the method used to evaluate the amount of radiation generated heat in concrete adjacent to the reactor and the criteria and methods used for the design of the reinforcement at this location.

5.33 Discuss the tornado protection provided for critical elements of Class I (seismic) structures, such as containment penetrations, locks, doors, large openings, spent fuel pool, diesel rooms, removable shielding blocks, etc. Indicate whether the tornado design considers possible increase of the tornado loading on the containment structure due to the funneling effect of the tornado wind blowing between the two containment structures.

5.34 Describe tie down arrangements made for all Class I equipment to resist seismic, jet and tornado forces. Restraints provided for equipment usually have gaps between them and the equipment. Indicate whether the seismic analysis considered these gaps and if the impact forces due to the gaps, acting on the equipment, the restraints and the structures were evaluated.

5.35 Present a list of typical missiles both accident and tornado generated, that have been considered in the overall design and indicate the nature of the missile, the weight, mass/cross-section ratio, shape, assumed point of impact, assumed impact velocity, and where and how originated. Indicate the criteria and the method of analysis used for checking the structure at point of impact. Indicate whether all Class I structures or parts of structures (such as spent fuel pool) have been checked for impact of missiles. Indicate whether the crane can generate internal missiles that may endanger the containment structure.

5.36 To evaluate the design adequacy of the fuel pool floor to withstand the effect of a fuel cask drop, submit the design criteria and applicable analyses used in the design of the floor. Indicate the maximum thermal stresses which can be developed in the spent fuel pool walls under the most adverse conditions. Describe what provisions have been made to control cracking in this structure.

5.37 Present a discussion of the adequacy of the steel structure over the spent fuel pool, since this structure has not been designed to resist tornadoes. Discuss its status under tornado conditions -- whether it will remain standing or partially collapse. Describe what will happen to the crane under these conditions.

5.38 Where yielding is permitted in structures, define the allowable limits and list the structures, parts of structures, and elements where yielding may occur and under what load combinations. Demonstrate how these limits meet acceptable design strain limits.

5.39 For the design of the containment structure and other Class I structures, describe the technique used for the proportioning of the reinforcement after completion of the computer analysis.

5.40 For several of the critical locations of Class I structures, including the containment, indicate separately the seismic stress contributions as a portion of the total stress.

5.41 State whether the seismic analysis covered the gradual or accidental deterioration of structures and systems. For instance, whether the analysis has considered the possible increase of reaction loads on structures due to ruptured pipes in the case of a postulated LOCA.

5.42 In the event of an OBE or DBE, torsional loads will be applied evenly to symmetrical structures (reference paper by N. M. Newmark, "Torsion in Symmetrical Buildings", Fourth World Conference on Earthquake Engineering, Santiago, Chile, 1969). Indicate whether torsional effects have been considered in the design of all Class I (seismic) structures and the method used for this analysis.

5.43 Describe the erection method associated with the tendon anchor, especially the means by which a good concrete bearing of the tendon bearing plate was achieved. Describe the quality control measures which were used to assure an adequate bearing.

5.44 Describe the arrangement provided to absorb the torsional moment generated in the tendon anchor because of the twisting of the tendon during erection, and its effect on the ultimate strength of the anchor. Indicate whether all the tendons or only the hoop and dome tendons have been twisted when erected to give a helical shape to the wires and equalize their lengths. Indicate the maximum eccentricity of the load which may exist because of erection inaccuracies and its effect on the ultimate strength of the anchor.

5.45 Furnish the pertinent design information contained in the procurement specification for the prestressing tendon ducts. Indicate the provisions made in the ducts to prevent in-leakage of mortar during concreting, and out-leakage of grease during the greasing operation. Indicate the measures used to prevent corrosion of the ducts, and what means have been used during erection to avoid water condensation in the ducts.

5.46 Furnish a typical chemical analysis of any fly ash used in concrete work and demonstrate that it will not increase the corrosion of the tendons by the inclusion of deleterious substances such as sulphur and chlorine.

5.47 Describe how the seal welds between the crane bracket and thickened liner plate are checked for leak tighness. These are working welds under normal operation and may have a substantial probability of failure relative to typical liner plate seam welds.

5.48 Indicate whether any welding has been used on reinforcing steel. If so, describe the design criteria for the weld and the quality control which has been used during welding operations. Also state whether any protective measures were taken during welding near unprotected prestressing tendons.

5.49 At what point between the mixing and placement of concrete were concrete compression and slump test samples taken during construction?

5.50 Indicate whether concrete in Class I (seismic) structures has been placed by pumping through an aluminum pipe and the measures used to prevent any such concrete from losing strength.

5.51 Describe the permanent provisions made for access to the exterior of the upper parts of the containment structure to facilitate periodic inspection and testing.

5.52 The acceptance pressure testing should provide a check on design and construction of the containment. In this context, justify the use of only two meridians for taking measurements during pressure testing, and demonstrate that the information thus collected will be sufficient to fulfill the test purpose. Indicate the tolerances on test acceptability. The reference to the Palisades and Turkey Point containments is not sufficient since the ARK-1 structure is a new type of containment with only 3 buttresses and 186 wire tendons.

5.53 Indicate whether periodic structural integrity tests are planned after the initial proof test. Discuss the expected results of these structural integrity tests and the testing procedure that will be implemented. These periodic tests may be necessary to check on possible gradual deterioration of the structure. Provide the frequencies at which containment structural integrity tests will be conducted during plant life, either separately or in conjunction with periodic containment leakage rate tests.

5.54 Present the program for surveillance of the containment structure concrete, tendons and liner during the lifetime of the plant, including a discussion of tendon inspection on the basis of a sampling procedure as specified in the MIL-STD-105D or MIL-STD-414, or similar standards. Describe the bases on which the decision was made to select three hoop, three vertical, and three dome tendons for surveillance. Furnish information as to whether the individual wires will be checked for breakage on the surveillance tendons and on other tendons. If they will be checked, indicate the procedure that will be followed. Reliance that button heads would "pop up", and remain so for an extended period, would be inadequate, as this visual evidence will not necessarily be present.

5.55 The grease used in prestressing ducts has a coefficient of thermal expansion much larger than that of the structure. It is pumped into the ducts at a temperature much higher than its operating temperature, and therefore will contract. Explain the means used to prevent void formation in grease because of this tendency to contract and the assurance that exists that this void formation will not accelerate corrosion phenomena.

5.56 It is not indicated whether a cathodic protection system will be provided for the corrosion protection of the steel liner, reinforcing bars and the tendon steel casings. Discuss the protection such a system may offer, especially to the prestressing tendons. Discuss also the detrimental effect the cathodic system may have when small faults occur, such as voids in tendon protective grease and discontinuities in the tendon ducts.

5.57 Indicate the design features provided to enable compliance with the AEC General Design Criterion 52, published in the Federal Register on July 7, 1971, in regard to ability to conduct periodic integrated leakage rate testing at containment design pressure.

5.58 Describe the design features of the containment airlocks that will permit testing of the airlocks at full accident pressure. Describe the test method that will be used to verify leak tightness of air lock doors, door penetrations and door gaskets.

5.59 Provide information that demonstrates that deformations associated with seismic loadings would not interfere with the proper operation and sealing of containment equipment penetrations.

5.60 Appendix 5K states that some of the liner plate anchor tests showed less load or displacement capacity than predicted values. Provide the test results in conjunction with the predicted values.

6.0 ENGINEERED SAFEGUARDS

6.1 Provide an analysis of potential post-LOCA hydrogen generation using the assumptions set forth in Safety Guide No. 7 "Control of Combustible Gas Concentrations in Containment Following a Loss-ofCoolant Accident".

6.1.1 Provide the predicted site boundary and LPZ doses that would result from containment purging in the event of the design basis loss-of-coolant accident using the assumptions set forth in Safety Guides 4 and 7.

6.2 Describe the design criteria that are applicable to the containment purge system including requirements for accident environment testing, redundancy, testability and single failure analysis. Indicate the means whereby the containment atmosphere will be mixed, sampled and monitored for radioactivity during purging operations.

6.3 Describe the inservice inspection program for fluid systems other than these composing the reactor coolant pressure boundary, including items to be inspected, accessibility requirements, and the frequency and types of inspection. The fluid systems to be considered are applicable engineered safety systems, reactor shutdown systems, cooling water systems, and the radioactive waste treatment systems.

6.4 Describe in detail the preoperational test program that will be performed on the installed ECCS system to demonstrate that the system will perform its design functions under accident conditions.

6.5 Provide an analysis which demonstrates that NPSH requirements for the Emergency Core Cooling and Containment Heat Removal System pumps are in conformance with Safety Guide No. 1.

6.6 Describe the design criteria that account for the potential effects of radiation resulting from both normal operation and accident conditions superimposed on long-term normal operation for electrical and mechanical equipment of the reactor protection system and engineered safety features.

6.7 Discuss the design of the control circuits for the motor-operated isolation valves located between the accumulator tanks and the primary coolant system, indicating the assurance provided by the design that the valves will be open when required. The inclusion of the following features would provide acceptable design:

(1) Valve position visual indication that is actuated by sensors on the valve ("Open" and "Closed").

(2) An audible alarm, independent of item (1), that is actuated by a sensor on the valve when the valve is not in the fully open position.

(3) A lock-out of power to the valve operator at any time the primary coolant pressure exceeds a preselected value (specified in the Technical Specifications) as an alternative consideration could be given to the incorporation of the following features in lieu of item (3).

(4) Automatic opening of valve when the primary system pressure exceeds a preselected value (specified in the Technical Specifications).

(5) A safety injection signal used to automatically remove (override) any by-pass feature that may be provided to allow a motor operated valve to be closed, for short periods of time, when the primary system is at pressure (in accordance with the provisions of the Technical Specifications).

7.0 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Provide the following information with regard to the protection systems that actuate reactor trip and engineered safety feature action:

7.1.1 A list of those systems designed and built by B&W that are identical to those of the Oconee Nuclear Station (as documented in the SAR) and a list of those that are different, with a discussion of the basis for the design differences;

7.1.2 A list of those systems and their suppliers that are designed and/or built by suppliers other than B&W; and

7.1.3 Identification of any features of the design that do not conform to the criteria of IEEE 279 or the Commission's General Design Criteria and an explanation of the reasons.

7.2 Provide the following information with regard to the control systems designed by B&W:

7.2.1 Identification of the major plant control systems (e.g., primary temperature control, primary water level control, steam generator water level control) that are identical to those in the Oconee Nuclear Station; and

7.2.2 A list and a discussion of any significant design features in these systems that are not identical to those used in the Oconee Nuclear Station. This discussion should include an evaluation of the safety significance of each design difference.

7.3 Describe the qualification testing requirements that have or will be used to assure that the seismic design criteria for the reactor protection system, engineered safety feature circuits, and the emergency power system are satisfied, and how these requirements are being imposed on equipment suppliers.

7.4 Submit the criteria and their bases that establish the minimum requirements for preserving the independence of redundant reactor protection systems, engineered safety feature systems and Class IE* Electrical Systems through physical arrangement and separation and assure minimum availability during any design basis event. The submittal should include a discussion of the administrative responsibility and control to be provided to assure compliance with these criteria during the design and installation of these systems. As a minimum the criteria and bases for the installation of electrical cable for these systems should address:

---

* Class IE electrical systems and design basis events are defined in the Proposed IEEE Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations (IEEE 308).

7.4.1 Cable derating.

7.4.2 Cable routing in containment, penetration areas, cable spreading rooms, control rooms and other congested or hostile areas.

7.4.3 Sharing of cable trays with non-safety related cables or with cables of the same system or other systems.

7.4.4 Fire detection and protection in the areas where those cables are installed.

7.4.5 Cable and cable tray markings.

7.4.6 Spacing of wiring and components in control boards, panels, and relay racks.

7.5 Identify all safety related equipment and components (e.g., motors, cables, filters, pump seals) located within the containment required to be operable during and/or subsequent to a loss of coolant accident or a steamline break accident. Describe the qualification tests that have been or will be performed on each of these items to assure their performance in a combined high temperature, pressure, and humidity environment.

7.6 What design criteria and/or operating limitations have been established to assure that loss of the air conditioning and/or ventilation system will not adversely affect the operability of safety related control and electrical equipment located in the control room or other equipment rooms. Describe the analysis performed to identify the worst case environment (e.g., temperature, humidity). State the limiting condition with regard to temperature that would require reactor shutdown, and how this was determined. Describe any testing (factory and/or onsite) that has been or will be performed to confirm satisfactory operability of control and electrical equipment under extreme environmental conditions.

7.7 Describe how reactor protection system and engineered safety feature equipment, including cables, will be identified physically as safety related equipment in the plant to assure appropriate treatment, particularly during maintenance and testing operations. Also, describe your identification scheme for distinguishing between redundant channels of the systems and discuss how it will be evident to the operator or maintenance craftsman, without consulting reference material, whether the equipment is safety related and which channel is involved. Evaluate your identification scheme with respect to compliance with the requirements of Section 4.22 of IEEE 279, Revision 1.

7.8 Describe the methods planned for periodic inservice testing of reactor protection system response times for the various reactor trip parameters. Include a discussion of the relationship of channel response time to safety limit and identify the margin in terms of time.

7.9 Describe the methods planned for periodic inservice testing of engineered safety feature instrumentation and control equipment. We interpret IEEE 279 to require the same high degree of inservice testability for engineered safety feature actuation as is required for the reactor trip system.

7.10 Identify the information readouts or indications provided to the operator for monitoring conditions in the NSSS and in the containment over the full operating range of the plant, including anticipated operational occurrences and accident and post-accident conditions. Your response should include the design criteria and their bases, the type of readout, number of channels provided, range, accuracy and location.

7.11 Discuss the indications available to the control room operator that would allow him to recognize that a protection system or subsystem has been placed in test, bypassed for operation or maintenance purposes, or removed from service for any cause. Also, include a listing of the operating bypasses and show that they have been designed to meet the requirements of Paragraph 4.12 of IEEE 279.

7.12 State the criteria and design bases that established the heat tracing requirements, temperature control, monitoring, and power requirements for the boric acid and the sodium hydroxide subsystems of the chemical addition system. State whether a single failure can cause a loss of function and provide justification for the cases where the single failure criterion is not satisfied.

7.13 The FSAR states (page 14-7) that a pump interlock system prevents startup of a pump in an idle loop when operating above 22% full power. The description of the system is not sufficiently detailed to permit evaluation. Submit a description of the circuitry to show that the requirements of IEEE-279 are satisfied.

7.14 Identify any reactor trip set point adjustments that must be made in the protection system when reducing the number of primary pumps in operation. Describe how these trip set point adjustments will be made, and verify these adjustments will be made in accordance with Section 4.15 of IEEE 279. Discuss the positive means that will exist for assuring that the most restrictive setpoint will be used when applicable and that improper use of a less restrictive setpoint will be prevented. Indicate how the single failure criterion associated with the reactor protection system is met during this operation.

7.15 Identify the reactor protection system and engineered safety feature instrumentation sub-systems for which the trip setpoints are within 5% of the high or low end of the calibrated range. Provide an error analysis for each such case to ensure that the required output signal is always conservative when viewed from a safety standpoint.

7.16 Describe the design of all instrument sensing lines that penetrate or are connected to the primary reactor containment, including the isolation valving and the valve position indication provided. Identify any features of the design that do not satisfy the recommendations of Safety Guide 11.

7.17 In your design, safety injection initiation is provided by either low reactor coolant pressure or high reactor building pressure. Initiation by either of these diverse signals, assuming failure of the other signal, will result in satisfactory ECCS performance. Since your analysis of the effectiveness of the ECCS performance design assumes a reactor trip, explain the bases for not using similar diverse signals to trip the reactor.

7.18 Your description of the logic required to actuate the reactor building spray system (page 7-12) implies that coincident signals from the two actuation channels are necessary to initiate operation of this system. Such a design is vulnerable to single failures in that the loss of either signal will preclude operation of this system. Provide a justification for your design or describe the design modifications you will make to correct the design in this regard.

7.19 The FSAR states that flexibility in controlling reactivity rate is provided by a patch panel that allows individual rods to be switched from one group to another. Discuss the interlocks and/or administrative procedures that will be used to assure that rod pattern changes made in this manner will be acceptable.

7.20 The FSAR does not explicitly discuss the design capability for shutting down the reactor from outside the control room. Mention is made of the ability to maintain the reactor in a safe shutdown condition from outside the control room, but this ability is contingent upon performing six discrete operations within the control room before external control can be made effective. Explain how the requirement to perform several operations prior to evacuating the control room is in conformance with General Design Criterion 11.

8.0 ELECTRICAL POWER SYSTEMS

8.1 Identify all aspects of the safety related portion of your standby electrical power system that do not conform to Safety Guides 6 and 9, and IEEE 308. Provide the bases for any deviations.

8.2 The FSAR states that either or both diesel generators may be used for system peaking. Section 5.2.1 of IEEE 308 reads, in part: "Sufficient physical separation, electrical isolation, and redundancy shall be provided to prevent the occurrence of common failure mode in Class IE systems". A subsequent part of this section is explicit with regard to common mode failures of the preferred and standby power supplies, by stating: "The preferred and the standby power supply shall not have a common failure mode. In addition, the generating sources shall not have a common failure mode for any design basis event." Therefore provide the following information:

8.2.1 Evaluate the degree of isolation and redundancy provided when either diesel generator, or both, are interconnected with the offsite system. Consider all applicable design basis events that can precipitate faults on any portion of the combined systems, and provide an analysis that delineates the consequences of such faults on the off-site power system and on the diesel generator system prior to the tripping of the 4160 V breakers. As a minimum, your response should: (1) provide the results of subtransient, transient and steady-state analyses that account for system faults and equipment malfunctions, (2) account for the effects of switching surges and fault clearing surges, (3) indicate the effects of lightning striking any portion of the combined system on the offsite and onsite power system, and (4) submit information on the surge protection devices that are integral to the diesel generator system such as neutral grounding transformers, lightning arresters, resistors, capacitors, and reactors.

8.3 Describe more fully the auxiliary equipment for the emergency diesel generator system. The description should include the fuel storage and transfer system, the source of power for control, the starting system and number of start attempts provided, method of cooling and warming the engine, the control and protection system including relevant schematic diagrams, logic diagrams depicting interlocks and related constraints and equipment protective interlocks.

8.4 The tabulation of connected loads to the diesels, as shown in Table 8-1 of the FSAR, is insufficient to evaluate the adequacy of the design. Provide data to verify that the design and rating of the diesel generators conform to Safety Guide 9. Data that are required include actual starting KVA loads (in addition to the connected loads shown), and the time required for the various loads to reach full speed. To correlate these data with the capability of the diesel generator, submit the following information:

8.4.1 A load profile during a LOCA showing the timing sequence and the time duration of the various loads subsequent to diesel start. Superimpose the anticipated starting KVA requirements for the various loads at the appropriate portion of the load profile, and indicate the computed duration of the transients and the calculated system voltage.

8.4.2 Indicate the maximum loads that can be incrementally added to the various block plateaus without exceeding the recommendations of Safety Guide 9.

8.4.3 Indicate the continuous, 2000 hr., and 30 min. rating of the diesel engine.

8.4.4 State the generator's KW, KVA, PF, X'_d, X_d, X''_d, and SCR. In addition, describe the type of excitation system provided and the response time of same for voltage regulation, and the unit's WR^2.

8.5 Discuss the design criteria for the diesel generator rooms with respect to seismic design and ability to prevent missiles, explosions and fires in one unit from affecting the other unit.

8.6 Identify the sources of control power to the 500 and 161 KV switchyard breakers. Submit an analysis to show that no single failure in these power supplies, control circuits and protective relaying will negate the ability to provide off-site power to the engineered safety features.

8.7 Describe the monitoring features that are provided to continuously indicate that the capability of a battery to supply power is not degraded. Consider the relevance of the monitored parameters to the actual charge stored in the battery and discuss the limitations of the system to ensure disclosure of battery degradation. What protection is provided against overcharging?

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.1 A review of the decay heat removal system description in Section 9.5 and Figure 9-12 of the FSAR indicates that overpressure protection of the 450 psig system from the 2500 psig reactor coolant system has been accomplished by valves interlocked to prevent opening with reactor coolant pressure above 450 psig and safety relief valves.

We have concluded that your present design and operating procedures do not provide for an acceptably low probability for inadvertent high pressurization of the low design pressure system. In our opinion the two valves should be provided with a highly reliable control system to assure that they will close automatically whenever the pressure in the primary coolant system exceeds the low pressure system maximum operating pressure. Although the FSAR states the valves will be interlocked to prevent opening at high pressure, there is no statement that the interlocks are designed to safety system standards and on the basis of different principles. Both the valve automatic closure system and the interlock system should conform to IEEE-279 requirements.

Provide either a description of the modifications you plan to make to the system to reduce the probability of an inadvertent pressurization of the low pressure system, or a comprehensive assessment of the facility design and operating procedures to convincingly demonstrate that the present design and proposed procedures provide an acceptably low probability for the overpressurization of the low pressure system.

9.2 Provide the following additional information with respect to the service water system:

9.2.1 Describe the system completely, including drawings that clearly illustrate the service water pump compartments and sluice gates in relation to the intake structure and the emergency pond. The drawings should show all (including redundant) pipe lines from and to the emergency pond.

9.2.2 Provide a summary of a failure analysis, including capability for emergency cooling if the intake structure is empty, (failure of Dardanelle Dam).

9.2.3 Provide the design criteria and bases for the sizing of the emergency pond, service water pumps and piping. Include all possible modes of operation for both Unit 1 and Unit 2, including modes in which an accident has occurred in either unit. Consider the effect of using either off-site or on-site power. Provide a summary of an analysis that demonstrates that the required performance will be provided for all potential conditions. Include relative elevations, emergency pond water temperatures, margins in net positive suction head for the service water pumps, flow rates and maximum operating time. How will maintenance outages for pumps and sluice gates be accommodated?

9.2.4 Describe in detail the design provisions to assure the availability of the required amount of water in the emergency pond. Include design criteria, drawings, soils characteristics, construction practices, in-service monitoring programs, and seismic design criteria and analysis methods. Provide results of test borings.

9.2.5 We understand that all sluice gates are to be "manually" operated. Discuss how you can assure that this operation would be accomplished in the event of a dam failure coincident with an accident, including how the operator will be advised of a failure of the Dardanelle Dam and the probable elapsed time, the minimum time at which water could no longer be provided to the service water pumps from the intake structure, the time required for personnel to operate the sluice gates manually and the radiation exposure these personnel would receive.

9.3 Describe in further detail the method you propose to use in handling the fuel shipping cask. Provide an analysis of the results of dropping a cask while passing over the control room, including assumptions and methods utilized. Repeat the analysis for the higher drop distance associated with the hatch opening to an elevation of 354 feet-0 inches. Include any potential radiological consequences associated with these events.

11.0 RADIOACTIVE WASTE AND RADIATION PROTECTION

11.1 Provide a summary and the results of an analysis to show that operating personnel in the control room are adequately shielded from direct radiation and airborne activity during accident conditions. Indicate how makeup air would be provided to the control room during the course of the accident and what time duration was assumed in your analysis.

11.2 Provide the locations, sensitivities and ranges of the area radiation monitoring system equipment.

11.3 Provide a description of the steam generator blowdown system and discuss how this effluent will be processed prior to release. Indicate how the blowdown line is monitored for radioactivity and specify the location, sensitivity and capability that this instrumentation will have for automatically isolating or diverting the blowdown steam.

11.4 Provide a diagram of the station vent plenum system showing the location and arrangement of all lines discharging into this plenum, filter locations, radiation monitoring equipment and other important features. Indicate how this system is interconnected with the plant vent and the reactor building flutes shown on Figure 9-13.

11.5 Describe the radiation monitoring system more fully including the design confidence level, the calibration methods to be employed and the in-service testability provided. Discuss the redundancy and independence of the effluent discharge monitors. Identify the type of sampler used in each monitor (on-line or off-line) and the sample drive means (pump or sampled system differential) of the off-line samplers. Identify the system status alarms. Indicate the radiation monitors on applicable P&I diagrams. Provide a block/logic diagram of the systems that includes the following:

a. Alarms and alarm logic

b. Control interlocks

c. Computer outputs

d. Number and type of detectors

e. Power requirements and sources

12.0 CONDUCT OF OPERATIONS

12.1 Provide the following information with regard to the corporate level technical staff that will support operation of ARK-1.

12.1.1 Resumes of each Production Department staff engineer directly involved in support of Unit 1, including previous experience, training, education and other pertinent background information.

12.1.2 A description of the duties and responsibilities of each staff member described in 12.1.1. Indicate where applicable, what functions these personnel will perform relative to the design and construction of Unit 2 while Unit 1 is in operation.

12.1.3 A description of the management transition of the facility from construction to operation, including any provisions for increased staffing at the corporate level prior to operation.

12.2 Provide the following information in regard to Middle South Services:

12.2.1 An organization diagram showing the various departments, services and areas of expertise available for consultation.

12.2.2 A description of the types of services and functions that Middle South Services will provide in support of Unit 1.

12.2.3 Resumes of the various personnel within this organization who will provide the services identified in 12.2.2.

12.3 Provide the following information in resume form for each individual assigned to initial operation of Unit 1.

12.3.1 Formal education including dates and general area of study.

12.3.2 Previous work experience including dates and general description of duties and responsibilities.

12.3.3 Training completed or projected to be completed prior to Unit 1 operation including dates, course content and method of instruction.

12.4 The independent review and audit function to be performed by the Safety Review Committee is considered to encompass many specialized areas of expertise, all of which are important to nuclear safety. These areas include: reactor operations, reactor engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety and mechanical and electrical engineering. Indicate how each of these areas of expertise (experience and/or training) are covered either collectively or individually by the proposed members of the Safety Review Committee.

12.5 The operations group proposed for Unit 1 operation consists of 4 shifts of 4 men apiece. Provide the following information:

12.5.1 Your basis for concluding that a 4 man shift can adequately and safely operate the facility during all normal, abnormal, and emergency conditions.

12.5.2 Describe the 4-shift rotational pattern that you intend to employ. State how vacations, illnesses, special tests and other unforeseen operational occurrences will be accommdated by your proposed 4 shift arrangement. How much regularly scheduled overtime will be required on a weekly and yearly basis for each of the 4 man shift members?

12.5.3 What provisions have been made to train and otherwise qualify supplemental personnel in the event additional shift crew personnel are required prior to startup.

12.6 It is our understanding that the nuclear engineer will be primarily a home office staff position that will provide support to the facility on an as needed basis. Which individual assigned to the plant on a <u>full time</u> basis will provide the 2 years minimum experience in reactor physics, core measurements, core heat transfer and core physics testing programs specified in ANSI-N18.1.

12.7 Either the Plant Superintendent or his immediate assistant should possess a Senior Operator's License prior to initial fuel loading. We note that Figure 12-2 shows the Assistant Plant Superintendent having such a license while paragraph 12.1.1 indicates neither of these individuals will be licensed until some time after initial operation is achieved. Please clarify your intention in this area.

12.8 The information presented in Section 12.3.1 in regard to emergency preparedness is insufficient. A more comprehensive plan must be developed in accordance with the requirements of 10 CFR 50.34, Appendix E. In addition, provisions to be employed during emergencies while Unit No. 2 is under construction should be included in the plan.

12.9 Provide information that outlines the security precautions that will be in effect during operation of ARK-1.

13.0 INITIAL TESTS AND OPERATIONS

13.1 Provide a brief summary of each test to be performed during the plant startup program that involves the operation or testing of systems and components having safety significance. Include the prerequisites, objectives, general method of performance, data required and acceptance criteria for each test identified. Indicate any special conditions that will be used to simulate abnormal plant conditions during testing.

13.2 The information presented concerning the proposed operating organization for initial plant startup is inadequate for our evaluation. Provide further details with regard to this group including an organization diagram that shows the relationship of the vendors and other support personnel to the AP&L operating organization, the number of shift personnel, technical support and contractor advisory personnel, and designate which positions will be staffed by licensed operators. Describe the specific duties, responsibilities and authority of each member of this initial operating organization. Indicate the method and timing of transition to the normal plant operating force.

13.3 Provide resumes for all contractor and vendor personnel identified as serving in the startup organization in 13.2 above, including position, education, training and experience.