JAN 28 1969

Docket No. 50-247

Frank Schroeder, Deputy Director, RL
R. S. Boyd, Assistant Director for Reactor Projects, RL
S. Levine, Assistant Director for Reactor Technology, RL
D. J. Skovholt, Assistant Director for Reactor Operations, RL

DRL REVIEW PLAN FOR REVIEW OF CONSOLIDATED EDISON'S
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

A proposed review plan and schedule for the Indian Point Unit No. 2 provisional operating license review was distributed and discussed between representatives of RP, RT, and RO. The attached review plan and schedule represents the review areas and assignments agreed to during these discussions. All concerned members of the staff are directed to proceed with the review as outlined.

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

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

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ATTACHMENT 1

BACKGROUND

The Consolidated Edison Company of New York submitted its Final Facility Description and Safety Analysis Report for the Indian Point Nuclear Generating Unit No. 2 on October 15, 1968. The application is based upon a 2758 Mwt reactor producing 873 Mwe net. (This is in accordance with the Construction Permit issued on October 14, 1966.)

The reactor site is in a relatively populated area north of New York City. As with Indian Point 3 and Zion, a considerable dependence is placed on iodine removal systems in the containment to lower offsite doses to acceptable values following a major accident.

Indian Point 2 is the first of the large core four-loop current generation Westinghouse plants employing their twelve foot long standardized zircalloy clad fuel rod and fuel rod assembly. The reactor core thermal output is derated some 18% below cores of similar design and size appearing in recent and current construction permit applications.

The Indian Point 2 plant is very similar to the Indian Point 3 plant for which the construction permit review is currently being completed. Results of this review should be used.

The Indian Point 2 plant engineering design is similar to the two-loop Ginna reactor currently undergoing the Preliminary Operating License review. As in Ginna, the core design incorporates zircalloy clad fuel rods; with burnable poison (in the first core loading), part length control rods, and two region long ion chambers to control anticipated axial xenon oscillations. Because of the larger diameter core, designated X, Y full length control rods have been included in the Indian Point 2 reactor in a control scheme to permit control of potential azimuthal or transverse xenon oscillations.

The Indian Point 2 reactor also includes an emergency core cooling system with accumulators and containment chemical spray and recirculating fan coolers with charcoal filters, all as in the Ginna plant.

REVIEW PROCEDURES

In organizing and conducting his evaluation, each reviewer should have in mind and be guided by the following primary elements of an operating license stage review:

(1) A review of how the design bases have been implemented in the final design. The final design of the structures, systems, and components of the facility should be evaluated to determine that performance requirements will be met especially for those systems which prevent or mitigate the consequences

of accidents. All safety analyses should be based on the final system design. All changes from the preliminary design should be identified and evaluated.

(2) A review of the as-built plant. The physical plant design should be determined to conform to the description and requirements of the Final Safety Analysis Report. The implementation of the quality assurance programs should be verified to provide assurance that a high quality product has been obtained.

(3) A review of the results of research and development programs. The results of these programs should provide assurance that safety questions identified at the construction permit stage have been resolved. The implementation of other specific design commitments made at the construction permit stage should be reviewed.

(4) A review of the operating organization. The operating philosophy of the applicant as reflected in the operating organization and plant design should be reviewed.

(5) A review of the proposed technical specifications. The technical bases for each specification should reflect the final design of the facility.

(6) A review of new safety issues. Safety issues that have been raised on other plants since the construction permit review should be evaluated to determine whether "backfit" of each item is required.

Each reviewer assigned to the project should (1) identify needs for additional information, (2) provide questions for additional information as required, (3) participate in technical meetings with the applicant, (4) evaluate answers and ask additional questions if necessary, (5) provide a report(s) on the review, with the conclusions reached and the bases for the conclusions, and (6) suggest recommendations for applicable portions to the Technical Specifications.

The review of each section should include where applicable:

1. Design basis events and stress or damage limits.

2. DRL evaluation model or criteria, codes, or standards used in review.

3. Performance requirements and equipment capabilities.

4. Mechanical design.

5. Integration of the system and its instrumentation, control, and power supply.

6. Pre-operational and inservice tests and inspections.

7. Failure mode analysis.

8. Determination of technical specification limits.

REVIEW BASES

Review and evaluation should be based on Criteria, Codes and Standards, and Reactor Technology Memoranda (RTM's) where applicable. Table I presents a compilation of these documents currently being used for guidance in DRL reviews. It should be realized that development of many of the items in the above table has proceeded concurrently with the Indian Point 2 design. It is expected, therefore, that the plant will be deficient with respect to these criteria, codes and standards, and RTM's in a number of respects. It is the responsibility of each reviewer to report to the project leader what in his judgment are considered to be non-conformance or marginal conformance conditions with real safety implications. This should be followed by an in-depth evaluation with a recommendation which can be presented to DRL management for resolution.

SCHEDULE AND REVIEW PROCEDURAL PLAN

The proposed schedule is presented in Table 2. Because of potential difficulties in meeting established criteria and codes as discussed above and because additional problems are anticipated with respect to satisfactory resolution of research and development items, a relatively early initial ACRS subcommittee meeting has been scheduled. The purpose of this meeting is to provide early recognition of major problem areas. Sufficient time should then remain for establishing an acceptable resolution without serious delays in initial plant operation. Success of this plan depends on implementation of the review plan by the project leader and the assigned reviewers leading to prompt preparation of ACRS Report No. 1.

TABLE I

EVALUATION BASES - CRITERIA, CODES AND STANDARDS, REACTOR TECHNOLOGY MEMORANDA, AND OPERATIONAL SAFETY GUIDES

A. CRITERIA

1. "General Design Criteria for Nuclear Power Plant Construction" U. S. Atomic Energy Commission, July 11, 1967.

2. "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems" Institute of Electrical and Electronic Engineers, August 1968.

3. "Supplementary Criteria for Auxiliary Electrical Power System for Nuclear Power Plants" U. S. Atomic Energy Commission, June 9, 1967.

4. "Regulatory Supplementary Criteria for ASME Code - Constructed Nuclear Pressure Vessels" U. S. Atomic Energy Commission, August 23, 1967.

B. CODES AND STANDARDS

1. ASME Boiler and Pressure Vessel Code Section III, Nuclear Vessels, 1968 Edition, American Society of Mechanical Engineers.

2. USA Standard, Power Piping, USAS B31.1 American Society of Mechanical Engineers, 1967.

3. Draft USA Standard, Nuclear Power Piping, USAS B31.7 (Issued for trial use and comment) ASME, February 1968.

4. Draft USA Standard, Code for In-service Inspection of Nuclear Reactor Coolant Systems, USAS ISI (Issued for trial use and comment) ASME, October 1968.

5. Reactor Containment Leakage Testing and Surveillance Requirements. U. S. Atomic Energy Commission, December 15, 1966.

C. REACTOR TECHNOLOGY MEMORANDA

1. "Tornado Considerations" - RTM No. 1, April 10, 1968.

2. "Nuclear Vessels - Supplementary Criteria" - RTM No. 2A, August 13, 1966.

3. "Seismic Design Criteria" - RTM No. 3 (Draft for comment), January 20, 1968.

4. "Emergency Core Cooling System Evaluation Guidelines" - RTM No. 4 (Draft for comment), January 1968.

5. "Reactor Vessel Irradiation Surveillance" - RTM (Draft for comment), June 27, 1968.

6. "Reinforcing Bar Cadweld Splices - Test Sampling Criteria" - RTM (Draft for comment), October 29, 1968.

7. "Flooding Considerations for Sites Along Rivers" - RTM (Draft for comment), May 28, 1968.

8. "Fission Products - Iodine Formation and Removal in Power Reactor Facilities" RTM (Draft for comment), September 20, 1968.

D. OPERATIONAL SAFETY GUIDES

1. "Proposed Guide for Operating Organizations at Nuclear Power Stations" - OSG No. 1, (Draft for comment), April 15, 1968.

2. "Proposed Supplementary Guide to the Content of Technical Specifications (Administrative Section)" - OSG No. 2, (Draft for comment), April 26, 1968.

3. "Proposed Guide for Emergency Preparedness at Power Reactor Facilities" - OSG No. 3, (Draft for comment), September 25, 1968.

ATTACHMENT 2

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

PRELIMINARY REVIEW PLAN FOR A PROVISIONAL OPERATING LICENSE

Review Items and Topics: The sections identified below refer to sections in the Final Safety Analysis Report.

1.0 GENERAL

[RP] 1.1 Quality Assurance Program

[RP] 1.1.1 Review the planning and organizational aspects of the quality assurance program utilized by the applicant.

[RP] 1.1.2 Evaluate the engineering design implementation of the quality assurance program involved in the construction and assembly of all systems, components, and structures significant to safety.

[RP] 1.1.3 Evaluate the implementation of the quality control methods and procedures used in plant component fabrication, assembly, and erection.

2.0 SITE AND ENVIRONMENT (Section 2)

[RT] 2.1 Evaluation of Consultants' Reports

2.1.1 Meteorology - Weather Bureau

2.1.2 Marine Ecology - Sport Fisheries and Wildlife

2.1.3 Seismic Design - Newmark and Hall

[RT] 2.2 Adequacy of meteorological data as a basis for determining offsite dose from accidents.

[RT] 2.3 Adequacy of environmental monitoring program.

[RT] 2.4 Evaluate proposed gaseous and liquid release rates.

[RT] 2.5 Evaluate potential effects of liquid and gaseous waste discharge on the local potable water supplies.

[RT] 2.6 Evaluate flooding potential.

[RT] 2.7 Evaluate extent of tornado protection provided for various portions of the plant and equipment.

3.0 REACTOR (Section 3)

[RP] 3.1 Review of Reactor Core Component Design

[RP] 3.1.1 Evaluate zirconium clad uranium oxide fuel based on irradiation test results for normal and design transient operation.

[RP] 3.1.2 Evaluate burnable poison rod design and results of irradiation testing.

[RP] 3.2 Review of Thermal Hydraulic Design

[RP] 3.2.1 Consider thermal hydraulic design in light of experience with San Onofre and Conn-Yankee and Westinghouse scale model tests.

[RP] 3.2.2 Evaluate design of reactor internals relative to loads during normal operation. Also consider potential vibration problems and adequacy of pre-operational tests for potential vibration problems.

[RT] 3.3 Review Reactivity and Power Distribution Control and Detection

[RT] 3.3.1 Review the adequacy of proposed nuclear instrumentation to detect mal-distribution of power including xenon oscillations with emphasis on:

(a) Ability to detect mal-distribution in power resulting from axial, azimuthal, and transverse xenon oscillations;

(b) Sensitivity and adequacy of out-of-core instrumentation;

(c) The potential need for permanent in-core instrumentation.

[RP] 3.3.2 Evaluate part length rods and review their use in the control of xenon oscillation with emphasis on:

(a) The rod drive mechanism; evaluate testing program

(b) Rod position indicators

(c) Detection of out of place rods

(d) Operating procedures followed for oscillation damping.

[RP] 3.3.3 Review the use and procedures proposed for X-Y control rods for the control of potential azimuthal or transverse xenon oscillations.

[RP] 3.3.4 Evaluate the need for automatic protection for bottomed or out of place control rods.

4.0 <u>REACTOR COOLANT SYSTEM</u> (Section 4)

[RP] 4.1 Review Adequacy of Primary System Leak Detection System and Makeup System.

[RP] 4.1.1 Establish reasonable criteria for leak detection capability.

[RP] 4.1.2 Review methods and systems proposed.

[RP] 4.1.3 Review adequacy of normal charging system.

[RT] 4.2 Review Adequacy of System Components in Meeting Design and Appropriate Code Requirements.

[RT] 4.2.1 Establish principal design requirements.

[RT] 4.2.2 Verify Code Requirements.

[RT] 4.2.3 Establish adequacy of structural design to meet Class I.

[RP/RO] 4.3 Evaluate Plans, Procedures, and Schedules for Inservice Inspection. Compare with the draft of the proposed USA Standard on Inservice Inspection of Nuclear Reactor Coolant Systems.

[RP] 4.4 Consider Adequacy of Primary System Design in Terms of Potential Vibration Problems. Evaluate the Adequacy of Proposed Testing Programs.

[RT/RP] 4.5 Review Problem of Radiation Embrittlement of the Pressure Vessel During Reactor Lifetime.

[RT] 4.5.1 Verify estimated fast neutron fluence of pressure vessel.

[RP] 4.5.2 Review heatup and cooldown procedures required to accommodate expected embrittlement of the vessel.

[RT] 4.5.3 Review the adequacy of vessel material surveillance program used to establish actual NDTT shift during the vessel lifetime.

[RP] 4.6 Review Adequacy of Methods Proposed for Fuel Element Failure Detection.

5.0 <u>CONTAINMENT</u> (Section 5)

[RT] 5.1 Review Adequacy of Containment Structural Design Including Seismic Design.

[RT] 5.2 Review the Adequacy of the Stress Analysis for the Containment Liner.

[RP] 5.3 Review Adequacy of Provisions for Leakage Tests and Proposed Inservice Inspection.

[RP] 5.4 Review the adequacy of isolation valve criteria and design and seal water injection system.

[RT] 5.5 Review missile protection provisions and their adequacy. Review design of pressure vessel pit for missile protection. Review potential effect of missiles from primary pump flywheel fracture and consider missile shielding requirements.

[RP/RT] 5.6 Consider the problem of hydrogen generation from various sources in the containment following a loss of coolant accident.

[RP] 5.6.1 Review and check calculations and analysis of hydrogen production in the containment.

[RT] 5.6.2 Review the adequacy of components, methods, or systems proposed to accomplish hydrogen concentration control.

6.0 <u>ENGINEERED SAFETY FEATURES</u> (Section 6)

[RT] 6.1 Determine the extent to which the emergency core cooling system meets the proposed ECCS criteria. Check the adequacy of the emergency core cooling system with respect to current design and performance criteria, and determine and note any deviations.

[RT] 6.1.1 Confirm functional performance requirements and equipment capacities for the emergency core cooling system and auxiliary systems required during long term recirculation.

[RP] 6.1.2 Review layout requirements with respect to needed flooding protection in the event of passive failure. Review location of local instrumentation and equipment controls. Review leak detection capability during recirculation.

[RT] 6.1.3 Check the effects of high flow velocities (water and nitrogen) from the accumulators, the moments in the piping, and the adequacy of anchoring the pipes.

[RP] 6.1.4 Review adequacy of system used for injection of concentrated boric acid. Check for adequacy of equipment design and inservice testing procedures in the prevention of line plugging with boric acid crystals.

[RP] 6.1.5 Review adequacy of the pre-operational test program for the emergency core cooling system and its components.

[RP/RT] 6.2 In connection with the review of potential thermal shock effects on the reactor vessel resulting from ECCS operation (see Item 14.3.4), determine the need for post-loss of coolant accident protection (PLOCAP) by incorporating provisions for cavity flooding.

[RP/RT] 6.3 Review the adequacy of the containment spray system.

[RP] 6.3.1 Confirm system performance with regard to heat removal. Verify necessary equipment capacities.

[RT] 6.3.2 Verify or determine the performance of the spray system with regard to iodine removal from the containment atmosphere following a loss of coolant accident.

[RP] 6.3.3 Review adequacy of pre-operational test program of system and/or components.

[RP/RT] 6.4 Review the adequacy of the air recirculation cooling and filtration system.

[RP] 6.4.1 Confirm system performance with regard to heat removal. Review heat transfer design and analysis. Review pre-operational testing.

[RP] 6.4.2 Verify the operability of the components under accident conditions.

[RT] 6.4.3 Verify or determine the performance of the charcoal filters with regard to iodine removal from the containment atmosphere following a loss of coolant accident.

[RT] 6.4.4 Review adequacy of inservice test program to assure availability of charcoal filters.

7.0 INSTRUMENTATION AND CONTROL (Section 7)

[RT] 7.1 Review and check the adequacy of instrumentation and appropriate circuits for reactor protection.

[RT] 7.2 Review and check the adequacy of instrumentation used to actuate engineered safety features.

[RT] 7.3 Review the adequacy of instrumentation employed in connection with charging of concentrated boric acid to the primary system. Use San Onofre experience as a guide.

[RT] 7.4 Review adequacy of instrumentation for containment. Check against needs during a loss of coolant accident and long term cooling and recovery operations.

[RP/RT] 7.5 Determine the capability for hot and cold shutdown of the plant from a location other than the control room. Review adequacy of instrumentation in connection with this type of operation.

8.0 ELECTRICAL SYSTEMS (Section 8)

[RT] 8.1 Review and analyze the network interconnections to determine reliability of power to the station.

[RT] 8.2 Check adequacy of the onsite emergency power system, and analyze the effects of the loss of offsite power. Review periodic testing procedures proposed to verify availability of onsite emergency power.

[RP] 8.3 Check tornado design protection of the onsite emergency power.

[RT] 8.4 Review adequacy of power and instrument cabling and check cable tray design, layout, and loading in light of San Onofre experience.

9.0 AUXILIARY AND EMERGENCY SYSTEMS (Section 9)

[RP] 9.1 Review the adequacy of components in meeting appropriate code requirements.

[RP] 9.2 Review the adequacy of the component cooling system and the service water system and check the extent to which they meet ECCS criteria for long term post-loss of coolant accident protection.

[RP] 9.3 Review the adequacy of the primary coolant charging system. Emphasize review of concentrated boric acid injection capability. Check for adequacy of equipment design and inservice testing procedures in the prevention of line plugging with boric acid crystals.

[RP/RO] 9.4 Evaluate procedures and equipment for fire protection and fire prevention.

10.0 STEAM AND POWER CONVERSION SYSTEM (Section 10)

[RP] 10.1 Review adequacy of normal and emergency steam generator feedwater system.

[RT] 10.2 Calculate the maximum permissible leakage rates from primary system to the steam system via the steam generators. Consider offsite dose resulting from radioactivity in the steam system (and primary secondary leakage) during normal operation and with steam line break accident.

11.0 WASTE DISPOSAL AND RADIATION PROTECTION (Section 11)

[RP] 11.1 Determine the adequacy of the provisions for monitoring discharge of radioactivity to the environment, and the effectiveness of the radwaste procedures in limiting the release of radioactive wastes to limits set by 10 CFR Part 20.

[RP] 11.2 Review the sources of tritium and determine the manner of release to the circulating water system and the environment.

[RP] 11.3 Review the adequacy of the present system to handle the anticipated waste load.

[RP] 11.4 Evaluate the provisions made to maintain the tritium inside the containment to a level acceptable to working personnel.

[RO] 11.5 Check the adequacy of onsite and offsite emergency radiation instrumentation.

12.0 CONDUCT OF OPERATIONS (Section 12)

[RO] 12.1 Review and evaluate the technical competence of the organization responsible for the operation of the plant.

[RO] 12.1.1 Evaluate the organizational breakdown and the associated distribution of responsibilities.

[RO] 12.1.2 Evaluate position minimum requirements.

[RO] 12.1.3 Evaluate training programs and other provisions for maintaining proficiency for the life of the plant.

[RO] 12.2 Evaluate operating conditions requiring adherence to detailed written procedures including as a minimum, startup, normal and abnormal operation, and shutdown of the plant and major systems, abnormal and emergency conditions, refueling, and maintenance operations which could affect the safety of the plant.

[RO] 12.3 Evaluate proposed technical specifications with respect to the administrative procedures section.

[RO] 12.4 Evaluate plant operation review and audit as performed by advisory boards and review committees. Evaluate methods for providing independent review of proposed changes to procedures, plant modifications, and audit of plant operations.

[RO] 12.5 Review for adequacy proposed plant operation records.

[RO] 12.6 Review emergency plans.

[RO] 12.7 Review medical preparedness for emergencies.

14.0 SAFETY ANALYSIS (Section 14)

[RT] 14.1 Establish acceptable assumptions concerning fission product release from fuel and leakage rate from the containment and auxiliary building following a loss of coolant accident for purposes of dose calculations. Calculate offsite doses, exclusion radius, and low population radius, and compare with applicant's calculations, and explain the difference. Compare the above analysis with that for Indian Point 3. State clearly any differences in equipment, operation, and assumptions used for Indian Point 2 and 3.

[RT] 14.2 Evaluate radiation exposure in the control room following a loss of coolant accident.

[RT] 14.3 For the loss of coolant accident, evaluate for the complete range of break sizes.

[RT] 14.3.1 Pressure, temperature, and water inventory transient during blowdown.

[RT] 14.3.2 Core thermal transient analysis during blowdown and subsequent water injection.

[RP] 14.3.3 Fuel rod and core integrity during the loss of coolant accident transient including recovery. Consider rod clad bursting, swelling, or shattering and establish preservation of coolable core geometry.

[RT] 14.3.4 Reactor internals integrity analysis during blowdown.

[RT] 14.3.5 Effect of loss of coolant and cold water injection on the integrity of the reactor vessel and its internals (thermal shock).

[RT] 14.3.6 Containment pressure and temperature history.

[RT/RP] 14.4 Evaluate the following accidents.

[RT] 14.4.1 Rod ejection (consider inserted part length rods and bottomed full length rods).

[RP] 14.4.2 Chemical and volume control system malfunctions.

[RP] 14.4.3 Loss of reactor coolant flow.

[RP] 14.4.4 Startup of an inactive coolant loop.

[RP] 14.4.5 Loss of external electrical load coincident with loss of offsite power.

[RP] 14.4.6 Fuel handling accidents.

[RP] 14.4.7 Accidental release of waste liquid.

[RP] 14.4.8 Accidental release of waste gas.

[RT] 14.4.9 Steam line break.

[RT] 14.4.10 Steam generator tube rupture with and without offsite power. Determine offsite dose and limiting conditions of operation.

[RP] 15.0 TECHNICAL SPECIFICATIONS

Review Technical Specifications for adequacy.

[RP] 16.0 CONFORMANCE TO PROPOSED 70 CRITERIA

Evaluate proposed plant in conformance with 70 criteria. Determine and note items in non-conformance or partial or marginal conformance.

TABLE II

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

PROVISIONAL OPERATING LICENSE

Item	Date
Application filed	10/15/68
Complete Preliminary Review	11/15/68
Issue Review Plan	1/20/69
Draft Consultant Reports Required	2/10/69
Complete Identification of Important Problem Areas by RT and RO in Writing	3/3/69
Meet with Applicant on Important Problem Areas	3/11/69
Complete Draft of ACRS Report No. 1	3/24/69
Issue Request for Supplementary Information	4/4/69
Issue ACRS Report No. 1	4/69
ACRS Subcommittee Meeting	4/69
Additional Meetings with Applicant Including Review of Technical Specifications	5/69 & 6/69
Additional Requests for Information	6/69 & 7/69
Develop Technical Specifications	8/69
Issue ACRS Report	8/69
ACRS Subcommittee Meeting	8/69
ACRS Meeting	9/69
Issue Operating License	10/1/69
Load Fuel (Applicant's Present Estimate)	12/1/69