2/6/71 Docket File

Those Listed Below

REVIEW PLAN AND REQUEST FOR TECHNICAL ASSISTANCE - CONSOLIDATED EDISON INDIAN POINT UNIT NO. 3, DOCKET NO. 50-286

The operating license review plan and our schedule of review for Indian Point Unit No. 3 is attached. DRS is requested to assist us in our review of the areas indicated in the review plan.

The responsibility for review of this facility is assigned to PWR Branch No. 1 with C. Hale assigned as Project Leader. Assisting in the review will be R. Lee. Personnel within DRL are hereby directed to conduct reviews as assigned in the attached review plan.

/S/

P. A. Morris, Director
Division of Reactor Licensing

Enclosure:
Review Plan for Indian Point Nuclear
Generating Unit No. 3

cc: C. K. Beck
M. M. Mann
S. H. Hanauer
P. A. Morris
F. Schroeder
T. R. Wilson
Assistant Directors, DRL
P. Howe
E. G. Case
R. R. Maccary
B. Grimes
R. Klecker
DRL Branch Chiefs
DRS Branch Chiefs
R. Lee
C. Hale
N. Brown

Returned 1/91
Remand 1/20

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P PDR

OFFICE ▶	RL:PWR-1	RL:PWR-1	AD/PWR	DRL	DRL	DRL
SURNAME ▶	Lee/ RSL Hale:lm	Muller	DeYoung	Wilson	Schroeder	Morris
DATE ▶	12/30/70	1/4/71	1/22/71	1/25/	2/6/	2/6/

Memo

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

February 6, 1971

Those Listed Below

REVIEW PLAN AND REQUEST FOR TECHNICAL ASSISTANCE - CONSOLIDATED EDISON INDIAN POINT UNIT NO. 3, DOCKET NO. 50-286

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OPERATING LICENSE REVIEW PLAN FOR

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

I. Introduction

Consolidated Edison Company of New York, Inc., (Con Ed) has submitted as Amendment No. 13 to its Application, the Final Facility Description and Safety Analysis Report. This document is in support of an application for a license to operate the Indian Point Unit No. 3 nuclear power plant (Docket 50-286, Permit No. CPPR-52). The Indian Point Unit No. 3 employs a pressurized water reactor nuclear steam supply system furnished by Westinghouse Electric Corporation, and is designed to operate at 3025 MWt (965 MWe net). A maximum power of 3216 MWt has been assumed for the design of engineered safety features and for the assessment of the fission product releases and radiation exposures associated with the design basis accident.

The 239-acre site is located in Westchester County, New York, on the east bank of the Hudson River at Indian Point, about 24 miles north of the New York City boundary line. The site is immediately adjacent to and south of the site of the existing Unit No. 1. The nearest city is Peekskill, 2.5 miles northeast of Indian Point with a population of about 19,000.

The design is essentially the same as Indian Point Unit No. 2 which has been licensed for construction by the Atomic Energy Commission at the same site. All functional and safety systems for Unit No. 3 will be independent of the other units at the site except for the common discharge canal. The experience gained in our review of Unit No. 2 will be reflected in our review of Unit No. 3. In particular, the review of subjects that are site related such as meteorology, geology, seismology, hydrology, environmental monitoring and emergency planning will be strongly based on our recent evaluation of Unit No. 2. Since the Zion facility is similar in many respects to Unit No. 3, and the schedules will coincide to a certain extent, we also intend to collaborate with the Zion reviewers in the evaluation of the two facilites.

II. Operating License Review Plan

The designation of the group having responsibility for review and preparation of comments or questions is indicated in parentheses beside each paragraph. Items for which comparative and/or concurrent reviews of Unit No. 2 or Zion can serve are indicated by an asterisk. Major paragraph members and subject titles coincide where possible to corresponding parts of the FFDSAR. The General Design Criteria referred to are those published in the Federal Register on July 11, 1967.

1.0 Introduction and Summary

(PWR#1)	1.1	Review and evaluate the overall station design and compare with Unit No. 2, reflecting design changes made as a result of the Unit No. 2 review.
(PWR#1)	1.2	Review the quality assurance program including the relationship of organizations responsible for design, construction and operation of the station.
(DRS)	1.3	Review and evaluate the quality assurance and quality control procedures as they apply to structures, components, and systems (electrical and mechanical) necessary to the safety of the plant.
(PWR#1/DRS)	1.4	Review status and results of research and development programs.
(PWR#1/DRS)	1.5	Determine the adequacy of information presented in Westinghouse topical reports as related to Unit No. 3.
(PWR#1)	1.6	Review those portions of the final design affected by the concerns expressed by the ACRS in their previous review of Unit No. 3.

2.0 Site and Environment

(SERSG)	2.1	Review and evaluate the population distribution data for significant changes since C.P. issuance. Rely on Unit No. 2 review.
(SERSG)	2.2	Review and evaluate environmental monitoring program and natural phenomena effects. Rely on Unit No. 2 review.
(SERSG)	2.3	Evaluate applicability of site meteorological data to radiological safety analyses. Rely on Unit No. 2 review.

(DRS)	2.4	Review and evaluate the seismic spectra selected for dynamic analyses of Class I (seismic) structures and systems considering comments from Newmark and Hall. Rely on Unit No. 2 review.
(SERSG)	2.5	Evaluate combined radiological effects of three unit operation on Indian Point site.
(PWR#1)	2.6	Assure conformance with General Design Criteria (GDC) Nos. 2 and 4.

3.0 <u>Reactor</u>

(PWR#1)	3.1	Review significant differences in the nuclear design from previously approved facilities.
(PWR#1)	3.2	Review significant differences in the thermal and hydraulic design from previously approved facilities.
(PWR#1)	3.3	Review the adequacy of the analytical methods used to calculate core thermal and hydraulic design characteristics of the plant. Ascertain the conservatism of the computational codes involved in multi-dimensional analysis of power distribution, particularly power mismatch.
(DRS)*	3.4	Review the design of the core internals, especially the capability to withstand blowdown forces and the design adequacy for elimination of unwanted vibration during normal operation.
(PWR#1)	3.5	Evaluate development of techniques for detection of failed fuel elements. Evaluate the adequacy of the action to be taken upon detection of failed fuel.
(DRS)*	3.6	Evaluate the analysis of the ability of the reactor, the reactor vessel, and the internals to withstand stresses imposed by earthquake loads and thermal shock.
(PWR#1)	3.7	Review and evaluate the performance of the boron carbide control rods.

(PWR#1)	3.8	Evaluate the proposed program for surveillance of pressurized fuel elements at high burnup with respect to assuring the fuel elements maintain their integrity while undergoing anticipated transients near the end of life.
(DRS)	3.9	Evaluate the potential consequences of the most reactive fuel assembly being inadvertently loaded into the most critical portion of the core.
(DRS)	3.10	Review and evaluate the intended use and performance of the proposed core instrumentation (e.g., incores, excores, and thermocouples) and the adequacy of the related proposed Tech. Specs.
(PWR#1)	3.11	Assure conformance with General Design Criteria Nos. 1, 2, 6, 7, 8, 10, 11, 12 13, 25, 26, 27, 28, and 29.

4.0 Reactor Coolant System

	4.1	Identify and evaluate all significant differences from previously approved systems pertaining to:
(DRS)*	4.1.1	Reactor vessel design, fabrication and installation, especially in the application of Section III and VIII ASME Boiler and Pressure Vessel Code, electroslag welding and any special problems associated with fabrication.
(DRS)*	4.1.2	Reactor primary coolant system including pumps, valves, pressurizer, steam generators, and instrumentation.
(DRS)	4.1.3	Review and evaluate the pressure relief systems.
(PWR#1)	4.2	Review and evaluate adequacy of primary system leak detection methods, and the response to be taken upon detection of leaks.

(DRS)*	4.3	Review and evaluate capability of the reactor coolant system including ECCS for in-service inspection conformance with Section XI, ASME Boiler and Pressure Vessel Code, January 1970.
(DRS)*	4.4	Review and evaluate the reactor pressure vessel for fast neutron fluence and corresponding NDT, vibration test programs, surveillance and in-service inspection programs.
(DRS/Newmark & Hall)	4.5	Evaluate seismic and thermal design of Class I (seismic) equipment and piping.
(DRS)	4.6	Determine the adequacy of missile and pipe whip protection for emergency core cooling systems and primary coolant system.
(PWR#1)	4.7	Assure conformance with GDC Nos. 1, 2, 5, 6, 9, 14, 15, 16, 29, 30, 33, 34, 35, 36 and 40.
(PWR#1)	4.8	Review adequacy of design, in-service inspection procedures, and quality control measures for primary coolant pump fly-wheels.
(PWR#1)	4.9	Review adequacy of fuel failure detection procedures.

5.0 Containment

	5.1	Review all significant design differences from previously approved facilities specifically, identify and evaluate significant differences in:
(DRS)	5.1.1	Containment structural design. Check the capability of the containment to accept thermal stresses and differential pressures calculated from the postulated LOCA.

(DRS)	5.1.2	Materials testing for installed equipment and building materials should be reviewed to assure appropriate code techniques were employed. Consider reinforcing steel, steel liner, concrete, Cadweld splices, sealants, insulation, vessels, valves, and pumps.
(DRS)	5.1.3	Penetration design and methods for assuring containment leak tightness at various times in plant life.
(PWR#1)	5.1.4	Review and evaluate the design and isolation criteria for air locks, pipe penetrations, instrumentation lines, ane electrical penetrations.
(DRS)	5.1.5	Containment leakage surveillance techniques.
(DRS/PWR#1)	5.2	Review and evaluate the calculations of peak pressure in the containment following the postulated LOCA.
(PWR#1)	5.3	Evaluate post-LOCA conditions and their long term effect on the containment. Include hydrogen build-up, potential metal-water reaction, contamination, and corrosive properties of containment materials.
(DRS)	5.4	Evaluate capability of the containment and other Class I structures to withstand effects without loss of integrity from: missiles and jets generated inside the structure, tornado-borne missiles, and hurricane or seismic forces.
(SERSG)	5.5	In conjunction with site analysis, determine the acceptability of a containment leak rate of not greater than "0.1 percent per day of the free volume at the peak calculated accident pressure." Rely on Unit No. 2 review.

(PWR#1)	5.6	Evaluate the systems provided to limit hydrogen build-up subsequent to a LOCA.
(DRS)	5.7	Evaluate adequacy and appropriateness of structural design criteria used to meet the requirements established during the C.P. review.
(PWR#1)	5.8	Assure conformance with General Design Criteria Nos. 1, 2, 3, 4, 5, 16, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 59, and 61.

6.0 Engineered Safety Features (ESF)

	6.1	Identify and evaluate significant differences from previously approved facilities in the following areas:
(DRS)	6.1.1	Design bases and performance of emergency core cooling system (ECCS); high head safety injection, low head safety injection, and accumulator injection systems.
(DRS)	6.1.2	Protection of ECCS piping and equipment from missile or pipe whip.
(DRS)	6.1.3	Assurance of adequate core cooling capability following a LOCA, including the adequacy of water supply for maintaining long-term cooling capability.
(PWR#1)	6.1.4	Adequacy of engineered safety features to depressurize containment and limit radiation doses.
(PWR#1)	6.1.5	Adequacy of accumulator isolation valve design.
(PWR#1)	6.1.6	Compatibility of ESF components in the accident environment.

(PWR#1)	6.2	Evaluate design margins on adequacy of NPSH for ECCS pumps and containment spray system pumps.
(PWR#1)	6.3	Evaluate the ESF design in conformance with criteria for "single failures" including "passive" component designations.
(DRS)	6.4	Evaluate the capability for and adequacy of any proposed in-service testing and inspection of ESF components.
(PWR#1)	6.5	Assure conformance with GDC Nos. 1, 2, 4, 5, 28, 32, 27 through 48, 58, 59, 60 and 61.

7.0 Instrumentation and Control

(DRS)	7.1	Evaluation of significant differences from previously reviewed plants of reactor protection system, control systems for the actuation and control of engineered safety features and process safety systems considering conformance with IEEE 279, including ability of system to withstand possible natural phenomena and the environment of postulated accidents, and diversity in actuation of the ECCS.
(PWR#1)	7.2	Identify and evaluate any significant differences from previously approved facilities for instrumentation and control systems associated with process radiation monitors, reactor building exhaust monitors, refueling interlocks, area radiation monitors, site radiation monitors, and containment remote monitoring instrument systems. Evaluate capability of these systems to provide necessary

surveillance and control to assure conformance with radiation exposure and radioactivity release limitations of 10 CFR Part 20 and 10 CFR Part 100.

(DRS)	7.3	Provide a review and evaluation of electrical and control schematics related to emergency power, reactor protection system, ESF systems, and containment isolation and atmosphere control systems.
(PWR#1)	7.4	Evaluate the adequacy of the control room and auxiliary control station for reactor operations under normal and abnormal conditions.
(DRS)	7.5	Evaluate the adequacy of the proposed testing of the control and protection system circuitry.
(DRS)	7.6	Review the potential for and consequences of common-mode failures in the reactor protection system.
(PWR#1)	717	Assure conformance with GDC Nos. 1 through 5, 7, 11, 12, 13, 17 through 28, 39 through 44, 46, 47, and 48.

8.0 Electrical Power Systems

(DRS)	8.1	Identify and evaluate all significant differences from previously designed facilities for offsite, onsite and d.c. power systems with particular attention to the adequacy of supplying power to protection and safety instrumentation and equipment.

(DRS)	8.2	Review and evaluate the plant's interaction with the external grid system and the relative independence of transmission lines. Rely on Unit No. 2 review.
(DRS)	8.3	Evaluate the onsite power system with respect to the switchyard bus and breaker arrangement, its connections and interlocks, and system independence. Rely on Unit No. 2 review.
(DRS)	8.4	Review and evaluate the plant's conformance to the auxiliary criteria for auxiliary electrical power systems (March 1, 1968). Consider also the switching circuits for load shedding from the emergency buses. Rely on Unit No. 2 review.
(DRS)	8.5	Evaluate the design capability of the emergency diesel generators. Check sizing and redundancy in supplying power to safety loads in conformity with current cirteria for rating of the emergency diesel generators.
(DRS)	8.6	Evaluate the conformity of Class IE electrical systems with the criteria in IEEE 279. Consider also the comments in a memo of R. L. Ferguson, October 8, 1969, to the IEEE/NSG/TCS/SC4 Auxiliary Power, subject; "IEEE-ANS Standards Program-Report #9" with enclosure titled: "Obtaining Power from the Transmission Network for Nuclear Fueled Generating Stations."

(DRS)	8.7	Review and evaluate the adequacy of the separation criteria employed for cables and penetrations of the reactor protection, engineered safety and emergency systems.
(PWR#1)	8.8	Assure conformance with GDC Nos. 1-5 and 39 through 48. Rely on Unit No. 2 review.

9.0 Auxiliary and Emergency Systems

Review and evaluate the design basis and performance of the auxiliary systems noted:

(PWR#1)	9.1	Fuel storage and handling including liner materials, corrosion potential, protection from missiles, effects of dropping fuel cask, handling equipment operations and interlocks.
(PWR#1)	9.2	Fuel pit cooling and cleanup system including water purity control, normal and maximum cooling capacity, level and water drain control and pool leakage effects.
(PWR#1)	9.3	Chemical and volume control system, boron recovery system, residual heat removal system and component cooling water system.
(PWR#1)	9.4	Station fire protection including areas of automatic coverage and emergency power backup.
(PWR#1)	9.5	Station instrument and service air system including emergency power sources.

(PWR#1)	9.6	Service water system including redundancy, emergency power sources and the effects of system failure.
(PWR#1)	9.7	Ventilation systems for the Containment Building, Turbine Building, Auxiliary Building and Control Room emergency power source requirements and monitoring and isolation capabilities of the Control Room and Fuel Building ventilation systems.
(DRS)	9.8	Equipment and floor drainage systems including reliability of level instrumentation and alarms, lower limits of leak detection capability of instrumentation and surveillance of activity of drains.
(PWR#1)	9.9	Communications system.
(PWR#1)	9.10	Assure conformance with GDC Nos. 1-5, 62-66, 67, 68 and 69.

10.0 <u>Steam and Power Conversion Systems</u>

Review and evaluate design bases and performance characteristics with respect to conformance to previously approved designs of the following:

(PWR#1)	10.1	Effects of turbine and generator trips with and without turbine by-pass operating.
(PWR#1)	10.2	Main steam bypass system capability with regard to load rejection and acceptance.
(PWR#1)	10.3	Auxiliary steam system.
(PWR#1)	10.4	Condensate and feedwater system capacity for residual heat removal.

(PWR#1)	10.5	Potential for turbine-generator missiles including degree of protection provided Class I equipment.
(PWR#1)	10.6	Assure conformance with GDC Nos. 17, 40 and 57.

11.0 Radioactive Waste Systems

(SERSG/ PWR#1)	11.1	Evaluate capability of the radioactive waste system to collect, confine, process, dispose, and monitor radwaste within the limitation of Title 10, CFR Parts 20, 50, and 100 under normal and abnormal operating conditions. Evaluate conformance with the intent of proposed changes of 10 CFR Parts 20 and 50 concerning radiation exposures and releases of radioactive materials to unrestricted areas.
(SERSG/ PWR#1)	11.2	Evaluate the estimated normal releases and possible additional means for reducing planned or accidental release of rad-waste to unrestricted areas.
(SERSG/ PWR#1)	11.3	Review and evaluate the ventilation system for the control room regarding isolation and/or filtering capability during a design basis accident.
(PWR#1)	11.4	Assure conformance with GDC Nos. 1-5, 11, 17, 18, 62-65, 69 and 70.

12.0 Structures and Shielding

(DRS)	12.1	Review and evaluate structural design of Class I (Seismic) mechanical, piping and structural systems considering seismic criteria and dynamic analyses (Reference Newmark and Hall reports and letters), loading (including turbine and tornado missiles), stress and deformation criteria.

(DRS)	12.2	Review and evaluate effects of wind and tornado loadings on Class I structures, particularly the containment building.
(DRS)	12.3	Review and evaluate significant containment and Class I structural design features including Class I-II interfaces and interaction and the effects on Class I systems from Class II failures.
(DRS)	12.4	Review adequacy of foundation grouting and foundation design under Class I structures, particularly with respect to seismic analyses of containment building.
(SERSG)	12.5	Evaluate the criteria for protection of personnel and selected equipment/materials against radiation during either normal or abnormal plant operating conditions.
(SERSG)	12.6	Evaluate the criteria for the radiation protection of control room personnel under normal and emergency conditions.
(PWR#1)	12.7	Assure conformance with GDC 68, 69 and 70.

13.0 <u>Conduct of Operations</u>

(PWR#1)

Review the following aspects of the station operation: (Rely on Unit No. 2 review).

a. Applicant's corporate and plant organization and responsibilities

b. Training of plant personnel

c. Pre-operational, startup and power test procedures

d. Westinghouse and its contractors' relationship to the applicant's organization, responsibilities for training plant personnel and conducting tests prior to commercial operation.

e. Preparation and maintenance of records

f. Normal operating procedures

g. Emergency plans (reference Appendix E to 10 CFR 50)

h. Review, approval, authorization and control of procedures, tests and changes thereto.

i. Pre-operational test program

j. Startup and power test program

k. Plant access control and provisions for controlling potential industrial sabotage.

14.0 Safety Analysis

(SERSG) 14.1 Re-evaluate the radiological consequences resulting from the design basis accidents using the final site LPZ distance and current DRL meteorological dispersion curves. Establish recommended limits on containment leakage to meet acceptable limits on radiological doses at the site boundaries.

(SERSG) 14.2 Review and evaluate the results of the analysis of the iodine removal capability of the sodium hydroxide sprays and charcoal filters.

(PWR#1/ 14.3 Review potential for and consequences of anticipated transients without reactor trip action.

(PWR#1) 14.4 Assure conformance with GDC No. 70.

15.0 Proposed Technical Specifications

(PWR#1)		Evaluate proposed technical specifications to assure conformance with 10 CFR 50.36. Use the Indian Point 2 Technical Specifications as a guide in the review and evaluation.

16.0 Environmental Statement

(PWR#1)	16.1	Coordinate environmental review with cognizant state and federal agencies.
	16.2	Verify the water quality standard certification required by the Water Quality Improvement Act of 1970.
	16.3	Prepare an environmental statement for transmittal to Federal agencies and subsequently to the Council on Environmental Quality.

OPERATING LICENSE REVIEW SCHEDULE - TECHNICAL

	TITLE	DATE
1.	Application submitted	December 4, 1970
2.	Initial meeting with applicant	January, 1971
3.	Initial draft request for additional information to PWR#1	April, 1971
4.	Technical meetings with applicant	June, 1971
5.	Formal request for additional information to DRL Management	July, 1971
6.	Formal request for additional information to applicant	July, 1971
7.	Interim report draft to RP branch	August, 1971
8.	Interim report to DRL Management	August, 1971
9.	Responses from applicant	October, 1971
10.	Second draft requests for additional information to RP branch	November, 1971
11.	Formal request for additional information to DRL Management	November, 1971
12.	Formal request for additional information to applicant	November, 1971
13.	Responses from applicant	December, 1971
14.	Draft sections of ACRS report to RP branch	December, 1971
15.	ACRS Report to DRL Management	January, 1972
16.	Transmit ACRS Report	February, 1972
17.	ACRS Meeting	March, 1972
18.	Safety Evaluation	April, 1972

OPERATING LICENSE REVIEW SCHEDULE - ENVIRONMENTAL

	TITLE	DATE
1.	Environmental Report from applicant	March, 1971
2.	Draft Environmental Statement and applicant's report out for comment	July, 1971
3.	Comments on Environmental Report from agencies	October, 1971
4.	Agency comments to applicant	October, 1971
5.	Applicant's reply to agency comments	December, 1971
6.	Environmental Statement	February, 1972