# Safety Evaluation

U. S. Nuclear Regulatory Commission

related to operation of

# Indian Point Nuclear Generating Unit No. 3

Office of Nuclear Reactor Regulation

Docket No. 50-286

December 1975

## Consolidated Edison Company of New York, Inc.

## Supplement No. 2

December 12, 1975

SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

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## 1.0 INTRODUCTION

The Nuclear Regulatory Commission's (Commission) Safety Evaluation Report in the matter of the application by Consolidated Edison Company of New York, Inc. (applicant) to operate the Indian Point Nuclear Generating Station Unit No. 3 (Indian Point 3) was issued on September 21, 1973. Supplement No. 1 to the Safety Evaluation Report was issued on January 16, 1975. We indicated in Supplement No. 1 that there were a number of outstanding issues which required completion. Some of these issues related to technical areas under review by the staff at the time that Supplement No. 1 was issued. Other issues related to technical areas which required additional information from the applicant to permit us to confirm that certain requirements would be met by the applicant.

This supplement is in support of our conclusions regarding a decision for issuance of an operating license authorizing fuel loading and subcritical testing of Indian Point 3. Therefore, this supplement includes a discussion of those matters that needed to be resolved to assure safe operation of Indian Point 3 during fuel loading and subcritical testing. The remaining matters which must be resolved before a decision can be made regarding power operation are also discussed. In those instances where the matter has been resolved, the results of our evaluation are presented in this supplement. In those instances where the matter is still unresolved, this supplement presents the bases for our conclusion that the matter need not be resolved prior to fuel loading and subcritical testing. We will report the final resolution of these matters in another supplement to the Safety Evaluation Report prior to a decision regarding power operation of Indian Point 3.

Each of the sections in this supplement is numbered the same as the section of the Safety Evaluation Report and Supplement No. 1 that is being updated, and is supplementary to and not in lieu of the discussion in the Safety Evaluation Report and Supplement No. 1. Appendix A is a continuation of the chronology of our principal actions related to the processing of the application.

We have reviewed the recommendations of the Office of Inspection and Enforcement and conclude that all items of construction and testing necessary for fuel loading and subcritical testing have been acceptably completed. We conclude that the issuance of an operating license authorizing fuel loading and subcritical testing of Indian Point 3 will not be inimical to the common defense and security, or to the health and safety of the public.

## 2.0 SITE CHARACTERISTICS

### 2.5 Geology, Seismology, and Foundation Engineering

On July 8, 1975 the Consolidated Edison Company reported to the Nuclear Regulatory Commission that faulting had been identified near Unit 3 at the Indian Point site. We visted the site on July 9, 1975 and confirmed the existence of faulting which appeared to pass through the foundation of Unit 3. We required the utility to conduct a geologic investigation of the site area sufficient to identify all significant faulting and provide information on the age of the most recent movement. A report of that investigation was submitted to the staff in draft form on November 26, 1975, and in revised form on December 5, 1975.

The Regulatory staff has completed its review of the draft report (received Nov. 26) pertaining to the supplemental geological investigations undertaken by Consolidated Edison at the Indian Point site; also, we have reviewed the revised report in meetings with the applicant on December 8, 10 and 11, 1975. We conferred (Dec. 10th and 11th) with the Applicant's consultants, Dames and Moore, the three members of the report review panel consisting of Drs. R. A. Price (Queen's University, Canada), D. R. Coates (State University of New York) and N. M. Ratcliffe (City College of New York); and Drs. J. F. Davis, R. H. Fakundiny, P. W. Pomeroy and R. J. Dineen of the New York State Geological Survey. In addition, we examined exposures of the faults at the Indian Point site and faulted Pleistocene materials about 12 miles northwest of the site. During our examinations of these exposures, we reviewed the genesis of the faults and the evidence for determining the time of most recent movements on them. With respect to the data reviewed, and our discussions with the above named individuals, we conclude that the investigations and evaluations are, for the most part, adequate. We consider these data and discussions to indicate that the conclusions pertaining to capable faulting made during the construction permit review for Unit 3 by our advisors, the U. S. Geological Survey and our predecessor, the U. S. Atomic Energy Commission, are still valid. To resolve some of the ambiguities in the data and to clarify some of the inferences and interpretations provided by the Applicant, additional data and additional documentation involving confirmatory data must be submitted. A final staff evaluation of a forthcoming report addressing these parts will be required before the issuance of a full power license.

The Applicant's evaluation of the age of most recent movement on the site faults is based on (1) genetic association with other regional structures, and (2) a determination of the minimum age of formation of undeformed calcite crystals taken from crosscutting faults. The determination of the age of crystal formation involves a determination of the temperature at which the crystals were formed, an evaluation of depth of burial at the time of formation, and an assessment of the rates of erosion in the area. The analysis appears to have been performed and the variables assessed in a conservative manner and indicates a minimum age of crystal formation of 330,000 years before present. Geologic evidence supports a much older age of formation, possibly in the late Mesozoic (65 mybp). This is suggested by both the time of the last known geothermal event in the region and by the last known regional tectonism.

The basis for our conclusion that the faults in the area of the site are non capable, include the following:

(1) The fluid inclusion - erosion data presented by the Applicant is considered to be valid. We believe that sufficient conservatism exists in the presented data that the possible minor errors in the methodology and laboratory techniques will not alter the conclusions as to the antiquity of the faults. However, documentation regarding the potential soruces of errors and the impact of these on the validity of the final conclusion must be submitted.

(2) An important piece of evidence supporting the antiquity of the faulting is the fact that the calcite deposited in the faults and joints at the site represents a pervasive regional hydro-thermal event, the conditions of which have probably not existed in the area of the site since late Mesozoic or early Tertiary time (ca. 37 to 65 mybp).

(3) Another strong line of supportive evidence lies in the relationship of the site's tectonic structures to those of the region. While the exact details of the regional picture are in dispute, there has been no geologic evidence discovered to date which indicates that relatively recent tectonic movements have occurred.

We have required the Applicant to provide additional documentation and supporting data on several aspects of the geological investigation as follows:

(1) With respect to the fluid inclusion age dating technique, additional information is to be presented on (a) the locations where the samples obtained for dating were taken, (b) the rates of erosion in the region and locally to the site, accounting for glacial activity and (c) the sources and magnitude of error in the analysis of the temperature of formation of the calcite crystals. In addition, we require supporting data on the textural relationships between the calcite crystals and the fault planes from which they were obtained.

(2) With respect to faulting in the site region, we require additional supporting discussion of (a) the genesis of Pleistocene faulting identified 12 miles northwest of the site and (b) an offset shown on the seismic reflection data in the Hudson River just north of the site. We require additional data to determine the cause of the river bottom anomaly.

Based on our review of the Applicant's investigations, our examination of exposure at the site and our discussions with experts in the geology of the region, we have concluded that the faults which pass through the Indian Point site have not moved in more than 300, 000 years. Moreover, we consider the faulting to be structurally associated with major northeast trending regional faults which have not experienced movement since the late Mesozoic (65 mybp). We therefore conclude that the faults are not capable as defined by Appendix A to 10 CFR Part 100.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

## 3.7 Seismic Design

In the Safety Evaluation Report we indicated that a plan for the utilization of any acquired seismic data would be developed before an operating license was issued.

In Supplement 32 to the Final Facility Description and Safety Analysis Report (FSAR), the applicant augmented the description of the plan for utilization of the seismic data obtained from the station seismic monitoring equipment following an earthquake. The plan describes the analyses which will be performed to compare the measured response spectra and the design response.

We have reviewed the applicant's plan for utilization of seismic data and conclude that the criteria and procedures for comparison of measured and predicted seismic responses are in accordance with Regulatory Guide 1.12 and are therefore acceptable.

## 3.9 Mechanical Systems and Components

We were informed on May 7, 1975 by a licensee of a pressurized water reactor, Virginia Electric and Power Company, that an asymmetric loading resulting from a postulated pipe rupture at a particular location in the reactor coolant loop had not been taken into account in the original design analysis of the reactor vessel support system for North Anna Units 1 and 2 (Docket Nos. 50-338 and 50-339). This loading results from the forces induced on the internals within the reactor vessel by transient differential pressure conditions within the vessel during the postulated pipe rupture. In addition, the asymmetric loading from the transient differential pressures that would exist around the exterior of the reactor vessel from the same postulated pipe rupture were not included in the original design analysis. However, the symmetric loadings from such a pipe rupture were included in the original analysis of the reactor vessel support system.

The maximum load calculated on the reactor pressure vessel supports occurs for a postulated instantaneous guillotine break at the cold leg reactor pressure vessel nozzle. This worst case break results in pipe whip, thrust, jet impingement, reactor cavity asymmetric pressure, and reactor internals response loads being developed that must be reacted by the vessel supports.

The term "reactor internals response loads" refers more specifically to the dynamic response of the reactor vessel internals to a very short time pressure differential that would develop across the core barrel if the postulated break occurred. This pressure differential would travel across and down the core barrel, and then up through the reactor internals resulting in the asymmetric loads on the core barrel. These loads are transferred through the core barrel flange to the vessel supports. In addition, the transient pressure distribution in the reactor vessel cavity results in asymmetric loads on the reactor pressure vessel which in turn are transferred to the supports.

It is the NRC staff's opinion that the question related to the adequacy of reactor vessel support systems could be generic in nature and may apply to all PWR facilities especially those for which the design analyses were performed some time ago. We have therefore initiated a systematic review of this matter to determine what, if any, corrective measures may be required for specific facilities.

The results of studies reported to date indicate typically that, although the margins of safety may be less than originally intended, the reactor vessel support system will retain essential structural integrity and that the ultimate consequences of this postulated accident which could affect the general public are no worse than originally stated.

We requested that the applicant provide additional information required for purposes of making the necessary reassessment of the reactor vessel supports for Indian Point 3. The applicant has submitted some of the requested information and has stated that additional analyses will be performed to: (1) determine the loads in the reactor vessel support system, (2) evaluate the full restraint capability of the support system, and (3) compute the safety margins of the support system. The applicant also stated that it is intended that the results of this work will be available by December 1, 1976. After we have reviewed this information we will determine what modifications to Indian Point 3, if any, are necessary to assure that acceptable margins of safety are maintained. If modifications are necessary we will require the applicant to make them.

Based on the results of our evaluation of this phenomenon to date and in recognition of the low probability of the particular pipe rupture which could lead to additional transient loads on the support systems, we conclude that reactor operation will not create undue risk to the health and safety of the public and is therefore acceptable until we complete our generic review.

## 5.0 REACTOR COOLANT SYSTEM

### 5.2 Integrity of Reactor Coolant Pressure Boundary

This section discusses the matter of microscopic cracks discovered in the cladding of the steam generator channel heads, and steam generator tube integrity concerns.

The reactor coolant system consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a recirculating pump and a steam generator. Each steam generator consists of three sections: an evaporator section, a steam drum section, and a primary coolant channel head. The steam generator is mounted on support pads which are an integral part of the channel head.

During plant operation, reactor coolant is circulated through the reactor vessel and steam generators. Reactor coolant enters the inlet side of the steam generator channel head and flows through the U-tubes to the outlet side of the channel head and leaves the steam generator. The channel heads are carbon steel castings with stainless steel cladding on the inside. The purpose of the cladding is to prevent corrosion of the carbon steel base metal of the channel head.

On April 5, 1975, the applicant notified the Office of Inspection and Enforcement that cracks had been discovered in the channel head cladding. The applicant conducted an investigation to determine the cause of the cracks and evaluated alternative actions regarding repair of the cladding and future inservice surveillance procedures. The details of the applicant's investigation program and proposed actions were submitted in its "Technical Report on Steam Generator Channel Head Cladding - Indian Point Unit 3", dated September 19, 1975.

Based on our review of the information submitted by the applicant, we conclude that operation of Indian Point 3 with the cladding on the steam generator channel heads in its present condition is acceptable. A detailed discussion of our evaluation is contained in Appendix B to this supplement. With regard to the inservice inspection program for continued surveillance of the steam generators, we informed the applicant of our position that its proposed inservice inspection program must be augmented to include surface examinations of the cladding for at least the first three refueling shutdown inspections. We also require that the development and system verification of the ultrasonic testing method proposed by the applicant be continued and that the results be incorporated into procedures to be used during the first refueling shutdown inspection. The applicant has agreed to modify the inservice inspection program to incorporate these requirements.

We conclude that operation of Indian Point 3 with the steam generator channel heads in their present condition will not create undue risk to the health and safety of the public and is therefore acceptable. Prior to power operation of Indian Point 3, the technical specifications will be revised to require performance of an inservice inspection program acceptable to the staff.

With respect to steam generator tube integrity concerns, the NRC staff is continuing to evaluate the measures that will be taken to assure that the tubes in the steam generators in the Indian Point 3 facility will not be subjected to conditions that will cause unacceptable degradation of integrity. Suitable technical specifications related to water chemistry, monitoring and remedial action requirements will be included in any license issued for power operation. The steam generators will not be utilized during fuel loading and subcritical testing and, therefore, there is no need for such technical specifications for this license.

## 5.7 Loose Parts Monitor

In Supplement No. 1 to the Safety Evaluation Report, we noted that a loose parts monitoring system would be functional by October 1, 1975.

We have confirmed that the system has been installed and therefore conclude that this matter is acceptably resolved.

# 6.0 ENGINEERED SAFETY FEATURES

## 6.2 Containment Systems

### 6.2.3 Containment Isolation Systems

During our review of the proposed technical specifications, the applicant proposed a provision that would permit certain non-automatic containment isolation valves to be open, either continuously or intermittently, during power operation or at other times whcn containment is required. At our request, the applicant provided a list of the specific valves, the bases for allowing the valves to be open, and the administrative procedures to be used to assure that the valves are closed when required.

In the Safety Evaluation Report for Indian Point 3, we stated that we had reviewed the isolation valve arrangements for conformance to General Design Criteria 54, 55, 56, and 57 and had concluded that the design meets the intent of the criteria. Based on our evaluation of the additional information submitted by the applicant, we find that the conclusions stated in the Safety Evaluation Report are unchanged.

### 6.2.5 Leakage Testing Program

In the Safety Evaluation Report we indicated that the provisions for leakage testing would permit containment leakage rate testing in compliance with Appendix J to 10 CFR Part 50. The preoperational tests required by Appendix J to 10 CFR Part 50 have since been completed and the results submitted by the applicant in a report entitled "Preoperational Integrated Leak Rate Test of the Reactor Containment Building." In the course of our review of the integrated leak rate test and the proposed technical specifications, we identified certain aspects which were unacceptable.

We concluded that the containment reduced pressure test did not conform to Appendix J requirements and therefore was unacceptable. The peak pressure test of the containment did meet Appendix J requirements and therefore we conclude it is acceptable.

By letter dated November 7, 1975, the applicant requested an exemption from those portions of Appendix J to 10 CFR Part 50 which relate to performance of a reduced pressure leak test. The applicant's bases for the exemption request were its belief that the successful performance of the preoperational peak pressure test verifies the acceptability of the Indian Point 3 containment, and that the purpose of Appendix J has been satisfied through the demonstration of the containment's leak tightness.

The purpose of the preoperational reduced pressure test required by Appendix J is to validate the results of future periodic containment leak rate tests performed at reduced pressure. However, the applicant desires to conduct future periodic leak tests at peak pressure and has proposed appropriate technical specifiations which will require future tests to be performed at peak pressure.

We, therefore, conclude that the purpose of the rules requiring performance of a preoperational reduced pressure test would not be served by applying those rules to Indian Pont 3. On December 12, 1975, the applicant's request for exemption was granted.

The applicant also proposed to use the weld channel and penetration pressurization system to meet the intent of the Appendix J requirement that air locks shall be leakage tested after each opening and at six-month intervals. With respect to the test after each opening, we conclude that use of the weld channel and penetration pressurization system, which pressurizes the space between the double seals on each air lock door to a pressure above peak accident pressure, meets the intent of Appendix J provided that repressurization of the seals is verified after each reclosing of the air lock doors. At our request, the applicant has modified the design to provide pressure gauges outside each air lock door to indicate the pressure between the double seals on each door. We conclude that the modified design is acceptable.

With respect to the periodic test conducted at six-month intervals, we conclude that since the weld channel and penetration pressurization system does not cover the entire air lock surface that would be exposed to post-accident containment pressure, use of this system does not meet the intent of Appendix J. Therefore we will require that the air lock be tested at six-month intervals in conformance with Appendix J.

We also requested that the applicant identify the valves needed to meet the containment isolation requirements specified in General Design Criteria 54, 55, 56, and 57 but which the applicant did not include among the valves to be tested in accordance with Type C test requirements specified in Appendix J. We have reviewed the information supplied by the applicant and conclude that the intent of the Type C requirements of Appendix J to 10 CFR Part 50 has been met.

We conclude that the containment integrated leak rate testing conforms to the intent of Appendix J to 10 CFR Part 50 and is therefore acceptable subject to resolution of the frequency at which air lock tests will be conducted.

## 6.3 Emergency Core Cooling System

### 6.3.2 System Design

In the course of our review of the emergency core cooling system in regard to determining whether the system meets the requirements of Section 50.46 and Appendix K to 10 CFR Part 50, the applicant informed us that certain valves located inside the containment may become submerged following a postulated loss-of-coolant accident. The applicant has identified the particular valves, evaluated the safety significance of the valves becoming submerged, and proposed design modifications to resolve the problem.

We are reviewing the information submitted by the applicant and will report the results of our evaluation and conclusions in another supplement to the Safety Evaluation Report. We conclude that this matter need not be resolved prior to fuel loading and subcritical testing of Indian Point 3 because, as discussed further in Section 6.3.3 below, the emergency core cooling system need not be operable prior to power operation of Indian Point 3.

### 6.3.3 Performance Evaluation

In Supplement No. 1 to the Safety Evaluation Report, we indicated that our evaluation of the emergency core cooling system performance in accordance with Section 50.46 and Appendix K to 10 CFR Part 50 would be reported in a future supplement to the Safety Evaluation Report. This section discusses our evaluation and conclusions regarding the emergency core cooling system performance for a decision regarding the issuance of a license authorizing fuel loading and subcritical testing.

The applicant has submitted analyses of the emergency core cooling system performance following a postulated loss-of-coolant accident during power operation. Until our evaluation of this information is completed, we cannot reach a conclusion that power operation would be in full conformance with Section 50.46 and Appendix K to 10 CFR Part 50. In the interim, there are a number of activities such as subcritical tests that must be performed before Indian Point 3 may begin power operation. We have therefore evaluated the emergency core cooling system performance required during fuel loading and subcritical testing. This evaluation was based on the full-power evaluations submitted for Indian Point 3 and other four-loop plants utilizing Westinghouse nuclear steam supply systems.

The initial core loading of Indian Point 3 will consist entirely o. new fuel. Therefore, in the event of a postulated loss-of-coolant accident the initial decay heat would result solely from the spontaneous natural radioactive decay of the fuel and is insignificant. No forced cooling of the fuel would be necessary to prevent exceeding the fuel clad temperature and other requirements of Section 50.46 and Appendix K to 10 CFR Part 50. Until the core is made critical and operated at power, there will be no significant increase in decay heat above that generated in the new fuel. The technical specifications will contain restrictions on operation during fuel loading and subcritical testing which will prevent achieving criticality, even in the event of an operator error or an equipment failure.

Based on our review of the technical specifications proposed by the applicant, which will be made part of the license, and our evaluation of the emergency core cooling system performance required during fuel loading and subcritical testing, we have concluded that the criteria of Section 50.46 and Appendix K to 10 CFR Part 50 will be met during fuel loading and subcritical testing.

## 7.0 INSTRUMENTATION AND CONTROL SYSTEMS

### 7.8 Seismic, Radiation, and Environmental Qualification

We have in the past found acceptable the analytical and testing programs used by Westinghouse Electric Corporation and described in its topical reports to qualify electrical equipment within its scope of supply, e.g., valve motor operators, for the design basis seismic event, and for the environmental conditions to which the equipment may be exposed in the event of a loss-of-coolant accident. These topical reports have been under generic review by the staff and Westinghouse for some time. Recently we have determined that there are certain deficiencies and Westinghouse is currently modifying the programs on an expedited basis.

The applicant has identified the equipment in the Westinghouse scope of supply which utilize Westinghouse topical reports as the bases for their seismic and environmental qualification. The applicant has also stated that when the results of the verification program to certify the adequacy of the Class IE electrical equipment within the Westinghouse scope of supply are available, they will be reviewed and corrective actions as necessary will be taken.

We conclude that this matter has been acceptably resolved. Although the modified qualification programs could result in identifying some deficiencies in the installed electrical equipment in Indian Point 3, we conclude that plant operation prior to completion of the modified qualification programs and the correcting of any deficiencies that may be identified, both of which are expected within one year, is acceptable because of the low probability of occurrence within that period of time of the environmental conditions that might adversely affect the operability of this equipment. If any defiencies in the installed equipment are identified, we will take appropriate action at that time.

## 9.0 AUXILIARY SYSTEMS

### 9.5 Other Auxiliary Systems

#### 9.5.4 Diesel Generator Cooling Water System

In Supplement No. 1 to the Safety Evaluation Report, we discussed the bases for our conclusion that the essential loads served by the service water system will receive adequate flow in the event of a service water system pipe break. This conclusion was based, in part, on the ability of the PIPEFLO computer program to calculate accurately the flow to the essential loads, including the diesel generators, in the event of a pipe break. Therefore, we indicated that we would require the applicant to submit a comparison of the functional test results of the service water system and the predicted test results presented in the FSAR.

The applicant has performed the requested preoperational tests of system pressures and flows to verify computer code accuracy for this system. The results were consistent with PIPEFLO predictions.

We conclude that the diesel generator cooling water supply from the existing service water system can accommodate the passive failure postulated in the Safety Evaluation Report and, therefore, is acceptable.

## 11.0 RADIOACTIVE WASTE MANAGEMENT

### 11.2 Liquid Wastes

#### 11.2.3 Steam Generator Blowdown

In Supplement No. 1 to the Safety Evaluation Report, we reported that the applicant had committed to installing an intertie between Indian Point 3 and Indian Point 1 to direct the blowdown flow from the Indian Point 3 steam generators to the Secondary Boiler Blowdown Purification system located at the Indian Point 1 plant.

We have confirmed that the intertie has been installed and, therefore, conclude that this matter is acceptably resolved.

### 11.3 Gaseous Wastes

#### 11.3.4 Steam Generator Blowdown

In Supplement No. 1 to the Safety Evaluation Report, we reported that the applicant had committed to installing a system for continuous monitoring of the blowdown effluent from the flash tank vents at Indian Point 1 and Indian Point 3.

We have confirmed that monitors have been installed on the blowdown flash tank vents of Indian Point 1 and Indian Point 3 and, therefore, conclude that this matter is acceptably resolved.

## 12.0 RADIATION PROTECTION

### 12.3 Health Physics Program

In the Safety Evaluation Report we discussed the bases for our conclusion that the planned radiation safety program of Indian Point 3 would assure that occupational exposures will be maintained within the established guidelines of 10 CFR Part 20. This section updates the Safety Evaluation Report with regard to the handling and storage of radioactive sources.

The applicant's radioactive materials safety program considers storage and shielding of sealed radioactive sources and the provisions taken to assure protection against undue exposure while handling them. The program also limits use of these sources to experienced and qualified personnel.

We have reviewed this program and conclude that adequate precautions have been taken with respect to the presence and location of each source, the adequacy of the physical safeguards, the administrative controls to control personnel exposure during the time the sources are being used, and the level of experience that has been acquired for handling them. All of these items will limit exposures to personnel from sealed radioactive sources to as low as practicable in accordance with 10 CFR Part 20 and, therefore, we conclude that the radioactive material safety program is acceptable.

## 18.0 THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

### 18.1 Isolation of Low Pressure Systems Connected to the Primary System

In Supplement No. 1 to the Safety Evaluation Report, we addressed the ACRS recommendation that the matter of testing of the proper positioning of check valves intended to isolate low pressure systems connected to the primary system be resolved in a manner satisfactory to the staff. We indicated that the applicant would provide a list of all check valves that are opened only during refueling and all check valves that open and close between refuelings. We also indicated that a procedure for testing the position of each of these check valves would be prepared by the applicant and reviewed by the staff.

The applicant has submitted proposed technical specifications which include a list of check valves that are opened only during refueling and will be tested at the conclusion of each refueling shutdown, and a list of check valves that are opened between refuelings and will be tested approximately midway between refuelings. The Office of Inspection and Enforcement has reviewed the procedures prepared by the applicant for testing the position of each of these check valves and concludes that the testing procedures are an acceptable method of verifying proper position of these check valves.

We conclude that this matter is acceptably resolved. The final technical specifications for power operation will include requirements for testing the position of the check valves at the intervals discussed above. The technical specifications attached to a license authorizing fuel loading and subcritical testing need not include these requirements because of the insignificant fission product inventory generated during such operation.

### 18.2 Turbine Overspeed

In Supplement No. 1 to the Safety Evaluation Report, we addressed the ACRS recommendation that the matter of design modifications to reduce the turbine overspeed be resolved in a manner satisfactory to the staff. We indicated that we would review the technical specifications and their bases prior to issuance of an operating license.

Since Supplement No. 1 was issued, the applicant has completed installation of the low pressure steam dump system which will extract steam from the supply lines to the moisture separators and route this steam to the condenser through dump valves. The applicant has also proposed technical specifications which require the submittal of a special report covering performance of the low pressure steam dump system during tests performed at a power level higher than 85 percent of the license application rating of 3025 MWt. The results of the tests will be extrapolated to verify performance at the design conditions for the license application rating.

We have reviewed the proposed technical specifications and conclude they are acceptable. In the event of a turbine trip at or below approximately 91 percent of the license application rating, the turbine will not experience excessive overspeed as has been demonstrated on Indian Point 2. Performance of the test on Indian Point 3 at a power level above 85 percent (but below 91 percent) of rated power will reduce the error that could otherwise be introduced by extrapolation from a lower level and will not result in excessive overspeed even if the low pressure steam dump system were to fail during the test.

We conclude that this matter is acceptably resolved. We will review the results of the test of the low pressure steam dump system when they are available. If any deficiencies are disclosed, we will take appropriate action before authorizing operation above 91 percent of rated power.

## 18.9 Administrative Controls to Prevent Overpressurization

In Supplement No. 1 to the Safety Evaluation Report, we addressed the ACRS concern regarding the adequacy of administrative procedures to prevent overpressurization of the reactor vessel below operating temperatures. We indicated that the applicant was developing operating procedures and that the Office of Inspection and Enforcement would review the final procedures prior to the issuance of an operating license.

The Office of Inspection and Enforcement has reviewed the operating procedures and administrative controls developed for Indian Point 3 and concludes that they provide adequate protection against over-pressurization of the reactor vessel below operating temperatures.

We conclude that this matter is acceptably resolved.

## 22.0 CONCLUSIONS

Based on our evaluation of the application as set forth in the Safety Evaluation Report, Supplement No. 1 to the Safety Evaluation Report, and in this Supplement No. 2 to the Safety Evaluation Report, we reaffirm our conclusions as stated in the Safety Evaluation Report and we conclude that fuel loading and subcritical testing of Indian Point 3 can be conducted without undue risk to the health and safety of the public.

## APPENDIX A

## CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW OF INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

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| November 8, 1974 | Meeting with applicant to discuss financial matters |
| December 10, 1974 | Letter to applicant requesting quality assurance information relative to WASH-1284, WASH-1309, and WASH-1283 |
| December 13, 1974 | Letter from Foster Associates, Inc. concerning financial qualifications of applicant |
| January 13, 1975 | Letter from applicant concerning quality assurance, in response to request of December 10, 1974 |
| January 16, 1975 | Issuance of Supplement No. 1 to Safety Evaluation Report |
| February 3, 1975 | Submittal of amended motion to extend construction completion date to July 1, 1975 |
| February 28, 1975 | Letter to applicant extending construction completion date to July 1, 1975 |
| March 14, 1975 | Letter from applicant transmitting Amendment No. 13 (Supplement No. 29) consisting of large break loss-of-coolant analysis and related proposed technical specifications |
| March 24, 1975 | Letter to applicant concerning QA program requirements |
| March 27, 1975 | Letter from applicant transmitting the following reports: "A Technical Report on Steam Generator Tubesheet Cladding," "Report on Consolidated Edison's Indian Point Unit No. 3, Containment Vessel Structural Integrity Test," and "Preoperational Integrated Leak Rate Test on the Reactor Containment Building" |
| April 11, 1975 | Letter to applicant concerning establishment of seismic monitoring stations |
| April 24, 1975 | Letter to applicant requesting information concerning ECCS |
| April 25, 1975 | Submittal by applicant and Power Authroity of the State of New York of "Application to Amend Operating License" to permit PASNY to purchase and acquire title to Indian Point 3 |
| May 1, 1975 | Letter from applicant confirming establishment of micro-earthquake monitoring network |
| May 5, 1975 | Letter from applicant providing an interim report on steam generator cladding |
| May 6, 1975 | Letter from applicant concerning quality assurance |
| May 12, 1975 | Letter from applicant transmitting revised pages to report, "Preoperational Integrated Leak Rate Test of the Reactor Containment Building" submitted March 27, 1975 |
| May 13, 1975 | Letter to applicant requesting additional information on report, "Containment Vessel Structural Integrity Test" |
| May 23, 1975 | Letter from applicant transmitting information regarding containment vessel structural integrity test |
| May 30, 1975 | Letter from applicant requesting extension of construction completion date to August 15, 1975 |

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| June 5, 1976 | Letter from applicant transmitting Amendment No. 14, (Supplement No. 30), consisting of information relative to ECCS, conduct of operations, and other changes |
| June 24, 1975 | Letter from PASNY transmitting engineering evaluation reports (financial information) |
| June 25, 1975 | Meeting with applicant to discuss post-LOCA loadings on reactor vessel supports, and seismic and environmental qualification of Westinghouse instrumentation and electrical equipment |
| June 25, 1975 | Letter from PASNY transmitting data concerning operating costs |
| July 7, 1975 | Letter from applicant transmitting preliminary geological report concerning fault systems near the site |
| July 15, 1975 | Submittal of motion to extend construction completion date to November 15, 1975 |
| July 22, 1975 | Meeting with applicant to discuss recently discovered fault at site, including applicant's plan for investigation of fault |
| July 22, 1975 | Letter to applicant regarding calculation of loads on the reactor pressure vessel supports |
| July 25, 1975 | Letter from applicant concerning program to investigate fault |
| July 30, 1975 | Letter from applicant regarding reactor vessel support loads |
| August 8, 1975 | Letter from Westinghouse transmitting acceptable instrument (transmitter) accuracy tolerances |
| August 15, 1975 | Letter from applicant providing information on reactor vessel support system |
| August 29, 1975 | Meeting with applicant to discuss generalized cracking in the cladding of the steam generator primary water-boxes |
| September 3, 1975 | Letter from PASNY regarding implementation of QA Program |
| September 4, 1975 | Letter from applicant providing information relative to reactor vessel support |
| September 16, 1975 | Letter to applciant concerning review of containment structural integrity test results |
| September 19, 1975 | Letter from applicant transmitting "Technical Report on Steam Generator Channel Head Cladding Indian Point Unit No. 3" |
| September 23, 1975 | Letter from applicant regarding deficiency in design of sample line penetration |
| September 30, 1975 | Letter from applicant regarding reactor vessel support loads |
| October 3, 1975 | Letter from applicant transmitting revised pages for report submitted September 19, 1975 |
| October 10, 1975 | Letter from applicant transmitting "Westinghouse ECCS 4 Loop 15 x 15 Sensitivity Studies" WCAP-8558-P and WCAP-8558 (proprietary and nonproprietary versions) |
| October 10, 1975 | Letter from applicant in response to request of April 24, 1975 concerning ECCS |
| October 14, 1975 | Letter transmitting electrical drawings as proprietary information |

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| October 14-15, 1975 | Meeting with applicant to discuss outstanding issues regarding Appendix J and proposed Technical Specifications |
| October 15, 1975 | Submittal of motion to extend construction completion date to February 15, 1976 |
| October 28, 1975 | Letter from applicant transmitting Amendment No. 15 (Supplement No. 31), consisting of information relative to organizational responsibility for preoperational and startup testing |
| November 3, 1975 | Letter to applicant's attorney concerning proprietary information submitted October 14, 1975 |
| November 5, 1975 | Letter to applicant concerning outstanding items to be resolved prior to issuance of license |
| November 7, 1975 | Letter from applicant requesting exemption from certain requirements of Appendix J to 10 CFR Part 50 relative to reduced pressure leak testing of containment |
| November 10, 1975 | Meeting with applicant, PASNY, and NELPIA to discuss financial protection and indemnity considerations |
| November 10, 1975 | Letter from applicant concerning investigation of faults |
| November 12, 1975 | Letter from applicant in response to letter dated November 5, 1975 |
| November 14, 1975 | Submittal by applicant and PASNY of "Application to Amend Construction Permit and Amendment No. 1 to Application to Amend Operating License" |
| November 17, 1975 | Letter from applicant in response to letter dated November 5, 1975 |
| November 26, 1975 | Letter from applicant in response to letter dated November 5, 1975 |
| November 28, 1975 | Letter from applicant transmitting Amendment No. 16 (Supplement No. 32), consisting of miscellaneous revisions and proposed technical specifications for fuel loading and subcritical testing |
| December 1, 1975 | Letter to applicant granting withholding of proprietary report submitted October 10, 1975 |
| December 5, 1975 | Letter from applicant transmitting report, "Supplemental Geological Investigation of the Indian Point Generating Station for Consolidated Edison Company of New York, Inc." |
| December 5, 1975 | Letter from applicant transmitting Amendment No. 17, consisting of Proposed Technical Specifications for Fuel Loading and Subcritical Testing |
| December 8, 1975 | Meeting with applicant to discuss onsite geologic faults |
| December 8, 1975 | Letter from applicant concerning report submitted December 5, 1975 |
| December 10-11, 1975 | Site visit |
| December 12, 1975 | Letter to applicant granting exemption requested by letter dated November 7, 1975 |

## APPENDIX B

## EVALUATION OF STEAM GENERATOR CHANNEL HEAD CLADDING REPORT

Plant Name: Indian Point Unit Number 3
Docket Number: 50-236
Report Number: Not Specified
Report Title: Technical Report on Steam Generator Channel Head Cladding
Report Date: September 19, 1975
Originating Organization: Consolidated Edison Company
Reviewed By: Materials Engineering Branch

### SUMMARY OF REPORT

The steam generators at Indian Point Unit Number 3 are Westinghouse Series "44" designed and manufactured in accordance with ASME Code Section III (Class A Nuclear Vessels), 1965 Edition including Addenda through Summer 1966. The channel heads are carbon steel castings (ASME SA 216, Grade WCC), clad on the interior surface by series submerged are weld overlay using 309 stainless steel weld metal. The objective of the cladding is to prevent the development of carbon steel corrosion products that would enter the primary coolant system.

During post hot functional examination, cracks in the cladding oriented parallel to the weld beads were identified by liquid penetrant examination. Attempts to remove the imperfections mechanically and restore the cladding were successful in one steam generator, but resulted in increasing the number of liquid penetrant indications in the three remaining steam generators. The existing surface area exhibiting dye penetrant indications has increased two-fold since the beginning of the field repair effort and now comprises approximately 10% of the total channel head surface.

A review of the fabrication history established that sea water leakage contaminated the cladding during barge transfer of sub-assemblies. To determine the origin of the indications, Westinghouse conducted a metallurgical examination program including weldability tests, macro and light microscopic examination, scanning electron microscopy incorporating energy dispersive analysis of the X-ray spectra, transmission electron microscopy incorporating electron diffraction analysis, electron microprobe analyses and chemical analysis. The conclusion was that the probable cause of the cladding cracks was a stress corrosion mechanism acting on an excessively dilute clad deposit.

Paragraph N-444 of the applicable ASME Code Section III specified that no structural strength shall be attributed to the cladding except where bearing stress is involved. Corrosion data derived from Westinghouse tests predicts a weighted corrosion rate of less than 2 mils per year and a total corrosion penetration in 40 years of service life of the order of 0.075 inches.

It is conservatively calculated that the critical flaw size for the SA 216-WCC channel head in the area of highest service stresses is no less than 2 inches in depth with a flaw taken as a discontinuity with a sharp, leading edge. With the most conservatively postulated corrosion of 0.075" considerably less than one-tenth the critical flaw, and the reasonable expections that the corrosion penetration into the base metal will be in the form of rounded pitting rather than a sharp discontinuity, it is concluded that the postulated corrosion penetration will not affect the integrity of the channel head.

Several alternative repair plans were investigated varying in degree of complexity from accepting the present condition to removing the steam generators and recladding all the heads at the fabrication plant. Repair programs, other than accepting the present condition of the cladding were determined to be not technically feasible, extremely costly and/or would result in extended down-time.

Consolidated Edison proposes the following channel head surveillance program:

1. Increase the Section XI, 1971 Edition, requirement for visual examination of 36 square inches to a general 100% visual examination with a TV camera of the interior water box surface each inspection interval.

2. Conduct a baseline ultrasonic examination from the exterior surface of selected relatively high stress areas near the biological shielding ledge and re-examine these areas during each of the next two refueling periods. The objective of the ultrasonic examination is to detect any unexpected extension of cracking into base metal.

EVALUATION OF REPORT

Our evaluation of the alternative repair plans presented by Consolidated Edison reached a similar conclusion that field repair of the existing cladding probably would not be successful and could result in additional degradation.

Although evidence of crack extension into the base metal was not detected, the observed rust colored deposits is indicative of corrosion of the casting surface. Based on the applicant's calculations of the critical flaw size in the area of highest service stresses of no less than 2 inches in depth assuming a sharp leading edge flaw, a localized total corrosion of the channel head casting during a 40-year service life of 0.075" should not adversely affect the integrity or operation of the unit. Long-term measurement of bare carbon steel surfaces exposed to PWR service environments are available from cases such as the clad void in the Yankee Rowe reactor.

Section 10 of the report entitled "Proposed Action" requires revision. The surveillance program should include surface examination in addition to the proposed ultrasonic examination during each of the first three refueling periods. Acceptance/rejection criteria for each examination technique should be clearly stated. The ultrasonic technique described in Appendix D is sufficient for interim use during a baseline examination. However, development and system verification of the ultrasonic method should be continued on specimens of a representative channel head casting. The system accuracy and minimum detectable base metal flaw size should be completely defined. The results of the development program should be incorporated into procedures used during the first refueling examination.

The technical basis for the recommendation of surface examination is that practically all existing cladding indications are too tight for visual detection. The applicant's position in report Section 4.5 that the cladding can be considered acceptable since the ASME Code Section III, 1965 Edition including Addenda through Summer 1966, does not require liquid penetrant examination is not sufficient. Section XI, 1971 Edition, paragraph IS-312 "Supplemental Examinations" requires that indications detected shall be evaluated by other nondestructive methods, where practical, to assist in the determination of the nature (size, shape, location, orientation) before final disposition is made.

REGULATORY POSITION

1. The subject report has been reviewed and found acceptable provided the augmented inservice inspection program in Section 10 is clarified and that additional surveillance examination requirements are implemented. The applicant should verify that a fatigue evaluation was performed to provide assurance that the existing cladding condition will not adversely affect the integrity as a result of corrosion assisted fatigue.

2. We agree with the applicant that field repair of the existing cladding probably would not be successful.

3. We have reviewed the applicant's calculation of the critical flaw size and expected total corrosion rate during a 40-year service life, and agree that the present condition of the cladding is acceptable.

4. The general 100% visual examination with a TV camera proposed by the applicant probably will not be effective unless supplemented by surface examination. We recommend that during the first three refueling shutdowns, surface examination be performed on representative sections of the cladding as follows:

A. Liquid penetrant examination.
B. Surface replication with a material such as RTV-11 silicone rubber.

5. The ultrasonic examination technique described in Appendix D is adequate for interim use during a baseline examination. We recommend that development and system verification of the ultrasonic method be continued with the results incorporated into procedures used during the first refueling examination. We recommend that the ultrasonic examination be performed during the first three refueling periods.