UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

JUN 30 1972

Docket No. 50-313

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

Dear Mr. Phillips:

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

In order to maintain our licensing review schedule we will need a completely adequate response by August 21, 1972. Please inform us within seven (7) days after receipt of this letter of your confirmation of the schedule or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our requests it is highly likely that the overall schedule for completing the licensing review for this project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the extent of extension will most likely be greater than the extent of delay in your response.

Sincerely,

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

Enclosure:
Request for Additional Informatio

cc w/encl:
Horace Jewell, Esq.
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1550 Tower Building
Little Rock, Arkansas 72201

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REQUEST FOR ADDITIONAL INFORMATION
ARKANSAS POWER AND LIGHT COMPANY
ARKANSAS NUCLEAR ONE - UNIT 1
DOCKET NO. 50-313

The requests for additional information listed herein are numbered by FSAR section in sequence with those previously made by our letters of November 1, and December 13, 1971 and April 6, 1972.

1.0 INTRODUCTION AND SUMMARY

1.9 It has been indicated in Section 1.7 of the FSAR that a barrier wall may be needed to close off the end of the Unit 1 control room if Unit 1 is ready to operate and the Unit 2 control room is not yet built. The FSAR specifies a concrete block temporary wall for this purpose. Explain the considerations which led to the choice of concrete block rather than some other form of barrier. Discuss, in particular, the tornado resistance and seismic qualities of this temporary barrier.

1.10 The organization chart (Figure 1-19) and the responsibilities of each department should be clarified as follows:

a. Define the geographical locations of the personnel and committees listed on Figure 1-19. Indicate whether the QA Committee, the Chief QA Coordinator, and the QA inspectors are located and their functions are performed at the Arkansas Power and Light Company home office or at the facility site.

b. Who prepares and approves the QA Program and the QA Procedures? How are these documents controlled and how are other affected departments made aware of them?

1.11 According to Figure 1-19 and the write up of responsibilities, the QC Engineers and the personnel performing inspections report to the Technical Support Engineer, therefore:

a. What other functions does the Technical Support Engineer perform?

b. What direct access do the QC Engineers and the personnel performing inspections have with the Chief QA Coordinator and upper management concerning QA and QC problem areas?

1.12 Describe the qualifications and training requirements for each position within your QA organization. Specify the approximate number of Quality Assurance Inspectors, Quality Control Engineers and inspection personnel required to implement the QA/QC program.

1.13 How will QA/QC participate in the Safety Review Committee and the Plant Safety Committee during plant operation? Explain in detail how the Quality Assurance Committee devises and implements a program to ensure that the plant is operated at an acceptable quality level throughout its operational life.

1.14 Will there be any subcontractors assisting in the activities of operating, maintaining, repairing, and modifying the facility? If so, define Arkansas Power and Light's position regarding use of such subcontractor organizations within the overall QA program. Specify the operational and organizational interfaces and procedures used to control interface decision making.

1.15 Describe the scope and depth of the QA Program for Operations.

1.15.1 If the program is not currently complete, indicate by a schedule when each portion of this program will be available.

1.15.2 Specifically identify those management policies, procedures and instructions to assure consistency and implementation of the QA/QC Program requirements. If these documents have not already been prepared, provide a schedule for their preparation. Describe how each of the document groups (policies, procedures, and instructions) are reviewed, approved, revised, distributed and controlled. Indicate how AP&L assures that applicable department personnel and subcontractors personnel implement these documents.

1.15.3 Define and describe the expected QA/QC activities during the operating license and what steps will be taken to assure that these activities meet the QA Program objectives.

1.15.4 Describe how and when AP&L reviews and audits the status and adequacy of their QA programs and QC procedures.

1.16 Provide a more definitive description of the design controls employed during plant operation and the associated QA requirements.

1.17 Provide a more definitive description of the Procurement Document Control activity. Describe the procedures which control this activity and the organization responsible for its implementation.

1.18 Define and describe all QA/QC activities at AP&L and the subcontractor's facilities which will be controlled by instructions, procedures and drawings.

1.19 Describe the procedures which assure that instructions, procedures and drawings and their revisions are reviewed for adequacy, approved, released and distributed. Indicate the disposition of obsolete documents. Who will perform these functions?

1.20 Describe how AP&L assures:

a. That purchased material meets procurement requirements both at source and at delivery.

b. That subcontractors or vendors have been selected properly.

c. That purchased material will not be installed or used until the required acceptance documents are at the site.

Describe the documents required to justify evidence of acceptance and the requirements for approval and control of the acceptance documents both at the subcontractors and vendor's facility and at the nuclear power plant.

1.21 Describe AP&L's method for identifying and controlling material, parts, components, structures, and systems starting from the raw material stages, thru the fabrication inspection, installation and operational life of the plant. Define the degree of identification (e.g., heat number, part number, serial number, etc.). Describe AP&L's method of identifying and controlling obsolete, defective and/or scrapped items and the measures which assure that these items are not installed or used in the plant.

1.22 Describe the anticipated special processes to be performed during the operational life of the plant. Describe the procedures which are required to control special processes and the qualification requirements of the personnel performing these special processes.

1.23 Describe the inspection operations which are expected to be required during the operational life of the plant and AP&L's procedures for generating inspection plans. Indicate:

a. Who prepares, reviews, approves and revises these plans?

b. By whom and how are inspection plans controlled and implemented?

1.24 What are the qualification requirements for the dimensional, ultrasonic, liquid penetrant, and radiographic inspectors in the event such services are required during the operational life of the plant?

1.25 Define what test programs will be developed and describe what QA/QC responsibilities will be required in the preparation, review, approval, control and implementation of such programs.

1.26 Provide the following information on control of measuring and test equipment:

a. Specify what type of measuring and testing devices will be used, controlled and calibrated under the QA program.

b. Describe the procedures that: (1) control the use and calibration of these devices both at AP&L's subcontractor's and vendor's facilities, (2) alerts QC that a particular item is due for calibration, and (3) prevents an item which is out of calibration from performing a particular service during the operational life of the plant.

c. Specifically state the calibration specifications used by AP&L and subcontractor personnel.

1.27 Describe AP&L's plans for generating procedures to control the handling, storage, shipping, cleaning, and preservation of material and equipment required during facility operation. Who is responsible for preparing, reviewing, approving, implementing and controlling these procedures?

1.28 Describe AP&L's and the subcontractor's procedures for marking (e.g., stamps, tags, etc.) (a) the inspection and test status on components, systems and structures and (b) the operating status of structures, systems and components during preoperational and operational stages. Who is responsible for tagging and controlling inoperative or malfunctioning components in such a manner that they will not be inadvertantly used?

1.29 Describe the procedures which establish requirements for the control, the review, the disposition, the required approvals of the disposition and the segregation of discrepant components structures and systems. Specify who is responsible for these activities and how affected personnel and/or organizations are notified of the discrepancy.

1.30 Present an outline of the organizational arrangements for the review and approval of discrepant hardware. If a subcontractor and/or vendor performs the above function, particular emphasis should be placed in describing the interface of the review, disposition and approval cycle between AP&L and the subcontractor and/or vendor.

1.31 Describe the procedures establishing the requirements for corrective action. Specify the type of discrepancy or malfunction requiring corrective action; the organizations responsible for the review and approval of the cause and corrective action. Provide the requirements for monitoring of components, structures, systems and activities to determine the need for corrective action. Indicate what management responsibility and involvement is involved in corrective action. If a subcontractor or vendor performs the above activity particular emphasis should be placed in describing the interface of the review and approval cycle with the AP&L organization.

1.32 Define specifically what Quality Assurance records will be maintained and filed at AP&L's, subcontractor's and vendor's facilities and the duration of record retention. Describe the procedure for record retention control with particular attention given to the following:

a. Who is responsible for controlling the record filing system?

b. Will there be a controlled access to these records and what assurance is there to prevent records from being misplaced or lost?

c. Define the method of identifying and retreiving records after they are filed.

1.33 Describe the audits to be performed to verify compliance with the QA program over the operating lifetime of the facility.

1.34 Describe the procedure for implementing audit activities with particular attention given to the following:

a. Who is responsible for preparing, reviewing, approving and implementing the audit procedures?

b. What determines the frequency with which audits are performed?

c. How are the results of audits documented and to what level of management are the results distributed?

d. What is upper management's responsibility in regard to deficiencies reported by audits?

e. How are corrective actions to deficiencies (as reported by audits) implemented and documented?

f. Where will the results of audits to be filed?

The definition and description of these audits policies and procedures should be detailed to the extent that it can be clearly understood what and where audits will be performed by the AP&L, subcontractors, and vendors and the responsibilities and interfaces between the responsible organizational departments.

2.0 SITE AND ENVIRONMENT

2.7 We understand that your new meteorological measurements program started in July 1971. Provide a description of this preoperational program and your proposed operational program including the type of measurements, locations and elevations of measurements, characteristics of the instruments, calibration and maintenance of instruments, data output and recording systems, and data analysis procedures.

As soon as possible after completion of one year of operation of the new meteorological measurements program, provide joint frequency distributions of wind direction, wind speed and atmospheric stability as determined from appropriate ΔT measurements. (See Safety Guide 23 for guidance). Specify percentages of data recovery for the period of record. Where periods of missing data are of days duration (as opposed to sporadic durations of a few hours at a time), specify the periods of missing data. Present evidence as to how representative is the period of data collection with respect to long period climatology.

3.0 REACTOR

3.3 The design and analysis of the fuel rod cladding is based on the fuel rod being initially pressurized with helium to about 400 psia. Describe the analysis which shows that the rod is adequately pressurized during the fuel rod assembly procedure.

3.4 Section 3.2.4.2.1 of the FSAR discusses the B&W High Burnup Irradiation Program for the fuel and indicates that the results thereof will be available prior to operation. Provide these results or a schedule of their availability.

3.5 Describe the dynamic system analysis methods and procedures which will be used to determine dynamic responses of reactor internals and associated Class 1 components of the reactor coolant pressure boundary which have an effect on the responses (e.g., analyses and tests). The preoperational test program as related to Safety Guide 20, Vibration Measurements on Reactor Internals should be described. If elements of the test program differ substantially from the guidance of Safety Guide 20, the basis and justification for these differences should be submitted.

3.6 Discuss the quality assurance programs and quality control checks that are designed to assure the mechanical integrity of your fuel over its anticipated lifetime including any design review effort, review and audit of quality assurance measures and your planned inspections of the fuel upon delivery. Indicate how your fuel design and manufacture will minimize possible failures from clad hydriding, UO_2 - clad interactions, and clad collapse.

4.0 REACTOR COOLANT SYSTEM

4.27 This request for information, which is related to inservice inspection of the reactor vessel and nozzles, was made in our letter of November 1, 1971. Your response, in Amendment 26 of April 21, 1972, indicated that you had not yet selected a vendor and equipment for this inservice inspection. A complete response to this request is needed for us to complete our evaluation.

4.28 The response to Request 4.20 contained in Amendment 25 requires further amplification.

4.28.1 Provide a description of the reactor coolant pump flywheels including drawings of the principal parts and assemblies with identification of dimensions and materials.

4.28.2 Provide a description of the flywheel assembly procedures including the quality controls used. The magnitude of bolting stresses and flywheel/spider interference fit should be discussed.

4.28.3 Provide a stress and fracture mechanics analysis which determines the relationship of critical crack size to the rotational speed of the flywheels. Cracks emanating from the bore radius or from other regions of possibly higher stress concentration (keyways or boltholes) should be considered.

4.29 Your analysis has assumed that the anti-reversing device on the reactor coolant pump precludes overspeed in the reverse direction due to a guillotine break in the pump suction. Provide a description and depiction of this anti-reversing device with an explanation of why you consider it capable of resisting the forementioned reverse blowdown.

4.30 The ASME Boiler and Pressure Vessel Code Section III requirements on fracture toughness have been revised recently. Consequently, to evaluate the adequacy of the proposed heatup and cooldown limits for this plant, provide the following information:

a. Regarding the fracture toughness data obtained for all pressure-retaining ferritic materials of the reactor vessel, state the degree of compliance with the acceptance criteria of the recently revised ASME Code Section III fracture toughness rules (Code Case 1514). These rules require determination of the following for the reactor vessel plates, forgings, and qualification welds:

(1) NDT temperatures obtained from dropweight (DWT) tests, and

(2) Temperatures at which "weak" direction Charpy V-notch

specimens exhibit at least 0.035 in. lateral expansion and not less than 50 ft-lbs absorbed energy.

b. In addition, for the materials of the reactor vessel beltline (including welds), provide the initial upper shelf fracture energy levels, as determined by Charpy V-notch tests. (In both directions, if available).

c. Provide proposed operating limitations during startup and shutdown of the reactor coolant system, using as a guide Appendix G, "Protection Against Non-Ductile Failure," of the recently revised ASME Code Section III fracture toughness rules (Code Case 1514).

d. The extent to which you have reviewed the design of affected systems and components to determine that annealing of the reactor vessel will be feasible should it be necessary because of radiation embrittlement after several years of operation. State the maximum reactor vessel temperature that can be obtained using an in-place annealing procedure.

4.31 Supplement the response to Request 4.3, Amendment No. 23 of the FSAR, by describing the acceptance program that will be implemented to determine the acceptance amplitudes of the vibration for confirming the structural integrity of the piping and pipe restraints.

4.32 Provide the dynamic testing procedures used in the design of Class I mechanical equipment (such as fans, pump drives, valve operators, heat exchanger tube bundles) to withstand seismic, accident and operational vibratory loading conditions, including the methods and procedures employed which consider the frequency spectra and amplitudes calculated to exist at the equipment supports. Where tests or analyses do not include evaluation of the equipment in the operating mode, describe the bases for assuring that this equipment will function when subjected to seismic accident loadings and vibratory loadings.

4.33 Clarify the response to Requests 4.4 and 4.5 by providing a summary of the dynamic analyses performed for Class I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break, including:

a. The locations and number of design basis breaks on which the dynamic analyses are based.

b. The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.

c. Descriptions of the forcing functions to be used for the pipe whip dynamic analyses. Include direction, rise time, magnitude, duration and initial conditions that adequately represent the jet stream dynamics and the steam pressure differences.

d. Typical diagrams of the mathematical models used for the dynamic analysis.

e. A summary of the analyses performed to demonstrate that unrestrained motion of ruptured lines will not sever adjacent impacted piping or pierce impacted areas of containment steel wall (or liner).

4.34 Describe the analytical methods used to evaluate stresses (e.g., elastic or inelastic) including a discussion of their compatibility with the type of dynamic system analysis. If inelastic component stress analyses and inelastic design stress limits are used in conjunction with an elastic dynamic system analysis, the bases upon which such design procedures may be justified should be discussed.

4.35 The response to Request 4.9 contained in Amendment 23 states that the pressurizer safety valves are specified to meet ASME Section III Article 9. This Article addresses only the functional requirements of safey valves. Provide the design criteria, such as stress and deformation limits, used to prove the structural adequacy of the safety valves.

4.36 Provide clarification of the response to Request 4.10 contained in Amendments 22 and 23 as to whether the design criteria, which have been used to take into account full discharge loads imposed on valves and on connected piping, consider the valves discharging concurrently.

4.37 Section 4.2.3.8 discusses detection of leakage from the primary system by radiation monitoring of the air in the reactor building. Provide information to indicate that an air particulate monitor will be included in this leakage detection system.

5.0 STRUCTURES

5.66 What precautions are being taken in the design and construction of the Unit 2 Auxiliary Building to assure that the seismic integrity of the Unit 1 structures are not compromised, especially when Unit 1 is operating during Unit 2 construction?

5.67 It is our understanding that you have experienced difficulties in the prestressing of the reactor building tendons, including breakage of the stressing jacks. Provide an evaluation of this problem with respect to the system design and expected service. Confirm that no damage to the tendons or structure occurred when the jacks broke.

5.68 The answers to Requests 5.9 and 5.42 do not cover the design for seismic torsion of seismic Class I symmetrical framed structures. Indicate the methods followed for the analysis and design of such reinforced concrete and structural steel structures. Provide a listing of these structures and indicate, for critical columns and girders, the increase of maximum stresses due to the seismic torsional load.

5.69 An amplification of your response to Request 5.19 is needed. For all seismic Class I structures which have been designed by using computer programs, indicate the following:

a. A listing of these structures.

b. Location of sections for which equilibrium checks have been made.

c. Results of equilibrium checks. Indicate the stress resultants computed, the stress resultants computed by hand and the corresponding resultants of the exterior loads at same location.

5.70 In your response to Request 5.20, it is indicated (page 5-15) that the maximum predicted principal tension in concrete, where shear reinforcing is not provided, for governing factored load combination is 181 psi. Indicate the corresponding tension for the same loading combination, but with all load factors equal 1.0.

5.71 In your response to Request 5.22, you discussed the radial reinforcing in the Reactor Building dome. The recommendations of ACI Committee 349 and the draft of the Joint ACI/ASME Code indicate that radial reinforcing should be provided wherever prestressing tendons are curved. Since this condition exists not only in the dome but also in the wall, justify the omission of this reinforcing in the wall. The high local tensile stress concentrations existing at the tendon ducts (hole effect) should be considered.

5.72 An amplification of your response to Request 5.30 is needed. For the Reactor Building interior structure indicate the differential design pressure and the design jet forces separately for:

a. The reactor cavity, lower part

b. The reactor cavity top and refueling canal

c. The steam generator cavity

d. The pressurizer cavity

e. The operating floor

Indicate the maximum stress in concrete and reinforcing and the corresponding locations. If the structure is allowed to go beyond the elastic range, indicate the ductility factor.

5.73 Demonstrate that the probability of a tornado missile damaging a penetration is negligible or provide missile protection for penetrations, since damage to a penetration may breach the containment.

5.74 In your response to Request 5.40, the seismic stress contributions in percentage of the total stresses are higher than expected for the reinforcing bars. Indicate the numerical values of the total stresses and, separately, the value of seismic stresses for cases (a) and (b). Also provide the same information for concrete.

5.75 Your response to Request 5.41 does not cover the case of concrete cracking due to shrinkage and thermal stresses. Discuss this case.

5.76 Your Response 5.48 does not answer our request which bears on welding of reinforcing bars or any other welding performed in the neighborhood of prestressing tendons before or during their erection.

5.77 Since the proposed Reactor Building acceptance testing deviates from Safety Guide #18, (your response to Requests 5.52 and 5.63) demonstrate that the offered acceptance testing will give information equivalent to the information which would be provided by testing following Safety Guide #18 without reference to the testing of any other plant.

5.78 Figure 5J-4 shows a stressing washer with stepped drilled holes for prestressing wires: the holes are drilled to 17/64" ϕ in front and to 11/32"ϕ in the back. During the CP review, stressing washers which had only smooth holes with uniform bores were approved. Demonstrate that the stepped holes did not damage the wires during erection.

5.79 The response to Request 5.1 contained in Amendment 23 is not satisfactory. An inadequate description of the mathematical models used for the seismic analysis of Class I structures, systems and components has been furnished. Provide a more detailed description of the mathematical models used including sketches where appropriate.

5.80 The response to Request 5.4 contained in Amendments 23 and 24 is not satisfactory. Provide details of the procedures used to incorporate soil-structure interaction in the seismic analyses of Class I structures. This should include a brief description of the methods, mathematical model and damping values (rocking, vertical, translation and torsion) that have been used to consider the soil-structure interaction.

5.81 Provide the basis for establishing the actual damping values which will be used to verify the response of different types of Class I systems and components under seismic excitation.

5.89 Discuss how the proposed containment leakage testing program complies with Appendix J of 10 CFR Part 50, Reactor Containment Leakage Testing for Water Cooled Power Reactors. Include a description of the test method(s) to be used and the associated leakage monitoring system(s), and a discussion of the expected accuracy of the test results.

6.0 ENGINEERED SAFEGUARDS

6.8 Discuss how a failed filter in the reactor building purge system would be cooled to prevent the exposed charcoal filter from reaching its ignition temperature. Provide a schematic of the purge system showing this capability.

6.9 Discuss the shielding design bases for the reactor building purge system and the accessibility of system components for maintenance under post-DBA conditions.

6.10 The statement is made in Section 6.6 of the FSAR that the hydrogen purge system is designed to limit the buildup of hydrogen in the reactor building to an average concentration of 4 $^{v}/_{o}$ (volume percent). A control limit lower than 4 $^{v}/o$, e.g., 3.5 $^{v}/o$, should be established to provide a margin between the lower flammability limit (see AEC Safety Guide No. 7) and the control limit. This margin should account for inaccuracies in the hydrogen monitoring system, incomplete mixing of the hydrogen in the reactor building, and should provide for the capability to interrupt or delay purging for a period of time. Therefore, provide the following information:

a. Specify and provide the justification for a hydrogen control limit that is below the lower flammability limit.

b. Provide a graph of post-accident hydrogen concentration as a function of time, showing the proposed duty cycle for the purge system.

6.11 Although the post-DBA hydrogen generation rate has been calculated in accordance with Safety Guide No. 7, it is not possible to determine from the information given that all potential sources of hydrogen have been considered. Therefore, provide the following information.

a. Identify the sources of hydrogen considered in the analysis.

b. Provide a listing of the aluminum components within the reactor building, as well as the mass and surface area of these components. Identify the component surfaces that have been painted. Discuss the assumptions made regarding the corrosion of the aluminum components.

c. Discuss how the use of zinc base paint has been factored into the analysis of hydrogen generation.

6.12 Describe the reactor building atmosphere monitoring system to be used in conjunction with the hydrogen control system, and address the following design considerations: the location of sampling lines within the reactor building to determine local hydrogen concentrations and the degree of atmosphere mixing, the isolation valving on the sampling lines and the return line, the provisions for filtering and drying the system flow stream, the monitoring equipment and readout locations, the hydrogen concentration measurement principle of the equipment, the expected accuracy and error band of the monitoring equipment, plans for equipment qualification tests, and the constraints on system operation (e.g., radiation level and moisture content of flow stream, and reactor building pressure). Will the reactor building atmosphere monitoring system be capable of performing its intended function without relying on purge system operation?

6.13 With respect to the reactor building spray system, provide the following design and performance information:

a. A description of nozzle design.

b. A discussion of the potential for nozzle clogging.

c. The expected spray drop size spectrum under design basis conditions.

d. The pressure drop across the spray nozzles under design basis conditions.

e. The flow rate per header for the injection and recirculation phases of system operation.

f. The fall height of the spray, and the spray heat removal efficiency.

g. The effective reactor building volume spray coverage.

h. The system design provisions to minimize spray trajectory overlapping.

6.14 Provide the rationale for establishing an actuation signal for the reactor building spray system of 30 psig. Discuss the delay times that are inherent in bringing the system into service.

6.15 Describe the procedure for transferring the spray system pump suction from the BWST to the reactor building sump. What information will be available to the operator to guide him in making a timely decision? Provide a malfunction analysis of the circuitry to assure that no single electrical fault will prevent the transfer to the recirculation mode from being made.

6.16 Under post-DBA conditions, the reactor building sump may contain debris that could be drawn into the systems taking suction from the sump unless preventive measures are taken. Therefore, describe the protective screen assembly that will be used to prevent debris, capable of clogging the reactor building spray system nozzles, from entering the sump suction piping. Provide assurance that the failure of a portion of the assembly will not negate the effectiveness of the entire assembly. Specify the screening surface area that will be available. Provide drawings showing the physical arrangement of the assembly within the sump.

6.17 Discuss the capability of the reactor building cooling system to thoroughly mix the building atmosphere, including the air within building compartments. Under post-DBA conditions, what effect would opening of the bypass dampers in the ventilation units have on the system capability?

6.18 Provide a more detailed schematic of the reactor building ventilation system showing the locations of the ventilation units within the reactor building and the routing of the air flow guidance ductwork. Discuss the design provisions to assure that the ventilation system ductwork remains intact following a design basis accident.

7.0 INSTRUMENTATION AND CONTROL

7.22 The response to Request 7.3 refers to the B&W Topical Report BAW-10003. This report has been revised to respond to AEC comments. Your letter of June 20, 1972 indicated acceptance of this revision. Your response to this question should state whether your equipment designs are consistent with this latest revision.

7.23 The information provided in response to Request 7.4 is not complete with regard to cable tray fill limitations, separation distances, restrictions on use of control room, cable spreading area to house fluid system and high power equipment, and spacing of wiring and components in control boards, panels, and racks. Your response should be modified to include this information.

7.24 Include in your response to Request 7.5 the length of time the reactor building pressure transmitters were subjected to the environment specified. Also indicate the schedule for completion of the tests of the reactor building coolers.

7.25 The information presented in response to Request 7.6 should discuss all equipment rooms to the same detail as that for the Control Room on Pages 9-29 and 9-30 of the FSAR.

7.26 Your response to Request 7.10 does not include secondary system instrumentation such as steam generator level, pressure, etc. Complete your response in this regard.

7.27 Confirm that your instrumentation design includes the feature of indicating bypasses at the system level whenever a component of that system is placed in a nonoperating condition for test or maintenance operation. The ESAS design is of particular interest. Your response to Request 7.11 appears to indicate that your design has this capability.

7.28 Your response to Request 7.12 does not include the information requested concerning the sodium hydroxide subsystem. Complete your response in this regard.

7.29 The information presented in response to Request 7.15 identifies the reactor protection system high temperature trip set point as being within 1°F of the maximum end of the calibrated range of 620°F. The maximum instrument error was stated to be ± 1°F. We conclude that this instrument design lacks adequate margin and should be corrected. Your response should be modified to describe the change you propose in this regard. Also state the proposed indicated range of the temperature instrumentation.

7.30 Confirm that your response to Request 7.16 considered the reactor building pressure instrumentation.

7.31 Confirm that analyses similar to that presented in response to Request 7.2, have been performed on all other instrumentation that serve both safety and control functions. In this regard, modify your response to include the results of the other analyses.

7.32 Your revisions to the discussion of the penetration room filtration and hydrogen control systems do not include the requirement that the initiation and control circuits meet IEEE-279. State and justify your design intent in this regard.

8.0 ELECTRICAL SYSTEMS

8.8 Seismic testing of d-c systems should include tests of the batteries (cells) as well as the auxiliary equipment and battery racks. Provide the results of such tests.

8.9 Describe the circuit breaker interlocks provided to ensure against interconnection of redundant emergency buses. In particular, identify and discuss the interlock and lockout features and relate these designs to the recommendations of Safety Guide No. 6.

8.10 Describe the high voltage transmission system which connects to the switchyard and to the plant. Identify and describe common rights-of-way and state whether a single structural failure could result in a loss of off-site power to the engineered safety features.

8.11 Describe the checks to be made on the completed cable installation to verify that it is in accordance with the physical separation criteria for redundant cables.

8.12 We have evaluated your responses to Requests 8.1 and 8.2. Your responses take exception to IEEE-308 for the purpose of gaining some small additional power during your system peak demand periods. We have concluded that the independence of redundant standby power supplies (diesel generator) must be maintained during all phases of plant operation; therefore, the use of both diesel generators for peaking purposes must be avoided. The use of one diesel generator for peaking could, under certain restricted conditions, provide the independence required in IEEE-308 and GDC 17. Your responses to these requests should be amended to reflect these comments.

8.13 Provide a piping and instrumentation drawing (P&ID), an arrangement drawing, and description of the diesel fuel transfer and storage system in sufficient detail to enable an independent analysis to be performed to determine that the system is capable of withstanding the effect of a single active or passive failure.

8.14 The FSAR indicates that the fuel oil bulk storage tank is a non-essential fuel supply and is designed accordingly. However, the emergency storage tanks inventories are replenished from this tank by gravity and feed through buried pipeline. Describe the location and type of isolation capabilities provided for these lines. Discuss the potential for fire or flood waters reaching the emergency storage tanks through these lines assuming the bulk storage tank fails.

8.15 Describe the protection provided for the diesel generators to preclude damage from environmental effects, such as tornado missiles, flooding by fire protection system, and excessive room temperatures, so that their performance will not be mitigated when called upon.

8.16 The description and physical arrangement of essential subsystems for the diesels have not been described in sufficient detail to permit an evaluation of single events that will not disable all emergency on-site AC power systems. Provide specific information for the following subsystems: (1) the air intake structure and its filtering system, (2) lubrication and its filtering system, (3) cooling water and its source, (4) the fuel oil filtering system, and (5) the starting systems.

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.7 With regard to the Makeup and Purification System, provide the following information lacking in the FSAR.

9.7.1 The functional performance requirements for operation of the system under accident conditions.

9.7.2 Requirements for testability and inservice inspection of the system.

9.8 The high pressure injection engineered safety system is an integral part of the Makeup and Purification System. Provide the following information for the dual function performance of equipment and piping.

9.8.1 Identify the safety related portions of the system that are necessary for safe shutdown including components, piping and valves.

9.8.2 Identify all safety related instrumentation and controls associated with the system and describe the means for actuation and the redundancy provided to assure safe shutdown.

9.8.3 Provide an evaluation of the effects of the seismic Class II equipment and component failure or malfunction on the siesmic Class I portion of the system.

9.9 The letdown flow rate during normal reactor operations will vary between 45 and 140 gpm, provide a description of the indications available that the reactor operator uses to determine these flow rates. Also provide the bases and description of indicated data for determining when the letdown flow should or should not be diverted to the waste disposal system.

9.10 In addition to the normal demineralization and filtration system, the makeup flow can be diverted through a pre-filter. Provide the bases for determining when the operator will use this pre-filter.

9.11 During abnormal reactor system conditions what determines the letdown flow rate? Does the purification system process the letdown flow or is a bypass situation achieved?

9.12 The letdown temperature in the letdown line downstream of the coolers is alarmed and provides an interlock for isolation to protect the purification system. Is the letdown temperature always indicative of the pressure associated with the letdown system? Discuss the effects that the interlock failure would have on the purification system. How would excessive temperature and pressure otherwise be detected?

9.13 Letdown flow rates are controlled by a fixed block orifice, a parallel remotely operated valve, and a second manually positioned valve also is parallel. Discuss the operation of these valves and describe the associated conditions required for operation. Consider also the effects on letdown flow and system pressure with either one of the valves open and with both valves open.

9.14 In addition to the normal makeup line, two alternate paths for adding boron to the reactor coolant system have been identified (Section 9.1.2.5). Determine the limiting condition for boration and provide the margin associated with the alternate injection method to maintain subcriticality during reactor cooldown.

9.15 Provide an evaluation of the effectiveness of the purification system to provide adequate cleanup capabilities. Include a discussion on the sluicing and backwashing operations, resin traps, the amount of radioactivity released to the radwaste system.

9.16 Provide a description, evaluation, and drawing of the Potable and Sanitary Water Systems. In your analysis consider the effects of: (1) the systems failure on safety related systems or components, (2) the means provided to preclude contamination of the system by radioactive materials, flammable materials, and chemicals, (3) describe the consequences of this contamination, and (4) determine the effects and adequacy of the system if shared between facilities.

9.17 Process sampling lines are connected to seismic Class I systems for sample purposes. Describe the codes and standards applicable to the sampling system and state the precautions taken to assure that failure of the sample line connections will not effect the integrity of the seismic Class I system.

9.18 Provide an evaluation to demonstrate that the sample taken is a true representative sample of gases, particulates or liquids and that the sampling and handling does not represent a hazard to the safety of the plant personnel. Have alternate paths been provided to obtain a sample from the reactor system or containment during accident conditions?

9.19 Figure 9-4 identifies two additional chemical addition systems that are not described in Section 9.2 of the FSAR. Provide a description and evaluation for the sodium hydroxide and nitrogen addition systems.

9.20 Describe the isolation capabilities of the chemical addition system. In the event of an uncontrolled release due to pipe or tank failure, determine what effects the release of the various chemicals and radioactivity would have on other systems and on the safety of the plant personnel working in the vicinity of the release.

9.21 Heat tracing in the chemical addition system is not redundant. Justify this aspect of your design. In your justification describe the effect of the loss of heat tracing on the system, the manner in which failures are detected, and the time required for corrective action to preclude an unsafe condition.

9.22 Provide an analysis for the chemical addition system to determine the effects of system malfunction or failure to operate related to the control of reactor coolant chemistry and shutdown margin.

9.23 Describe the requirements for the detection and control of leakage, inservice inspection, and testing of the chemical addition system.

9.24 Provide a description, evaluation and drawing of the equipment and floor drainage system. The information provided should include sump locations, capacity, sump pump redundancy and pumping capacity, instrumentation and alarms, leak detection measurements taken, radioactivity surveillance and potential for the inadvertent transfer of contaminated fluids.

9.25 Section 9.3.2.1 provides information pertaining to service water system cooling requirements during normal and emergency operations. Figure 9-6 does not appear to be applicable to either condition. Provide a service water system P&ID for each condition stated in the above referenced section. Also check Figures 9-7 and 9-8 for completeness of equipment listed.

9.26 Provide the following information with respect to the service water system:

a. Describe the flood protection provided for the service water pumps (intake structure). Consider individual compartments, elevation, access opening, and other protective features.

b. In response to Request 9.2.3, Table 9-9.2.3 indicated that the minimum NPSH for the service water pumps is 3.9 ft. Provide the pump characteristic curve. Specify the NPSH available during extreme low reservoir level conditions (severe drought conditions).

c. Identify any seismic Class II equipment or components located within or adjacent to the seismically designed Class I service water system and determine their effect on the system due to failure.

d. Provide the requirements for testability and inservice inspection for the service water system.

9.27 The design bases for the condenser circulating water system states that "if a steam generator has developed a leaky tube, one circulating water pump can be manually connnected to an unloaded essential electrical power bus to permit steam bypass to the condenser in lieu of direct release to the atmosphere during plant cooldown." Define unloaded essential electrical power bus. What is the bases for determining the use of this procedure? Describe the procedure including location, time required for hookup, and functional performance with use of one pump. Also explain what is meant by the following statement: "the cycle cooldown requirements can be adequately handled by other means" (Section 9.3.1-d).

9.28 Provide a description and evaluation of the condensate storage facility including drawings. The information provided should discuss environmental design considerations, requirements for leakage control, limits of radioactivity concentrations, code design requirements, and capability to provide a minimum supply of condensate for emergency purposes.

9.29 Describe the design features that protect against the possible release of radioactivity to the service water system during normal shutdown if there is a leak in the decay heat removal coolers.

9.30 With respect to the sizing of the decay heat removal (DHR) system, both coolers and pumps are required to cool reactor coolant temperature from 280°F to 140°F in 14 hours. What is the time requirement for one pump and cooler. The DHR system is also used to maintain reactor coolant and fuel transfer canal refueling water temperatures at 140°F during refueling operations. What are the requirements for pumps and coolers under these conditions. Assuming only one is available, determine the temperature of the reactor coolant and the refueling water and describe how it would affect the refueling operation.

9.31 One pump of the decay heat removal system is used to perform other operations at various times (such as filling the transfer canal, pumping water back to storage tank). Describe these operations in detail and discuss the effects of these operations on maintaining safe shutdown conditions.

9.32 During cooldown of reactor coolant from 280°F to 140°F, the pressurizer is cooled by spray from the decay heat removal system. Discuss the effects of a single failure in the single spray line indicated on Figure 9-12.

9.33 The Borated Water Storage Tank (BWST) is located outside the reactor and auxiliary buildings. In light of the safety function of this tank, provide the following additional information: (a) discuss the effects of a BWST heater failure, (b) describe heat tracing requirements for the system, (c) provide the limits for radioactivity concentration in the tank, (d) describe the requirement for leak detection and leakage control, and (e) describe inservice inspection requirements.

9.34 Provide a description of the new fuel storage racks and the new fuel storage area (vault). Include in the description: (a) types of material used in construction, (b) all external design loading considered in their design, (c) codes and standards governing their design, and (d) location of the storage vault in the station complex. Provide an evaluation of the measures taken to preclude flooding and the effect of adjacent equipment failure.

9.35 With respectto the spent fuel storage facility provide the following information:

a. Provide the degree of subcriticality provided by fully loaded spent fuel storage racks when stored in both borated and unborated water.

b. Describe the spent fuel storage racks and their arrangement in the storage pool.

c. Describe the ability of the storage racks to withstand all external loads and forces, the impact forces due to dropped objects, and design codes and standards complied with.

d. Describe the minimum shielding requirements provided by the spent fuel pool and the provision incorporated into the design to assure that the minimum requirements will be met.

e. Describe the instrumentation and controls provided for the spent fuel pool, specifically for radiation, water level, and component failure. Provide the level at which these alarms are actuated and describe the action taken for each should they alarm.

f. Describe the means provided to detect and control leakage from the spent fuel pool.

g. Describe the methods used to assure that material compatability requirements are met for the storage facilities and the handling equipment.

9.36 Section 9.4.2.2 states that if all cooling is lost to the spent fuel pool, the time required for the pool to reach 205°F for 1/3 core and 1-1/3 cores is 19 hours and 5 hours, respectively. Specify the arrangments that would be made during that time to provide emergency cooling.

9.37 For each of the spent fuel storage conditions listed in Section 9.4.2.1.2 of the FSAR, provide a heat load, temperature vs. time figure for each.

9.38 Describe the design provisions that have been incorporated into the spent fuel cooling system to preclude coolant being lost due to failure of the cooling system.

9.39 Discuss the ability of the spent fuel cooling system to maintain uniform pool water conditions (such as, water temperature including mixing ability, control of fission and corrosion products, and pool clarity).

9.40 Section 9.4.2 of the FSAR describes the pumps associated with the spent fuel cooling system; Figure 9-11 does not identify these components with the same nomenclature. Clarify this apparent discrepancy.

9.41 Describe the procedure and the associated pumps, piping and valves used to supply the spent fuel pool with a seismic Class I makeup source from the borated water storage tank and the emergency pond.

9.42 Describe the procedure for filling and draining the fuel transfer canal and instrumentation tank. Identify the pumps, piping, and valves necessary to accomplish this operation.

9.43 The borated water recirculation pumps can be used to circulate borated water storage tank and fuel transfer canal water through the spent fuel pool demineralizer. State the conditions when this procedure will be used.

9.44 Provide a list of all tools and servicing equipment necessary to perform the various reactor vessel servicing and refueling functions and indicated whether each is designed to seismic Class I requirements.

9.45 Provide a discussion, with the aid of drawings, of the principles of operation of the spent fuel shipping cask crane and the containment crane plus any special handling fixtures employed in handling their respective loads such as the reactor vessel head, reactor vessel internals and the spent fuel shipping cask. Describe in detail the applicable codes and standards used in the design, fabrication, installation and testing of the crane, rails, supporting structures, bridge, trolley, hoists, cables, lifting hooks, special handling fixtures and slings. For each, list its design load rating, preoperation test load, maximum operating loads and the test loads that will be used throughout the life of the facility.

9.46 Describe the modes of failure that were considered in the design of the spent fuel cask crane and containment crane such as breaking of cables, lifting slings, sheared shafts, keys, stripped gear teeth, and brake failures. Also discuss the limitations and control that will exist in handling objects over an opened reactor vessel.

9.47 Assuming the maximum drop height, discuss with the aid of drawings where appropriate, the consequences of dropping the following:

a. The reactor vessel head on to an opened reactor vessel.

b. The upper core barrel assembly in an opened reactor vessel.

9.48 Provide the design bases for the air treatment system for the control room and auxiliary building, including (1) temperature requirements, (2) requirements for radiation protection and monitoring, and (3) environmental design requirements.

9.49 Provide a description of the ability of the control room ventilation system to detect air-borne contaminants, specifically smoke and radiation, and preclude their admission to the control room. Include in the description the detection methods, closure times of isolation valves, and time required to expedite the discharge of contaminants from the control room.

9.50 Provide a failure mode and effects analysis to demonstrate the ability of the control room and auxiliary building ventilation system to meet single failure criterion.

9.51 Describe the design parameters used to determine sizing of the normal and package air conditioning systems for the control room, computer room and any other area serviced by these systems.

9.52 Figure 9-13 indicates that the normal control room and computer room return air flow is recirculated. Justify this aspect of your design.

9.53 Describe the air supply to the control room, computer room, and electrical room as indicated on Figure 9-13. How does the redundant air supply train supply the control room? Describe the isolation capabilities of the power operated dampers indicated for the system. Describe the effects of emergency isolation of the control room on the air supply and return to the cable spreading and relay rooms.

9.54 Section 9.7.2 of the FSAR states that the normal air conditioning system is automatically de-energized for isolation upon signals from high radiation, excessive pressure drop through filters, and for fan failure. Provide the actuation level for each signal that implements isolation.

9.55 Discuss the seismic design classification of the Auxiliary Building Ventilation System and bases therefore.

9.56 Based on the information provided it appears that the fuel handling ventilation system does not meet single failure criteria. Discuss the effects of component failure with respect to the consequences of a fuel handling accident.

9.57 Confirm that the ventilation air from the fuel handling and radwaste area is continuously discharged through the filters and discuss the potential for clogging or overheating of the filters.

9.58 Section 9.7.2 of the FSAR states that two independent air supply units serve the radwaste ventilation area. Figure No. 9-13 does not confirm this design aspect. Clarify this apparent inconsistency.

9.59 Provide the bases for sizing the filter bank in the emergency ventilation system in the control room, the fuel handling area and the radwaste area in the auxiliary building.

9.60 Provide a process and instrumentation diagram of the Fire Protection System (FPS) which details the main loop and branch piping, valving, pumps, tanks and hydrant locations.

9.61 Specify the extent of coverage of the FPS as to plant areas protected, types of extinguishing agents, manual or automatic system actuation, types of detection equipment, and the location where detectors annunciate.

9.62 Provide the design parameters (capacity, operating pressure, head, type of drive, and automatic or manual start) of the fire pumps.

9.63 Provide specific reference to all applicable fire codes and standards to which the FPS has been designed.

9.64 Specify the seismic classification of the FPS components and piping. Discuss how the design assures that failure of any part of the FPS not seismic Class I will not damage or prevent fire protection to a Class I structure, system or component.

9.65 Discuss the construction techniques which have been used to minimize the potential for fires such as non-combustible material, fire walls and doors, and spatial separation.

9.66 Specify the primary and alternate sources of fire protection water and the maximum quantities required from each source.

9.67 Discuss the accessibility, with respect to radiation and toxic combustion products of all areas which rely on manual fire protection and identify these areas.

9.68 Discuss the extent to which the FPS can operate with any single failure.

9.69 For all tanks that contain gas under pressure (such as nitrogen, hydrogen, oxygen, air, and CO_2 tanks) provide the following: (1) the design and operating pressure, (2) the maximum pressure of the gas supply, (3) the location of the tank, (4) the maximum total energy if the tank should rupture, (5) the possibility of the tank or parts of the tank to act as a missile, (6) the protective measure taken to prevent a tank failure, and (7) the protective measures taken to prevent the loss of function of adjacent equipment essential for a safe shutdown condition.

9.70 Describe the functional performance requirements for the compressed air system under accident and normal conditions; include limiting condition of operation, operating pressures and temperatures, and quality or purity of air in terms of freedom from moisture, oil, and foreign particles.

9.71 The compressed air system capabilities do not include the ability to withstand the effects of a single active or passive failure. Provide a listing of all safety related equipment necessary for safe shutdown that the compressed air system supplies and describe each component function. Provide an analysis that demonstrates that in the event of system failure, each component will be in the fail-safe mode.

9.72 Describe the means provided for the detection and isolation capabilities of that portion of the compressed air system with leakage or malfunctions. Also describe the actuation and redundancy requirements for the instrumentation and controls of the system.

9.73 Describe the requirements for testing the automatic transfer of the standby compressors and any in-service inspection requirements subsequent to the initial inspection of the compressed air system.

9.74 Provide a description and evaluation for the lighting and communication systems for your facility. The discussion should include: (a) ability of the systems to operate under emergency conditions, (b) requirements for inspection and testing, and (c) ability to communicate with on-site personnel and off-site.

9.75 Provide the following information to substantiate your conclusions of the adequacy of the emergency pond:

a. Provide a centerline profile of the cooling pond spillway exit channel to Dardanelle Reservoir.

b. Describe the type of construction to be used for the emergency pond exit channel to Dardanelle Reservoir.

c. Based on the information available to us at this time, we do not agree that the spillway you propose for the emergency pond is adequate to pass the flood you have selected for design purposes without potential loss of the pond. We note the elevation and location of the channel sill downstream of the spillway crest could effectively reduce the outlet capacity of structure. Furthermore, we note that backwater from the spillway crest, exclusive of the effects of the downstream sill, could cause overtopping of the adjacent dike, and we can not agree that no freeboard on the dikes is acceptable. Provide inflow, outflow and pond level hydrographs for your design flood. Substantiate your spillway rating curve. Discuss the reductions in pond spillway capacity which can be caused by submergence effects of the downstream sill, and backwater effects between the crest and the pond.

d. Provide a topographic map of the emergency cooling pond and surrounding area which shows the location of the inlet and outlet structures illustrated on Figure 9.5.1-1, any exit channels, and the dikes. Describe the extent of the diversion dike to be located around a portion of pond periphery.

9.76 If one of the service water lines inside the reactor building is ruptured during a LOCA, diluting water from the emergency pond might be pumped into the reactor building at a very high rate. It is possible to isolate such a break in one of the two service water mains but there is little evidence of such leakage available to the operator. What means are available to ensure that the operator will detect a break of this type before it releases an unacceptably large amount of water into the reactor building?

9.77 In response to Request 9.3 regarding the fuel cask drop, you did not consider the dose rate to reactor operators in the control room. Estimate this dose rate predicated on the fifty-foot drop accident using a conservative source-detector distance of thirty feet. State all assumptions used in your calculations for this estimate.

11.0 RADIOACTIVE WASTE AND RADIATION PROTECTION

11.7 List (preferably on a diagram of the primary system) the concentration of N-16 in all piping and equipment outside of the reactor building at full power operations. In this regard, state the total linear footage of the reactor coolant letdown line and coolant velocity in this line in containment.

Show specifically the routing of this line into the purification system in the auxiliary building. Give the N-16 concentrations and shielding associated with this line in the auxiliary building.

11.9 What is the airflow and thermal neutron flux in the space between the standoff insulation and reactor cavity shield wall? What Ar-41 activity and subsequent whole-body dose rate to plant personnel during plant operation is expected from this source in containment? What dose rate is experienced if the highly enriched Ar-41 air within the standoff insulation leaks into the cooling stream?

How will the above argon problem be treated during a normal purge to keep effluents as low as practicable?

11.10 Discuss the programs you intend to implement during operation and shutdown to reduce in-plant personnel exposure. Some details on special shielding features of waste treatment components, filters, valves, might be appropriate. Cover specifically items related to:

a. The estimated contact and area dose rate of the control drive mechanism unit atop the reactor vessel.

b. Any program on the primary coolant circuit to study the generation and plate-out mechanism of radioactive crud with the objective of predicting, controlling or minimizing this source.

c. Any special maintenance or servicing programs: e.g., full-scale mockup of known troublesome zones of the steam generator requiring long repair time so that maintenance crews can most effectively simulate actual working conditions, anticipate problems and layout of temporary shielding.

d. The decontamination procedures anticipated for radwaste systems in general. Describe your decontamination plans for the once-through steam generators to accomplish testing, plugging or major repairs.

11.11 Discuss the capabilities of your proposed plant process monitoring system in light of the guidelines of AEC Safety Guide No. 21, "Measuring and Reporting of Effluents from Nuclear Power Plants." Discuss any capabilities which do not meet these guidelines.

11.12 Provide an analysis of the production rate, distribution and discharge of tritium from the plant. Discuss the methods that will be used to evaluate and account for the trituim contained in effluents from the plant. In addition, provide the following:

a. The uncertainties associated with estimating the amounts of tritium generated.

b. The special precautions to be taken during refueling with respect to plant personnel and releases to the environment.

c. In regard to b above, large dilutions may be required toward the end of life to reduce tritium content before pulling the reactor head. Show that your tankage could handle a cold shutdown and head pulling to accomplish the necessary dilution.

d. How do you intend to monitor the tritium vapor above the spent fuel pool?

11.13 Estimate the dose rate from all sources inside the proposed guard shack. Are visitor film badges issued or returned to this shack? If so, what precautions are being planned for accurate reading, etc.?

11.14 Estimate the dose rate from all sources to general areas in the administration building (e.g., the office area, lunchroom). Specify the sources and assumptions used in your estimate.

11.15 Can any liquid or gaseous wastes be discharged from the plant via a path which is not monitored by a radiation detector?

11.16 Figure No. 11-3 of the FSAR is inconsistent with your response to question 11.6 concerning the gaseous waste system. Figure 11-3 shows effluents from the waste gas decay tanks normally going through one charcoal filter, and then to the station vent plenum. Your assumptions for the waste gas decay tank releases (Page 11.6b) take credit for two filters. Resolve this discrepancy.

11.17 Justify the filter decontamination factor values taken in the gas waste system as given in your response to question 11.6.

11.18 Include in your Table 11-6, "Waste Disposal Systems Component Data," for tanks, pumps, filters, demineralizers and degassifiers the following:

a. Operating temperature (°F).

b. Design codes and standards.

c. Seismic design class.

11.19 For piping and valves in the waste disposal systems provide design information as in your Table 11-6 (with the additional information requested in Request 11.18.)

11.20 Describe in detail the flow path of gas from the decay tanks to the station vent to eliminate discrepancies between assumptions made in Amendment 25, Page 11.6b and Figure 11-3. Also describe the "gaseous waste discharge filter" and "gas collection header filter," Figure 11-3. Submit corrected drawing, if required.

11.21 Figure 11-1 shows that gas from the degassifier is pumped to the auxiliary building gas header; this disagrees with Figure 11-3 which shows the gas going to the waste gas surge tank. Clarify and provide revised drawings to reflect any changes.

11.22 Figure 11-1 shows the nitrogen cover gas from the clean waste receiver tanks discharging to the gas collection header. This conflicts with the statement on Page E-9 of the Environmental Report that all nitrogen cover gas will be compressed in the gas decay tanks. Clarify the discrepancy and derive your estimated 25,000 scf (Page E-9) annual gaseous radioactive waste processed in the waste gas decay system.

11.23 Inconsistencies exist in the four references to liquid wastes destined for complete degassification; namely, Table 3-4a, Table 3-4b and Page F. 5-2 (all in the Environmental Report) and in Amendment 25, Page 11.6b in the FSAR. Provide your best estimate of the volume most appropriate for calculating the annual curies of radioactive gases discharged to the atmosphere and justify your selection if it is less than 625,000 gallons - the largest volume stated.

11.24 Potentially contaminated gases, apparently, are collected in the gas collection header and discharged to the station vent through a filter. Will this gas discharge through a monitoring station? If so, what is the sequence of events if a high radiation alarm occurs? Describe the filter and revise Figure 11-3 to reflect your description.

11.25 The annual radioactive liquid waste discharges as estimated in the Environmental Report (Table E-1, Table E-3, Page F 5-2) and Page 11-9 of the FSAR do not agree. Clarify and provide information to justify using the lowest of the four volumes for calculating your annual curie release in liquid wastes.

11.26 Amendment 25, Page 11.6e, states, "Unit 2 liquid waste processing systems have been designed with sufficient capacity to enable the additional treatment of Unit 1 waste effluents and will be installed for Unit 1 use as soon as scheduling allows." Provide an estimate of the period Unit 1 will operate without use of this equipment and describe the flow paths between the two units including the effect of Unit 1 wastes on the processing time of wastes in the Unit 2 systems.

12.0 CONDUCT OF OPERATIONS

12.11 Provide a list of titles of safety-related plant operating, maintenance and testing procedures. Provide clear indication of their purpose and applicability as well as a description of the review, change and approval methods for administering these procedures.

14.0 SAFETY ANALYSIS

14.1 The Steam Line Failure analysis in Section 14.2.2.1 of the FSAR indicates that the reactor can return to criticality after it has tripped, because of the cooldown (with a negative moderator temperature coefficient). It appears from the discussion in the Arkansas 1 FSAR and in Supplement 3 of the Oconee FSAR that the reactor's uncontrolled return to power is stopped by boron addition, particularly from the core flooding tanks. The return to power is by nature a pressure-increasing phenomenon and the release of boron from the core flooding tanks can be stopped by increasing pressure or will not start at all unless a low pressure is reached. Explain how the prevention of uncontrolled return to high power is assured.

14.6 In analyses related to steam generator tube leakage or tube failure, it is assumed that the plant operator can isolate the affected steam generator at an appropriate time. This assumes that the operator can identify both the fault and the affected steam generator promptly. Describe the means by which the operator can make this judgement quickly. Discuss in greater detail what the consequences will be if the operator is unable to identify which steam generator is leaking.

14.7 Provide a detailed discussion of the methods and procedures which will be followed in the event of an accident to determine the magnitude of the accident and to determine what steps may be needed to mitigate the consequences of the accident. In your discussion, include the following information:

a. The method to be used for sampling the containment and other plant building atmospheres to determine the magnitude of any radioactivity release to the containment or other buildings.

b. The method to be used for determination of the rate of any releases of radioactivity from the containment or other plant buildings, and the location of possible release points.

c. The instrumentation which will be available for achieving the objectives in a and b above. Include instrument types, ranges, locations and method of calibration. Describe the bases used for selection of the types of instruments to be provided.

d. The methods, procedures and instrumentation to be used to determine the direction and rate-of-movement of effluents released to the environs, and the quantities or concentrations of any radioactive materials in areas beyond the plant boundaries.

14.8 Section 14.2.2.5.5 (see page 14-62) acknowledges that reactor building compartment differential pressure analyses were made with the Bechtel computer program COPRA. However, the results of the analyses were not discussed. Provide the following information:

a. Identify the reactor building compartments that were analyzed.

b. Discuss the results . . the differential pressure analyses performed for each compartment, including the maximum absolute and differential pressures attained.

c. Discuss the structural design capability of these compartments to withstand the energy releases that might result from design basis accidents.

14.9 For the purpose of performing an independent assessment of the reactor building pressure response under LOCA conditions, provide the following information:

a. Specify the minimum reactor building operating temperature.

b. Specify the assumed efficiency of the spray droplets in cooling the reactor building atmosphere.

c. Provide the mass flow rates, corresponding thermodynamic fluid states, and energy release rates to the reactor building from additional energy sources following the completion of blowdown such as boil-off of the reflooding water and core decay heat.

APPENDIX 5 STRUCTURES

A5.1 Section 5A.5 of the FSAR identifies two penetration areas that are not located within a tornado resistant structure or protected by missile shielding. Describe their inherent missile resistant capabilities as presently designed.

A5.2 Section 5.A.5 and 5.1.1.2.6 of the FSAR provides the design bases for tornado loads for seismic Class I structures. Describe the adequacy of the design of these structures to preclude the generation of secondary missiles within the structures from tornado missiles. Determine the effects of any secondary missiles on safety related equipment and systems. In addition to the design basis tornado missiles, consider also tornado missiles of lesser size that could produce a more damaging secondary missile.

A5.3 Seismic Class I structures are designed for the maximum probable flood level of 361 feet (MSL). Identify all exterior openings and penetrations below this level and describe the type of flood protection provided for each.

A5.4 Provide a list of all seismic Class I systems and equipment that are located below elevation 361 feet (MSL).

A5.5 Assuming that the pressure, resulting from the maximum probable flood hydrostatic head, is acting on saturated soils, determine the amount of seepage expected through: (a) the seismic Class I structures walls, (b) an assumed crack in the concrete, and (c) a poorly constructed water stop at a construction joint.