APR 7 1969

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CONSOLIDATED EDISON COMPANY, INDIAN POINT NUCLEAR GENERATING UNIT NO. 2, DOCKET NO. 50-247; QUESTIONS RELATING TO INSTRUMENTATION AND ELECTRICAL POWER SYSTEM

Please include the following questions among those now in preparation for transmittal to the applicant. In addition, one item was noted which Projects might wish to pursue. Figure 1.2-5 of the FSAR shows the three safety injection pumps in a common enclosure or room. The review was performed by Mr. O. D. Parr of the I&PT Branch.

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INDIAN POINT #2

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

   a. For those systems designed and built by Westinghouse identify which are identical to Ginna Station. Discuss any design differences.

   b. Where systems are designed and/or built by other than Westinghouse identify the supplier of the system. Also identify any areas which conflict with IEEE 279 and the G.D.C. Justify any conflicts.

2. In regard to the Westinghouse designed control systems, the following information is requested:

   a. Identify the control systems which are identical to Ginna Station.

   b. List and discuss any design differences. This discussion should include an evaluation of the safety significance of each system change.

3. Where reactor protection and engineered safety feature signals feed annunciators and/or a data logging computer, the design criterion used to assure isolation should be described and evaluated.

4. What are the design bases for the reactor protection system, the instrumentation and controls for engineered safety features, and

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the emergency electric power with regard to earthquakes? Will the systems 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, will emergency core cooling be interrupted? What tests and analysis were performed to assure that the seismic design bases are met?

5. Please describe the quality control procedures which apply to the equipment in the reactor protection system, engineered safety feature 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, component checkout, system check out; and (b) records pertaining to (a) above.

6. Please provide your reactor protection system and engineered safety feature installation design criteria being utilized to protect the redundancy of the reactor protection system and engineered safety feature circuits (power, control and instrumentation). While the FSAR provides a few criteria for certain systems (Page 8.2-5), please provide the criteria that apply to all safety related systems. For the purpose of cable installation, the protective function circuits should be

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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.
   (2) Power cables separate from control and instrumentation.

B. Cable intermixing
   (1) Different plant parameter signals in same wireway.*
   (2) Instrumentation and control cables in same wireway.

C. Containment penetrations
   (1) Separation of penetration areas

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*Wireway is used here to include trays, ladders, junction boxes, conduit, etc.

(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

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

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(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. Figures 1.2-3 and 5.1-4 of the FSAR shows that there is a single electrical penetration area. Figures 1.2-3 and 1.2-6 show a single electrical tunnel. A description of these two areas is requested which provides sufficient detail to show the installation of electrical cables associated with protection systems. The answer to this question might be combined with Question 6.

8. Various pages of the FSAR discuss the design of electrical equipment inside containment (e.g., Page 6.1-6). Other pages discuss the proposed testing of some components (e.g., Page 6.2-35a). Please identify all equipment and components (e.g., motors, cable, etc.) located in the primary containment which are required to be operable during and subsequent to a loss of coolant or a steam line break accident, and describe the qualification tests which have been or will be performed on each of these items to insure their availability in a combined high temperature, pressure and humidity environment.

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

10. Figure 7.2-8 of the FSAR shows two bypass breakers (AB-1 and AB-2) in the scram circuit. During the Ginna review that applicant decided to utilize one bypass breaker as a safeguard against inadvertently having both bypass breakers closed at the same time. What is your intent with regard to this problem? What information is provided to the control room operator which allows him to evaluate that the breaker(s) is installed properly?

11. Describe what information is available to the control room operator which would allow him to recognize that the door to the reactor protection system panels have been opened improperly (e.g., two doors are open at the same time). Page 7.2.3.3-3 of the Beaver Valley PSAR indicates each panel has an associated annunciator.

12. The following logic diagrams supplied with the Ginna FSAR are are requested:

   a. Activation logic diagram for Safety Injection, Containment Spray and Isolation similar to Figure 7.2-6 (Ginna FSAR).

   b. Analog channel testing arrangement similar to Figure 7.2-7 (Ginna FSAR).

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c. Actuation circuits of engineered safety feature circuitry similar to Figure 7.2-15 (Ginna FSAR).

13. Page 7.2-26 of the FSAR indicates that the Low Reactor Coolant Flow Trip has been modified from the Ginna design (a two loop plant) to provide a direct reactor trip on undervoltage on 2/4 reactor coolant pump buses. This would indicate that three loop operation is permissible. Are any manual adjustments required to the reactor protection system for the three loop mode of operation? If so, please identify the adjustments and show that unsafe operation cannot result in their misadjustment.

14. Page 7.2-1 of the FSAR discusses the Control Room. Please provide the following additional information:

a. Communications available to the Control Room

(1) Operational control (e.g., page party)

(2) Administrative control (e.g., Bell system)

(3) Special purpose (e.g., sound powered phones)

(4) Emergency (e.g., radio)

b. Lighting - what facilities for emergency lighting are provided?

15. For the process instrumentation which provides signals to the reactor protection and engineered safety feature actuation circuitry, please provide a table which lists the following information:

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a. The parameter being sensed

b. The type of sensor (e.g., Foxboro pressure)

c. The type of readout (e.g., indicating, blind, etc.)

d. The type of power required (e.g., external, self)

e. Use of channel

f. Failure mode

g. Sensor errors

h. Number of sensors on common line or penetration

16. Page 6.2-12 of the FSAR states that during safety injection the first low level alarm ($\approx$ 15 minutes) on the refueling water storage tank sounds and the operator should take appropriate action to assure that only a certain number of pumps are operating. On a second alarm ($\approx$ 22 minutes) the operator switches to the recirculation mode. Figure 6.2-1 shows one level indicator and one level alarm on the refueling water storage tank. Please describe the instrumentation provided to the operator in sufficient detail to show that a single failure cannot lead to improper operation or an interruption of cooling to the core.

17. Page 7.3-10 of the FSAR states that the a.c. power supply for the rod drive system uses a single overhead run of enclosed duct which is bolted to and therefore comprises part of the power cabinet arrangement. What is the distance for this plant?

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18. Please perform an analysis to demonstrate the ability of IPP #2 to meet Criterion 39 recognizing that a single 138 kv line connects the station to the Buchanan substation, a distance of approximately 3/4 mile. This analysis should include:

   a. The ability of the grid to meet Criterion 39 while lossing the IPP #2 generating capacity or the next largest unit on the grid.

   b. The degree of backup being supplied by the 13 kv line to include:

      (1) Capacity of 13 kv line with regard to engineered safety feature and safe shutdown loads; to include line and 13/6.9 kv transformer.

      (2) Starting characteristics of the gas turbine; what signals are provided to start the generator? What are onsite fuel storage details and fuel resupply for the generator?

      (3) What are siting details and protection provided for the 13 kv line and the gas turbine?

      (4) Under what conditions is power from this source required for IPP #1?

19. The three diesel generators are located in a common enclosure which does not appear to meet the present day tornado requirements, nor is there any protection between the machines in case of missiles. Please explain the rationale for placing the

diesel generators in a common enclosure. Also discuss the rationale for providing fuel oil for 54 hours of operation for two diesels.

20. From the information contained on Page 8.2-14 of the FSAR, it has been determined that the IPP #2 onsite power system is designed identical to that originally proposed for IPP #3 and commented on in the ACRS letter. If the proposed design is to be retained, please perform an analysis to show that the independence of the onsite power is not compromised by the use of automatic breakers between essential buses.

21. Please provide an analysis to show that a single failure in the d.c. power to the station switchyard will not prevent the offsite power from meeting Criterion 39.

22. What are your design bases for the safety related electrical equipment (control room or other equipment rooms) which take into account the loss of the air conditioning and/or ventilation system(s)? Please describe the analysis performed to identify the worst case environment (e.g., temperature, humidity) for the instrumentation, control and electrical equipment. What is the limiting condition with regard to temperature that would require reactor shutdown? What is your basis? Describe any testing (factory and onsite) which has been or will be performed to determine the equipment characteristics for this environment.

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23. What are your design bases for reactor protection system and engineered safety feature related equipment located in containment or elsewhere in the plant (e.g., sensors, cabling, pumps) which take into account the effect of radiation on the component? Please describe the analysis or testing performed to determine the long term effect and the DBA condition superimposed on the long term effect.

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