Docket

Docket No. 50-286 AUG - 2 1971

Consolidated Edison Company
of New York, Inc.
ATTN: William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

On June 19, 1971, the AEC adopted interim acceptance criteria for the performance of emergency core cooling systems (ECCS) in light-water nuclear power plants. A copy of the Commission's interim policy statement on this matter is enclosed for your information. In accordance with Section IV.B. of the interim policy statement, before we can complete our evaluation of the ECCS for the Indian Point Nuclear Generating Unit No. 3 we will need information to show that the system meets the general criteria of Section IV.A. using a suitable evaluation model. Appendix A, Part 3 of the interim policy statement describes an evaluation model acceptable to the Commission for plants incorporating a nuclear steam supply system designed by the Westinghouse Electric Corporation. We have discussed this request with representatives of the Westinghouse Electric Corporation and we understand that appropriate analyses have been or are being performed for your plant.

The information that we need regarding these analyses is:

(1) For the break size range, location and type mentioned in Appendix A, Part 3, of the interim policy statement, provide information pertaining to (a) the system pressure, (b) the hot-spot clad temperature, local mass velocity, fluid temperature, and heat transfer coefficient, (c) the core pressure drop, quality, and mass velocity, (d) the heat flux distribution in the hot channel, (e) the flow rates in the upper and lower plenums, (f) the flow rates in the broken and intact cold-leg and hot-leg piping, (g) the flow rate out of the break, and (h) percent clad metal-water reaction.

(2) Provide a detailed discussion of the calculation used to predict heat transfer during the reflood portion of the transient.

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(3) Discuss in detail any deviations in the evaluation model used in the foregoing studies from that described in Appendix A, Part 3 of the interim policy statement.

In addition, you should submit for our review any changes to the Technical Specifications for the plant that may be required on the basis of the results of your analyses.

When this information has been prepared, please submit it as an amendment to your application.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
AEC Interim Policy Statement

cc: Leboeuf, Lamb, Leiby & MacRae
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Washington, D. C. 20036

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REQUEST FOR ADDITIONAL INFORMATION

CONSOLIDATED EDISON OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 GENERAL

1.3 In the event that the results of tests planned for Indian Point Unit No. 2 are unavailable at the time Unit No. 3 is ready for operation, indicate how you will obtain the data needed to evaluate core stability.

3.0 REACTOR DESIGN

3.12 Describe the system that you plan to use to monitor the distribution of power in the X-Y plane and specify the operational limits and reactor trip limits associated with this system as they will appear in the Technical Specifications.

3.13 References to model studies and tests in Section 3 of the SAR indicate that Indian Point Unit No. 2 is the prototype plant that will provide vibrational test data for evaluating the adequacy of the Indian Point Unit No. 3 core support structures to withstand flow induced vibration effects. However, the use of prototype results are valid only if the analytical methods and procedures employed for the prototype have been confirmed by an acceptable preoperational vibration test program. Provide the test data and supporting analyses that validate the use of Indian Point Unit No. 2 as the prototype for Unit No. 3.

4.0 REACTOR COOLANT SYSTEM

4.27 Indicate the extent to which use of sensitized stainless steel piping in Unit No. 3 conforms to the following criteria:

(1) Sensitized non-stabilized stainless steel piping with a carbon content equal to or less than 0.03% is acceptable.

(2) Sensitized non-stabilized stainless steel with a carbon content greater than 0.03% may be used in any system related to safety provided that:

(a) the piping is in a system that is not connected to or that can be isolated from the reactor coolant pressure boundary by two or more isolation valves;

(b) the piping is accessible for inservice inspection and the frequency of inservice inspection will be greater than normally required; and

(c) a single failure in the piping would not preclude adequate protection for the reactor core should the failure occur following a loss-of-coolant accident.

(3) Sensitized non-stabliized stainless steel piping that does not meet Criterion (1) or (2) above, must be replaced unless it can be demonstrated to be adequate for the intended service on the basis of supporting data or tests.

5.0 STRUCTURES

5.30 With regard to Section 5.2 of the SAR provide the following information on the containment isolation system:

5.30.1 Identify those valves that are locked closed, or otherwise closed and under administrative control, that you consider as automatic trip valves in your isolation system.

5.30.2 Six classes, or categories, of lines penetrating the containment are defined in Section 5.2.2 of the SAR. Table 5.2-1 in the SAR identifies each containment penetration and provides information regarding the isolation of that penetration. Identify the class or category assigned to each of the items listed in Table 5.2-1.

5.30.3 Table 5.2-1 identifies the isolation valves whose position will be indicated in the control room. Describe the indication that will be available, its location in the control room, the planned monitoring frequency, and the audible alarm features that will be provided. State the criteria that you used to determine which valves would have positions annunciated in the control room.

5.30.4 Correct the inconsistent information presented in Table 5.2-1 and Figure 5.2-3 in the SAR with regard to line No. 8 (the letdown line) and line 41 (service air).

5.30.5 Some of the figures presented in the SAR for the containment piping penetrations (for example Figures 5.2-8 and 5.2-9) make no reference to seismic design (the notation S-1, is not on every drawing). Indicate the seismic design specification on these figures.

5.30.6 Identify all containment penetrations, except electrical, that are not now listed in Table 5.2-1 in the SAR (for example, the personnel and equipment hatches). Describe these penetrations, including the instrumentation and administrative controls that will govern their use.

5.31 Regarding the post-accident containment venting system described in Section 5.4 of the SAR, describe in detail the major components of this system, the seismic design, and the redundancy requirements.

5.32 With regard to the hydrogen recombiner system, provide the following information:

5.32.1 Normal and emergency power requirements for the recombiners.

5.32.2 Provisions for minimizing a local explosion in the area of the combustors due to an excessive concentration of gaseous hydrogen.

5.32.3 Conservatively calculated personnel doses that may be experienced in the area of the control panel during equipment maintenance. Discuss the potential for airborne radioactivity.

5.32.4 The time required to analyze a hydrogen sample of the containment atmosphere.

5.32.5 Extent to which the corrosion of zinc has been considered in your hydrogen evolution analysis. Indicate the amount of this material that would be exposed to spray inside the Unit No. 3 containment.

5.33 Provide the following information for the selected recombiner design:

5.33.1 Describe the test program that has been or will be performed to support the adequacy of the recombiner design and to demonstrate its performance capability over the expected range of combustible gas concentrations.

5.33.2 Describe the test program that has been or will be performed to demonstrate that the recombiner system will function reliably in an accident environment considering applicable conditions such as elevated pressures, temperatures, alkaline spray water impingement effects, radiation exposure, fission gas concentrations, and the turbulent conditions that may be created by the other operating safety features.

6.0 ENGINEERED SAFETY FEATURES

6.11 Provide the bases for maintaining the motor operated valves in the recirculation suction lines from the containment sump in a normally open status as indicated on Figure 6.2-1 in the SAR. Include consideration of the following in your response:

6.11.1 If the sump valves are to be normally open during plant operations, describe the provisions that have been made to prevent flow of water into the containment sump from the refueling water storage tank.

6.11.2 If the sump valves are to be normally closed, describe how the valves will be exercised and tested and the frequency of these functions. Relate this frequency to plant operation, including refueling, shutdown, and normal operating conditions. Describe how inadvertent valve opening will be prevented.

6.11.3 A certain amount of stagnant refueling water under the hydrostatic head from the refueling water storage tank may be present in the sump lines during sump valve tests. How will this water be handled? Are drain provisions required, and if so where will they be located? Provide a discussion on how these test conditions will be controlled.

6.12 With regard to the containment air recirculation cooling and filtration system (Section 6.4 of the SAR) provide the following information:

6.12.1 Describe the verification tests that have been or will be performed on the components of this system under the combined environmental effects of high humidity, pressure, temperature, radiation, and applicable chemical concentrations (as opposed to tests being performed separately). Indicate whether an assembled system of fans, motors, coolers and filters will be tested under these environmental conditions.

6.12.2 Present more detail on the system and component preoperational and postoperational testing that will be performed. Clarify the sequence of the tests and the planned frequency of testing.

6.13 Discuss the potential for and consequences of the pH becoming extremely low or high following an accident and how your design reflects consideration of these consequences. Consider the following in your response:

6.13.1 With reference to the pertinent accident analyses in Section 14 of the SAR, provide an evaluation of the consequences of eductor maloperation permitting: (1) the entire contents of the sodium hydroxide tank to enter the containment immediately upon actuation of containment spray; and (2) the water in the refueling water storage tank to be exhausted before the proper amount of sodium hydroxide has been introduced into the containment by the sprays.

6.13.2 The SAR indicates that a sump pH of 10 is most desirable. What range of pH values will be allowed for the injection water and the recirculation water? How will the pH be determined and what design features will be available that will permit an adjustment of the sump pH?

6.14 Certain components of the hydrogen recombiner system, such as the combustor unit, will be exposed to both the post loss-of-coolant accident containment environment and to high operating temperatures. Section 6.8 of the SAR states that the surfaces of such components will be protected against any corrosive environment by a Phenoline-305 paint over a CZ-11 primer. In this regard describe the protection against corrosion afforded these coated surfaces including the results of tests that have been performed to give assurance of this protection and the extent to which these coatings have been or will be used on equipment within the reactor containment building.

6.15 Certain portions of the engineered safety feature systems are located outside the containment to facilitate maintenance on system components following an accident. Provide your design bases for the shielding needed to limit the total dose to maintenance personnel who may be required to operate valves or repair equipment following a loss-of-coolant accident. Describe the bases for the selected maintenance personnel exposure times that would establish the allowable dose rates for the shielding design.

7.0 INSTRUMENTATION AND CONTROL

7.18 Table 7.2-1 in the SAR lists those parameters that would cause actuation of the safety injection system following a loss-of-coolant accident; these are (1) high containment pressure and (2) the coincidence of low pressurizer pressure and water level. Justify the inclusion of the low pressurizer water level as a coincident signal when a loss-of-coolant accident could (1) occur such that it would not result in a low pressurizer water level, or (2) occur at a location where low pressurizer water level would not be detected.

9.0 AUXILIARY SYSTEMS

9.8 Describe in greater detail the boric acid batch integrator discussed on pages 9.2-14 and -15 of the SAR. For example, present the accuracy of the instrument, the method of calibration, the provision and means provided to periodically check the operation of the integrator, and an evaluation of the effect of possible maloperation of this integrator. Discuss the basis for the selection of the specified method over other methods available for determining boric acid concentration in the reactor coolant system. Indicate the capability and availability of the specified system to perform during and following an accident.

9.9 Assuming normal primary coolant activity (following normal clean-up of the primary water prior to fuel handling operations), indicate what the expected dose rates to operations personnel would be during fuel handling operations from the activity of the water alone.

9.10 Describe in more detail the design provisions for maintaining the water temperature above 32°F in the refueling water storage tank and the exposed piping. Include the source of the steam heating provided and the assurance that steam will be there when needed.

10.0 STEAM AND POWER CONVERSION

10.3 Describe the design bases for the protection of the main steam and feedwater lines from reaction forces following failures in the reactor coolant system pressure boundary.

10.4 If the condenser heat sink is not available during a turbine trip, excess steam will be discharged to the atmosphere by twenty safety valves having a total capacity of 15 million pounds per hour (4 relief valves are also provided with a capacity of 1.4 million pounds per hour of steam at operating pressure). This atmospheric dump capability constitutes a path for the anticipated release of potentially contaminated effluents to the atmosphere. Provide a discussion of this system including consideration of the following:

10.4.1 What percent of the rated steam flow does the relief capacity to the atmosphere represent? What is the total amount of steam expected to be released per year of plant operation? Indicate the maximum quantity of radioactivity that could be released via this path assuming the plant is operating with the radioactivity concentration in the secondary system at (1) the maximum level to be permitted by the technical specifications and (2) the anticipated level assuming some tube leakage and fuel failures.

10.4.2 In Section 10.1.3 of the SAR it is stated "Monitors will ensure that any activity discharged will be within 10 CFR Part 20 limits." Describe these monitors, the system in which they are to be installed, the manner in which the cumulative release of radioactive material will be able to be determined, and the tests that have been or will be performed to demonstrate that the monitors to be used will reliably detect the appropriate range of possible radioactive releases.

10.4.3 Describe, in terms of potential radioactive releases, the practicality of modifications that could be made in the steam relief system to reduce the amount relieved to the atmosphere.

10.5 Discuss the potential for and the effects and consequences of a turbine building collapse on safety related systems and structures that would be vulnerable (e.g., main steam lines, feedwater lines, control room, service water lines, auxiliary building, diesel generators).

11.0 WASTE DISPOSAL AND RADIATION PROTECTION

11.8 It is not clear how the radiation monitors in the plant vent and containment ventilation systems will be utilized to detect iodine in the effluents. Provide more design information on how your gas monitoring system will detect iodine in these areas. Include in your discussion consideration of such means as isokinetic sampling to establish the extent to which the sample activity as seen by the detector will represent the activity concentration released to the environment.

11.9 Expand the information contained in Tables 11.1-1, 11.1-4, 11.1-5, 11.1-6 and 11.1-7 in the SAR to include, as applicable, the following information:

11.9.1 In each table include the comparative values for Units No. 1 and No. 2. Where actual data exist, so indicate.

11.9.2 List the releases for each unit as percentages of 10 CFR Part 20 limits.

11.9.3 Include all sources (for example, steam generator blowdown) in the listings of estimated and actual liquid discharges. When considering gaseous discharges include containment venting, pressure relieving operations, and secondary system relief to the atmosphere.

11.9.4 Describe the procedures and administrative controls that will coordinate and control the release of all radioactive materials from each of the Indian Point units.

11.9.5 Describe the processing of each liquid or gaseous effluent stream before its release to the environment and justify not providing some treatment for each stream.

11.10 Provide the following information with regard to the handling of gaseous radioactive materials:

11.10.1 Indicate the maximum concentrations of radioactivity expected off-site during venting of the gas decay tanks, the duration of such concentrations, and the number of operations per year. Provide the same information with regard to containment venting operations. What release limitations will be imposed on the timing and duration of the releases by meteorological considerations?

11.10.2 Provide a description of the in-line monitors that will be used to detect specific isotopes, including I-131 the controlling iodine isotope, in the presence of noble gases. Describe the method to be employed to calibrate instruments to be used in quantitatively measuring radioactive noble gases, iodines, and particulate matter.

11.11 For conditions that will exist during normal reactor operations, describe the procedures that will be employed prior to and during containment purging and venting operations and the isolation protection to be required during such operations. Describe fully the radiation detection system to be used in such operations including the system design, the alarm and isolation settings, the availability requirements for the system, the location of the detectors, and the applicable technical specifications. Indicate the sensitivity of the instrumentation to radiation levels in the range that might result from various fuel handling accidents within the containment during refueling and the ability of the system to isolate prior to the release of radioactivity from such an incident.

11.12 With regard to the Unit No. 3 sampling system, when samples from common headers (for example, containment air coolers) are analyzed, what assurance is there that samples from each of the systems are being taken and one system is not plugged?

14.0 SAFETY ANALYSIS

14.4 Provide an analysis of a loss-of-coolant accident (LOCA) that occurs while the containment is being purged, and is at a point in the reactor coolant system that is not immediately reflected by a change of the water level in the pressurizer. (Note that safety injection is actuated by a coincidence of low level and low pressure signals.) With these assumptions, safety injection will be delayed until the high containment pressure trip point is reached. However, during purging operations, two 36-inch diameter lines will be open to the atmosphere, thereby delaying pressure build-up in the containment. The only signals that will isolate the containment purge ducts are the Phase A isolation signal (initiated by the safety injection signal) or a high radiation signal in the purge ducts by a detection system for which additional information is required (see Question 11.8). The information provided should include:

14.4.1 The delay, and consequences of delay, in the start of safety injection following the LOCA while purging, using various break sizes for the LOCA up to and including the double-ended break and allowing safety injection initiation from the high containment pressure signal only.

14.4.2 The resulting off-site doses with containment isolation available only from the safety injection signal, delayed by those times determined in part 14.4.1 above.