Docket

POOR ORIGINAL

September 16, 1970

Docket No. 50-289

Metropolitan Edison Company
ATTN: Mr. J. G. Miller
Vice President and Chief Engineer
P. O. Box 542
Reading, Pennsylvania 19603

Gentlemen:

In our continuing review of your application for a Provisional Operating License for the Three Mile Island Nuclear Station, Unit 1, we find that we need additional information as described in detail in the enclosure. The requests have been categorized into groups which correspond to sections in your Final Safety Analysis Report (FSAR). Most of these requests were discussed with your representatives in a meeting held at Bethesda on August 5, 1970. We understand from a meeting with your representatives on May 19, 1970, that you intend to amend the FSAR to include additional information regarding onsite meteorology. Accordingly, our review of this area will continue after receipt of your amendment.

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. If such is the case, you may wish to incorporate the information by reference.

Some of our questions concern a Gilbert Associates Report GAI-1720, which you have incorporated into your FSAR by reference, and which has been placed in the Public Document Room. Your answers to these questions may be included in the FSAR.

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

Sincerely,

1437 007

Original Signed By
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Add'l Info Requested

cc:

page 2

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Form AEC-318 (Rev. 9-53)

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ADDITIONAL INFORMATION REQUESTED
FOR
THREE MILE ISLAND UNIT 1

2.0 SITE

2.1 The purpose of the environmental monitoring program as stated in Section 11 of the FSAR is to measure radiation of significance to people in the vicinity of and attributable to operation of the Three Mile Island Unit 1 facility. The program as described appears to be inadequate to meet the stated objective. The modifications that we believe to be necessary are the inclusion of river sediment and aquatic plant sampling at locations reasonably close to the point of effluent discharge, and the inclusion of certain minimal sampling, at least on a quarterly basis, without consideration of the extent to which the actual releases are within the Technical Specification limits. Accordingly, provide the following additional information concerning your environmental program:

2.1.1 An identification of the location, preferably on a map, of each type of sample to be taken or measurement to be made.

2.1.2 A revision of Table 11-13 in the FSAR to include sampling of river sediments and aquatic plants.

2.1.3 A revision of Table 11-13 in the FSAR to include, under Regime I, a full complement of samples at a frequency of at least four times per year.

2.1.4 A description of the analyses to which each type of sample will be subjected. Include information to establish that the analyses will be sensitive enough to detect levels in environmental media which indicate the likelihood of public intakes in excess of 3% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II of 10 CFR Part 20.

2.1.5 A statement as to whether sample collection will be coordinated with waste discharge periods.

2.1.6 Provide additional information on the studies which you state have been started to identify pathways of potential exposure via human food.

(1) What studies are underway to identify the critical population groups?

(2) What studies are underway to identify possible terrestrial pathways?

(3) What are your criteria for choice of fish types and specific food crops to be studied?

(4) What information is to be gathered in these studies that will be useful in assessing exposure of critical population groups?

2.2 The FSAR (Section 2) states that flood protection for the plant will be provided for the probable maximum flood (PMF) as recently revised by the Corps of Engineers. Provide the following information:

2.2.1 A discussion of the basic analyses, and a presentation of the resulting estimated PMF hydrograph (unregulated and regulated) for the Susquehanna River at the Three Mile Island plant.

2.2.2 A discussion of the backwater analyses including the bases for the roughness coefficients, and the significance of channel and overbank flows during extreme flood events. Present the stage-discharge curve for the plant site for discharges up to the PMF. To supplement the discussion and substantiate the analyses of backwater computations, include pertinent maps, tables, and graphs to indicate:

(1) The locations of river cross sections used in the analyses,

(2) The Manning "N" coefficients assumed for each cross section,

(3) River flood profiles, flood discharges of record, and hypothetical floods (indicate all record highwater marks on the profiles),

(4) Flood level contours, and

(5) Comparisons of backwater computed stages to the stages of record such as those for the 1936 and 1964 floods.

2.2.3 A discussion of the flood protection to be provided for the engineered safety features.

2.2.4 A discussion of the operational procedures and actions to be taken in advance of and during floods. The discussion should consider the information and instructions to be given to the operations staff, actions to be performed by the staff, and the estimated times of occurrence, in sequential order, of the occurrence of various river flow conditions, up to the attainment of river flow stages at which decisions must be made. If a gauge station upstream of the site is to be used to monitor river elevations and required actions are to be initiated on the basis of measurements made at the station, provide the relationship between the elevation of the river at the gauge station and the elevation of the river at the reactor site.

3.0 REACTOR DESIGN

3.1 Discuss the probability for axial flux shapes that are tilted toward the upper half of the core. Provide the axial flux distribution for the worst case predicted and the effect that this flux distribution would have on the thermal design margins, including those associated with the departure from nucleate boiling ratio and the maximum fuel rod temperature.

3.2 Provide an analysis of the effects of changes in power level, including the design overpower level, on the axial flux shape. To what degree will the in-core detectors and the external detectors indicate axial shape changes as a function of power level?

3.3 Provide additional information on fuel swelling as a function of burnup. The information should be the latest information available and should relate more closely to your present fuel pellet design that that presented to date.

4.0 REACTOR COOLANT SYSTEM

4.1 With reference to Section 4.1.2.5, of the FSAR, Seismic Loads and LOCA Loads, describe the extent to which the reactor pressure vessel and the reactor internals differ from those for the Duke Power Company Oconee units.

4.2 Provide a discussion of effect on the structural integrity of the reactor pressure vessel cavity of the rupture of a pipe having a double-ended break area of 14.1 ft^2 within the cavity, and of the potential for the generation of missiles and interference with continued core cooling in the event of such a rupture.

4.3 Provide a discussion of the design procedures used to establish the thicknesses of the special missile shields provided in the plant. Discuss the typical missiles assumed for design purposes and the methods used to calculate concrete penetration.

4.4 Provide a pressure-time history for the primary coolant system for the postulated event of one safety valve sticking in the open position.

4.5 For all components that are part of the reactor coolant pressure boundary, identify those stainless steel parts of the pressure-containing boundary, or those load-bearing stainless steel members which are vital to the structural integrity of the reactor vessel and core, that became furnace sensitized during the fabrication sequence. Also identify any stainless steel type 304 or 316 for which nitrogen was added to enhance the strength of the material. Describe any plans you have with respect to replacement of these materials or other corrective measures in recognition of the susceptibility of furnace sensitized stainless steel and of nitrogen-enhanced stainless steel to stress corrosion cracking.

4.6 If the process of electroslag welding was used in the fabrication of components within the reactor coolant boundary, identify such components and describe the process specifications, control of variables, and quality control procedures which were applied in production to achieve the physical properties in the welds and heat affected zones comparable to those obtained in the weld procedure qualification tests.

4.7 Identify the edition of the applicable Codes, Addenda, and Code Cases, which have been or are being applied to pressure vessels, piping, valves, and pumps of the reactor coolant pressure boundary.

4.8 We understand that vibration test data obtained at other reactor plants will be used to establish the adequacy of the Three Mile Island Unit 1 core support structure to withstand vibrations. Identify the reactor plants at which these data will be obtained. Describe the vibration surveillance program that will be used in the Three Mile Island Unit 1 facility to confirm that the vibration of the core support structure is comparable to that measured in the other plants.

4.9 With respect to seismic ground motion and differential relative settlement, describe the design criteria which were employed for critical piping and tunnels buried or otherwise located outside of the containment. Particular emphasis should be given to the criteria used for the piping at the point of entry into various structures.

4.10 To permit an assessment to be made as to whether the seismic design bases are correctly translated into the required specifications, drawings, procedures, and instructions so that the necessary structures, systems, and components can withstand seismic loads combined with the other appropriate concurrent loads, provide the following information:

4.10.1 A description of the design organizations that are involved in the seismic design of all structures, systems and components of the plant that are related to safety.

4.10.2 A description of the responsibilities of the involved design organizations in connection with the seismic design, the extent to which these responsibilities have been promulgated to the organizations in writing, and the identification of the design organization that has been assigned overall responsibility for the adequacy of the seismic design.

4.10.3 A description of the documented procedures that have been or will be promulgated to provide for the interchange of needed design information and changes thereto and the coordination of the various facets of the seismic design among the involved design organizations.

4.10.4 A description of the manner by which you assure that the design procedure described in 4.10.3 above has been or will be followed.

4.10.5 A description of the design control measures that have been or will be instituted to verify or check the adequacy of the seismic design. Indicate by whom they will be performed, and describe the design procedures that have been or will be promulgated to provide for these measures.

4.10.6 A description of the requirements that are or will be included in the purchase specifications for safety-related equipment to assure that this equipment is adequately designed to withstand and can function under the seismic design conditions, and of the provisions that have been included in the purchase specification to permit the purchaser to verify that these requirements are satisfied.

4.11 Identify the Class I piping systems and equipment, other than the primary coolant system, which have been designed to withstand the loads that would result from combining the seismic loads due to the Design Basis Earthquake (identical to the Maximum Hypothetical Earthquake as used in the FSAR) and the pipe rupture loads due to the Design Basis Accident. Identify the loading combinations and the stress and deformation limits applicable to these loading systems.

4.12 Provide the sensitivities and response times of the leak detection systems described in Section 4.2.3.8 of the FSAR.

4.13 The FSAR states that the reactor pressure vessel material surveillance program will be in accordance with the integrated program described in Topical Report BAW-10006, Reactor Vessel Material Surveillance Program. We understand that you intend to modify your program for Three Mile Island Unit 1, so that it will conform to the more stringent programs established for other recently licensed facilities. Provide a discussion of the revised program indicating in particular the independence of the program from all other programs and the planned withdrawal schedule for a minimum of four capsules.

4.14 If the estimated inservice transition temperature shift of the reactor vessel beltline material is based on data which are related to control of residual elements (specified in ASTM-E-185-70, Section 3.1.3), including control of copper and vanadium, specify the results of the chemical tests of these vessel materials, including the residual element content in weight percent to the nearest 0.01%.

4.15 Provide a discussion of the proposed preoperational and inservice inspection programs for the reactor coolant pump flywheels, with particular emphasis on the bases for the acceptance criteria established for inspections and the actions to be taken in the event that these criteria are not met.

5.0 STRUCTURES

5.1 The use of steady-state temperature profiles, such as those presented in Figure 5-17 of the FSAR for normal operating conditions and in Figure 5-19 for accident conditions, may not necessarily result in the maximum temperature stresses that could be predicted for the structure. Discuss the transient thermal gradients during startup, shutdown, and post-accident conditions and provide information to demonstrate that the structure, as a whole, can accommodate the resulting stresses within the permitted safety margins. The statement, in Section 5.2.2.3.1 of the FSAR, that transient and shutdown conditions have been considered seems to apply only to the design of the buttresses and not to the entire structure.

5.2 The use of the specified working stress design method and the ultimate strength design method, as presented in ACI 318-63 Code, is not directly applicable to structures where two- and three-dimensional stress fields are predominant. Discuss the manner in which this problem has been taken into account, including consideration of the following:

(1) The allowable stresses and the ultimate strength for those cases where one principal stress is compression and the other principal stresses are tensile.

(2) The influence of shearing forces on the allowable stresses and the ultimate strength of concrete in compression and in tension for those cases where the direction of the mild steel reinforcing does not coincide with the direction of the principal stresses.

(3) The influence of cracking on the allowable stresses and the ultimate strength of concrete between cracks.

(4) The anchorage and bond properties of mild steel reinforcing when cracks occur in a direction parallel with the reinforcing.

The discussion should also consider that, for some load combinations with load factors of 1.0, the design has no safety margin on actual strength with respect to the nominal ultimate strength of the material.

5.3 The reinforcement and the concrete in the anchorage zone have been designed in accordance with ACI and PCI Codes and Recommendations, and the work of Y. Guyon and F. Leonhardt, as stated in Section 5.2.2.3.1 and on page 5 D-2 of the FSAR. However, these codes, recommendations, and studies do not necessarily apply to structures where a three-dimensional stress field exists which includes high shear, thermal and tensile shrinkage stresses, and is influenced by major cracking patterns and stress redistributions due to creep and thermal stresses that develop gradually during the lifetime of the structure.

Provide a discussion to demonstrate that the anchorage zone can accommodate the most unfavorable stress combination with acceptable safety margins when the above-mentioned factors are taken into account. This discussion should address the design of the buttresses and the design of the anchorages of the vertical tendons and the dome tendons. In addition, the bases for design of the mild steel reinforcing for anchorage and bond should be explained.

5.4 Provide information to demonstrate that the finite element approach used for the design considers the cracked state of concrete, and the influence of the prestressing tendon holes on the surrounding concrete, in an acceptable manner.

5.5 The information presented in Section 5.2.2.2 of the FSAR indicates that the mild steel reinforcing in the reactor building has been designed in accordance with ACI 318-63. In addition, a minimum amount of mild steel reinforcing (0.15%) has been provided for crack control. Since ACI 318-63 is not directly applicable to the containment structure, discuss the design approach and justify the adequacy of the reinforcing provided for bonding, shear, and crack control. Also, describe the manner in which the crack-controlling reinforcing is spliced and anchored, and present the basis for the omission of such reinforcing at the liner.

5.6 One hole in the anchorage hardware is used to accommodate an unstressed surveillance wire, as stated in Section 5.2.2.3 of the FSAR. Discuss the following:

(1) The ability to pull a wire out of a tendon for inspection, despite the interference with other wires in the tendon.

(2) The degree of confidence that the state of wires that are highly stressed can be assessed reliably by the inspection of an unstressed wire.

5.7 The flanged joints of the equipment and personnel hatches are designed with double gasketed seals, as stated in Section 5.2.2.4.8(a) of the FSAR. Describe the initial testing and the surveillance procedures which will be employed for these joints to assure leak-tight integrity over the service life of the containment structure.

5.8 Describe the nondestructive testing procedures which were employed to detect laminations in the load-carrying steel plates that were welded into the containment liner in order to transfer loads normal to their surface, to serve as anchorage bearing plates, and to provide essential steel members for equipment supports.

5.9 Appendix 5E of the FSAR presents the response of the containment structure that is predicted to occur during the pressure proof-testing of the structure. With respect to the proof-testing:

5.9.1 Describe the measurements that will be made before the containment pressure proof testing in order to define the actual geometry of the structure and to provide correct bases for the interpretation of the test results.

5.9.2 Describe the procedures that will be used during the containment pressure proof testing in order to separate the influence of the temperature variations from the influence of the mechanical loads.

5.9.3 Indicate the acceptance tolerances beyond which the test results will be considered unacceptable and describe the corrective measures which will be followed in the event that the tolerances are not met.

5.9.4 Indicate how the safety margin that is provided in the completed structure will be established and compared with the design margin.

5.10 List all Class I structures, systems, and components. Indicate the method of seismic analysis (modal analysis response spectra, modal analysis time-history, equivalent static load analysis, empirical test analysis or other method) used for the design of each item, and the applicable stress and deformation criteria and the damping values used in each analysis. Provide a brief description of all the methods that were used for the seismic analysis of the Class I items.

5.11 Two methods of seismic analysis that were used in the design of Class I piping are described on pages 5-75 and 5-76 of the FSAR and are characterized as a response spectrum approach and a lumped mass modal analysis technique. Identify all Class I piping that was analyzed by either or both of these methods.

5.12 The information in Section 5 of the FSAR indicates that constant vertical load factors and horizontal multi-mass dynamic analysis was used in lieu of a combined vertical and horizontal multi-mass dynamic analysis. This approach may not have been sufficiently conservative. Provide the basis for determining the combined vertical and horizontal response loads for the Class I structures, systems, and components.

5.13 The following requests relate to Gilbert Associates Report GAI-1729 which you have incorporated by reference into your application:

5.13.1 The comparison for the single degree of freedom model method that is described in "Seismic Analysis of Equipment Mounted on a Massive Structure," J. M. Biggs and J. M. Roesset (referenced in GAI-1729) may not be conservative for multi-mass systems. Provide the basis for, and indicate the conservatism in, the use of this method by demonstrating its equivalency to a multi-mass time history method. Alternatively, other theoretical methods of experimental analyses and tests may be submitted in justification of the use of the designated design methods.

5.13.2 Provide the design criteria that apply to the dynamic analysis of valves and other in-line system components that, by virtue of geometry and size, introduce significant torsional moments into the piping system.

5.13.3 Provide the design criteria and analytical procedures that apply to piping, and that take into account the differential movement between floors at different elevations.

5.13.4 The predominant mass and compliance of structures, systems, and components will affect the response of the system. Provide the criteria that were used in formulating the system analysis, and that considered the coupling of the predominant mass and foundation compliance effects in the system dynamic analysis.

5.13.5 In-phase or out-of-phase displacements at different elevations between equipment, piping, and buildings may produce high stresses or deformations in connecting piping. Provide the criteria and analytical procedures that were used in the system analysis to account for in-phase or out-of-phase displacement of the building equipment and piping.

5.13.6 Provide the design methods that were used to compute shears, moments, stresses, deflections, and accelerations for each mode, and for the combined total response, including the procedures for combining closely spaced modal frequencies (e.g., number of modes having frequencies close together).

5.13.7 The presentations of the methods of dynamic analysis are general and the precise techniques used for the system and piping analysis are not described sufficiently to permit evaluation of their applicability and design conservatism. Provide additional detail concerning the methods and procedures used.

5.13.8 Identify the codes referred to in the topical report that were used for the combination of seismic and other stresses.

5.14 The material presented in Section 2 of the FSAR suggests that all Class I structures are founded on bedrock, since bedrock is quite near the surface. Provide, for each of the Class I structures, a tabulation indicating the type of foundation employed for the structure, the elevation of the foundation, the foundation medium, and other pertinent information.

5.15 A summary of the consideration of aircraft impact in the design of vital structures is presented in Appendix 5A of the FSAR. However, the combined effect of separately considering the simultaneous or non-simultaneous localized engine impact, and the impact of a Class D aircraft (200,000 lb aircraft at 200 knots) is not examined explicitly. More precisely, certain assumptions made regarding the loss of engines, outer portions of wings, and fuel, require further assessment. If the aircraft remains whole (no loss of weight), the peak load would be 18 x 10^6 lb. The area under the load-time curve (impulse) would be equal to the initial momentum of the 200,000 lb aircraft with a 200 knot velocity. The curve in Figure 5A-1 and Table 5A-1 of the FSAR show a peak load of 15 x 10^6 lb, assuming the loss of engines, fuel, and wing structure. This is inconsistent with the peak load of 16.2 x 10^6 specified in Table 5A-6. Explain this inconsistency and indicate the correct load.

Since no discussion is presented to show when these pieces are assumed to break away from the aircraft, it is not clear how this lost weight is handled or how the detached pieces of the aircraft are treated as additional colliding masses on the containment structure at some location removed from the main fuselage loading. Provide additional discussion and information based on a comprehensive investigation of:

(1) The impact of the whole aircraft.

(2) The simultaneous impact of the partially disabled aircraft and the detached elements (engines, wings, tips, etc) on separate locations.

(3) The impact of the partially disabled aircraft on the structures as damaged by the previous impact of the detached elements.

Explain the assumptions used in considering the times for the outboard engines, outer portion of wings, and fuel, to break away during impact. The discussion requested for the containment should be expanded to include all structures designed for aircraft impact (see page 5-11 of the FSAR).

5.16 Provide additional discussion of the method of calculating the fundamental frequency discussed in Section 3.2.1 of Appendix 5A of the FSAR. The frequency of a linear elastic flexural system, such as that assumed, varies directly with the square root of the modulus of elasticity, E, which for concrete is substantially changed near its ultimate strength. Indicate the actual values of E used in frequency calculations, and show that this value is conservative with respect to variations in E versus load.

.17 Provide an evaluation of the influence of a variation in the fundamental frequencies on the response of the structure and on the dynamic load factors, and a discussion of the influence of changes in assumed edge conditions.

5.18 Section 2 of Appendix 5A in the FSAR indicates that a single-degree-of-freedom analysis was employed to determine hypothetical aircraft impact loadings. On the other hand, in the analysis for Case C impact loading also discussed in Appendix 5A, there is an indication that a modal analysis procedure involving multiple-degree-of-freedom considerations was employed. It is not clear whether the dynamic load factors originally referred to were employed with that analysis technique or whether some other approach was used, since the forcing functions are not defined. Submit the following additional information to permit an evaluation of the analysis carried out and the significance of the results:

5.18.1 The justification of the adequacy of modeling a multiple-degree-of-freedom system as a single-degree-of-freedom system for concentrated load considerations, as indicated at the beginning of Appendix 5A. This justification should include consideration of the structure-soil interaction, especially when the impact occurs near an edge. Indicate to what degree the use of a single-degree-of-freedom system affects the natural frequency and the dynamic load factor.

5.18.2 A further explanation of the technical analysis employed for Case C and any correlations between the findings there and those reported in the earlier sections of Appendix 5A.

5.18.3 A discussion of the applicability of the penetration formulas especially with respect to the empirical material coefficients and the variation in these that might be expected to be applicable in this case; and, the interaction of the penetration with the overall flexural response. Indicate how these were considered to be interrelated.

5.18.4 The impact effects on equipment and components within the structure subjected to shock. Any significant shock effects carried through the structures should be identified, and the provisions employed to alleviate damage and the effects of the shock should be described.

5.19 The following structures, as stated on page 5-11 of the PSAR, are designed for aircraft impact:

Reactor building
Fuel handling building
Portions of auxiliary building as shown in Figure 5-42
Portions of intermediate building as shown in Figure 5-42
Control building
Intake screen house and pump house
Heat exchanger vault
Air intake structure
Access tunnel - vault to auxiliary building

For each of these structures, identify equipment such as inserts, instruments, hangers, brackets, and lighting fixtures that are attached to the walls and roofs and which, upon the impact of an aircraft with the structure, could conceivably be ejected as missiles capable of significantly damaging Class I equipment located within these structures. Indicate the design criteria and quality control procedures that were used to determine that these elements will not become missiles, even for the cases where extensive spalling of concrete might be expected to occur. Discuss the probability of such spalling with particular emphasis on the effects of potential concentration of dynamic compressive, tensile, and shear stresses, and on the specific design features which limit or prevent significant spalling.

5.20 Provide additional information on the provisions utilized in the design of the dome-region liner anchors to prevent possible spalling effects. The discussion should address the following items:

(1) The possible local concentration of dynamic stresses, including shear stresses, due to elastic shock waves in concrete.

(2) The adequacy of the liner anchors when located in parts of the structure where concrete is subject to tensile stresses or cracking.

(3) The possible rupture of the liner plate.

(4) The design criteria, quality control measures, and safety margins that were used in the design and construction of the liner anchors.

5.21 Additional information is needed concerning the likelihood and consequences of loss of prestress tendons and anchors during an aircraft impact incident, particularly in view of the fact that the concrete under the anchor bearing plates will be in tension and very likely will be completely cracked. Provide a discussion of this matter, considering the following aspects:

(1) The potential for increased tensile stresses under the bearing plates that should be added to the bursting force and temperature stresses.

(2) The potential for additional shear stresses.

5.22 Provide additional details on the flat-slab wall and roof reinforcing including:

(1) The design criteria for this reinforcing, and specifically the design basis for the main reinforcing, the shear reinforcing, the provisions for anchorage of the reinforcing bars, and the allowable bond stresses.

(2) An explanation of whether welding has been used to improve the anchorage of bars, or whether any other means, such as the installation of anchor plates located in concrete under tension, has been used for this purpose.

(3) A description of the quality control procedures applied to any special construction methods that may have been employed.

5.23 In the discussion of the stress analysis of the buttresses for the containment presented in Appendix 5B of the FSAR, the maximum shearing stress for a number of different loading conditions is not included. Provide the sizing criteria used in designing the reinforcing steel and the shearing stresses determined by the analysis.

5.24 The structural integrity testing of the reactor containment structure is described in Appendix 5B of the FSAR. It is not apparent from the descriptions what specific measurements are to be made at, and in the vicinity of, the prestressing anchorages in order to demonstrate the adequacy of the design. In view of the high stresses that will exist in these areas by virtue of the prestressing, the additional stress and strains arising from the internal test pressure loading may not be significant. Monitoring of the response of the structure during the test, including visual observations and inspections, seems appropriate in order to provide further insight into the validity of the structural analysis and the predicted structural behavior. Provide a discussion which addresses this subject and describes the proposed monitoring program planned for the test.

6.0 ENGINEERED SAFETY FEATURES

6.1 Describe the sump pump and level alarm systems for the engineered safety feature compartments, including the arrangements made for providing emergency power to the alarm systems.

6.2 Discuss the procedures to be employed to verify the accumulator flow rate versus time predicted by analysis.

6.3 Describe the level monitoring systems for the reactor building sump and discuss the provisions made in the design to ensure that the systems would operate as required in the environment following a loss-of-coolant accident.

6.4 Describe the procedures to be followed to isolate the core flooding tanks during routine depressurization and to ensure that the isolation valves cannot inadvertently close during normal operation, or following a loss-of-coolant accident.

6.5 Discuss the criteria that were used for the design of the reactor protection system, and for the electrical and mechanical components of engineered safety features, to ensure that these systems and components will be capable of performing their design functions in the radiation field to which they will be exposed during normal operation and in the event of an accident. Describe the analyses and tests performed or to be performed to verify compliance with the design criteria.

6.6 Identify all components, including motors, pumps, fans, cables, valves, filters, pump seals, and instruments that are located within the containment structure and that would be required to be operable during and subsequent to a loss-of-coolant accident or a steam-line-break accident. Describe the qualification tests which have been or will be performed on each of these items to ensure its capability to perform its design function in a combined high temperature, pressure, and humidity environment.

6.7 In our accident evaluations we assume that up to 5% of the iodines available for leakage following the postulated loss-of-coolant accident may be in particulate form not affected by the containment spray system. Explain the reasons for not providing high efficiency particulate adsorber (HEPA) units in the reactor building air recirculation and cooling system, and discuss the feasibility of including such adsorber units in the Three Mile Island Unit 1 facility.

6.8 Discuss the overpressure that may occur in the ventilation tunnel as a result of an airplane crash, and the related design capability of the fire protection damper in the tunnel.

6.9 Describe the design features of the fire damper and its control and instrumentation which minimize spark formation.

6.10 Explain why the complete Halon system is not activated simultaneously with a signal from any one zone.

6.11 Submit a diagram that shows the location of the Halon reservoirs with respect to the ventilation tunnel.

6.12 Explain the notation in Table 9-13 which describes water deluge in zone 1 only, upon detection of a temperature rise in zone 3.

6.13 Provide a complete description of the types and locations of combustible gas detectors.

7.0 INSTRUMENTATION AND CONTROL

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

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

7.1.2 Identification of the supplier for those systems that are designed and/or built by suppliers other than B&W.

7.1.3 Identification of those features of the design which do not conform to the criteria of IEEE-279 and the Commission's proposed General Design Criteria, and an explanation of the reasons for any aspects of nonconformance.

7.2 In regard to the B&W designed control systems, provide the following information:

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

7.2.2 A list and a discussion of the design differences in those systems not identical to those used in the Oconee Nuclear Station, including an evaluation of the safety significance of each design change.

7.3 State the seismic design criteria for the reactor protection system, engineered safety feature circuits, and the emergency power system, including the station batteries. The criteria should address: (1) the capability to initiate a protective action during the Design Basis Earthquake, and (2) the capability of the engineered safety feature circuits to withstand seismic disturbances during post-accident operation. Describe the qualification testing requirements which will be used to assure that the criteria are met and the means by which these requirements have been imposed on equipment suppliers.

7.4 Describe the quality assurance procedures which apply to the equipment in the reactor protection system, engineered safety feature circuits, and the emergency electric power system. This description should include the quality assurance procedures used during equipment design, fabrication, shipment, field storage, field installation, and system component checkout, and the records pertaining to each of these.

7.5 State the criteria which 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 and 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) which has been or will be performed to confirm satisfactory operability of control and electrical equipment under extreme environmental conditions.

7.6 Describe how reactor protection system and engineered safety equipment will be physically identified as safety-related equipment in the plant.

7.7 Describe the methods for the periodic testing of engineered safety features, instrumentation and control equipment and discuss the degree of consistency of the design with the requirements of IEEE-279. We interpret IEEE-279 to require the same high degree of on-line testability for engineered safety feature equipment as is required for the reactor-trip system.

7.8 Describe the information that will be available to the operator to identify all reactor protection system and engineered safety feature channels that are either in test or undergoing maintenance. Provide descriptions of the indications available, down to the channel level, to identify which instruments initiate a protective action. These descriptions should be in sufficient detail to permit a determination of the system's compliance with Sections 4.13 and 4.19 of IEEE-279.

7.9 Describe any rod speed limiting features that will prevent a withdrawal rate in excess of 30 inches per minute.

7.10 Do the circuits which prevent improper sequencing of the control rods conform to the provisions of IEEE-279?

7.11 Do the circuits which automatically terminate dilution of the primary coolant conform to the provisions of IEEE-279?

7.12 Provide a description of the instrumentation systems included in your design for the remote monitoring of post-accident conditions within the primary containment. Provide an analysis to show that these systems provide adequate information for the full spectrum of postulated accidents.

7.13 Identify the electrical and pneumatic components, including valves, and pumps, of the auxiliary cooling systems which should be considered as portions of the engineered safety features. Do the criteria that were used for the design of the associated instrumentation and power systems for operation of these components conflict in any way with the requirements of IEEE-279 or the proposed AEC General Design Criteria?

8.0 ELECTRICAL SYSTEMS

8.1 Provide the following information with respect to the onsite electrical power systems:

(1) The fraction of the rated load for continuous operation of each diesel generator that is required by the electrical loads associated with the engineered safety features. The continuous rating is defined as that continuous load for which the supplier guarantees operation at a 95% availability with an annual maintenance period.

(2) The 2000-hour and the 30-minute diesel generator overload ratings.

8.2 What was the basis used to size the station batteries to operate for 2 hours without the benefit of any station power?

8.3 Discuss the analyses that have been or will be performed to demonstrate that neither the loss of Unit 1 nor the loss of the largest generating unit in the grid will negate the ability to provide offsite power to this station.

8.4 Provide an analysis to demonstrate that no single failure within any dc system (e.g., station battery) will adversely affect the shedding of loads and/or the opening of supply breakers to such an extent that adequate diesel generator operation will be prevented.

8.5 To supplement the information presented on page 8-6.7 of the FSAR, submit the cable installation design criteria intended to preserve the independence of redundant reactor protection systems and engineered safety feature circuits (instrumentation, control and power). For the purpose of cable installation, the protection system circuits should be interpreted in their broadest sense to include sensors, signal cables, control cables, power cables (both ac and dc), and the actuated devices (e.g., breakers, pumps). The discussion of cable installation criteria should include:

(1) Cable separation in terms of space and/or physical barriers between redundant cables. The separation of power cables from those used for control and instrumentation, the intermixing of control and instrument cables within a tray (or conduit or ladder), the intermixing within a tray of cables for different protection channels, and the intermixing of non-vital cables with protection system cables.

(2) Criteria for the separation of penetration areas, the grouping of penetrations in each area, and the separation of penetrations which are mutually redundant.

(3) Cable tray loading, insulation, derating, and overload protection for the various categories of cables.

(4) Criteria with respect to fire stops, protection of cables in hostile environments, temperature monitoring of cables, fire detection, and cable and wireway markings.

(5) Administrative responsibility for, and control over, all of the foregoing (1 - 4) during design and installation.

(6) Design criteria for locating the process instrumentation inside containment, including separation of redundant sensors and sensing lines, and protection provided to cables between sensors and electrical penetrations.

8.6 Are the battery rooms separately ventilated?

8.7 With respect to both the ac and dc emergency power systems, describe the electrical interlocks that prevent improper operation (e.g., onto a fault) of the manual cross connections between redundant buses.

8.8 Identify any heat tracing circuits vital to the operation of the engineered safety features (e.g., circuits provided to ensure that boron remains in solution). For each circuit identified, provide a description of the circuit and its power source, and explain the criteria used in the design of the circuit and its power source.

8.9 Submit a one-line diagram, similar in format to Figure 8.3 in the FSAR, showing the assignment of engineered safety feature equipment to emergency buses ID and IE.

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.1 Discuss your evaluation of the makeup and purification system, with regard to identification of any single failures that could lead to excessive makeup (overfill) or deficient makeup.

9.2 How long could the control rod drives function without cooling from the intermediate cooling system?

9.3 Provide the following additional information with respect to the heat removal capacity of the spent fuel coolers for the storage of 1-1/3 cores:

(1) The delay time for unloading the core.

(2) Does each spent fuel cooler transfer 25.85 megawatts of heat during storage of 1-1/3 cores? What are the associated heat exchanger fluid inlet and outlet temperatures?

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Describe the calculational method used to estimate the iodine concentration and inventory in the secondary coolant system, and discuss the assumptions made with respect to the primary coolant iodine concentration, leakage of primary coolant into the secondary coolant system, blowdown, carryout due to entrained moisture, partitioning, and radioactive decay. Relate the results of the calculation to the activity release values given in Table 14-19 of the FSAR for the postulated steam-line-break accident.

10.2 List the valves that must be open in order for the emergency feedwater pump to supply water to the steam generator. Identify those valves that are open in normal operation and which must remain open, and those that are provided with operators for remote opening in the event that it is required. Identify the control systems which control the remote opening of the valves. Discuss the extent to which actions by the plant operator are required.

10.3 Describe the operating characteristics of the turbine-driven emergency feedwater pump in terms of steam requirements versus water delivered. What is the lower limit of operation?

10.4 Following a turbine trip with loss of offsite power, will the reactor continue to operate at some low power level? If so, what is the minimum secondary system water inventory that must be maintained in order that procedures for cold shutdown can be implemented if and when necessary?

11.0 RADIOACTIVE WASTE SYSTEMS

11.1 Section 1.3.2.16 of the FSAR refers to a design change involving a separate condenser off-gas vent pipe. This vent pipe will apparently be monitored by subsystem RM-A4, described in Section 11 (page 11-19) of the FSAR. What were the criteria for the selection of the monitor sensitivity and the alarm set point? If leakage occurs between the primary and secondary systems, how will the monitor detect possible iodine releases via this route, and how will such releases be controlled?

11.2 Provide an analysis of the concentration of radioactivity in the river, as a function of distance downstream of the plant for short-term releases of radioactive wastes. Consider various river flow conditions, estimated waste volumes and radioactivity concentrations, flow rates, and expected discharge times.

11.3 Discuss your criteria and intended operating procedures for maintaining radioactive effluent releases to values as low as practicable. Include information on anticipated minimum holdup times for liquid and gaseous wastes, and your criteria for minimum processing requirements.

11.4 Table 11-4 of the FSAR indicates that some components of the waste processing system are vented to the atmosphere. Discuss the possibility for and the expected magnitude of radioactive effluent releases via this route.

11.5 Provide an estimate of the annual discharge of significant amounts of non-fission-product radionuclides. Include a tabulation, similar to Tables 11-6 and 11-7 of the FSAR which relate to fission-product radionuclides, and a discussion of the method used to estimate the quantities predicted to be discharged.

12.0 CONDUCT OF OPERATIONS

12.1 Describe the procedures to be used to restrict access to critical structures, systems, and components of Unit 1 during continued construction work on Unit 2 after Unit 1 is licensed for operation.

12.2 Indicate the extent to which the emergency planning complies with 10 CFR 50.34(a)(10) and amended 50.34(b)(6)(v), as published in the May 21, 1970 Federal Register.