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S. Rivenbark DoR #24
Vollmer
Adm
& dist

February 27, 1979

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
ATTN: Mr. Victor Stello, Jr., Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

As discussed with your staff, we have decided to pursue reracking the Indian Point Unit No. 2 spent fuel storage pool instead of the modifications described in our May 4, 1978 submittal.

The Attachment to this letter presents proposed design criteria for reracking the Indian Point Unit No. 2 spent fuel pool with high density spent fuel racks. The present schedule is based on submittal to the NRC of a detailed design report by September, 1979. Con Edison requests that the NRC comment on the design criteria presented in the Attachment by April 16, 1979.

Should you or your staff have any questions, please contact us.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

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ATTACHMENT

PROPOSED DESIGN CRITERIA FOR RERACKING INDIAN POINT UNIT NO. 2 SPENT FUEL POOL

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
February, 1979

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1. Criticality Analysis

1.1 Criticality Criteria

The spent fuel storage racks will be designed to limit K_{eff} to ≤ 0.95. The criticality calculations will be based on a maximum U-235 enrichment of 3.5 w/o. Credit will not be taken for borated water in the spent fuel pool during normal conditions.

1.2 Calc lational Methods

Appropriate computer codes, will be used to obtain broad neutron cross sections for input into a diffusion theory, transport theory or Monte Carlo code for criticality calculations. If a diffusion code is used in the calculations, it will be benchmarked to a transport theory and/or Monte Carlo code.

2. Thermal - Hydraulic Analysis

2.1 Spent Fuel Cooling Capability

The adequacy of the spent fuel cooling system will be determined. The impact of the proposed modification on the cooling capacity of the spent fuel cooling system will be evaluated for spent fuel discharges to the spent fuel pool from normal refueling operations and full core discharge. The decay heat loads will be calculated using computer codes such as ORIGEN or by other accepted methods.

2.2 Spent Fuel Assembly Heat Removal Capability

The proposed spent fuel racks will be designed with adequate flow paths between the racks, pool walls, etc., to ensure natural thermal circulation of water in the spent fuel pool.

3. Mechanical Analysis

3.1 Design Criteria

The spent fuel racks will be designed to resist combined loadings of the dead weight of the cells, the weight of the spent fuel assemblies and seismic loads. The spent fuel racks will be designed to be Seismic Category I structures in accordance with Regulatory Guide 1.29. The spent fuel racks will be designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III or AISC Specification for Design, Fabrication and Erection of Structural Steel for Buildings, as described in the NRC Guidance document entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications" (sent to all power reactor licensees by NRC letter dated April 14, 1978 and modified by NRC letter dated January 18, 1979). In addition, the spent fuel pool structure will be evaluated to determine maximum stresses in the pool walls and floor.

3.2 Method of Analysis

A dynamic analysis of the spent fuel storage racks will be performed. All appreciable loads, including seismic will be considered. Stresses will be evaluated against code allowables for both normal load conditions & accident conditions. Load combinations presented in Section 3.8.4 of the Standard Review Plan will be used in the analysis.

4. Spent Fuel Storage Racks

The spent fuel assemblies will be stored vertically in a more closely spaced lattice than the presently installed racks. It is estimated that the spent fuel storage capacity will be increased to at least 840 assembly storage locations (depending on final selection of poison, material and vendor).

The racks will be manufactured of type 304 stainless steel and will rest on the stainless steel liner plate at the bottom of the spent fuel storage pool.

The design and fabrication of the spent fuel storage racks shall comply with:

(1) The requirements pertaining to fuel storage racks contained in ANSI N18.2.

(2) The requirements pertaining to spent fuel storage racks contained in ANSI N210.

(3) The requirements contained in 10CFR50 Appendix A General Design Criterion 62.

(4) The structural design shall be in accordance with SRP 3.8.4 (see Section 5 Codes, Standards and Design Criteria).

(5) Spent fuel storage racks will be designed as a Seismic Category I structure as designated in Regulatory Guide 1.29.

5. Codes, Standards and Design Criteria

The abbreviations listed below have the following meanings:

AISC - American Institute of Steel Construction

AISI - American Iron and Steel Institute

ANS - American Nuclear Society

ANSI - American National Standards Institute

ASME - American Society of Mechanical Engineers

ASNT - American Society for Nondestructive Testing

ASTM - American Society for Testing and Materials

AWS - American Welding Society

CFR - Code of Federal Regulations

R.G. - Regulatory Guide of the USNRC

SRP - Standard Review Plans of the USNRC

SSPC - Steel Structures Painting Council

USNRC- United States Nuclear Regulatory Commission

When and if any of the following codes, technical or trade standards, regulations and specifications, such as AISC, ANSI, ASME, ASTM, SRP, AWS, USNRC Regulatory Guides, Federal Specifications shall be employed during this proposed project the latest edition and latest addenda thereto, to the extent applicable to this project, shall be used:

AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings

AISC	Code of Standard Practice for Steel Buildings and Bridges
ANS 5-1	Proposed ANS Standard Decay Heat Release Following Shutdown of Uranium Fueled Thermal Reactors
ANSI N16.1	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
ANSI N18.2	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants and Supplement N18.2A-1975
ANSI N45.2.-197 1:	QA program requirements for Nuclear Power Plants
ANSI N45.2.1-1973:	Cleaning of Fluid Systems and Associated Components During the Construction Phase of Nuclear Power Plants
ANSI N45.2.9	Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants
ANSI N210	Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants
ANSI N45.2.12	Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants

ANSI N45.2.13	Quality Assurance Requirements for Control of Procurement of Equipment, Materials and Services for Nuclear Power Plants
ASME B&PV Code	Section II, Material Specification, Parts A, B and C and all addenda
ASME B&PV Code	Section V, Nondestructive Examination and all addenda
ASME B&PV Code	Section VIII, Division 1, Pressure Vessels and all addenda
ASNT (SNT-TC-1A)	Recommended Practice for Nondestructive Testing, Personnel Qualification and Certification
ASTM-A262	Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steel
AWS D.1.1	Structural Welding Code
USNRC	Guidance document entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications " (sent to all power reactor licensees by USNRC letter dated April 14, 1978 and modified by USNRC letter dated January 18, 1979).
ASME B&PV Code	Section IX, Welding and Brazing Qualifications and all addenda.

SSPC-SP-10	Surface Preparation Specification, Near-White Blast Cleaning
SSPC-Vis-1	Surface Preparation Specification, Pictorial Surface Preparation Standards for Painting Steel Structures
R.G. 1.13	Spent Fuel Storage Facilities Design Basis
R.G. 1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors
R.G. 1.29	Seismic Design Classification
R.G. 1.31	Control of Stainless Steel Welding
R.G. 1.48	Design Limits and Loading Combination for Seismic Category I Fluid Systems Components
R.G. 1.50	Control Preheat Temperature for Welding of Low Alloy Steel
R.G. 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants
R.G. 1.71	Welder Qualification for Areas of Limited Accessibility
R.G. 1.92	Combining Model Responses and Spatial Components in Seismic Response Analysis

R.G. 3.15	Standard Format and Content of License Applications for Storage Only of Unirradiated Reactor Fuel and Associated Radioactive Material
R.G. 3.4	Nuclear Criticality Safety in Operation with Fissionable Materials Outside Reactors
R.G. 8.8	Information Relevent to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactors)
SRP 3.7.2	Seismic System Analysis
SRP 3.8.4	Other Category I Structures
SRP 9.1.2	Spent Fuel Storage
SRP 9.1.3	Spent Fuel Pool Cooling and Cleanup System
10CFR50, Appendix A	General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling