DISTRIBUTION:
AEC Document Rm.
Docket File
DR Reading
DRL Reading
RPB-2 Reading
C. K. Beck
M. M. Mann
R. S. Boyd
CO (2)
H. Steele (2)
H. Denton
Branch Chiefs, RP
B. Grimes
bcc: J. R. Buchanan, ORNL

MAY 1 9 1969

Docket Nos. 50-237
and 50-249

Commonwealth Edison Company
72 West Adams Street
Chicago, Illinois 60690

Attention: Mr. W. B. Behnke
Assistant to the President

Gentlemen:

We will need additional information on certain design aspects of Units 2 and 3 of the Dresden Nuclear Power Station. In particular, many of the answers to previous questions on facility seismic design matters, which were provided in Amendments 7/8 and 11/12, are incomplete. Therefore, as discussed with you at a meeting held on April 23, 1969, we are not able to complete our review in this area until the necessary information is provided.

Further, as discussed with you previously at meetings held on April 16, 17, 29 and 30, 1969, we will need additional information on certain other aspects of the facility design. These include matters related to your startup organization, the high pressure coolant injection system (HPCIS), proposed fuel operating conditions, quality assurance, tornado damage, leakage detection capabilities in the drywell, and the potential consequences of a loss of offsite power.

The specific information that is requested on the foregoing matters is described in the questions listed in the attachment to this letter. Your prompt and complete response to these questions will be necessary to complete our review for initial operation of Unit 2.

Sincerely yours,

ORIGINAL SIGNED BY
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Attachment:
Request for Additional
Information

5/15

OFFICE ▶	DRL:RP	DRL:RP	DRL:RT	DRL:RP	DRL	DRL
SURNAME ▶	HRDenton/dj	RLTedesco	RDeYoung	RSBoyd	FSchroeder	PAMorris
DATE ▶	5/6/1969	5/6/69	5/6/69	5/7/69	5/16/69	5/19/69

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION

UNITS 2 AND 3

DOCKET NOS. 50-237/249

REQUEST FOR ADDITIONAL INFORMATION

A. SEISMIC

A review of the information presented in Amendments 7 and 11 (Unit 2), Amendments 8 and 12 (Unit 3) and in Appendix D, Exhibit H, Earthquake Analysis of the Reactor Pressure Vessel, indicates that the following clarifying information is still needed to complete our evaluation.

1. With regard to Question 2.7, please provide additional information on the seismic design of the torus itself and on the dynamic interaction between the torus and the suction header. Critical stresses should be listed for both earthquakes and their location indicated for the torus, the header, and all supports. The snubbers should be specified and their effectiveness justified. Indicate whether the time history or the response spectrum has been used in the seismic analysis and include a discussion of the seismic design of the ring girder inside the torus, the downcomers and the vents with their bellows (see also the partial answer presented to Question 2.10).

2. Concerning Question 2.9, indicate what part of the main steam lines has been analyzed dynamically and what part was designed statically, and describe in detail how the anchor block between the two parts has been designed. Indicate the critical stresses in the anchor block. Explain whether the differential motion of the two adjacent buildings has been considered in the design of the main steam lines.

   Explain the use of only seven modes for the three piping systems analyzed in this facility. Specify the design basis of the hangers, anchors, stops and snubbers, and indicate the safety factors provided in the design. Specify the snubbers, and indicate their effectiveness and their maintenance problems.

   Specify whether time history or response spectrum has been used.

   For Class I piping designed statically, explain the basis for a deflection limit of $\frac{L}{480}$, justify the horizontal load of only 0.5g used for the design and indicate the design basis, the design methods and the critical stresses for pipes, all hangers, stops, anchors, and snubbers.

Explain on what basis the original snubber locations are set, and by what means these locations are verified or revised. Differentiate between systems dynamically designed and systems which are not dynamically designed.

3. To complete your answer to Question 2.13, present sketches or drawings showing the details of the reactor supporting structure including bolting, ring girder, concrete pedestal, and details of horizontal stabilizers near the top of the reactor. Discuss how the horizontal shears are carried through the bolts (with oversized holes) and at the interfaces of different elements and how the reliability and level of friction during earthquake can be justified.

   We understand you have made no provisions to prevent sliding of the inside concrete structure on the inside face of the steel plate of the drywell bottom. Please present an analysis and evaluation to justify this approach.

   Indicate critical stresses in all parts of the supporting structure: reactor skirt, steel ring girder, bolts, concrete pedestal; include thermal, seismic and jet stresses.

   Discuss the influence of the stretching of bolts on the rocking response of the reactor.

4. Regarding Question 2.14(b), we note that the value of $E=3 \times 10^6$ has been used in design. In view of the fact that this will change after the nominal 28-day period and for dynamic loads, what effect will this have on the resultant spring values and on the validity of the dynamic analysis?

5. For Question 2.14(e), justify the assumption that the drywell bottom is fixed in the foundation concrete. Indicate the stresses in the skirt anchoring the drywell to the foundation concrete, and in the drywell plates at the connection with the skirt.

6. Explain how the two response spectra plotted in Figure II.A.1 have been used. Specifically list all structures and equipment for which the actual response spectrum or the smooth spectrum have been used and where the actual time history of the El Centro earthquake has been used. How have these two response spectra been established?

7. Discuss whether the use of any one of these inputs: time history, actual El Centro response spectrum, or the smooth spectrum, may result in underestimating the response.

8. In view of the total interval of record time for the El Centro earthquake, please present your basis for using only a 10-second interval (Question II.A.1).

9. We understand that one seismic input was used for the design of the reactor vessel and another input for the design of the dry-well and reactor building.

   Please demonstrate that this use of two different inputs does not result in an inconsistency. (Question II.A.1)

10. In answer to Question II.A.2, the stack base was considered to be rigid. Please justify this assumption. Indicate clearly what seismic input has been used and what damping. Indicate which condition is critical, i.e., small earthquake, large earthquake, or wind.

11. To complete the answer to Question II.A.3, clarify that the drywell is considered to be free to move except where it is attached to the concrete around it at two points: El. 575 ft 2 in., and at the bottom.

    How sensitive to the choice of damping coefficients is the value of forces due to interaction between the reactor and supports, drywell, reactor building, and turbine building? Please define exactly the values of damping factors used in the design.

    List in detail what damping coefficients have been used for each of these buildings and structures, or parts of them.

    Since the drywell, the reactor building, and the turbine building have been analyzed as one interconnected structure, justify the neglect of any possible torsional effect due to lack of symmetry of the whole system.

12. We understand that the reactor and its concrete pedestal have been considered as a one-mass system; discuss the validity of this assumption.

B. GENERAL

1. The response to Question B-1 given in Amendment 1 to your Unit 3 application indicated that you would investigate the facility stack to assure that stack failure would not occur for tornado winds of 300 mph. Please describe the results of this investigation including assumptions, allowable stress limits, and margins available against failure.

2. Describe in detail the startup organization for Unit 2, including the authorities and responsibilities of Commonwealth Edison and General Electric and their personnel, the availability and utilization of specialized technical support, the role of licensed operators, and provide resumes of all startup personnel.

3. Because Unit 2 will be operated prior to completion of Unit 3, describe the provisions which have been made to establish physical, electrical and mechanical independence between Units 2 and 3. For each system in Unit 2 which is common to or interconnected with a counterpart in Unit 3, state the minimum completion and performance requirements and provide an evaluation showing that design bases will not be compromised by Unit 3 construction activities. Also, discuss provisions for controlling personnel access to Unit 2 from Unit 3.

4. In answers 5.2 - 5.7, Amendments 7 and 8, and their supplements, you indicated that an analysis has been performed to predict moisture carryover to the turbine and that carryover is within turbine limitations when the HPCIS is required to operate. You further indicated that failures of the turbine when slug flow occurs would not result in excessive radiation releases. To support these conclusions, please provide the following information:

   a. A description of the model used to predict moisture carryover and slug flows for steam and liquid breaks and an analysis of expected carryover and slugging as a function of break size.

   b. Turbine design specifications and limitations with respect to moisture carryover and water slugging. Relate expected performance to break sizes.

   c. The analytical and experimental bases for assuming that carryover and slugging will not cause gross failure of the HPCIS pressure boundary.

   d. Your model for calculating radiation releases in the event of pressure boundary failures, including assumptions for the source terms (steam activity and leakage rate).

   e. The design requirements and specifications of the steam line isolation valves with respect to slug flow.

f. An analysis of the ability of the steam flow sensors to reliably detect moisture carryover and slugging and an identification of the break sizes for which the sensors would initiate HPCIS isolation. Show that failures of any single component will not prevent isolation of the system.

5. Provide the results of an evaluation regarding the likelihood of actuation of the automatic pressure relief system following a loss of offsite power. Describe drywell pressure and reactor vessel water level as a function of time, and consider single system failures. Include the effects of reduced diesel availability in the event of a design basis accident in one of the units. Based on our preliminary review of this transient, it appears to us that reliance is placed on utilization of engineered safety features to prevent actuation of the auto-relief system for anticipated transients. What consideration has been given to incorporation of an alternate reactor water level actuation signal that was identified as triple low level for the Oyster Creek and Nine Mile Point plants?

6. The proposed fuel operating conditions for Units 2 and 3 reflect linear power generation rates and exposures higher than those experienced previously in BWR plants. A review of available experimental data* indicates inadequate justification to support the combinations of fuel linear power generation rates and exposures requested considering both normal and anticipated transient modes of operation. Please discuss this matter in detail including considerations for developing sufficient data and possible operating limits that may be included in the Technical Specifications for assurance of fuel integrity under normal and anticipated transient modes of plant operation. (*APED-5608 and Amendment 22 for the Oyster Creek facility.)

7. Describe your plans for in-service inspection and means of local leakage detection for assurance that the likelihood of failure of the main steam and feedwater lines is small. In addition, describe in detail the quality assurance and control measures that were employed for these lines and in particular to those portions located in the drywell.

8. The answer to Question III.K, regarding the control room, in Amendments 9 and 10 is incomplete. In view of the complexity of the geometry involved, provide drawings to show that the stated amount of shielding is available, <u>and</u> that the geometry does not permit significant leaks of radiation from critical directions.

   Indicate on the drawings the directions from which radiation reaches the control room when there are radiation sources present in the Unit 2/3 reactor building. Describe the design features of the control room ventilation system with regard to margin to accommodate various radiation sources.

9. For the Emergency Gas Treatment System, provide the geometry of the charcoal beds, the charcoal content in pounds, and the EVESR charcoal adsorption data referenced on page 14.2-26. A schematic for the normal reactor building ventilation system, indicating flow distributions and pressure distributions, and identifying the exhaust point should also be provided.

10. The information and analysis presented on the record concerning reactor building leakage in the emergency mode under various wind conditions is incomplete. Provide a complete description of the analytical model used to analyze the performance of this system and to produce the curve in Figure III.E.1 of Amendments 9 and 10. This should include the allowances for leakage in specific areas, the exposed external surface areas of metal siding on each side of the reactor building, the assumed wind pressure on each such surface, pressure drops in major components of the exhaust system at design flow, mode of system control, changes in the pressure drop distribution which occur when the filters reach the maximum design value of pressure drop. Include a basis for the 850 cfm infiltration rate at zero wind speed (Figure III.E.1), and a discussion of system conditions which cause flow increases with wind.

11. We understand that a year of onsite meteorological data taken at elevations up to the top of the new stack is available. In view of the complexity of the stack emission limits requested for the site, this information should be provided. The data should include, if available, a joint frequency distribution of wind speed, velocity, and atmospheric stability at a level near the top of the main stack, as well as at any other levels where significant releases are anticipated. In connection with the stacks, the accurate distance from each to the nearest edge of the restricted area to the south and northwest should be supplied, as well as the distance from each to the nearest corner of the Thorsen farm and to the edge of the bluff across the river. The status, with respect to Part 20 definitions of a restricted area, of the corridor to Dresden Dam designated on the revised site drawing should also be provided.

12. Describe the post-accident recovery provisions for the drywell including instrumentation which would be available to assist in the evaluation of necessary action measures.

13. At various meetings we discussed the flooding capability of the jet pump configuration employed in the reactor pressure vessel. In particular, matters related to your pre-operational testing plans to measure leakage from the jet-pump-shroud assembly were considered. Please describe these plans and state the basis for and the allowable leakage limits considered acceptable for these tests.

14. Describe your plans with regard to improving present methods of leakage detection in the drywell, study of possible means for continuous monitoring for vibration or for the presence of loose parts in various portions of the reactor coolant system including the reactor vessel, means to detect early signs of gross failure of fuel elements, and periodic in-service inspection of the high pressure coolant system including welds in the reactor pressure vessel.

15. Your discussion of brittle fracture is limited to the reactor vessel. For the ferritic steels in the reactor coolant and associated systems, provide the material fracture properties and relate to pressurization temperatures. Include a discussion of provisions in the feedwater systems to accommodate the thermal transients expected during HPCIS operation.

16. The information contained in answer I.A.7 in Amendments 9 and 10 does not indicate whether quality assurance measures beyond applicable code requirements were followed in the design, fabrication, inspection, and installation of piping and other components within the primary coolant pressure boundary. Discuss in detail any additional measures which were specified in this regard.

17. a. The location of the radiation monitors in the refueling area should be identified. Further, the manner in which these and the monitors in the building exhaust duct are connected into the building isolation trip circuit, as well as the interconnection of their power supplies, should be described. Specify whether the design can accommodate the effects of single failures.

    b. The transit time for activity released in the reactor cavity or fuel pool to reach the normal ventilation exhaust point should be supplied. The time for radioactivity to reach the refueling monitors and to complete building isolation should also be supplied.

18. Your response to Question IV.4.5 of Amendments 9 and 10 did not contain sufficient information regarding single failure design protection of the RBM. Please identify in detail the areas of the RBM design which lack adequate redundancy or testability, and discuss the quality and equipment qualifications of the system components.

19. We understand from the Division of Compliance that certain materials and fabrication methods used for the main steam line flow restrictors are considered questionable with regard to acceptability on the basis of applicable codes. We will need your evaluation of this matter including your justification of the acceptability of the "as built" units considering their safety significance.