MAY 2 1973

R. C. DeYoung, Assistant Director for Pressurized Water Reactors, L

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC., INDIAN POINT NUCLEAR GENERATING UNIT NO. 3; SAFETY EVALUATION OF THE INSTRUMENTATION; CONTROL AND ELECTRIC POWER SYSTEMS; DOCKET NO. 50-286

Plant Name: Indian Point Nuclear Generating Unit No. 3
Docket Number: 50-286
Licensing Stage: Operating License
Responsible Branch and Project Manager: PWR-1, H. Specter
Description of Response: Safety Evaluation of the Instrumentation, Control and Electric Power Systems
Requested Completion Date: May 1, 1973
Applicant's Response Date Necessary for Completion of Next Action Planned on Project: Prior to ACRS Meeting
Review Status: Complete except for Technical Specifications and Supplemental Safety Evaluation Report

The enclosed report was prepared by the L:RS, Electrical, Instrumentation and Control Systems Branch for use in the Safety Evaluation Report for Indian Point Nuclear Generating Unit No. 3. The report is based on a review of the Final Facility Description and Safety Analysis Report (FSAR) through Supplement 16 and schematic diagrams of the reactor trip system, engineered safety feature systems and safety-related electric power systems. The review was performed by R. D. Pollard, L:EI&CS and is based, in part, on the review performed by D. F. Sullivan prior to his transfer to the Directorate of Regulatory Standards.

Original Signed by
Victor Stello

Victor Stello, Jr., Assistant Director
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Directorate of Licensing

Enclosure: Safety Evaluation Report

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Memo

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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ESB-84

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Enclosure: Safety Evaluation Report

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CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

Docket No. 50-286

Instrumentation, Control and Electric Power Systems

Safety Evaluation

7.0 Protection and Control Systems

7.1 General

The protection and control systems for the Indian Point Nuclear Generating Unit 3 have been evaluated against the Commission's General Design Criteria as published July, 1971 and the Institute of Electrical and Electronics Engineers standard, IEEE 279, "Criteria for Nuclear Power Plant Protection Systems", dated August, 1968.

The evaluation of the Indian Point Unit 3 plant was accomplished by comparing its design with that of the previously evaluated Indian Point Unit 2 plant. In addition to the Final Facility Description and Safety Analysis report, various electrical diagrams were reviewed to determine that the final design conforms to the design criteria. The specific diagrams reviewed and other documents used in the review are listed in the Appendix to this report.

7.2 Reactor Trip System

The design of the reactor trip system is virtually identical to that of Indian Point Unit 2. The basic design has been reviewed extensively in the past and we conclude that the design for Indian Point Unit 3 is acceptable.

During our review we considered the adequacy of reactor protection for operation with less than four coolant loops in service. When operating with one of the coolant loops out of service the reactor is normally automatically limited to 60% of rated power. However, by manual adjustment of several protection system setpoints, adequate reactor protection can be provided for operation up to 75% of rated power. We have concluded that this aspect of the design does not conform to the requirements of IEEE Std 279-1968. However, since the need for manual adjustments during reactor power operation is expected to arise infrequently and the Technical Specifications will require adjustment of overtemperature ΔT setpoints prior to increasing the power level limit, we have concluded that the design is acceptable for the Indian Point Unit 3 plant.

7.3 Initiation and Control of Engineered Safety Feature Systems

The design of the protection systems for initiation and control of the operation of the engineered safety feature systems is functionally identical to the design for Indian Point Unit 2. The basic design has been reviewed extensively in the past and we consider it to be acceptable. Therefore, our review of the Indian Point Unit 3 design concentrated on those aspects of the design that are different than those of Unit 2.

We have reviewed the capability for testing the engineered safety feature circuits during reactor power operation. The design has been changed to permit more complete testing of the circuits during reactor operation. To prevent actuation of the associated engineered safety feature systems during the tests, operation of certain circuits is blocked. The continuity of the circuits that are not operational during the tests is verified using permanently installed equipment. Use of an ohmmeter is not necessary. Since automatic initiation of one train of engineered safety feature equipment is disabled during these tests, it is necessary to test the two logic trains one at a time. At our request, separate annuciators have been installed on the main control board to provide unique identification of the logic train being tested. Manual initiation of safety injection is not blocked during these tests. We have concluded that this testing capability is acceptable.

We have reviewed the procedure and circuits used to change operation of the safety injection system from the injection phase to the recirculation phase following a loss-of-coolant accident. To facilitate the change in operating modes of the system, a series of eight swithces are provided and these would be operated in a sequence depending on whether the high pressure injection pumps were needed in the recirculation phase. The original design was such that premature operation of certain recirculation switches could prevent operation of redundant safety injection system components. At our request, the design was modified to prevent the loss of redundant functions due to the malpositioning of any single recirculation switch while there is a safety injection signal present. We have concluded that this approach is acceptable but we have not completed our review of the necessary circuit changes. Prior to the issuance of the operating license, we will review the applicable schematic diagrams to verify that no single malpositioned recirculation switch will disable redundant functions when a safety injection signal is present.

We have also requested that the applicant re-examine the adequacy of the information available to the reactor operator during the change-over to the recirculation phase. The present procedure requires the operator to manipulate the recirculation switches in either of two sequences depending on the indicated flow in 3 out of 4 low pressure injection lines. With the present design of the power supplies for these flow instruments, a single failure could result in loss of two flow instruments. We have informed the applicant of our requirement that there must be sufficient information available to the operator to complete correctly the change-over following a loss-of-coolant accident, even in the event of any single failure. Prior to the issuance of the operating license, we will review the applicant's modifications to assure that this requirement is met.

We reviewed the design of the engineered safety feature systems to insure that the design conformed to the single failure criterion. The design of the high pressure injection system required automatic operation of the discharge valves for pump 32 if either of the other two pumps failed to start. We concluded that the design was not in conformance with the single failure criterion because of the lack of independence between the otherwise redundant pumps. In response to our requirement that the system be designed in accordance with the single failure criterion, an additional orifice was installed to provide the correct flow distribution to both injection headers without repositioning any valves, even in the event of failure of any single pump. The applicant has proposed to leave the discharge valves for pump 32 open and remove the power to the valves. We find this proposal acceptable provided that the position indication for the valves in the main control room remains operable with the power to the valves removed. An acceptable alternative would be to remove the existing automatic control circuits for the discharge valves. Prior to issuance of the operating license, we will require that one of these modifications be incorporated in the design. If the applicant's proposal is adopted in the final design, we will include a requirement in the Technical Specifications that power to the valves must be locked out.

We reviewed the design to assure that all operating bypasses conform to the requirements of IEEE Std 279-1968. At our request, an additional bypass switch was installed to provide assurance that no single failure would result in a bypass of the low pressurizer pressure/low pressurizer level signal in both safety injection logic trains. We conclude that the modified design is acceptable.

With the exceptions of the final design details discussed above, we have concluded that the design of the protection systems for initiation and control of the engineered safety feature systems conforms to the requirements of the Commission's General Design Criteria and IEEE Std 279-1968 and is therefore acceptable.

7.4 Systems Required for Safe Shutdown

The instrumentation and control systems provided for safe shutdown have been reviewed, with the exception of those associated with the auxiliary feedwater system, and we have concluded that their design is acceptable. The controls for the service water system were found acceptable provided the essential header is isolated from the conventional header during reactor operation. The Technical Specifications will require that this condition exist during reactor operation.

The applicant has stated that the auxiliary feedwater system was not established as an engineered safety feature in the Indian Point Unit 3 PSAR. In response to our request to provide the design criteria used for the instrumentation, control, and power systems associated with the auxiliary feedwater system, the applicant stated only that the system shall provide a reliable source of high pressure feedwater for plant loads below 3%. We are continuing to evaluate the safety significance of the auxiliary feedwater system. If we conclude that the system is necessary to adequately protect the health and safety of the public, we will review the design of the associated instrumentation systems to insure that the requirements of the appropriate General Design Criteria and the requirements of IEEE Std 279-1968 are met prior to issuance of the operating license.

We have reviewed the instrumentation and controls provided outside the control room and determined that they are identical to those provided for Indian Point Unit 2 and are acceptable.

7.5 Safety Related Display Instrumentation

We have reviewed the instrumentation systems that provide information to enable the operator to perform required safety functions throughout all operating conditions of the plant and to monitor the course of accidents. Except as discussed above in Section 7.3, we have concluded that the safety related display instrumentation is acceptable.

7.6 RHR System Interlocks

We are continuing our evaluation of the design of the interlocks used to prevent overpressurization of the residual heat removal system. The applicant is opposed to providing an interlock to automatically close the RHR shutdown cooling valves should primary system pressure increase. Two occurrences of pressurization of the primary system above the technical specification limit have occurred in the Indian Point 2 plant. Applicant believes that addition of an automatic closing function could cause similar problems in Unit 3. We will report the results of our review on this subject in a supplementary safety evaluation report.

7.7 Control Systems Not Required for Safety

The applicant has stated that the functional design of the reactor control systems for Indian Point Unit 3 is the same as that for Indian Point Unit 2 with the exception of minor changes in equipment. With the exception of the auxiliary feedwater system controls, we have found that such equipment changes have not changed the functional design or degraded the safety of this plant and concluded that these control systems are acceptable. The final acceptability of the overall control system scheme is predicated on the resolution of the safety significance of the auxiliary feedwater system as discussed in Section 7.4 of this evaluation.

7.8 Seismic, Radiation, and Environmental Qualification

The seismic design criteria for the reactor protection system and engineered safety feature circuits are that the equipment does not lose its capability to perform the required safety functions during or following a safe shutdown earthquake. Type tests have been performed to demonstrate conformance with the seismic design criteria. We conclude that the seismic qualification program is acceptable.

The design criteria for safety-related equipment installed inside the containment structure are that the equipment shall be capable of functioning under the post-accident temperature, pressure, humidity and radiation conditions for the time periods required. Type tests have been performed to demonstrate conformance with these design criteria. We conclude that the environmental and radiation qualification program is acceptable.

7.9 Common Mode Failures and Anticipated Transients Without Scram

In connection with our review of potential common mode failures, we have considered the need for means of preventing common mode failures from negating protective functions and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. This concern is applicable to all light water cooled power reactors.

This problem is being studied on a generic basis and we have not completed our evaluation on the schedule we had expected and indicated in our previous Safety Evaluation Report, dated February 20, 1969, on this plant. If the probability of any of the events considered is determined to be sufficiently high to warrant consideration as a design basis for plants having a nuclear steam supply system similar to Indian Point Unit 3, suitable design modifications to reduce the probabilities or to limit the consequences to acceptable levels may be necessary.

Although the regulatory staff has not completed its evaluation of this general question, we conclude that it is acceptable for the Indian Point Unit 3 reactor to operate at power levels up to rated power while final resolution of this matter is made on a reasonable time scale.

8.0 Electric Power

8.1 General

The design of the safety-related electric power systems for Indian Point Unit 3 is similar to that for Unit 2. Therefore, our review concentrated on those aspects of the design that have changed since our evaluation of Unit 2 and those aspects of the design affected by changes in regulatory requirements.

8.2 Offsite Power

Two 138 kilovolt (Kv) circuits connect the Buchanan switchyard to the Millwood Substation which is connected to the Consolidated Edison, Niagara Mohawk, and Connecticut Light and Power transmission networks. Two additional 138 Kv lines, using separate routes from the first two lines, connect the Buchanan switchyard to the Orange and Rockland system.

Two 138 Kv circuits connect the Indian Point station and the Buchanan switchyard. These circuits carry the output power from Indian Point Unit 1 and supply power to the station auxiliary transformers for Units 2 and 3. The normal source of power for startup of Unit 3 and the preferred source of power in the event of an accident is the station auxiliary transformer. A second source of offsite power is available to Unit 3 via two underground 13.8 Kv circuits from the Buchanan switchyard. In addition to power from the transmission network, power is available from two gas turbine generators, one located in the Buchanan substation and one located on the Indian Point site, which can be connected to the 13.8 Kv circuits.

We conclude that the offsite power system conforms to the requirements of General Design Criterion 17 and is acceptable.

8.4 Onsite Power

8.4.1 A-C Power Systems

Emergency a-c power is supplied by three physically and electrically independent diesel generator sets. The redundant engineered safety feature and safe shutdown loads are arranged in three groups, each group powered from its assigned diesel generator in the event of loss of offsite power. Any two of the three load groups and their associated diesel-generator sets are adequate to mitigate the consequences of an accident. No manual or automatic interconnections or transfers are necessary. We conclude that the design of the onsite a-c power system is acceptable.

8.4.2 D-C Power Systems

The applicant originally proposed the use of two d-c power systems and automatic transfer devices to supply power to the three engineered safety feature load groups. We concluded that such a design could unduly compromise the independence of redundant safety systems. At our request, the applicant modified the design to eliminate the need for automatic transfers between redundant power sources. This was accomplished by the addition of a third d-c power system.

We conclude that the modified design of the d-c power system is compatible with the a-c power system, meets the regulatory positions of Regulatory Guide 1.6 (formerly Safety Guide 6), and is acceptable.

8.4.3 Instrument Power Supplies

We have not completed our review of the power supplies for the four vital instrument buses. As a result of the changes in the design of the onsite d-c power systems discussed in Section 8.4.2 above, it is expected that the instrument power supplies will be changed. We have informed the applicant of our requirement that the power supplies for the protection system must be designed in accordance with IEEE Std 279-1968. Prior to issuance of the operating license, we will review the design changes to assure that the requirements of IEEE Std 279-1968 are met.

8.5 Separation and Identification of Redundant Protection and Emergency Power Systems

We have reviewed the means used to provide physical separation between redundant protection and emergency power systems.

The diesel generators and their local panels are located in three separate rooms of a Class I structure. Two batteries are located in separate battery rooms with no other equipment. The third battery (and its associated equipment), which was added to comply with our request, is located in the room with the diesel generator to which it supplies power. The applicant has examined the environmental conditions associated with this location and has found that operation of the battery and the diesel generator will not be adversely affected at this location.

The applicant's criteria for installation of cables and cable trays require a minimum of one foot between redundant circuits spaced either horizontally or vertically except that a minimum of three feet is required between redundant heavy power circuits space vertically. Where these distances are not provided, fire barriers are installed between redundant circuits. Two cable tunnels are provided between the control building and the containment penetration area and separation is provided by locating redundant channels on opposite sides of the tunnels.

The identification methods used to distinguish between safety and non-safety equipment and between redundant channels of safety systems are color and numeric codes.

We conclude that the identification and separation of redundant protection and emergency power systems is acceptable for this plant.

8.6 Diesel Fuel Oil System

We reviewed the design of the power and control systems for the diesel fuel oil system and concluded that the design originally proposed by the applicant was unacceptable. Specifically, all three fuel oil transfer pumps were powered from non-safety buses, their power supplies were disconnected in the event of a loss of offsite power, and the control system was vulnerable to single failures. At our request the system was modified so that the control system would meet the single failure criterion. Two fuel oil transfer pumps are now powered from safety related load centers that are automatically energized by the diesel generators. The power supply for the third pump remains unchanged from the original design.

We conclude that the control and power systems for the fuel oil transfer system are acceptable even though one pump is powered from a non-safety bus. The bases for this conclusion are:

(1) Using manual control, either of the other two transfer pumps can supply the fuel demands of all three diesels concurrently;

(2) The diesel can operate for a minimum of 55 minutes before manual control of the transfer pumps is required;

(3) Alarms are provided in the control room to indicate low level and low-low level in the diesel day tanks; and

(4) The technical specifications will require onsite storage of seven days' fuel supply in tanks other than the storage tank served by the subject transfer pump.

APPENDIX

This Appendix lists the documents used by R. D. Pollard in the preparation of the Safety Evaluation Report for Indian Point Nuclear Generating Unit No. 3.

1. 10 CFR Part 50 and Appendix A to 10 CFR Part 50.
2. Regulatory Guides 1.6, 1.9, 1.22 and 1.32.
3. Indian Point Nuclear Generating Unit No. 3 Final Facility Description and Safety Analysis Report (FSAR) through Amendment 31, FSAR Supplement 16.
4. The following drawings:

   A. Elementary Wiring Diagrams:

   Westinghouse Electric Corporation Drawing Number 500B971,

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B. Westinghouse Electric Corporation Schematic Diagrams

U.E.&C. Drawing No. 9321-LL-31173

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U.E.&C. Drawing No. 9321-LL-31183

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U.E.&C. Drawing No. 9321-LL-31333

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C. Westinghouse Electric Corporation,

U.E.&C. Drawing No. 9321-F-30493, Revision 9.

D. Westinghouse Electric Corporation, Drawing No. 113E303

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8	6

5. Letter dated April 2, 1973, from William J. Cahill, Jr. to R. C. DeYoung.

6. The following Institute of Electrical and Electronics Engineers (IEEE) Standards:

   IEEE Std 279-1968

   IEEE Std 308-1970