Docket No. 50-331

MAY 27 1969

Mr. Duane Arnold, President
Iowa Electric Light and
Power Company
Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Arnold:

In order that we may complete our review of your application for a construction permit, additional information on the design and safety of the proposed Duane Arnold Energy Center is needed. Most of the areas requiring additional information were discussed with your representatives at technical meetings on February 25 and 26, 1969.

Areas of principal concern relate to release of radioactive material to the environment, reliable supply of emergency cooling water, instrumentation and control, and seismic design. Our review of your quality assurance program and conduct of operations have not been completed, since your third amendment, which discusses these topics, has just been received.

The enclosed questions have been grouped by sections which correspond to the chapters in your Preliminary Safety Analysis Report. We prefer that the appropriate sections of the Preliminary Safety Analysis Report be revised as you have done in the previous amendments. Some of the requested information may be available as part of the public record connected with the review of other plants and may be incorporated by reference in your application. For any information that you plan to submit later than July 1969, state the nature of the information and provide a schedule for submittal.

Please contact us if you desire clarification or additional discussion of the information requested.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Additional Information

cc: See attached

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cc: Mr. Robert Lowenstein
Lowenstein and Newman
Suite 340
1100 Connecticut Avenue, N. W.
Washington, D. C. 20036

bcc: AEC PDR
Docket
DRL Reading
RPB-5 Reading
P. A. Morris, DRL
F. Schroeder, DRL
Assistant Directors, DRL
Branch Chiefs, DRL
S. M. Kari (2)
N. M. Newmark (2)
S. Forbes, (2)
ACRS(18)
W. Minners

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SURNAME ▶	WMinners:emh x7791	DFKnuth	RSBoyd	SLevine	FSchroeder	PAMorris
DATE ▶	5/14/69	5/15/69	5/20/69	5/20/69	5/22/69	5/21/69

Request for Additional Information

Iowa Electric Light and Power Company
Duane Arnold Energy Center

1.0 General

1.1 Provide a status report, including results where available and planned completion dates for each of the development programs identified in Section 1.10 of the PSAR.

1.2 List all of the design differences, which may affect safety, between this plant and any recently reviewed General Electric boiling water reactor, such as the Edwin I. Hatch Nuclear Power Plant of the Georgia Power Company (Docket No. 50-321).

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2.0 Site and Environment

2.1 Within the low population zone, provide population distribution data for one mile increments for each 22 1/2° sector. Provide the data for 1960, 1970, and 2010 and the basis for the projections. Also state the location of each hospital and school within this zone and how they will be considered in your emergency plans. Identify the nearest airfield used for commerical or military traffic and state its location relative to the plant.

2.2 Provide a correlation between river stage and flow for normal and low flows and for floods. Provide and justify the values of the minimum flow and maximum temperature of the river which are to be used in the design of the emergency safeguard systems cooled by service water.

2.3 Evaluate the release of radionuclides from the stack during both normal operations and accident conditions considering the effect that the cooling tower may have on the atmospheric diffusion of the release.

2.4 Confirm that the pre-operational environmental monitoring program will include sampling of food crops. State the minimum number of each type of sample that will be taken during the program. Describe the methods which will be used to determine the radionuclides in the samples. Provide your estimated schedule for reporting the results of the program.

2.5 Describe the means of controlling access to the plant for the purpose of protecting individuals from exposure to radiation and radioactive materials, and of providing plant security protection.

2.6 Provide your basis for the statement on page 2.6-50 "that rock inspection, exploratory drilling to detect cavities, and remedial grouting of cavities are not necessary" for the Class IM radwaste building, diesel generator building, and stack.

2.7 Provide the results and analyses of the rock exploration program described on page 2.6-51 of the PSAR.

2.8 Describe how any cavities underlying the buildings will be treated to assure the adequacy of the foundations.

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3.0 Reactor

3.1 Elaborate on the safety evaluation of the mechanical design of core internals and reactivity control to consider the effects of a rapid depressurization of the core concurrent with the incidence of the design basis earthquake. Specifically, discuss:

a. The number of fuel pins which would come in contact with deformed fuel element channels.

b. The range of deflections which would cause breaching of the fuel pin cladding irradiated to levels representative of end-of-core life.

c. The number of fuel pins which would be breached.

d. The ejection or damage of the temporary control curtains.

3.2 Relate the general discussion on vibration testing given in section 3.3.6 to the Duane Arnold reactor. As indicated, vibration testing would be performed if this plant represents a significant departure from design configurations previously tested and found to be acceptable. Although all the details of your vibration analyses and testing programs may not be available for review at the Construction Permit review stage, we request you consider the following in as much detail as is possible:

a. Outline your proposed program for vibration testing of the primary coolant system and reactor internals, or state which reactor will be used as your point of comparison. Justify that the reactor used for comparison is not significantly different from the Duane Arnold reactor.

b. For the Duane Arnold reactor or the reactor of comparison, describe the model and full-scale tests and vibration analyses which will be performed. Provide analyses for normal, abnormal and emergency operating modes; model scales and flow conditions; full-scale test instrumentation and test conditions. Also provide bases and methods for correlating test data with the various operating modes.

3.3 Clarify how the general loading criteria of Appendix C will be applied to the design of components.

a. For each of governing loading conditions given on page C.0-11, state which of the limits given in Tables C.0.1, C.0.2, and C.0.3 applies. If any selected limit is different from current code values (e.g., ASME Section III) justify the use of the selected limits.

b. State the basis for determining loss of function, i.e., loss of clearance, binding, or deformation. Where empirical data are used as a basis, provide details of the tests.

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c. State the stresses or deflections which are considered to result in loss of function.

3.4 On page 3.3-5, it is stated that "welds procedures and welders are qualified in accordance with the intent of Section IX of the ASME Boiler and Pressure Vessel Code." Clarify the meaning of the word "intent" by contrasting the procedures which will be followed with those procedures which are usually considered to be in full accord with Section IX of the ASME Code.

3.5 On page C.0.2 certain exceptions are taken to the items which will be treated as Class IM. State the extent and the specific findings of the analyses which provide the basis for assuring that failure of these items will not degrade any Class I systems, e.g., the effect of failure of the steam separators on the core spray internal pipe.

3.6 Several thermal and hydraulic parameters of the Duane Arnold core have been changed compared with earlier high power density reactors of the same size, such as Vermont Yankee. Although the recent Bell design incorporates most of these changes, the reactor is larger, has lower jet pump capacities and is therefore not directly comparable. For the Duane Arnold plant discuss the consequences of the increased feedwater temperature, the decreased recirculation flow and the related effects on the core hydrodynamic stability and critical heat flux ratio.

3.7 Describe the method used to determine the fuel element orifice size, the procedures which assure correct orificing, and the consequences, during both normal and abnormal operation, of incorrect orificing.

3.8 What is the value of the power coefficient at end-of-life (19,000 MWD/T)?

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4.0 Reactor Coolant System

4.1 Provide the following information concerning the system which will be designed to detect leakage from the nuclear system.

a. Estimate the sensitivity of the system to detect leaks from small incipient cracks. Include in the estimate the minimum leak rate from an unidentified source that can be detected and the crack size corresponding to this leak rate.

b. Justify the statement on page 4.10-4 that "corrective action could be taken before the integrity of the barrier is threatened with significant compromise."

4.2 Specify the inservice inspection program to be used for the Duane Arnold plant. Compare this program with the draft USA standard "Code for Inservice Inspection of Nuclear Reactor Coolant Systems", dated October 1968, and discuss the reasons for variances.

4.3 Provide the following information concerning irradiation of the reactor vessel:

a. Does the surveillance program meet the requirements of ASTM-E-185-66?

b. What is the schedule for withdrawal of the samples?

c. Describe the source and history of the surveillance specimens. Will the specimens be from the weld, heat affected, and plate areas of the vessel as fabricated in the field?

4.4 Provide a tabulation of all the nuclear pressure vessels in the facility that are designed as Class A vessels in accordance with ASME Section III. The tabulation should include a notation of whether the vessel design is complete, the stage of fabrication of the vessel, and the extent to which each of the vessels will comply with each of the 34 supplementary criteria contained in the "Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Nuclear Pressure Vessels" issued by AEC Press Release No. IN-817 dated August 25, 1967.

4.5 Describe the seismic and vibration design of the main recirculation piping system including main pumps and their supports, and valves and their supports, particularly with respect to motor or fluid induced vibration effects and methods of control.

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4.6 To reduce the probability of the fracture of any component or piping in the reactor coolant pressure boundary which could initiate a loss-of-coolant accident, provisions to prevent brittle fracture should be applied over the entire boundary. Please provide information on the manner in which you intend to provide for this prevention in terms of material selections, operating stress, and temperature controls.

4.7 In Appendix A "Pressure Integrity of Piping and Equipment Pressure Parts", it appears that piping lines connecting to the primary system and which can be isolated by two valves would be designated Class M. State the design requirements for such piping (e.g., main steam line, reheater steam line, turbine bypass line, HPCI and RCIC turbine supply, and core spray piping).

4.8 Specify which codes and standards apply to the design, fabrication, testing and inspection of the main steam line flow restrictors. Since neither the ASME boiler or ~~USAS~~ piping codes apply specifically to the design of the restrictors, how will the provisions of these codes be applied? Describe the analyses and tests which will be done to demonstrate that the design will not fail due to vibration, transient thermal stresses, or combined blowdown and earthquake loads. How will the sensitization of the stainless steel be prevented during fabrication?

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5.0 Containment

5.1 It appears from the PSAR that no positive anchorage system is provided between the interior concrete structure supporting the reactor, the lower part of the drywell shell, and the concrete foundation under the drywell. The dynamic characteristics of these three structures are very different and their response to a seismic disturbance will be dissimilar. In view of this, provide for the design earthquake and the maximum earthquake, a discussion of the following:

a. Is the usual assumption that the three structures are rigidly connected at the foundation level, a realistic assumption? It may be that differential sliding of the structures, which may rotate about the common center of rotation (center of the spherical part of the drywell) will occur, rupturing the bond between the steel and the concrete.

b. If such sliding occurs, what are the consequences from the point of view of dynamic stresses and strains for the three structures?

c. If the bond between the steel and concrete is broken, can corrosion of the steel shell occur? Consider in this discussion, the effect of the temperature gradient, and of the resulting thermal expansion of the shell, on the bond between steel and concrete.

d. Provide diagrams, showing, for both earthquakes, and separately, for each of the three structures:

- the earthquake accelerations, at different levels;
- the overturning, or bending moments and the shears;
- the relative displacements.

5.2 For the gap between the drywell and the concrete, describe (a) how the gap will be maintained, and to what tolerances, during construction (b) the effect of jet forces impinging on the drywell at the gap, (c) how the gap will be ventilated and drained, (d) the effect on the drywell during normal operation or accident if there is a sharp, local blockage in the gap.

5.3 Show how the interior support walls and columns will transfer their loads and shears through the drywell into the foundation.

5.4 Describe the effect of an earthquake on a flooded drywell.

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5.5 Describe the dynamic interaction during earthquake between the reactor building and the soil, as the reactor building foundations extend more than 40 feet below grade. Also describe how the layer of till between foundation and bedrock will be taken into account in foundation design and dynamic analyses.

5.6 Describe how the torus is designed for earthquake forces.

5.7 For drywell penetrations show:

a. That the drywell reinforcement about a penetration, as in Figure 5.2.3, complies with the ASME Boiler and Pressure Vessel Code requirements for Class B vessels.

b. How the potential reaction forces on the penetrations are computed and accommodated in the design.

c. How the stress levels in the drywell shell openings at penetrations are computed and what limits are established.

5.8 Describe the material used for the principal Class I component supports (e.g., drywell, reactor, torus, main pumps, etc.) and give the stress/strain limits and thermal gradients used. Provide a detailed sketch of the reactor and drywell supports showing concrete reinforcing, connection details and sizes, and loads, especially shears, on these connections.

5.9 For the strength and leak rate test of the drywell and pressure suppression system, discuss the following:

a. How will the drywell and the torus be supported during these tests? If temporary supports are used, describe the type to be used and provisions which will be made for removal of the temporary supports after the tests are completed.

b. The lower part of the drywell shell is sandwiched between the concrete foundation under it and the slab supporting the interior concrete structure; it will therefore be impossible to inspect this part of the steel shell after the construction is completed. Is it planned to extend the soap bubble test to the full area of the steel plates and not only to the welded seams?

5.10 Provide the criteria for splicing of reinforcing bars.

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5.11 Provide the design bases for the sizing of the standby gas treatment system. Describe how leakage of air into the reactor building due to cracking of the concrete and deterioration of the siding sealant will be controlled.

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6.0 Core Standby Cooling System

6.1 Discuss the capabilities of the Low Pressure Coolant Injection System and the Core Spray System to accommodate a single passive failure in the pump suction lines. Include in the discussion:

a. The design of the piping from the suppression chamber to the pumps.

b. The capability within each system for detection and isolation of leakage

c. The protection of the pumps from flooding by the leakage

d. The ability to maintain the required NPSH for the pumps

6.2 What value of mixing efficiency was assumed in calculating the depressurization effect of the High Pressure Coolant Injection System?

6.3 Will the design be modified to include changes in the auto-relief logic which have been made on similar plants? These changes are:

a. Actuation of auto-relief coincident with core spray and LPCI to prevent repressurization.

b. Removal of the HPCI-low-flow permissive so that auto-relief actuation occurs for low-low vessel level and high drywell pressure.

c. Provision ~~for an interlock~~ to prevent actuation of auto-relief in the event ~~power is not~~ available ~~to drive the LPCI pumps.~~ low pressure core cooling capability is not

6.4 If not provided in the answer to question 3.3, describe the design criterion which will be applied to the Core Standby Cooling System equipment for the condition of concurrent occurrence of the design basis earthquake and the worst pipe rupture.

6.5 Provide information on the performance of the high pressure coolant injection pump turbine under accident conditions.

a. Provide the most probable and lower limit of quality of the steam flowing to the turbine for the full range of main steam line break sizes.

b. Describe HPCI turbine performance over the range of qualities calculated, particularly at the largest break size for which the HPCI system is designed.

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c. Evaluate the possible damage to the HPCI turbine due to the slugs of water which will carryover during a break of the main steam line.

d. Evaluate the offsite radiological doses if the HPCI turbine is damaged.

6.6 Describe the inservice inspection program and leak detection procedures planned for the Core Standby Cooling Systems.

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7.0 Instrumentation

7.1 In regard to the protection systems which actuate reactor trip and engineered safety feature action the following information is requested:

a. For those systems designed and built by General Electric identify which are identical to a recently reviewed plant (e.g., the Bell Station). Discuss any design differences.

b. Where systems are designed and/or built by other than General Electric, identify the supplier of the system.

c. Identify any features of the design which differ from the criteria of IEEE 279 and the GDC. Explain the reasons for such differences.

7.2 What criteria are used to determine whether instrumentation systems are considered to be "safety related" or "operational" types, as indicated on page 7.101 of the PSAR? Does this categorization represent a change from that used for recent GE plants (i.e., Bell, Hatch, and Brunswick)?

7.3 Describe and provide a list of all interlock functions performed by the mode switch. Why are the interlock functions not considered a part of the reactor protection system as stated on page 7.2-14 of the PSAR? How is channel separation and independence maintained?

7.4 Provide a list of all manual and automatic reactor trip bypasses and describe how each is reinstated in accordance with section 4.12 of IEEE 279.

7.5 Provide a list of all electrical equipment which will be tested under simulated accident condtions, as stated on page 7.3-37 of the PSAR. Discuss what accident conditions will be considered during the tests and to what extent combinations of environmental conditions will be applied simultaneously.

7.6 What are your seismic design bases for the reactor protection system, the emergency electric power system, and the instrumentation and controls for both the engineered safety features and the decay heat removal system? Will the systems be designed to be capable of actuating reactor trip or engineered safety feature action during the maximum peak acceleration? If a seismic disturbance occurred after a major accident, would emergency core cooling be interrupted? What tests and analysis will be performed to assure that the seismic design bases are met?

7.7 Please describe the quality control procedures which apply to the equipment in the reactor protection system, the engineered safety features and containment isolation systems, and the associated emergency power systems. This description should include, but not necessarily be limited to, (a) quality control procedures used during equipment fabrication, shipment, field storage, field installation, and system component checkout; and (b) records pertaining to (a) above.

7.8 Please provide your cable installation design criteria being used to protect the redundancy of the reactor protection system and engineered safety feature circuits (power, control and instrumentation). While the PSAR provides a few criteria for certain systems, please provide the criteria that apply to protection systems. For the purpose of cable installation, the protective function circuits should be interpreted in their broadest sense to include:

1. Sensor to protective device (scram breakers, solenoids, pumps, valves, valve limit switches, etc.).

   a. Instrumentation cables

   b. Control cables

2. Power from source through controller to protective device.

   a. d-c power from batteries to protective device

   b. a-c power from diesel to protective device

   c. Include starting or switching circuits where appropriate (e.g., diesel starting, battery switching).

The cable installation design criteria and bases should include but not be limited to the following:

A. Cable separation

   (1) Redundant protective circuits separated by space and/or steel or concrete. Discuss cable installation in sufficient detail to show that no physical event considered credible could disable redundant channels in an unsafe direction.

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(2) Power cables separate from control and instrumentation.

B. Cable intermixing

(1) Different plant parameter signals in same wireway (including trays, ladders, junction boxes, conduit, etc.),

(2) Instrumentation and control cables in same wireway.

C. Containment penetrations

(1) Separation of penetration areas

(a) Distance and/or steel or concrete.

(2) Grouping of penetrations in each area

(a) Protection between penetrations.

(3) Separation of protective functions

D. Design and spacing of wireways

(1) Trays

(a) Loading

(2) Ladders

(3) Conduit

(4) Other

E. Types of cables - power, control and instrumentation

(1) Insulation

(2) Derating

(3) Other

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F. Overload and short circuit considerations

G. Special considerations

(1) Fire stops

(2) Cables in hazardous areas

(a) Containment

(b) Diesel generator area

(c) Protection from hot weld material or missiles

(3) Temperature monitoring

(4) Fire detection and protection

(5) Nonvital cabling - Describe in sufficient detail to show that installation of nonvital cabling does not compromise protective function cabling.

(6) Cable and wireway markings

(7) Administrative responsibility for, and control over, the foregoing during design and installation.

7.9 Provide a safety evaluation of the Neutron Monitoring System in the same detail and format as is provided for the Core Standby Cooling Systems.

7.10 Supply the physical grouping and cabinet arrangement for the SRM's, LPRM's, APRM's and RBM's in the same detail as was provided for the IRM's on page 7.5-11 of the PSAR.

7.11 Describe the basis for locating two of the three IRM channels from one of the two groups in a common cabinet with two from the other group while the remaining channels are paired in a second cabinet. Can protection be lost as the result of a fault in the common cabinet when a channel in the second cabinet is bypassed? Please show how the IRM design meets paragraph 4.11 of IEEE 279.

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7.12 Describe in detail the qualification testing of the temperature equalizing columns and condensing chambers as stated on page 7.2-26 of the PSAR.

7.13 Please provide an evaluation of your rod control system to show that you meet your criterion that single failures will not negate the effectiveness of a reactor trip. Describe the design features which prevent a single failure from allowing the simultaneous withdrawal of more than one control rod.

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8.0 Electrical Power System

8.1 Please describe how the emergency diesel generators are started automatically. Provide information as to where the signal or signals are sensed and what combination of conditions have to exist before starting each or all diesels.

8.2 Provide a description and Piping and Instrument Diagram of the Diesel Fuel Oil Transfer System.

8.3 How long can the emergency diesels run fully loaded without cooling water? If the diesels should be needed immediately after completing a fully loaded routine test and the water would be hot at this time, would sufficient cooling water be available soon enough to prevent damage?

8.4 Provide a discussion of the effect that tripping of the main generator will have on system stability and loss of offsite power.

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9.0 Radioactive Waste System

9.1 Identify the systems and tanks that will contain radionuclides and which will not be designed to prevent release of radionuclides in the event of a tornado, the maximum probable flood or the design basis earthquake. List the maximum quantity of radionuclides that would be contained in each system or tank. Provide analyses which show the whole body and critical organ doses that could result from the release of the radionuclides to unrestricted areas as a result of the tornado, flood, earthquake. Identify all factors and assumptions used in the analyses.

9.2 State the maximum stack gas release rate, in curies per second, that will be proposed as a technical specification limit. This is needed in order to evaluate the main steam line break as a design basis accident.

9.3 Evaluate the radiological consequences of a hydrogen explosion in the off-gas piping and the resulting damage to the particulate filters with simultaneous failure of the isolation valve.

OK 9.4 Justify the design of the radwaste system as a Class II system. implied Verify that all equipment is housed in Class I structures, as is [redacted] on page 9.1.7. What fraction of the total capacity of the radwaste system can be retained by the secondary containment enclosures? Discuss the effects of evaporation and seepage through cracks on the offsite dose in the event of failure of the system.

9.5 Describe the intended method of using the two 20,000 gpm service water pumps for dilution of the radioactive liquid effluent. What frequency of use is anticipated? Will an adequate supply of river water be available when needed? How is this assured (see also question 2.2)?

9.6 What number of leaking fuel pins and quantity of released activity are used as the liquid and gaseous waste systems design bases?

9.7 Describe the operation, flow paths, capacities, flow rates and anticipated concentrations in the radwaste system shown in Figure 9.1.1. Describe the processing of high activity wastes; for example, floor drains when there is a fuel leak.

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10.0 Auxiliary Systems

10.1 Describe the design features which will prevent loss of the fuel pool water as a result of tornado generated winds or missiles, main turbine missiles, or a dropped fuel cask. What means are provided to maintain adequate cooling of stored fuel in the event fuel pool water should be lost?

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10.2 How is the supply of water to the RHR and Emergency Equipment Service Water System assured? Give the design bases and material data for the intake channel and pipes. Relate the pump elevation to flood stages and low river levels. Discuss how the formation of ice on the intake will be prevented during both normal operating and design basis accident conditions.

10.3 Since the fire protection system is Class II and not protected against tornadoes, show that failure of this system will not affect Class I systems or the ultimate safe shutdown of the plant.

10.4 Since the service water is used for liquid waste dilution, verify that an adequate quantity will be available. (See also question 2.2).

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11.0 Power Conversion

11.1 Evaluate the potential for a buildup of radioactivity in the circulating water and subsequent release to the environment. Consider recriculation of the plant radioactive discharge to the cooling tower makeup water and condensate leakage during shutdown or condenser leak testing. Evaluate the necessity for monitoring the circulating water to detect unexpected activity.

11.2 Provide a description of the cooling tower which is sufficient to permit evaluation of its effect on dispersion of the stack release. Also include a tabulation of those design parameters (e.g., air flow rate, evaporation losses) which would be required to evaluate the release of radioactivity from the circulating water.

11.3 Describe the inservice inspection program and leak detection procedures planned for the main steam lines.

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12.0 Structures

12.1 Provide a list of any Class I or IM items or structures combined with, housed in, adjacent to, or supported by Class II structures, and describe the protection given these items against failure of the Class II structures. Indicate whether the collapse of the stack or cooling tower can affect Class I or IM structures or equipment.

12.2 Refering to your definition of Class I and Class IM structures, systems or components (Section 12.4.2), clarify whether the failure of any Class I or IM structure, system or component can result in a release of activity greater than the guideline values of 10 CFR 100.

12.3 Where Class I or IM tunnels, and underground piping and cables enter or leave a structure, the seismic response inside and outside the structure is quite different. Show how this will be accounted for in the designs of these systems.

12.4 Where Class I or IM equipment and piping systems are directly connected to Class II systems, show if and how the influence of seismic activity of the Class II portions on the Class I or IM sections will be considered in the Class I or IM designs, so that failure or excessive movement of Class II systems will not adversely affect Class I or IM.

12.5 Provide the seismic criteria to which the station batteries, battery racks and Class I and IM instrumentation must conform. Can commercially available equipment be purchased to these criteria? How will the seismic criteria be presented in the procurement specifications for equipment; how will specific details such as instrument mountings, wire runs, cable trays, and pressure tubing be designed to meet these criteria?

12.6 State what deformation limit criteria will be applied to the reactor vessel support pedestal under D+H+R+P+T+E' (PSAR, page 12.4-8). Also describe how the term "No Functional Failure" (page 12.4-3) will be related to allowable stresses and deformations.

12.7 Paragraphs 12.4.4 and 12.4.6 indicate that the primary containment and the reactor vessel support pedestal are designed for the simultaneous occurrence of the design basis earthquake and the design basis loss-of-coolant accident, while the other Class I and IM structures are not. Justify this inconsistency.

12.8 What is the expected response of the main steam line, the turbine stop valves, and the associated supports and structures when subjected to the operating or design basis earthquake?

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12.9 Discuss the manner in which seismic loads are considered in the design of the steam line anchor points with special attention to the amplified loads imposed by the Class II piping that are attached in the vicinity of the anchor point structure.

12.10 Provide the seismic design criteria used for other piping and associated equipment whose failure could either result in or initiate an accident with consequences equivalent to the break in a main steam line.

12.11 In order to evaluate the adequacy of Class I and IM systems, provide:

a. A list of Class I and IM Systems which are designed dynamically.

b. The method of dynamic analysis (modal or time-history) which is to be used.

c. How the input will be established for the different parts of the system.

d. The damping factors to be used for the different parts of the system.

e. How the supports, shops, anchors, tiedowns, and snubbers will be located and designed.

f. The limiting stresses and strains for safe shutdown.

12.12 If the time history method is to be used in dynamic analysis as noted in Section 12.3.2.5, provide the following information:

a. The specific time history input chosen to be consistent with the earthquake hazard and the earthquake response spectra used.

b. Details of the mathematical idealization or modeling of the reactor building, drywell and reactor vessel, and other structures and equipment for dynamic analysis.

c. The specific technique by which the design forces are computed.

12.13 Describe the structural models to be used in earthquake design and the modal analysis by which the reactor building, drywell, and drywell internals design will be made. Include the number of modes which will be used. Include the applications of the damping factors. State whether stresses due to horizontal and vertical seismic force components will be combined linearly and directly.

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12.14 In piping seismic analysis, if a three-dimensional pipe run has two or more modes which are very nearly equal, clarify how they would be classified in the categories given on page 12.3-4 of the PSAR. Also, further clarify how equipment supported on multiple support points will be treated, and whether different input conditions are to be assumed at the various supports.

12.15 Paragraph 12.4.6.3 b. states that stresses may exceed the yield point and Limit Design may be used. Clarify how far beyond the yield point this design may go. Also indicate for what structures and at what points such an approach will be acceptable, and whether locally overstressed areas may result. Provide a clearer definition with boundary values and safety margins of the concluding sentence "The resulting distortion is limited to assure no loss of function and an adequate factor of safety against collapse."

12.16 For the torus dynamic design, furnish information as to whether it is treated as a single-or multi-degree-of-freedom system, and the way in which the stiffness and location of the supports is treated.

12.17 Describe the manner in which the reactor building and diesel generator building are to be connected and the effect of an earthquake on the structural integrity of each.

12.18 The High Pressure Core Injection Pump Room is appended structurally to the containment. Describe how its leaktight integrity, and functional continuity will be ensured even under seismic effects. Will the full thickness of the torus base slab be extended at this point?

12.19 Clarify if your criterion of 300 mph winds with a 60 mph translational velocity considers an effective wind loading of 360 mph on the structural steel components of the reactor building.

12.20 Evaluate the margin of safety provided by your design against tornado effects by indicating the maximum tornado winds speeds which would not affect the safety or shutdown of the reactor systems.

12.21 The stress limits presented for tornado loading are 90% of yield stress in reinforcing steel and 85% of ultimate stress in concrete. These limits provide acceptable safety margins when used in multiple combinations of loads, or with conservative load factors. Under a combination of only normal loads and tornado loads on Class I and IM structures with an assumed tornado load factor of 1.0 (see the next question), state if the more equivalent set of limits of 90% of yield stress of reinforcing and 75% of ultimate stress in concrete will be used?

12.22 Provide the load factors to be used in limit design for the various load combinations listed in Section 12.0.

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12.23 Describe the means by which all Class IM structures and emergency safety features equipment will be protected from the maximum probable flood.

12.24 Clarify which structures, equipment and systems are to be designed as tornado resistant and which are Class I or IM. Specifically consider the following:

a. The Nuclear Steam Supply System and all reactor piping up to and including the first isolation valve external to the primary containment. (Please provide a consistent and complete definition of the Nuclear Steam Supply System since the definitions on pages 1.2.3, 12.4.2, and C.0.2 are incomplete and inconsistent).

b. The Standby Liquid Control System and the test tank.

c. The Diesel Generator System, fuel tanks, and fuel supply system.

d. The Standby Gas Treatment System, holdup pipe and filters.

e. The Radiation Monitoring System.

f. The fuel pool, handling equipment, storage racks and cooling system.

g. List the instruments and controls which are to be designed to seismic or tornado criteria.

12.25 State what information will be available to quantitatively determine the loadings experienced by Class I structures and the reactor coolant system due to an earthquake. Indicate whether accelerometers or other devices will be used and how they will be placed. State the criteria you will use to determine the need for and extent of a post-earthquake survey to determine whether continued operation is safe.

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14.0 Safety Analysis

14.1 Provide the following information concerning the detection of radioactivity released in the design basis refueling accident:

a. Provide plan and elevation drawings which show the location of the fuel storage pool, the primary containment drywell and the reactor vessel in relation to the reactor building ventilation exhaust ducts and radiation monitors.

b. Specify the time required to 1) switch from normal ventilation to the standby gas treatment system and 2) reduce the secondary containment pressure to the design value. Indicate the assumptions made in this calculation, such as, monitor response time, vent closure time, and fan startup time.

14.2 In order that the possible effects of radiolysis of water during a loss-of-coolant accident may be evaluated, provide the following information:

a. A summary of the results of applicable analytical and experimental work completed to date on radiolytic decomposition of water, and indicate areas which are not yet complete.

b. A discussion of the R&D effort contemplated which would provide information on the areas which are not yet complete. Indicate the time schedule of this work.

c. An evaluation of the safety significance of radiolysis products in the containment vessel after an accident. Include the buildup of radiolysis products as a function of time, the effects of metal-water reactions, the effects of additives to the suppression pool water.

d. The criterion which will be used to determine whether equipment to mitigate the presence of radiolysis products will be required and the design of this equipment.

14.3 Analyze the accidental dropping of a fuel cask. Estimate the radiological dose resulting from damage to spent fuel in the pool. Also estimate the dose if the cask is dropped onto a concrete floor. Provide all factors and assumptions used in the calculation.

14.4 Furnish additional discussion regarding the adequacy of the design bases and performance requirements of the engineered safety features to demonstrate that the offsite doses in the event of an accident will be within the guidelines of

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10 CFR Part 100 using the assumptions of TID 14844. In particular, consider the following:

a. The engineered safety feature equipment considered in the analyses of doses to individuals offsite for the four design-basis accidents described in Section 14.6, (loss of cooling, refueling, control rod drop, and steam line break). Relate the design bases and performance requirements of such equipment to the assumptions used in the accident analyses.

b. The design bases for the control room ventilation system mentioned on page 10.11-4 and for control room shielding discussed in Section 12.5.

c. The design-basis leak rate for the secondary containment and the design bases for the standby gas treatment system. Include in the discussion, the effects of exfiltration up to the design wind speed, the effect of the standby gas treatment fan capacity and the effect of time delays in starting this fan.

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