UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 31, 1996

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

SUBJECT: ORDER TO AUTHORIZE DECOMMISSIONING AND AMENDMENT NO. 45 TO LICENSE NO. DPR-5 FOR INDIAN POINT UNIT NO. 1 (TAC NO. M59664)

Dear Mr. Bram:

The Commission has issued the enclosed order to authorize decommissioning of Indian Point Unit No. 1. Also enclosed is Amendment No. 45 to License No. DPR-5, which revises the license to possession-only status, revises the Technical Specifications (TSs) and renews the license until October 14, 2006. The order and amendment respond to your application of October 17, 1980, as revised October 13, 1981; July 31, 1986; March 28, 1988; August 10, 1989; March 28 and July 17, 1990; February 5, April 2, July 31, September 20, and October 12, 1993; May 13 and August 11, 1994 and July 19, 1995. The order has been forwarded to the Office of the Federal Register for publication.

As required by 10 CFR 50.82(d), and the enclosed order authorizing decommissioning, Con Edison must submit a detailed dismantling plan for NRC review and approval prior to major dismantlement activities at IP-1.

A Notice of Consideration of Issuance of Amendment to License and Opportunity for Prior Hearing related to the requested action was published in the FEDERAL REGISTER on December 31, 1985, (50 FR 53407). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Notice stated a license renewal date of October 14, 2006 to coincide with the planned permanent shutdown of Indian Point Unit 2 (IP-2). Subsequent to the December 31, 1985 notice, the license for IP-2 was extended to September 28, 2013 and you informed us, in your letter dated March 28, 1988, of your intention to delay dismantlement of IP-1 to after that date. The enclosed safety evaluation and environmental assessment of your decommissioning plan are consistent with the 2013 date. However, we have renewed License No. DPR-5 to October 14, 2006, to be consistent with the license renewal application as noticed in the December 31, 1985 FEDERAL REGISTER Notice in order to put new TSs for the current shutdown condition in place.

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Copies of the related Safety Evaluation, the Environmental Assessment supporting the order, and Amendment No. 45 are enclosed. Also enclosed is a copy of the Notice of Issuance of Environmental Assessment and Finding of No Significant Impact, which was published in the FEDERAL REGISTER on

Sincerely,

ORIGINAL SIGNED BY

Peter B. Erickson, Senior Project Manager
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket No. 50-3

Enclosures:
1. Order Authorizing Decommissioning
2. Amendment No. 45 to License No. DPR-5
3. Safety Evaluation
4. Environmental Assessment (EA)
5. Notice of Issuance of EA

cc w/enclosures: See next page

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Mr. Stephen B. Bram
Consolidated Edison Company
of New York, Inc.

Docket No. 50-003
Indian Point Nuclear Generating
Station, Unit No. 1

cc:

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
CONSOLIDATED EDISON COMPANY OF NEW YORK INC.	)	Docket No. 50-003
	)	
(Indian Point Unit No. 1)	)	

ORDER APPROVING DECOMMISSIONING PLAN AND

AUTHORIZING DECOMMISSIONING OF FACILITY

By application dated October 17, 1980, as revised October 13, 1981; July 31, 1986; March 28, 1988; August 10 1989; March 28 and July 17, 1990; February 5, April 2, July 31, September 20, and October 12, 1993; May 13 and August 11, 1994; and July 19, 1995; Consolidated Edison Company of New York, Inc. (the licensee) requested the U.S. Nuclear Regulatory Commission (the Commission, NRC) to approve its proposed Decommissioning Plan (Plan) for Indian Point Unit No. 1 (IP-1) and an amendment to Provisional Operating License No. DPR-5 and the associated Technical Specifications (TSs) to make them consistent with the Decommissioning Plan. The Decommissioning Plan proposes long-term safe storage (SAFSTOR) of IP-1 spent fuel and residual radioactivity until the adjacent Indian Point Unit No. 2 (IP-2) has been permanently shut down. The licensee must submit a detailed dismantling plan for NRC review and approval prior to major dismantlement activities at IP-1.

A Notice of Consideration of Issuance of Amendment and Opportunity for Prior Hearing was published in the FEDERAL REGISTER on December 31, 1985, (50 FR 53407). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has reviewed the application with respect to the provisions of the Commission's rules and regulations and has found that decommissioning as stated in the Plan is consistent with the regulations in 10 CFR Chapter I and will not be inimical to the common defense and security or to the health and safety of the public. The basis for these findings is given in the concurrently issued Safety Evaluation by the NRC Office of Nuclear Reactor Regulation.

The Decommissioning Plan supplements the IP-1 Safety Analysis Report. Accordingly, a license condition has been added allowing the licensee to make changes to the Decommissioning Plan and Safety Analysis Report after performing a review based upon criteria similar to the criteria of Title 10 of the Code of Federal Regulations (10 CFR 50.59) to ensure that such changes do not involve an unreviewed safety question.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact for the proposed action. The Commission has determined that the proposed action will not result in any significant environmental impact and that an environmental impact statement need not be prepared. The Notice of Issuance of Environmental Assessment was published in the FEDERAL REGISTER on January 31, 1996.

Accordingly, pursuant to Sections 103, 161b, 161i, and 161o of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.82, the Commission approves the proposed Decommissioning Plan, dated October 17, 1980, as revised, and authorizes decommissioning of the IP-1 facility in accordance with the Decommissioning Plan and the Commission's rules and regulations, subject to the following conditions:

(a)(1) The approved Decommissioning Plan supplements the Final Safety Analysis Report (FSAR) and the licensee may (i) make changes in the facility or procedures as described in the FSAR or the Decommissioning Plan and (ii) conduct tests, or experiments not described in the FSAR or Decommissioning Plan, without prior Commission approval, unless the proposed changes, tests or experiments involve a) a change in the Technical Specifications (TSs) incorporated in the license or b) an unreviewed safety question, or c) major dismantlement activities such as removal of the reactor pressure vessel or other major radioactive components.

(2) A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR and/or the Decommissioning Plan may be increased or (ii) if the possibility of an accident or malfunction of a different type than evaluated previously in the FSAR and/or the Decommissioning Plan may be created; or (iii) if the margin of safety as defined in the basis for any TS is reduced.

(b)(1) The licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section if these changes constitute changes in the facility or procedures as described in the FSAR or Decommissioning Plan. The

licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records must nclude a written safety evaluation that provides the basis for determining that the changes, tests, or experiments do not involve an unreviewed safety question.

(2) The licensee shall annually submit, as specified in 10 CFR 50.4, a report containing a brief description of any changes, tests, and experiments, including summaries of the safety and environmental evaluation of each.

(3) The licensee shall maintain the records of changes in the facility until the date of termination of the license and shall maintain the records of changes in procedures and records of tests and experiments for 3 years.

(c) If the licensee desires (1) a change in the TSs, or (2) to (i) make a change in the facility or the procedures described in the FSAR or Decommissioning Plan, or (ii) conduct tests or experiments that are not described in the FSAR or Decommissioning Plan, and such changes, tests, or experiments involve an unreviewed safety question, a change in the TSs, or major dismantlement activities, the licensee shall submit an application to amend its license pursuant to 10 CFR 50.90.

For further details with respect to this action, see: (1) the licensee's application for authorization to decommission the facility, dated October 17, 1980, as revised; (2) Amendment No.to License No. DPR-5; (3) the related NRC Safety Evaluation; and (4) the NRC Environmental Assessment and

Finding of No Significant Impact. These documents are available for public inspection at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC, and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York.

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia Jr.

Frank J. Miraglia Jr., Acting Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 31st day of January 1996

Appendix A to

Provisional Operating License DPR-5

for the

Consolidated Edison Company of New York, Inc.

Indian Point Station

Unit No. 1

Docket No. 50-3

Amendment No. 45

Date of Issuance: January 31, 1996

2. Accordingly, Provisional Operating License No. DPR-5 is amended by revising the indicated paragraphs as follows:

   2.A. Pursuant to Section 104b. of the Act and Title 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations described in the application and this license;

   2.B. Pursuant to the Act and 10 CFR Part 70, to receive and possess up to 1918 kilograms of contained uranium-235 previously received for reactor operation;

   2.E. Pursuant to the Act and 10 CFR Parts 30 and 70, to receive and possess, but not to separate, such byproduct and special materials as were produced by the prior operation of the facility;

   3.A. Maximum Power Level

   The licensee is prohibited from taking the reactor to criticality, and the facility shall not be operated at any power level.

   3.B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, are hereby incorporated in the license. The licensee shall maintain the facility in accordance with the Technical Specifications.

3. This amended license is effective as of its date of issuance, shall be implemented within 30 days, and shall expire at midnight, October 14, 2006.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield, Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Attachment: Appendix A Technical
Specification Changes

Date of Issuance: January 31, 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT UNIT NO. 1

DOCKET NO. 50-3

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 45
License No. DPR-5

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to Provisional Operating License No. DPR-5 filed by Consolidated Edison Company of New York, Inc. (the licensee), dated October 17, 1980, as revised October 13, 1981; July 31, 1986; March 28, 1988; August 10, 1989; March 28 and July 17, 1990; February 5, April 2, July 31, September 20, and October 12, 1993; May 13 and August 11, 1994; and July 19, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will be maintained in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance that (i) the activities authorized by this amendment can be conducted without endangering the health and safety of the public and (ii) such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

ATTACHMENT TO LICENSE AMENDMENT NO. 45

PROVISIONAL OPERATING LICENSE NO. DPR-5

DOCKET NO. 50-3

Replace all of the pages of the Appendix A Technical Specifications with the enclosed pages.

TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

Appendix A to

Provisional Operating License DPR-5

For the

Consolidated Edison Company of New York, Inc.

1.0 GENERAL INFORMATION

The facility, known as the Consolidated Edison Indian Point Station Unit No. 1, is located on the 235 acre site in the Village of Buchanan, Westchester County, New York. The Consolidated Edison Indian Point Station Unit No. 2 and the New York Power Authority Indian Point Station Unit No. 3 share this site.

Indian Point Unit No. 1 includes a pressurized water reactor which operated with an authorized maximum steady state power level of 615 thermal megawatts until October 31, 1974. Pursuant to a June 19, 1980 Commission Order Revoking Authority to Operate Facility and a Decommissioning Plan for Indian Point Unit No. 1 submitted by Con Edison to NRC on October 17, 1980 in accordance with that Order, the reactor remains in a defueled status and the unit continues to operate as a support facility for overall Con Edison site operations. Unit No. 1 and Unit No. 2 are physically contiguous and share a number of systems and facilities as well as a common operating organization. The technical specifications contained herein recognize this commonality as well as the intended use of the Unit No. 1 facilities to support Unit No. 2 until retirement of that unit, and contain specific references to Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26. Unit No. 1 contains radioactive waste processing facilities which provide waste processing services for both Unit No. 1 and Unit No. 2. Radiological effluent limits are met on an overall site basis and specific operating limits and surveillance requirements for effluent monitoring instrumentation, including stack noble gas monitoring, are discussed in Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

1.1 Definitions

1.1.1 Operable-Operability

A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its intended safety function(s). Implicit in this definition shall be the assumption that necessary instrumentation, controls, electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its safety function(s) are also capable of performing their related support functions.

1.1.2 Member(s) of the Public

Member(s) of the Public includes all persons who are not occupationally associated with the site. This category does not include employees of either utility, their contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries.

1.1.3 Offsite Dose Calculation Manual (ODCM)

The Offsite Dose Calculation Manual contains the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

1.1.4 Process Control Program (PCP)

The Process Control Program is a manual containing and/or referencing selected operational information concerning the solidification of radioactive wastes from liquid systems.

1.1.5 Site Boundary

The Site Boundary is that line beyond which the land is neither owned, leased, nor otherwise controlled by either site licensee.

1.1.6 Solidification

Solidification is the conversion of wet wastes into a form that meets shipping and burial ground requirements.

1.1.7 Unrestricted Area

An Unrestricted Area is any area at or beyond the Site Boundary, access to which is not controlled by either site licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

1.2 Exclusion Distance and Restricted Area

1.2.1 The minimum distance from the reactor facility to the nearest land boundary of the exclusion area, as defined in Part 100 of the Commission's regulations, shall be 1400 feet.

1.2.2 The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone as defined in 10 CFR 100.3 is 460 meters and 1100 meters, respectively. For the purpose of satisfying 10 CFR Part 20, the Restricted Area is the same as the Exclusion Area defined in Figure 2.2-2 of Section 2.2 of the IP#2 FSAR.

1.3 Principal Activities

1.3.1 The principal activities carried on within the Exclusion Area shall be the generation, transmission and distribution of steam and electrical energy (except by gas-fired power plant); associated service activities; activities relating to the controlled conversion of the atomic energy of fuel to heat energy by the process of nuclear fission; and the storage, utilization and production of special nuclear, source and byproduct materials. Transmission and distribution of natural gas shall be through the use of facilities located as described in the application as amended.

2.0 REACTOR FACILITY DESIGN PERFORMANCE REQUIREMENTS

2.5 Electrical Power Supply

2.5.1 Major Supplies

2.5.1.1 Power for electrical equipment shall normally be supplied by at least two independent transmission feeders from the Consolidated Edison system. If power is lost to the spent fuel storage area radiation monitor, a portable monitor will be promptly set up in the spent fuel storage area.

2.10.2 Fuel Storage

2.10.2.1 No fuel other than irradiated fuel from Indian Point Unit No. 1 shall be stored in the Unit No. 1 spent fuel storage area. No fresh fuel shall be stored at Unit No. 1.

2.10.2.2 Spent fuel storage shall be provided in the storage pools in the Fuel Handling Building. The Fuel Handling Building and the spent fuel storage pool will contain the spent fuel until such time as offsite spent fuel management facilities are provided for, and the spent fuel is transferred to the Department of Energy, or as authorized by 10 CFR Part 72.

2.10.2.3 Spent fuel storage shall be provided with racks that shall limit the effective multiplication factor to less than 0.75.

2.10.2.4 Radiation levels in the spent fuel storage area shall be monitored continuously with a high level alarm indication in a location manned by a licensed operator* whenever there is irradiated fuel stored therein. If the monitor is inoperable, a portable monitor may be used. In such cases, provisions shall be made for prompt notification of a licensed operator upon actuation of the portable monitor's high level alarm.

2.10.2.5 If a spent fuel pool contains spent fuel, the spent fuel cask shall not be moved over that pool or within a distance of that pool such that the cask could strike the pool if it fell or tipped.

2.10.2.6 A dead-load test shall be successfully performed on the fuel handling building crane before fuel movement begins. The load assumed by the crane for this test must be equal to or greater than the maximum load to be assumed by the crane during the fuel handling operation. A thorough visual inspection of the crane shall be made after the dead-load test and prior to fuel handling.

2.11 Fire Protection

Overall site fire protection is provided by a fire protection system which is common to both Unit No. 1 and Unit No. 2. Operation, maintenance and testing are controlled by station procedures.

Fire protection and detection systems provided for protection of Indian Point Unit No. 2 safe shutdown systems are addressed in Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

° Licensed Operator for IP-2

3.0 ADMINISTRATIVE AND PROCEDURAL SAFEGUARDS

3.1 Responsibility

3.1.1 The Vice President Nuclear Power shall be responsible for overall facility activities and shall delegate in writing the succession to this responsibility during his absence.

3.1.2 The General Manager Nuclear Power Generation shall be responsible for facility operations and shall delegate in writing the succession to this responsibility during his absence.

3.2 Organization

3.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR for Indian Point Unit No. 2.

b. The General Manager-Nuclear Power Generation shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President-Nuclear Power shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager, however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

e. The operation of the facility, the operating organization, the procedures for operation, and modifications to the facility shall be subject to review by the Station Nuclear Safety Committee. The committee shall report to the Vice President, Nuclear Power.

f. The Nuclear Facilities Safety Committee shall function to provide independent review and audit of designated activities in areas of nuclear engineering, chemistry, radiochemistry, metallurgy and non-destructive testing, instrumentation and control, radiological safety, mechanical and electrical engineering, administrative controls and quality assurance practices, and radiological environmental effects.

g. All fuel handling shall be under the direct supervision of a licensed operator.*

h. The Senior Watch Supervisor is responsible for operations at the Unit No. 1 facility.

i. The Operations Manager shall hold a senior reactor operator license.*

* Licensed operator for IP-2

3.3 Operating Instructions and Procedures

3.3.1 No fuel will be loaded into the reactor core or moved into the reactor containment building without prior review and authorization by the Nuclear Regulatory Commission.

3.3.2 Detailed written instruction setting forth procedures used in connection with the operation and maintenance of the nuclear power plant shall conform to the Technical Specifications.

3.3.3 Operation and maintenance of equipment related to safety when there is no fuel in the reactor shall be in accordance with written instructions.

4.0 OPERATING LIMITATIONS

4.1 General

4.1.1 Whenever any operation is being performed that could result in the release of radioactivity or create a change in radiation levels, supporting facilities shall be maintained and operated as required in these Technical Specifications.

4.1.2 The concentration of radioactive materials released in liquid or gaseous form to unrestricted areas shall not exceed the limits specified in 10 CFR Part 20. Release of radioactive liquids and gases shall also be consistent with the requirements of 10 CFR Part 50, Appendix I, as specified in Specifications 3.9 and 4.10 of Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

4.1.3 All radioactive waste material shall be handled in accordance with 10 CFR Part 20. In addition, solid radioactive waste shall be controlled as specified in Specifications 3.9.D and 4.10.D of Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

4.1.4 Radiation monitoring systems shall be maintained operable for: (1) nuclear services building sewage, (2) sphere foundation sump, (3) secondary purification blowdown cooling water, and (4) area radiation monitors. If monitoring systems are not operable, effluent sampling and/or local monitoring shall be accomplished to replace the non-operating system. In addition, Unit 1 radioactive effluent monitoring instrumentation shall be operable as specified in Specification 3.9 of Appendix A to Indian Point Unit No. 2 Facility Operating License No. DPR-26.

4.1.5 The Indian Point site meteorological monitoring system shall be maintained and operated as specified in Specifications 3.15 and 4.19 of Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

4.1.6 The Indian Point site Radiological Environmental Monitoring Program shall be conducted as specified in Specification 4.11 of Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

4.1.7 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

4.1.8 High Radiation Area

4.1.8.1 As an acceptable alternate to the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 4.1.8.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry to such areas and the keys shall be maintained under the administrative control of the Radiation Protection Manager and/or the Senior Watch Supervisor on duty.

4.1.9 Spent Fuel Storage and Handling

4.1.9.1 All irradiated fuel shall be stored in the racks provided in the Fuel Handling Building Storage pools, with sufficient shielding that ensures that the radiation level on the operating deck is $\leq$ 15mr/hr. Should the radiation level be found to be above 15 mr/hr, corrective action shall be initiated to restore the level to $\leq$ 15 mr/hr.

4.1.9.2 Whenever, spent fuel storage pool water inventory is provided for personnel shielding, the normal water level shall be maintained at or above elevation 48 feet (approximately 6 feet above the top of the spent fuel racks). Any pool in which spent fuel is stored shall be subject to weekly verification of water level. Should the water level be found to be below elevation 48 foot, both pool level and radiation level on the operating deck shall be verified daily. Should the water level be found to be below elevation 47 foot, corrective action shall be initiated to investigate the reason for the reduced level and restore the level to ≥ 48 foot.

4.1.9.3 Water chemistry in any spent fuel storage pool containing spent fuel shall be maintained within the following limits:

Chlorides:	≤ 1.5 ppm
pH:	4.0 - 8.0
Conductivity	≤ 20 μs/cm

Should any of the above parameters be found to deviate from the specified limits an effort shall be promptly initiated to investigate the cause of the deviation and a process to restore the parameter to within the applicable limit shall be established in a timely fashion.

4.1.9.4 Ventilation capable of directing all Fuel Handling Building airborne effluents through monitoring pathways shall be available during any fuel movement or other activity that might potentially damage spent fuel assemblies.

5.0 MAINTENANCE

5.1 General

5.1.2 Components addressed in these technical specification requirements which have been repaired, replaced, or otherwise subjected to temporary or permanent modification shall be tested in accordance with procedures

which are appropriate in view of the nature of the repair, replacement, or modification, and the condition of the system.

5.2 Testing

5.2.5 Functional radiation monitoring systems (only for the following: nuclear services building sewage, sphere foundation sump, and secondary purification blowdown cooling water) and area radiation monitoring systems shall be:

(a) qualitatively checked daily to verify acceptable operability of instrument channel behavior during operation, and

(b) tested quarterly by injection of a simulated signal into the instrument channel to verify that it is operable, including alarm and/or trip initiating action. The quarterly interval is defined as quarterly plus or minus 25% of the quarter.

5.2.6 Unit 1 radioactive effluent monitoring instrumentation shall satisfy the surveillance requirements as specified in Specification 4.10 of Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

5.3 Spent Fuel Storage Pool Sampling

Any spent fuel storage pool containing spent fuel stored in water shall be sampled monthly for chloride level, pH and Cesium 137 activity. If Cesium 137 activity is found to be elevated above normal levels, an effort shall be promptly initiated to investigate the cause of the elevated level and take subsequent corrective action, as appropriate.

5.4 Sealed Sources

All sealed sources located on the Consolidated Edison Indian Point Station Site are maintained under the Indian Point Unit No. 2 Facility Operating License No. DPR-26 and surveillance and use of such sources are addressed in Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

6.0 PLANT REPORTING REQUIREMENTS

6.1 Routine Reports and Reportable Occurrences

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator-Region I, unless otherwise noted.

6.1.2 Annual Radiological Environmental Operating Report[1]

6.1.2.1 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

6.1.2.2 The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including (as appropriate) a comparison with preoperational studies, operational controls and previous environmental surveillance reports; and an assessment of the observed impacts of the plant operation on the environment. The report shall also include the results of land use censuses required by Specification 4.11.B. of Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

The Annual Radiological Environmental Operating Report shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements as described in the ODCM. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

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1 A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps[3] covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program; discussion of all deviations from the sampling schedule; and discussion of all analyses in which the LLD required was not achievable.

6.1.3 Radioactive Effluent Release Report[1]

6.1.3.1 Routine Radioactive Effluent Release Reports covering the previous 12 months of operation shall be submitted by May 1 of each year.

6.1.3.2 The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in the Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted by May 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability.[4] This same report

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[1] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

[3] One map shall cover stations near the site boundary; a second shall include more distant stations.

[4] In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods are acceptable. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109 Rev. 1, October 1977.

The Radioactive Effluent Release Report shall include the following information for each class of solid waste (in compliance with 10 CFR Part 61) shipped offsite during the report period:

a) Container volume.

b) Total Curie quantity (specify whether determined by measurement or estimate),

c) Principal radionuclides (specify whether determined by measurement or estimate),

d) Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottom),

e) Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f) Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Report shall include a list and description of unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Report shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 4.11.B of Appendix A to the Indian Point Unit No. 2 Facility Operating License No. DPR-26.

6.2 Special Reports

6.2.1 Reports of major safety-related corrective maintenance shall be submitted to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the Regional Administrator - Region I, no later than 6 months following completion of such maintenance.

6.2.2 Each such report shall include a description of any major safety-related corrective maintenance performed including the system and component involved.

6.3 Reportable Event Action

6.3.1 A Reportable Event is defined as any of the conditions specified in 10 CFR 50.73.a(2).

6.3.2 The following actions shall be taken in the event of a reportable Event:

a. A report shall be submitted to the Commission pursuant to the requirements of 10 CFR 50.73 and

b. Each Reportable Event report submitted to the Commission shall be submitted to the NFSC Chairman, and the Vice President-Nuclear Power and be reviewed by the SNSC.

6.4 Any references to the term "Safety Analysis Report", "SAR" or "FSAR" for Indian Point Station, Unit No. 1, shall be deemed to refer, as appropriate, to the following exhibits which are a part of the application: K-5 (Rev.-1) Figures 1-2, 1-3, 3-14 only, K-5A11 Section 3.7.2 pages 171 through 176 only.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AN ORDER AUTHORIZING DECOMMISSIONING AND

AMENDMENT NO. 45 TO LICENSE NO. DPR-5

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT UNIT NO. 1

DOCKET NO. 50-3

1.0 INTRODUCTION

Indian Point Nuclear Generating Station, Unit No. 1 (IP-1), is a four-loop pressurized water reactor with a thermal rating of 615 MW. IP-1 is owned and operated by Consolidated Edison Company of New York, Inc. (Con Edison), and is located in Westchester County, New York, near the village of Buchanan, New York. IP-1 operated from August 1962, until it was permanently shut down on October 31, 1974, because the plant emergency core cooling system did not meet the current regulatory requirements.

IP-1 is situated between Indian Point Unit Nos. 2 and 3 (IP-2 and IP-3), two operating nuclear power plants. IP-2 and IP-3 are owned and operated by Con Edison and the Power Authority of the State of New York, respectively. By January 1976, all spent fuel from the IP-1 reactor had been moved to the IP-1 west spent fuel pool and is expected to remain there until a Federal repository is available to receive it.

IP-1 encompasses many systems and buildings that are required for operation of IP-2. Except for the fuel-handling building which houses the spent fuel, all other major buildings, including the IP-1 containment building, contain common facilities that will continue to be used to support IP-2 operations. This situation is expected to continue throughout the life of IP-2.

On October 17, 1980, as revised October 13, 1981, Con Edison submitted a Decommissioning Plan (SAFSTOR option) for IP-1 in response to an NRC Order of June 19, 1980, which also revoked Con Edison's authority to operate the reactor. Con Edison plans to maintain IP-1 in a safe-storage condition until after IP-2 ceases to operate, to dismantle both units at the same time, and to remove residual radioactivity so that Con Edison property can be released for unrestricted use and both licenses terminated. In response to questions and comments from the NRC, Con Edison revised the Decommissioning Plan,

provided supplemental environmental information, and submitted proposed Technical Specifications (TSs) by letters dated July 31, 1986; March 28, 1988; August 10 1989; March 28 and July 17, 1990; February 5, April 2, July 31, September 20, and October 12, 1993; May 13 and August 11, 1994; and July 19, 1995.

A Notice of Consideration of Issuance of Amendment and Opportunity for Hearing was published in the FEDERAL REGISTER on December 31, 1985 (50 FR 53407). The notice stated that the NRC was considering (1) approval of the IP-1 Decommissioning Plan, (2) amendment of License No. DPR-5 to a possession-only status, (3) revision of the TSs to be consistent with SAFSTOR status, and (4) renewal of the IP-1 license to be consistent with the expiration date of the IP-2 license. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Notice stated a license renewal date of October 14, 2006 to coincide with the planned permanent shutdown of Indian Point Unit 2 (IP-2). Subsequent to the December 31, 1985 notice, the license for IP-2 was extended to September 28, 2013 and Con Edison informed us, in a letter dated March 28, 1988, of its intention to delay dismantlement of IP-1 to after that date. This safety evaluation and the enclosed environmental assessment of the decommissioning plan are consistent with the 2013 date. However, we have renewed License No. DPR-5 to October 14, 2006, to be consistent with the license renewal application as noticed in the December 31, 1985 FEDERAL REGISTER Notice in order to put new TSs for the current shutdown condition in place.

2.0 EVALUATION

This evaluation considers the possession-only license amendment, safety issues related to SAFSTOR of IP-1 to September 28, 2013, and the Con Edison financial assurance plan.

2.1 Possession-Only License Amendment

Sections 2.A, 2.B, 2.E, and 3.A of License No. DPR-5 are changed to clarify that the provisional operating license is being modified so that the licensee may possess IP-1, but not operate it at any power level; may possess, but not use, the uranium fuel previously used for reactor operation; and may possess byproduct and special nuclear materials that were produced by prior operation of IP-1. Issuance of the possession-only license amendment involves no unreviewed safety issues and does not change the operating status of IP-1 because the NRC order of June 19, 1980, already revoked the authority of Con Edison to operate IP-1. Therefore, issuance of the license changes to clarify IP-1 possession-only status is acceptable.

2.2 Spent Fuel Storage

The IP-1 spent fuel pool system consists of east and west fuel pools, a failed fuel pool, a cask loading pool, and a transfer pool in the fuel handling building (Figure 1). All of the spent fuel (160 assemblies) is stainless steel clad and is stored in the west pool, as are some control rods, follower rods, and hold-down columns. The east pool contains control rods, follower rods, flux depressors, shim rods, filler rods, dummy fuel elements, and a fuel element top nozzle, but contains no fuel. The failed fuel pool, the disassembly pool, and the cask loading pool contain no radioactive components or spent fuel. The east and west pools have been isolated from the other pools with two gates in series in the gate openings between the west pool and the transfer pool and between the west pool and the disassembly pool. Subsequently, the transfer pool, the failed fuel pool, the disassembly pool and the cask loading pool have been drained and the radioactively contaminated sludge material has been removed.

The fuel is stored under a minimum of 5.0 feet of water for shielding, and pool covers keep debris and other objects out of the pools. The spent fuel pool cooling system is no longer in use, and water temperature measurements are not required because the decay heat generated by the spent fuel is now significantly reduced and the pool water is now more than adequately cooled by natural convection. In addition, calculations by the licensee and NRC staff (February 11, 1980, "NRC Order to Show Cause") showed that, although water is required for shielding, the fission products in the spent fuel had decayed enough that air cooling of IP-1 fuel would be adequate in the event of a complete loss of water from the pool. Continued storage of spent fuel since 1980 has resulted in additional fission product decay and reduced heat load from the spent fuel.

2.2.1 Spent Fuel Pool Water Level Monitoring

Con Edison has installed a high-low level alarm system and a recorder to monitor and record pool water level. The high and low level alarms are indicated on an IP-1 panel in the always manned IP-2 control room and locally near the spent fuel pools. In addition, the radiation monitor in the fuel storage building would activate a local alarm and an alarm in the IP-2 control room if pool water level dropped significantly. East and west pool water level is recorded locally and station operating procedures require checking of pool water level every 6 hours. The staff has determined that Con Edison has adequately addressed unexpected changes in spent fuel pool water level.

2.2.2 Technical Specifications (TSs) for Spent Fuel Pool and Fuel Handling

The proposed SAFSTOR TSs for spent fuel storage (Section 2.10.2) are not significantly changed from current TSs. Additional requirements are specified for pool water chemistry, fuel handling, building ventilation, and pool water sampling. The proposed TSs provide criticality limits and requirements for radiation and water chemistry monitoring, fuel handling, and cask movement near the fuel pool. Key TS requirements are as follows:

(1) Spent fuel storage shall be provided with racks that shall limit the effective multiplication factor to less than 0.75;

(2) Radiation levels in the spent fuel storage area shall be monitored continuously with a high-level alarm indication in a location manned by a licensed operator (licensed operator for IP-2);

(3) No fuel will be loaded into the reactor core or moved into the reactor containment building without prior review and authorization by the Nuclear Regulatory Commission;

(4) If a spent fuel pool contains fuel, the spent fuel cask shall not be moved over the pool or within such a distance of that pool such that the cask could strike the pool if it fell or was tipped;

(5) All fuel handling shall be under the direct supervision of an IP-2 licensed operator;

(6) Any pool in which spent fuel is stored shall be subject to weekly verification of water level; and

(7) Water chemistry in any spent fuel storage pool containing spent fuel shall be maintained within the following limits:

Chlorides:	1.5 ppm
pH:	4.0 - 8.0
conductivity	$\leq$ 20 micromhos per cm

Should any of the above parameters be found to deviate from the specified limits an effort shall be promptly initiated to investigate the cause of the deviation and a process to restore the parameter to within the applicable limit shall be established in a timely fashion.

(8) Any spent fuel storage pool containing spent fuel shall be sampled monthly for chloride level, pH and cesium-137 activity. If cesium activity is found to be elevated above normal levels, an effort shall be promptly initiated to investigate the cause of the elevated level and take subsequent corrective action.

Portable demineralization and filtration equipment now purifies and filters the pool water. Cesium monitoring and water chemistry controls will adequately protect the stainless steel cladding of the spent fuel. Also, cesium-137 measurements will show whether there are any leaks in the cladding.

The staff has reviewed the proposed Decommissioning Plan and TS for spent fuel and finds that the requirements for storage and handling are sufficient to ensure the spent fuel will be safely stored and handled, and that fuel pool water chemistry will be adequately monitored.

2.2.3 IP-1 Resolution of Bulletin 94-01

On April 14, 1994, the NRC staff issued Bulletin 94-01, "Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1," to inform licensees of NRC concerns relative to the storage of spent fuel at permanently shutdown plants and to decommissioning. The NRC staff also initiated special team inspections at permanently shutdown facilities, including IP-1, to determine if there were related problems at these facilities.

The special team inspection of IP-1 was conducted from December 12 through 15, 1994 to review Con Edison actions to implement Bulletin 94-01. The results of the team inspection (J. H. Joiner, NRC, letter to S. E. Quinn, Con Edison, February 10, 1995) are summarized below.

2.2.3.1 Management oversight

The inspection team conducted interviews with the plant management, plant employees, and security officers and concluded that management oversight of plant activities was evident. Interviews with radiation protection (RP) technicians, contractors, and maintenance personnel confirmed that site management periodically assessed the status of their work. IP-1 is physically located directly adjacent to the operating IP-2. IP-1 is used in support of IP-2 activities, such as waste processing and handling and storage of radioactive waste generated from IP-2 operations. As a result of the close proximity to the operating unit and the routine presence of personnel performing work in the IP-1 buildings, management oversight of IP-1 activities was continuing and was at an appropriate level. Management awareness of and concern for problems identified in IP-1, such as spent fuel pool leakage, were good.

2.2.3.2 Organization

The IP-1 project manager reports to the Vice President of Nuclear Power through the Manager, System Engineering and Analysis. The project manager has primary responsibility for all IP-1 activities, including those associated with the spent fuel pool system, maintenance, and repair in IP-1, preparation of facilities for possible use for interim low-level waste storage, and future decommissioning plans. The appointment of a project manager for IP-1 demonstrates additional support for IP-1 activities.

2.2.3.3 Training and equipment for spent fuel handling

Con Edison does not anticipate any movement of IP-1 spent fuel until after the retirement of IP-2, when all spent fuel from the site will be shipped to a Federal repository. In its submittal dated September 20, 1993, Con Edison outlined its expected actions should fuel movement be desired or become necessary. The proposed plan and TSs require fuel movement to be conducted under the direct supervision of a licensed IP-2 operator who is trained on the necessary skills, knowledge, procedures and equipment for IP-1 fuel movement

in accordance with the training program accredited by the Institute of Nuclear Power Operation (INPO). The Con Edison training will ensure that fuel movement will be conducted in a timely manner and without threat to the public health and safety.

Con Edison stated that since cessation of power operations, IP-1 has increasingly been utilized for the operational support of IP-2 and that IP-1 is an integral part of power generating operations at the Indian Point site and will remain so throughout the service life of IP-2. IP-2 personnel, trained and qualified using training programs based on a systems approach to training (SAT), routinely enter IP-1 to perform site-related duties and to monitor and inspect the facility. IP-1 systems are used in accordance with site radiological protection, effluent control, and environmental monitoring programs.

A tool for handling fuel assembly baskets is on site and available for use. Con Edison has manufactured special tools for use during refueling operations. If the need arises, Con Edison has the ability and experience to construct and use new tools or procedures for fuel handling. Con Edison also has a contract with Westinghouse to procure special fuel-handling tools.

Con Edison has stated that Procedures 0.18.G and 18.7 are currently available for IP-1 fuel handling and that a plan to move IP-1 fuel could be developed in a timely manner without any threat to the public health and safety. Although inactive, the procedures can be updated and reviewed in sufficient time to support fuel movement. In addition, Station Administrative Order (SAO) 202 specifies management supervision, support, and controls required to safely perform this infrequent activity.

Training in supervising fuel movement and in operating cranes is conducted as part of the IP-2 training program. The inspection team determined Con Edison also has the option of using qualified outside contractors to effect fuel movement at IP-1 should it be deemed necessary. If outside contractors were used, all fuel handling would be under the direct supervision of an IP-2 licensed operator trained for fuel handling at IP-1.

The fuel-handling and bridge crane is adequate for fuel and cask handling and is used frequently in IP-2 activities. Crane operability and crew training were demonstrated to the inspection team by the removal and replacing of the steel cover plates on each of the six IP-1 pools to allow inspection of the pools and stored fuel. Con Edison stated that the crane will continue to be used until after the decommissioning of IP-2.

Based on the team inspection and this evaluation, the staff concludes that the Con Edison plan to train IP-2 licensed operators or others for handling fuel at IP-1 is consistent with, and meets the intent of, the regulations and is, therefore, acceptable.

2.2.3.4 Self-assessments and audits

TS 3.1.1.e. in the current TSs and 3.2.1.e in the proposed TSs requires that the Station Nuclear Safety Committee (SNSC) independently review designated activities and report to the Vice President, Nuclear Power. The SNSC met and reported on its activities monthly. The inspection team reviewed "Nuclear Safety Group Functions and Responsibilities" and the monthly reports produced by the SNSC.

The inspection team reviewed the Con Edison self-assessment and audit programs to verify their adequacy to identify developing problems and implement timely corrective actions. The inspection consisted of selected interviews with plant personnel, reviews of audits, selected reviews of the Onsite Review Committee meeting minutes, and independent verification of related plant activities.

Interviews with licensee personnel and review of a sampling of quality assurance reports indicates that quality assurance reviews and audits of IP-1 operations have been adequately performed on a regular basis and that the audit program implementation is good.

Con Edison also has a mechanism to confront significant station safety issues. Station Administrative Order-132 (SAO-132) reports are generated when an issue has been identified as warranting additional resources. SAO-132 reports heighten management's awareness of issues that need to be resolved. The inspection team concluded that Con Edison was complying with the IP-1 TSs and that management oversight was evident in the self-assessment activities.

2.2.3.5 Safety review program

The inspection team reviewed the Con Edison administrative procedure associated with the 10 CFR 50.59 safety review process and also evaluated a number of the Con Edison safety reviews. Changes at IP-1 that may involve an unreviewed safety question or a change in the TS were evaluated with respect to the FSAR as defined by TS 6.4. The FSAR comprises the Final Hazards Summary Report for Core B, as amended, and additional exhibits as specified in TS 6.4. A complete copy of the three-volume FSAR was given to the inspection team. The decommissioning plan is a supplement to the FSAR and may also be changed in accordance with criteria similar to 10 CFR 50.59 requirements as specified in the decommissioning order.

10 CFR 50.71 and its statement of considerations (45 FR 30614, May 9, 1980) specifically excludes IP-1 from 10 CFR 50.71 requirements to submit revised pages of the FSAR to the NRC. Con Edison is, however, required to comply with 10 CFR 50.59 requirements to update the FSAR and have it available for review by NRC inspectors. The inspection team found that 50.59 safety reviews were complete and in compliance with 10 CFR 50.59 requirements. Con Edison safety reviews of the reduction of spent fuel pool water level, the pool isolation

gate replacement, and the draining of unused pools were particularly applicable to Bulletin 94-01 and were found to be acceptable. The inspection team determined that Con Edison properly used the 50.59 safety review process and has conducted consistently excellent safety assessments.

2.2.3.6 Technical specification compliance

The inspection team reviewed the IP-1 TSs currently in place and selected applicable implementing and surveillance procedures, records, and logs documenting that Con Edison had complied with the TS requirements. The inspection team determined from these reviews that Con Edison had adequate programs and procedures in place and effectively implemented the TS requirements. No violations of NRC requirements were identified.

The inspection team also reviewed the proposed SAFSTOR TSs and determined that they provide adequate requirements for criticality, radioactivity and water chemistry monitoring, fuel handling, and cask movement near the fuel pool.

2.2.3.7 Special nuclear material accountability

The inspection team reviewed the Con Edison special nuclear material accountability records and recent audits and determined that fuel was being stored in accordance with licensing requirements and applicable procedures.

Station Administrative Procedure SAO-411, "Special Nuclear Material Accounting and Handling," and an inventory of the components stored within the east and west fuel pools were also reviewed. The inspection team verified that in accordance with the Procedure SAO-411, a physical inventory of special nuclear material was conducted annually. No deficiencies were identified in special nuclear material accountability.

2.2.3.8 Spent fuel pool water chemistry

Fuel pool water is monitored monthly for pH, conductivity, chloride, sulfate, boron, and radioactivity. Portable demineralization and filtration equipment currently purifies and filters the pool water to maintain the chemistry within limits established by Con Edison and within the proposed SAFSTOR TSs. Allowable cesium-137 in the spent fuel pool system is based on as-low-as-is reasonably-achievable (ALARA) considerations. Exposure to workers near the contaminated pools and potential exposure to the public from pool leaks are considered as are exposures received from operation of a cleanup system and potential exposures to radioactivity in stored resins.

Sampling specifications in Procedure IPC-S-012, "Chemistry Specifications" were reviewed by the inspection team. The procedure specifies the chemistry sampling program for both IP-1 and IP-2. Spent fuel pool water analysis currently being conducted is consistent with requirements in the proposed SAFSTOR TS. The inspection team considered the sampling program for the IP-1 spent fuel pool to be good. Chloride concentration is limited by TS 4.1.9.3 to 1.5 parts per million (ppm), pH to the range of 4.0 to 8.0 and conductivity

to 20 micromhos per cm. The inspection team verified that water quality was sampled and analyzed every month and when the above limits were approached the spent fuel pool water was processed through an ion exchange system to bring water quality within the above requirements.

2.2.3.9 Spent fuel pool condition

Covers were lifted from each of the six pools in the IP-1 spent fuel pool system, including the west fuel pool containing the spent fuel assemblies and the east fuel pool containing irradiated hardware. With the exception of the latter two pools, in which the water level had been lowered to about the 49-foot elevation, the pools have been drained. The inspection team observed no continuing through-wall leak indications between the west fuel pool and adjacent empty pools, although pool wall colorations did indicate that leakage may have occurred in the past. The outside walls of the east fuel pool cannot be easily observed. Several of the "empty" pools did have moisture or water on the floor. The source of this water was unknown, but the licensee suspects that water vapor from the warm (approximately 85°F at the time of inspection) water in the east and west pools is carried over into the empty pools and condenses on the cooler pool walls. The inspection team noted that water vapor rose from the empty pools when the covers were lifted during the inspection. The inspection team also noted that the licensee completed installing vapor barriers between the pools on September 9, 1994, to minimize water vapor carryover between the pools.

Before draining the pools, Con Edison calculated the stresses on the north and south walls of the west fuel pool with water on one side only. These calculations showed that the walls would support the increased hydrostatic load. The inspection team reviewed the calculations with Con Edison engineers and discussed ongoing engineering reviews of the spent fuel pool walls to determine areas of high stress that may be susceptible to cracking and leakage. Con Edison is proceeding with finite element analysis, mapping, and additional physical investigation of the walls. The inspection team determined that Con Edison has provided adequate evaluation and analysis of the fuel pool structure to ensure the safety of fuel storage.

2.2.3.10 Spent fuel pool water inventory

The inspection team reviewed the status and results of the licensee's efforts to quantify water losses from the spent fuel pool system, to identify water loss pathways and mitigate losses, and to assess the consequences of any losses on public health and safety and the environment. This area was previously reviewed in NRC Inspections 50-03/94-01 (May 23-25, 1994 and June 16-17, 1994) and 50-03/94-02 (August 11-12, 1994). The team noted that the mass balance data based on boron and tritium measurements are in good agreement with the "apparent" pool losses based on spent fuel pool level decreases.

Con Edison installed plastic vapor barriers over and between the pools to reduce the effect of evaporation and thereby allow a more accurate estimate of pool leakage. After vapor barrier installation, the estimated water loss rate declined to about 25 gallons per day (gpd) and has remained there. This rate also agrees fairly closely with the effective pool release rate calculated by mass balance techniques using the boron and tritium measured in the determined leakage pathways.

Con Edison plans to further evaluate pool evaporation losses versus pool leakage by lowering the spent fuel pool (SFP) water temperature from the current 85°F to approximately 55°F by use of a heat exchange unit. The residual loss rate should then more nearly approximate the actual pool leak rate. Additional details on pool leak rate measurements are provided in NRC Inspection Report 50-03/94-80 of February 10, 1995.

In conclusion, the NRC staff finds that the projected 25 gpd water loss rate is a sufficiently accurate measurement and that if all of the 25 gpd loss were to ground water it would not cause a significant offsite exposure (see Section 2.2.3.16).

2.2.3.11 Siphon concerns

The inspection team checked to see whether Con Edison had identified any temporary or permanent piping or hoses connected to the SFP that could serve as a siphon path and, if necessary, that corrective actions had been taken to prevent pool siphoning. IP-1 building and installed plant piping was inspected to verify plant drawings against actual plant configurations.

The inspection team confirmed that no active siphon paths were created by the existing equipment or piping configurations and that procedures were adequate to assure that any temporary hoses or piping would not result in siphoning.

2.2.3.12 Freeze concerns

The east and west fuel pools (all spent fuel is in the west pool) have been isolated from the other pools, and the other pools, including the transfer pool, and have been drained. The transfer pool and its associated transfer tube are dry. As a result, there can be no loss of water from the spent fuel pool system due to freezing of the transfer tube.

Most of the auxiliary steam system has been isolated. At the time of the inspection, only the steam heating lines to the outside waste distillate tanks were in service. Con Edison is installing 40-kilowatt electric heaters in the spent fuel building floor area to replace the auxiliary steam as a source of heating. Normal building ventilation heats the remainder of IP-1 with warm air from the fuel building floor. Supplemental operator building logs were initiated to monitor ambient temperatures in applicable areas of IP-1 and temporary heating was installed to preclude freezing events in IP-1 until the permanent system was completed.

2.2.3.13 Emergency preparedness

The inspection team reviewed the Con Edison Emergency Plan, Emergency Operating Procedures, and Emergency Plan Implementing Procedures to determine if they were adequate to respond to potential accidents for the permanently defueled condition of IP-1 with stored spent fuel on site.

The IP-1 facility is included in the IP-2 Emergency Plan, and the inspection team verified that IP-1 had been included in some of the recent emergency drills. The emergency response capability for the Con Edison site has been routinely reviewed during past NRC inspections and exercise observations. Inspection findings typically indicated excellent response capability. The inspection team noted that IP-1 did not appear vulnerable to any accident scenarios that were outside the scope or severity of IP-2 scenarios.

Because radiation decay of the IP-1 spent fuel has gone on for more than 20 years, the total loss of spent fuel pool water would not result in fuel overheating or fuel cladding degradation. High radiation levels would occur directly above the fuel pool; however, the levels would be manageable by licensee personnel and no significant offsite dose consequences would be expected. The inspection team further noted that the Con Edison response organization was the same for both units and was well trained to cope with IP-1 emergency events.

2.2.3.14 Radiation protection

The inspection team reviewed the adequacy of the IP-1 radiation protection program for normal and accident conditions to determine Con Edison compliance with TS requirements. This inspection effort included a review of the radiation protection organization, procedures, survey records, ALARA reviews, and the radiological surveillance and controls for plant personnel.

The Con Edison radiation protection program is common to IP-1 and IP-2. The radiation protection group dedicates one health physics (HP) supervisor and two HP technicians to provide the radiological surveillance and work coverage for IP-1. Con Edison has established a schedule to survey all accessible areas of IP-1 on a 6-month to 2-year frequency. The clean area walkways are scheduled for contamination surveys on a daily or weekly basis.

Con Edison survey records show that approximately half of the rooms in IP-1 were contaminated to more than 100,000 dpm/100 cm^2, there were several high radiation areas and the drained fuel handling pools were contaminated to more than 3 million dpm/100 cm^2. The contaminated rooms were well posted and the drained fuel handling pools were covered and locked with HP-controlled locks. The IP-1 HP staff demonstrated good knowledge of work occurring in the facility and provided the appropriate radiological controls.

The inspection team concluded that Con Edison provided good radiological protection for personnel at IP-1 and met all conditions of 10 CFR Part 20 and the TSs.

2.2.3.15 Material condition and maintenance

The inspection team observed the material condition of IP-1 Vapor Containment, Fuel Handling Facility, Chemical Systems Building, and Nuclear Services Building. Facility housekeeping, lighting, fire protection services, ventilation, and general building integrity were assessed. The overall physical plant was neat, clean, and orderly.

The lower elevations of the Chemical Systems Building indicated continued rain water intrusion from the IP-1 bioshield structure, and limestone mineral deposits on all of the axial joints in the concrete dome inside the bioshield structure were evidence of leaking over some period of time. The inspection team noted that the steel containment sphere surrounding Unit 1 inside of the concrete dome shield was in good condition. The intruding rain water flowed down walls into the Unit 1 bioshield annulus and also outside the bioshield wall and down into the Chemical Systems Building. This leaking created the potential for migration of contamination from contaminated areas and for additional radwaste cleanup. Con Edison has repaired the roof area over the annulus, and is evaluating the repair of less significant leaks in axial joints of the concrete dome.

2.2.3.16 Radiological effluent and environmental monitoring

Indian Point Units 1, 2, and 3 have a common radiological environmental monitoring program (REMP). Consequently, IP-1 has been monitored by a complete REMP for an operating facility and has been routinely inspected by the NRC over the years. The team reviewed the licensee's actions and program enhancements implemented as a result of the suspected spent fuel pool leakage. These actions included the addition of routine boron and tritium analyses for the sphere foundation drain (SFD), and more frequent analyses of the SFD and north curtain drain (NCD) effluents. (The NCD had not been included in the TS requirements.) These enhancements were more fully described in Inspection Reports 50-03/94-01 and 50-03/94-02. In addition, Con Edison has added a sampling and monitoring point in the Offsite Dose Calculation Manual/REMP, for the onsite stream. This stream is at a lower elevation than the groundwater at the plant and would be a probable location for detecting any activity that entered the groundwater from IP-1. Tritium analyses are to be performed quarterly on these samples, when water is available. Further, Con Edison added annual sediment and tritium sampling from the onsite storm sewers. No environmental samples have indicated elevated measurements as a result of IP-1 pool leakage.

The inspection team reviewed the Con Edison measurements, calculations and assessments of the loss of pool water and environmental impacts. Using conservative assumptions and NRC Regulatory Guide 1.109 methodology, Con Edison calculated that daily releases of 100 gallons per day from the fuel pools would result in approximately 0.1 percent of the routine plant releases. The annual dose commitment from such releases is primarily due to tritium via the fish pathway and is of the order of 0.1 microrem/year (whole body) to an individual. Actual water releases have been determined to be approximately 25 gallons per day (Section 2.2.3.10).

The inspection team reviewed the airborne release pathways from IP-1 and determined that adequate monitoring is incorporated into the IP-2 TSs and has been routinely reviewed by the NRC in conjunction with effluents inspections at IP-2. The last inspection of this area is documented in Inspection Report 50-247/93-11. No problems were detected.

The inspection team determined that all applicable documentation relative to leakage of radioactive materials from the SFP and other onsite sources has been maintained pursuant to 10 CFR 50.75(g). These documents will be important during decommissioning activities to assure all residual licensed radioactivity is appropriately addressed.

The inspection team concluded that Con Edison has been responsive to concerns of pool leakage and its impact on the public health and environment. The implemented program appears to incorporate all reasonable pathways of release and the data demonstrates no offsite impacts from IP-1 pool leakage.

2.2.4 Potential Criticality from Seismic Events

All 160 spent fuel assemblies are stored in a stainless steel rack as described in the Con Edison report entitled "Indian Point Unit No. 1 Spent Fuel Racks Earthquake Analysis" (December 5, 1975). The Con Edison report concludes that the racks are capable of withstanding the postulated earthquake forces, including gravity loads, and that under these postulated loads there is no loss of function. An NRC report dated November 29, 1974, concluded that the most significant local fault (the Ramipo) is not active within the meaning of Appendix A of 10 CFR part 100 and that 0.15g is an adequately conservative value for the reference acceleration for seismic design. This safety evaluation reviews the rack and the spent fuel only with respect to the potential for seismically induced criticality. Significant related information from Con Edison is as follows:

(1) The fuel rods are core B type and are 12-percent cold-worked stainless steel tubes containing slightly enriched uranium dioxide fuel. The wall thickness of the tubes is designed to make the cladding free standing under all reactor operating conditions (Con Edison letter dated December 11, 1964, page 1.0-1).

(2) All fuel is stored in the west fuel pool. U-235 enrichments for IP-1 varied from 2.86 w/o to 4.08 w/o without any burnup assumed (Con Edison letter dated December 11, 1964, table 3-3).

(3) Each storage location may hold one fuel storage basket containing four fuel assemblies. Each basket is essentially open on all four sides with a center divider of 1/8-inch thick boron-stainless steel plates (Con Edison letter dated October 20, 1975).

(4) The storage locations are formed by a gridwork of 10-inch high stainless-steel plates welded together to form funnel-shaped square openings around each fuel basket at the top and the bottom. The plates are continuous across the width of the pool and are fastened to the wall at each end,

(Con Edison letter dated October 20, 1975, Figures 3 and 4). The structure thus formed is supported by a 6WF25 column to which the guide funnels are welded. Each column is bolted to the pool floor by two 3/4-inch anchor bolts. The fuel assembly is supported at the lower nozzle by the basket, which rests on ledges welded to the lower guide funnel with about a foot and a half of clearance between the bottom of the fuel assemblies and the floor of the pool. On either side of each group of storage locations is an aisle about 2.5 feet wide. All storage baskets are at least 8.5 inches apart.

(5) Con Edison concluded in its March 28, 1988, submittal, page 14 that the most reactive fuel assemblies (4 assemblies) contained in the storage baskets would have a K_{eff} of less than 0.69 with no boron in the pool water. The current and proposed TS 2.10.2.3 limits the effective multiplication factor to less than 0.75.

We have, therefore, determined that the IP-1 spent fuel is adequately protected from earthquake induced criticality for the following reasons:

(1) Fuel rack is constructed to maintain spacing of at least 8.5 inches between each four-assembly basket. Fuel assemblies in each basket are separated by boron-stainless steel plates. The spent fuel, as stored, has a K_{eff} of less than 0.69 with no credit for the boron in pool water.

(2) A design basis earthquake may cause some damage to the IP-1 spent fuel rack but not enough to remove boron plates from baskets or to significantly reduce the 8.5 inch spacing between baskets, and the boron-stainless steel plates and remaining separation of fuel assemblies would be sufficient to prevent criticality even without credit for the boron in the pool water.

(3) The licensee's analysis indicated that, in a postulated design basis earthquake, the racks would be able to sustain the load, including the gravity load, without exceeding material yield stress. Thus, the basic geometry of the racks would be maintained and the rack structure would protect the fuel assemblies and boron plates, preserving adequate geometry from a criticality point of view.

2.3 Financial Considerations

Con Edison's financial assurance plan and funding report for the decommissioning of IP-1 and IP-2 are contained in a letter of July 17, 1990. Therefore, this Con Edison submittal met the July 26, 1990, submittal date required by 10 CFR 50.33(k)(2).

Con Edison had previously established an internal funding method to retain the funds collected for IP-1 and IP-2 decommissioning. On June 28, 1988, Commission regulations (10 CFR 50.75) were revised to establish requirements for reasonable financial assurance for reactor facility decommissioning. Since the licensee financial surety method was not consistent with these regulations, Con Edison established a separate external decommissioning

sinking trust fund for each of the two nuclear units. In 1989, certain amounts of those internal fund contributions were deposited into an external trust fund. As of May 31, 1990, Con Edison had contributed $34.9 million to the external fund, and the market value of the balance in the external fund was $37.1 million.

Con Edison calculates the funding levels prescribed by the Commission regulations to be about $218.1 million at the time of decommissioning for both reactor facilities, adjusted to 1989 dollars. Although Con Edison did not submit detailed individual decommissioning cost estimates for each unit, it did specify the total amount of funds required to decommission both units at the same time.

The staff compared the $218.1 million total estimated decommissioning costs to the minimum amounts required by the regulations of 10 CFR 50.75 and has determined that sufficient funds will be available to meet the decommissioning costs. Thus, Con Edison has demonstrated that there is reasonable assurance of adequate funds for decommissioning both of the reactor facilities in a manner that protects public health and safety.

Present contributions by Con Edison into the external fund match the amounts for decommissioning authorized by the New York Public Service Commission (NYPSC) as part of its electric rates. The NYPSC policy is to allow recovery of decommissioning costs at levels calculated to comply with NRC requirements.

On January 24, 1992 (P.S.C. Case No. 91-E-0462), the NYPSC approved a higher recovery rate of decommissioning costs for IP-1 and IP-2 in an "Agreement and Settlement Concerning Electric Rates of Consolidated Edison Co. of New York, Inc." The NYPSC-approved higher rate for recovery of decommissioning costs gives funds to Con Edison to cover the IP-1 decommissioning costs that are required by the Commission regulations in 10 CFR 50.75(c). From this agreement, the staff has determined that the rate of collection of decommissioning funds is based primarily on the rate treatment allowances set by NYPSC for the estimated decommissioning costs of the two nuclear units. The licensee has stated that all decommissioning funds for IP-1 will be collected by 2013, when the IP-2 operating license will expire. The IP-1 license is renewed until 2006 with this action (see Section 1.0). Since the licensee plans to dismantle IP-1 and IP-2 simultaneously some time after 2013, none of the decommissioning funds will be needed until after 2013, when dismantlement actions begin.

The July 9, 1992, final rule (10 CFR 50.82(a)) on decommissioning funding for prematurely shut down plants (57 FR 30383) states that the collection period for funds will be determined on a case-by-case basis, taking into account the specific financial situation of each licensee. In the supplementary information accompanying the publication of this rule, the Commission stated that licensees with bond ratings of A or better would be allowed to fund into the safe-storage period. The Commission also stated that its general policy would be to require all funds needed for decommissioning to be available before final dismantlement starts.

Con Edison currently has an AA bond rating and thus meets the NRC screening criterion for allowing accumulation of funds during the safe-storage period. The Con Edison funding plan also ensures that all funds will be available before final dismantlement starts. The staff concludes that the licensee's method for providing financial assurance is consistent with current Commission funding guidance and is, therefore, acceptable.

3. Final Dismantlement

As required by 10 CFR 50.82(d) and in the order approving the IP-1 Decommissioning Plan, Con Edison will submit a detailed dismantling plan for NRC review and approval prior to final IP-1 dismantling. Con Edison plans to start final dismantlement after permanent shut down of IP-2.

4.0 STATE CONSULTATION

In accordance with NRC regulations, the New York State official was notified of the proposed issuance of the order, license renewal, and amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This action involves an order that authorizes Con Edison to decommission IP-1 in accordance with its Decommissioning Plan and an amendment to License No. DPR-5. The amendment revises License No. DPR-5 to a possession-only status; renews the license to October 14, 2006, and amends the TS requirements to reflect the permanently shut down and defueled status of IP-1. Pursuant to 10 CFR 51.30, 51.32, 51.35, and 51.95, an Environmental Assessment has been prepared by the staff and a Notice of Issuance of Environmental Assessment and Finding of No Significant Impact has been published in the FEDERAL REGISTER (61FR3496, January 31, 1996). On the basis of the Environmental Assessment, the Commission concluded that the proposed order and amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by decommissioning IP-1 in the proposed manner, (2) such activities will be conducted in compliance with Commission regulations, and (3) the issuance of the order and amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: January 31, 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENVIRONMENTAL ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION REGARDING ORDER AUTHORIZING FACILITY DECOMMISSIONING AND AMENDMENT OF PROVISIONAL OPERATING LICENSE NO. DPR-5

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT UNIT NO. 1, DOCKET NO. 50-3

1.0 IDENTIFICATION OF PROPOSED ACTION AND SUMMARY

The Indian Point Unit No. 1 (IP-1) is a 615 MW thermal, pressurized water reactor (PWR) that was operated by Consolidated Edison Company of New York, Inc. (Con Edison), from March 26, 1962, to its final shutdown on October 31, 1974. IP-1 is situated between Indian Point Unit Nos. 2 and 3 (IP-2 and IP-3), two operating nuclear power plants. IP-2 and IP-3 are owned and operated by Con Edison and the Power Authority of the State of New York, respectively.

IP-1 was shut down because its emergency core cooling system did not meet the current regulatory requirements. Con Edison has proposed to decommission IP-1 and to renew and amend License No. DPR-5 as specified in its proposed Decommissioning Plan of October 17, 1980, as revised by letters dated October 13, 1981; July 31, 1986; March 28, 1988; August 10 1989; March 28 and July 17, 1990; February 5, April 2, July 31, September 20, and October 12, 1993; May 13 and August 11, 1994; and July 19, 1995. The Con Edison submittal of March 28, 1988, is a supplement to its June 1973 Environmental Report for IP-1.

A Notice of Consideration of Issuance of Amendment and Opportunity for Hearing was published in the FEDERAL REGISTER on December 31, 1985, (50 FR 53407) which stated that the NRC was considering: (1) approval of the IP-1 Decommissioning Plan; (2) amendment of License No. DPR-5 to a possession-only status; (3) revision of the Technical Specifications (TSs) to be consistent with SAFSTOR status; and (4) renewal of the IP-1 license to October 14, 2006 (consistent with the expiration of IP-2 license). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Notice stated a license renewal date of October 14, 2006 to coincide with the planned permanent shutdown of Indian Point Unit 2 (IP-2). Subsequent to the December 31, 1985 notice, the license for IP-2 was extended to

September 28, 2013 and Con Edison informed us, in a letter dated March 28, 1988, of its intention to delay dismantlement of IP-1 to after that date. The enclosed safety evaluation and this environmental assessment of the decommissioning plan are consistent with the 2013 date. However, we have renewed License No. DPR-5 to October 14, 2006, to be consistent with the license renewal application as noticed in the December 31, 1985 FEDERAL REGISTER Notice in order to put new TSs for the current shutdown condition in place.

All spent fuel assemblies have been removed from the reactor and placed in the west IP-1 spent fuel storage pool in the fuel handling building. The spent fuel will remain on site until a Federal repository becomes available.

On June 27, 1988, the NRC clarified environmental review requirements for decommissioning licensed nuclear facilities by amending the regulations in "General Requirements for Decommissioning Nuclear Facilities" (53 FR 24018). In so doing, the NRC eliminated the mandatory requirement for an Environmental Impact Statement (EIS) after considering decommissioning impacts in its Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities (FGEIS) (NUREG-0586). The FGEIS addresses the decommissioning of commercial nuclear facilities (both PWRs and boiling-water reactors (BWRs)) and applies to IP-1.

Environmental assessments (EAs) are required by 10 CFR 51.95(b) for power reactor decommissioning. According to the statement of considerations for the amendment to the decommissioning regulations (53 FR 24039), "if the impacts for a particular plant are significantly different from those studied generically because of site-specific considerations, the environmental assessment would discover those and lay the foundation for the preparation of an EIS." If the impacts for a particular plant are not significantly different or do not significantly affect the human environment, a Finding of No Significant Impact is prepared. The decommissioning of IP-1, a PWR, therefore requires this site-specific EA to determine whether or not the potential impacts are significantly different from those discussed in the FGEIS and whether or not any such differences have significant impact.

2.0 PURPOSE OF AND NEED FOR PROPOSED ACTION

The proposed action is needed to allow Con Edison's to retain IP-1 in a SAFSTOR status until after IP-2 is permanently shut down and to keep the spent fuel pools in operation until the spent fuel has been shipped to a Federal repository.

By allowing dismantlement to be delayed, granting of the proposed action would significantly reduce the gamma radiation exposure rate to workers involved in the final decontamination and dismantlement activities, would gain the safety and the environmental advantage of not interfering with IP-2 operations, and would gain the cost advantage of dismantling both units at the same time.

3.0 ALTERNATIVES INCLUDING PROPOSED ACTION

As discussed in the FGEIS, the three decommissioning alternatives for reactor facilities are DECON, ENTOMB, and SAFSTOR. Con Edison has chosen the SAFSTOR alternative for IP-1. These alternatives, as well as the no-action alternative, are discussed below.

3.1 DECON - Prompt Dismantlement

The DECON (decontamination) option involves promptly removing equipment, structures, and other portions of the facility containing radioactive contaminants or decontaminating them to a level that permits the facility to be released for unrestricted access. Con Edison has not selected the DECON option for IP-1. The safe operation of IP-2 depends on the continued operation of numerous systems in buildings that surround the IP-1 containment building. Many of these buildings and systems would have to be removed to decontaminate and dismantle IP-1. These buildings and systems cannot be removed without disabling support operations for IP-2. If such dismantling occurred, similar systems would have to be reconstructed to provide the required support for IP-2 operations resulting in increased costs for IP-2 operations.

3.2 ENTOMB - Entombment

The ENTOMB option involves encasing radioactive contaminants in a structurally long-lived material, such as concrete. The entombed structure must be appropriately maintained, and continued surveillance is required until the radioactivity is removed from the site or decays to a level that permits unrestricted use of the property. In the ENTOMB option, with the reactor internals and their long-lived activation products entombed, the security/access-control of the site could not be assured for the thousands of years necessary for radioactive decay, so this option is not considered viable (GEIS page 4-17).

Con Edison has not selected the ENTOMB option for IP-1. Con Edison has determined that even after 100 years of decay, sufficient radioactivity from these long-lived radionuclides will be present in reactor structures to preclude the release of IP-1 for unrestricted use. Furthermore, the congestion that would occur if entombment construction activities were initiated while IP-2 was operating could adversely impact the safe operation of IP-2. An entombment structure would make removal of that residual radioactivity more difficult during final dismantlement.

3.3 SAFSTOR - Deferred Dismantlement

In the SAFSTOR alternative, a facility is placed and maintained in a condition that allows it to be safely stored and subsequently decontaminated to levels that permit release of the property to unrestricted use. The facility may be left intact, except that all fuel must be removed from the reactor core and radioactive fluids and wastes must be removed from the site. Con Edison selected the SAFSTOR alternative because the IP-1 buildings and systems used

for IP-2 operations could continue to be used and there would be no increased cost to replace these needed buildings and systems. Furthermore, by delaying the dismantlement, the licensee may be able to reduce the radiation exposure to workers. Maintaining the facility in SAFSTOR will not cause significant environmental effects.

3.4 No Action

A radioactive facility is decommissioned in order to restore it to a condition that offers reasonable assurance that it will not adversely impact public health and safety and the site may be released for unrestricted access. Some action is required to ensure that the risk from a facility at the end of its life is within acceptable bounds. Thus, independent of the type of facility and its level of contamination, no action, implying that the licensee would simply abandon or leave the facility after ceasing operations, is not permitted by NRC regulations and, thus, is not a permissible alternative for any nuclear facility that is permanently shut down (FGEIS Section 2.4.1).

4.0 AFFECTED ENVIRONMENT

In a letter of February 11, 1992, the New York Power Authority (the IP-3 licensee) submitted 1990 population data based on the 1990 census that showed a population of 237,338 in a circle with a 10-mile radius around the Indian Point site. The 1990 data show the actual population to be 42 percent below the population projection in the IP-3 Final Environmental Statement (FES) of February 1975. As discussed below, SAFSTOR activities at IP-1 are very local and passive and will have no significant impact on that population.

5.0 ENVIRONMENTAL IMPACT OF THE PROPOSED ACTION

The staff has evaluated the proposed decommissioning, the renewal, and the amendment of the IP-1 license with respect to the environmental considerations in 10 CFR 51.45. Environmental considerations are discussed in the sections that follow. Tables 1 through 3 present the radionuclide inventories due to neutron activation of the reactor vessel, vessel internals, and bioshield for the years 1974 (reactor shutdown), 1988, and 2013. Table 4 presents the inventories of radionuclide contamination of the primary system calculated by Con Edison for the years 1987, 1988, and 2013. Table 5 presents calculated inventories of radionuclide contamination of auxiliary systems for the years 1982, 1988, and 2013.

The data on radionuclide inventories for activation and contamination show that cobalt-60 is the dominant gamma-emitting radionuclide. The 378,000 curies of cobalt-60 in the reactor vessel and its internals at reactor shutdown in 1974 will decrease to 2,390 curies in 2013. The data on primary system contamination (Table 4) show that the inventory of cobalt-60 will decrease from 198 curies in 1988 to 7 curies in 2013 and that cesium-137 (another gamma emitter) will decrease from 23 curies to 13 curies over the same period of time. Table 5 data on auxiliary systems contamination show that cobalt-60 will decrease from 29 curies in 1988 to 1.1 curies in 2013 and cesium-137 will decrease from 3.9 curies to 2.2 curies. Cesium-137 is the

dominant gamma-emitting fission product at IP-1, as it is at other light-water power reactors being decommissioned, and it appears as a contaminant with the cobalt-60 in systems and components (NUREG/CR-4289, "Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants," February 1986).

5.1 Status of Federal, State, and Local Environmental Permits

Federal, state and local environmental permits are established for the IP-1 and IP-2 site. IP-2, an operating plant, dominates permit issues. IP-1 has no active construction or other work that could significantly impact the environment or require a separate environmental permit.

5.2 Local Short-Term Uses Versus Long-Term Productivity

Many IP-1 systems and buildings are needed for continued power operations of IP-2, and Con Edison intends to continue to use them until IP-2 ceases to operate. The licensee has no plans for this site other than electrical power production during the SAFSTOR period of IP-1, and there is no advantage in making this area available earlier. Therefore, the local short-term use and the long-term productivity of the site are not in conflict.

5.3 Irreversible and Irretrievable Commitments of Resources

The proposed SAFSTOR period of IP-1 followed by dismantling would not involve any change in commitment of resources other than the IP-1 land area. This land area, however, is committed to the purpose of supporting power generation at the site regardless of the IP-1 status. The amount of land used for IP-1 will remain the same. The land could ultimately be returned to unrestricted use after decommissioning and decontamination or dismantling of all structures. Delay in dismantling the IP-1 facility may result in a reduced volume of radioactive waste to dispose of at the end of the SAFSTOR period because radioactivity decay would have gone on longer. Therefore, with less radioactive waste, the required burial space at a low-level waste burial site may be reduced (GEIS page 4-16).

5.4 Potential Exposure to Workers

The results of Con Edison calculations (Tables 1 through 5) show that delaying the final decontamination and dismantlement of IP-1 an additional 18 years to 2013 will reduce the gamma-radiation levels from the activated steel components by an order of magnitude from current, 1995 values. The reduction in radiation levels will significantly reduce radiation exposures to workers doing the final dismantlement of IP-1. Although Con Edison does not estimate the total exposures for the SAFSTOR alternative at IP-1, the reduced radionuclide inventory, the lower power level, and the smaller component size demonstrate that potential exposures during the SAFSTOR period and during final dismantling at IP-1 would be bounded by the 333 person-rem estimate in the FGEIS for a much larger plant. All work associated with radioactive components will be accomplished in accordance with radiation work permits that are issued for each activity.

5.5 Radioactive Waste

Although the IP-1 radwaste system will remain in operation during the IP-1 SAFSTOR period, more than 95 percent of the waste being processed will originate from the operating plant, IP-2. A small amount of radwaste will be produced in cleaning up the water in the spent fuel pools at IP-1 and decontaminating some areas for operational reasons. The radionuclide inventory at IP-1 is considerably less than it is at the standard PWR evaluated in NUREG 0586 because IP-1 operated at 615 MW thermal or less for 12 years, compared to 3436 MW thermal for 30 years assumed for the standard PWR. The integrated neutron flux at the standard plant would therefore be more than 12.5 times that of IP-1, as would the radionuclide inventory. The components and piping systems at the standard plant are also much larger than at IP-1. Therefore, the radioactive waste curies and waste volume for IP-1 during SAFSTOR and final DECON are bounded by the FGEIS.

5.6 Nonradiological Issues

NRC approval of the SAFSTOR Decommissioning Plan for IP-1 is not expected to result in any new work activities that would significantly change air quality, exhaust emissions, asbestos removal, chemical or biocide discharge, sanitary discharge, or noise. IP-1 will continue to be used to support IP-2 operations and will be maintained and monitored in accordance with the decommissioning plan and the TSs. The NRC approval will, therefore, have no significant impact on endangered species or the socioeconomics of the area. Since the impact from nonradiological issues will not significantly change, the SAFSTOR of IP-1 will fall within the bounds of the FGEIS.

5.7 Accident Analysis

Spent fuel accidents represent the controlling accidents during the SAFSTOR period. Con Edison conservatively estimated a worst case scenario as a bounding perspective. The scenario assumed a cladding failure of all of the fuel in the spent fuel pool and a release of the gap activity in accordance with the assumptions of Regulatory Guide 1.25. The total body dose (due to krypton-85) was determined to be 0.027 rads at the site boundary. This dose estimate is considerably less than protective action guidelines specified by the Environmental Protection Agency in EPA-520/1-75-001. This scenario is more conservative than a fuel handling accident that would involve failure of only a few spent fuel assemblies as a result of dropping a cask or a fuel assembly onto other assemblies.

In addition, TS 2.10.2.5 prohibits the movement of the spent fuel cask over the spent fuel pool. Calculations by the licensee and NRC staff ("NRC Order to Show Cause," February 11, 1980) show that the fission products in the spent fuel had decayed sufficiently so that air cooling of IP-1 fuel would be adequate in the event of a complete loss of water from a spent fuel pool.

Fission products have decayed about 15 years since the 1980 calculations. Consequently, the probability of a significant release of radioactivity from IP-1 spent fuel during the SAFSTOR period or any significant environmental impact from the spent fuel is acceptably small.

5.8 Radiological Environmental Monitoring

TS 4.1.6 requires that Radiological monitoring for IP-1 be conducted in accordance with IP-2 TS 4.11 and the Indian Point site Radiological Environmental Monitoring Program, which applies to all three units at the site. An annual Radiological Environmental Operating Report is submitted jointly by Con Edison and the New York Power Authority. Direct radiation, as well as airborne and waterborne pathways for radioactivity, are monitored.

Direct radiation is measured at 160 locations, 1 to 5 miles from the site, by means of thermoluminescent dosimeters, which cumulatively measure background radiation, plus any exposures from onsite operations. Air is monitored continuously at nine locations, 0.25 to 20 miles from the site, using fixed air particulate filters and in-line charcoal cartridges. Gamma spectroscopy is performed quarterly on air particulate filters. Cartridge samples are analyzed weekly for gross beta activity.

Water samples are collected continuously from the Hudson River. Both the upstream intake structure (control location) and the downstream discharge canal (indicator location) are located on site. Weekly river water samples are combined for monthly gamma spectroscopy analysis and quarterly for tritium analysis. Any accidental releases of radioactive liquids at the Indian Point site will flow to the Hudson River. The releases would not affect any nearby wells because of rock formations and closeness of the plant to the river (Draft Environmental Statement for IP-1, dated December 18, 1973, p. II-6 and V-15). Monthly drinking water samples are taken from the Camp Field Reservoir (3.5 miles NE) and are analyzed for gamma-emitting radionuclides and quarterly for tritium. Vegetation, precipitation, and soil from the site, and fish, aquatic vegetation, and bottom sediment from the Hudson River are also sampled and analyzed for radioactivity.

5.9 Unavoidable Impacts

During the SAFSTOR period, IP-1 will continue to occupy a restricted area located between two operating nuclear generating stations, IP-2 and IP-3. However, IP-1 facility systems continue to be needed for IP-2 radioactive liquid-waste processing, solid waste volume reduction, water treatment, an alternate safe-shutdown system, site emergency facilities and office space. Therefore, based on these considerations and our review above of other environmental issues of the SAFSTOR option selected by Con Edison, we have determined that there is no significant unavoidable impact in retaining IP-1 in SAFSTOR.

6.0 FINAL DISMANTLEMENT

As required by 10 CFR 50.82(d) Con Edison will submit a detailed dismantling plan for NRC review and approval prior to final IP-1 dismantling. Con Edison plans to start final dismantlement after permanent shut down of IP-2.

7.0 AGENCIES AND PERSONS CONSULTED

The NRC staff reviewed the Con Edison request and consulted with an official of the State of New York. The State official had no comments.

8.0 FINDING OF NO SIGNIFICANT IMPACT

On the basis of the foregoing environmental assessment, the Commission has concluded that the environmental impacts of decommissioning IP-1 are adequately bounded by the environmental impacts of decommissioning light-water reactors as analyzed in the FGEIS and that the proposed action will not have a significant impact on the quality of the human environment. Accordingly, the Commission will not prepare an environmental impact statement for this proposed action.

Attachment: Tables 1 through 5

Date: January 25, 1996

TABLE 1

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME=REACTOR SHUTDOWN

COMPONENT	RADIAL FLUX REDUCTION	AXIAL FLUX REDUCTION	SPECTRAL FLUX FACTOR	TOTAL FLUX REDUCTION	VOLUME (CM**3)	MATERIAL	H-3 (CURIES)	C-14 (CURIES)	CL-36 (CURIES)
SHIM RODS CORE A ONLY	1.00E+00	1.00E+00	1.00E+00	1.00E+00	5.11E+04	STAINLESS STEEL	7.46E-08	1.02E+00	1.92E-02
CRB CORES A&B	1.00E+00	1.00E+00	1.00E+00	1.00E+00	2.09E+05	STAINLESS STEEL	3.01E-05	1.93E+01	3.43E-01
CRB FOLLOWERS CORES A&B	1.00E+00	1.00E+00	1.00E+00	1.00E+00	2.09E+05	ZIRCALOY-2	1.93E-03	2.82E+00	3.43E-06
FILLER RODS CORE A ONLY	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.57E+05	ZIRCALOY-2	4.36E-05	4.60E-01	1.05E-07
FILLER RODS CORES A&B	5.00E-01	1.00E+00	1.00E+00	5.00E-01	1.66E+05	ZIRCALOY-2	7.65E-04	1.12E+00	1.36E-06
HOLDOWN COLUMNS CORES A&B	2.40E-02	1.00E+00	1.00E+00	2.40E-02	2.50E+06	STAINLESS STEEL	8.64E-06	5.54E+00	9.84E-02
FILLER BARS CORES A&B	2.40E-02	1.00E+00	1.00E+00	2.40E-02	5.07E+04	STAINLESS STEEL	1.75E-07	1.12E-01	2.00E-03
CORE SHROUD	6.30E-03	1.00E+00	5.00E+00	3.15E-02	6.68E+05	STAINLESS STEEL	3.03E-06	1.94E+00	3.45E-02
UPPER INTERNALS	1.00E+00	1.00E-02	5.00E+00	5.00E-02	9.30E+05	STAINLESS STEEL	6.70E-06	4.30E+00	7.63E-02
LOWER INTERNALS	1.00E+00	1.00E-02	5.00E+00	5.00E-02	9.30E+05	STAINLESS STEEL	6.70E-06	4.30E+00	7.63E-02
THERM. SHLD. #1	1.38E-03	1.00E+00	5.00E+00	6.90E-03	1.12E+06	STAINLESS STEEL	1.11E-06	7.14E-01	1.27E-02
THERM. SHLD. #2	5.05E-04	1.00E+00	5.00E+00	2.53E-03	1.88E+06	STAINLESS STEEL	6.84E-07	4.39E-01	7.79E-03
RX VESSEL CLAD	2.15E-04	1.00E+00	5.00E+00	1.08E-03	7.54E+04	STAINLESS STEEL	1.17E-08	7.49E-03	1.33E-04
RX VESSEL CORE HEIGHT	4.50E-05	1.00E+00	1.00E+00	4.50E-05	4.51E+06	CARBON STEEL	2.19E-08	3.35E-03	1.91E-04
RX VESSEL 3' BELOW CORE	4.50E-05	2.00E-01	1.00E+00	9.00E-06	1.60E+06	CARBON STEEL	1.56E-09	2.38E-04	1.35E-05
RX VESSEL >3' BELOW CORE	4.50E-05	2.00E-01	1.00E+00	9.00E-06	6.94E+06	CARBON STEEL	6.75E-09	1.03E-03	5.87E-05
RX VESSEL 3' ABOVE CORE	4.50E-05	1.00E-01	1.00E+00	4.50E-06	1.60E+06	CARBON STEEL	7.78E-10	1.19E-04	6.77E-06
RX VESSEL >3' ABOVE CORE	4.50E-05	1.00E-01	1.00E+00	4.50E-06	4.36E+06	CARBON STEEL	2.12E-09	3.24E-04	1.84E-05
INSULATION	9.65E-06	1.00E+00	1.00E+00	9.65E-06	2.10E+06	ALUMINUM	5.29E-10	6.02E-21	0.00E+00
NST I.D. WALL	9.00E-06	1.00E+00	5.00E+00	4.50E-05	7.85E+05	CARBON STEEL	3.82E-09	5.83E-04	3.32E-05
NST O.D. WALL	2.68E-13	1.00E+00	1.00E+01	2.68E-12	1.34E+06	CARBON STEEL	3.88E-16	5.93E-11	3.38E-12
BIOSHIELD	2.68E-05	1.00E+00	1.00E+01	2.68E-04	1.00E+08	CONCRETE	7.75E-06	1.99E-09	1.26E-10
						TOTALS	2.80E-03	4.21E+01	6.70E-01

TABLE 1
(continued)

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME=REACTOR SHUTDOWN

COMPONENT	AR-39 (CURIES)	CA-41 (CURIES)	MN-54 (CURIES)	FE-55 (CURIES)	CO-60 (CURIES)	NI-59 (CURIES)	NI-63 (CURIES)	ZN-65 (CURIES)	NB-94 (CURIES)	MO-93 (CURIES)
SHIM RODS CORE A ONLY	0.00E+00	0.00E+00	5.47E+03	3.93E+04	1.75E+04	6.85E+00	8.64E+02	0.00E+00	1.95E-02	1.97E-02
CRB CORES A&B	0.00E+00	0.00E+00	3.26E+04	3.26E+05	1.89E+05	1.11E+02	1.52E+04	0.00E+00	3.47E-01	3.72E-01
CRB FOLLOWERS CORES A&B	0.00E+00	0.00E+00	5.68E+01	5.77E+02	1.10E+03	4.56E-01	6.23E+01	2.22E-01	3.78E-01	0.00E+00
FILLER RODS CORE A ONLY	0.00E+00	0.00E+00	2.92E+01	2.14E+02	3.12E+02	8.64E-02	1.09E+01	1.79E-02	6.53E-02	0.00E+00
FILLER RODS CORES A&B	0.00E+00	0.00E+00	2.26E+01	2.29E+02	4.35E+02	1.81E-01	2.47E+01	8.80E-02	1.50E-01	0.00E+00
HOLDOWN COLUMNS CORES A&B	0.00E+00	0.00E+00	9.36E+03	9.36E+04	5.42E+04	3.19E+01	4.35E+03	0.00E+00	9.96E-02	1.07E-01
FILLER BARS CORES A&B	0.00E+00	0.00E+00	1.90E+02	1.90E+03	1.10E+03	6.46E-01	8.82E+01	0.00E+00	2.02E-03	2.17E-03
CORE SHROUD	0.00E+00	0.00E+00	3.28E+03	3.28E+04	1.90E+04	1.12E+01	1.53E+03	0.00E+00	3.49E-02	3.75E-02
UPPER INTERNALS	0.00E+00	0.00E+00	7.25E+03	7.25E+04	4.20E+04	2.47E+01	3.37E+03	0.00E+00	7.72E-02	8.28E-02
LOWER INTERNALS	0.00E+00	0.00E+00	7.25E+03	7.25E+04	4.20E+04	2.47E+01	3.37E+03	0.00E+00	7.72E-02	8.28E-02
THERM. SHLD. #1	0.00E+00	0.00E+00	1.21E+03	1.21E+04	6.99E+03	4.10E+00	5.60E+02	0.00E+00	1.28E-02	1.38E-02
THERM. SHLD. #2	0.00E+00	0.00E+00	7.41E+02	7.41E+03	4.29E+03	2.52E+00	3.44E+02	0.00E+00	7.88E-03	8.45E-03
RX VESSEL CLAD	0.00E+00	0.00E+00	1.26E+01	1.26E+02	7.33E+01	4.30E-02	5.88E+00	0.00E+00	1.35E-04	1.44E-04
RX VESSEL CORE HEIGHT	0.00E+00	0.00E+00	4.40E+01	4.36E+02	1.66E+01	7.14E-03	9.80E-01	0.00E+00	7.49E-05	0.00E+00
RX VESSEL 3' BELOW CORE	0.00E+00	0.00E+00	3.12E+00	3.10E+01	1.18E+00	5.07E-04	6.96E-02	0.00E+00	5.31E-06	0.00E+00
RX VESSEL >3' BELOW CORE	0.00E+00	0.00E+00	1.36E+01	1.34E+02	5.10E+00	2.20E-03	3.02E-01	0.00E+00	2.30E-05	0.00E+00
RX VESSEL 3' ABOVE CORE	0.00E+00	0.00E+00	1.56E+00	1.55E+01	5.88E-01	2.53E-04	3.48E-02	0.00E+00	2.66E-06	0.00E+00
RX VESSEL >3' ABOVE CORE	0.00E+00	0.00E+00	4.26E+00	4.22E+01	1.60E+00	6.91E-04	9.48E-02	0.00E+00	7.24E-06	0.00E+00
INSULATION	0.00E+00	0.00E+00	1.38E-02	1.37E-01	5.94E-04	0.00E+00	3.83E-01	5.69E-02	0.00E+00	0.00E+00
NST I.D. WALL	0.00E+00	0.00E+00	7.67E+00	7.59E+01	2.89E+00	1.24E-03	1.71E-01	0.00E+00	1.30E-05	0.00E+00
NST O.D. WALL	0.00E+00	0.00E+00	7.79E-07	7.72E-06	2.93E-07	1.26E-10	1.73E-08	0.00E+00	1.33E-12	0.00E+00
BIOSHIELD	1.12E-07	1.48E-08	6.97E-07	6.89E-06	1.24E-10	2.08E-11	2.53E-09	0.00E+00	0.00E+00	0.00E+00
	1.12E-07	1.48E-08	6.76E+04	6.60E+05	3.78E+05	2.18E+02	2.97E+04	3.84E-01	1.27E+00	7.26E-01

TABLE 1
(continued)

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME=REACTOR SHUTDOWN

COMPONENT	TC-99 (CURIES)	SB-125 (CURIES)	I-129 (CURIES)	BA-133 (CURIES)	SM-151 (CURIES)	EU-152 (CURIES)	EU-153 (CURIES)	EU-154 (CURIES)	TOTAL CURIES
SHIM RODS CORE A ONLY	8.84E-04	2.27E-02	2.10E-17	2.58E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.31E+04
CRB CORES A&B	1.50E-02	2.36E-01	6.98E-15	3.72E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.63E+05
CRB FOLLOWERS CORES A&B	1.31E-05	0.00E+00	3.39E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.80E+03
FILLER RODS CORE A ONLY	1.07E-07	0.00E+00	1.16E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.67E+02
FILLER RODS CORES A&B	5.21E-06	0.00E+00	1.34E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.13E+02
HOLDOWN COLUMNS CORES A&B	4.30E-03	6.78E-02	2.00E-15	1.07E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.62E+05
FILLER BARS CORES A&B	8.71E-05	1.37E-03	4.06E-17	2.17E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.28E+03
CORE SHROUD	1.51E-03	2.38E-02	7.03E-16	3.75E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.67E+04
UPPER INTERNALS	3.33E-03	5.25E-02	1.55E-15	8.28E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.25E+05
LOWER INTERNALS	3.33E-03	5.25E-02	1.55E-15	8.28E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.25E+05
THERM. SHLD. #1	5.53E-04	8.73E-03	2.58E-16	1.38E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.08E+04
THERM. SHLD. #2	3.40E-04	5.36E-03	1.59E-16	8.45E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.28E+04
RX VESSEL CLAD	5.80E-06	9.16E-05	2.71E-18	1.44E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.18E+02
RX VESSEL CORE HEIGHT	8.58E-17	2.29E-04	3.71E-18	1.96E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.98E+02
RX VESSEL 3' BELOW CORE	6.09E-18	1.63E-05	2.64E-19	1.39E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.53E+01
RX VESSEL >3' BELOW CORE	2.64E-17	7.06E-05	1.14E-18	6.03E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.53E+02
RX VESSEL 3' ABOVE CORE	3.05E-18	8.14E-06	1.32E-19	6.96E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.77E+01
RX VESSEL >3' ABOVE CORE	8.30E-18	2.22E-05	3.59E-19	1.90E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.81E+01
INSULATION	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.92E-01
NST I.D. WALL	1.49E-17	3.99E-05	6.46E-19	3.41E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.67E+01
NST O.D. WALL	1.52E-24	4.06E-12	6.57E-26	3.47E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.81E-06
BIOSHIELD	0.00E+00	0.00E+00	0.00E+00	2.76E-09	1.17E-09	5.17E-07	7.34E-08	3.19E-09	1.61E-05
	2.93E-02	4.71E-01	4.86E-13	7.32E+00	1.17E-09	5.17E-07	7.34E-08	3.19E-09	1.14E+06

TABLE 2

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME= 1988

COMPONENT	RADIAL FLUX REDUCTION	AXIAL FLUX REDUCTION	SPECTRAL FLUX FACTOR	TOTAL FLUX REDUCTION	VOLUME (CM**3)	MATERIAL	H-3 (CURIES)	C-14 (CURIES)	CL-36 (CURIES)
SHIM RODS CORE A ONLY	1.00E+00	1.00E+00	1.00E+00	1.00E+00	5.11E+04	STAINLESS STEEL	1.05E-08	1.02E+00	1.92E-02
CRB CORES A&B	1.00E+00	1.00E+00	1.00E+00	1.00E+00	2.09E+05	STAINLESS STEEL	1.44E-05	1.93E+01	3.43E-01
CRB FOLLOWERS CORES A&B	1.00E+00	1.00E+00	1.00E+00	1.00E+00	2.09E+05	ZIRCALOY-2	9.22E-04	2.82E+00	3.43E-06
FILLER RODS CORE A ONLY	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.57E+05	ZIRCALOY-2	3.88E-06	4.57E-01	1.05E-07
FILLER RODS CORES A&B	5.00E-01	1.00E+00	1.00E+00	5.00E-01	1.66E+05	ZIRCALOY-2	3.66E-04	1.12E+00	1.36E-06
HOLDOWN COLUMNS CORES A&B	2.40E-02	1.00E+00	1.00E+00	2.40E-02	2.50E+06	STAINLESS STEEL	4.13E-06	5.53E+00	9.84E-02
FILLER BARS CORES A&B	2.40E-02	1.00E+00	1.00E+00	2.40E-02	5.01E+04	STAINLESS STEEL	8.27E-08	1.11E-01	1.97E-03
CORE SHROUD	6.30E-03	1.00E+00	5.00E+00	3.15E-02	6.68E+05	STAINLESS STEEL	1.45E-06	1.94E+00	3.45E-02
UPPER INTERNALS	1.00E+00	1.00E-02	5.00E+00	5.00E-02	9.30E+05	STAINLESS STEEL	3.20E-06	4.29E+00	7.63E-02
LOWER INTERNALS	1.00E+00	1.00E-02	5.00E+00	5.00E-02	9.30E+05	STAINLESS STEEL	3.20E-06	4.29E+00	7.63E-02
THERM. SHLD. #1	1.38E-03	1.00E+00	5.00E+00	6.90E-03	1.12E+06	STAINLESS STEEL	5.32E-07	7.13E-01	1.27E-02
THERM. SHLD. #2	5.05E-04	1.00E+00	5.00E+00	2.53E-03	1.88E+06	STAINLESS STEEL	3.27E-07	4.38E-01	7.79E-03
RX VESSEL CLAD	2.15E-04	1.00E+00	5.00E+00	1.08E-03	7.54E+04	STAINLESS STEEL	5.58E-09	7.47E-03	1.33E-04
RX VESSEL CORE HEIGHT	4.50E-05	1.00E+00	1.00E+00	4.50E-05	4.51E+06	CARBON STEEL	1.05E-08	3.35E-03	1.91E-04
RX VESSEL 3' BELOW CORE	4.50E-05	2.00E-01	1.00E+00	9.00E-06	1.60E+06	CARBON STEEL	7.42E-10	2.38E-04	1.35E-05
RX VESSEL >3' BELOW CORE	4.50E-05	2.00E-01	1.00E+00	9.00E-06	6.94E+06	CARBON STEEL	3.22E-09	1.03E-03	5.87E-05
RX VESSEL 3' ABOVE CORE	4.50E-05	1.00E-01	1.00E+00	4.50E-06	1.60E+06	CARBON STEEL	3.71E-10	1.19E-04	6.77E-06
RX VESSEL >3' ABOVE CORE	4.50E-05	1.00E-01	1.00E+00	4.50E-06	4.36E+06	CARBON STEEL	1.01E-09	3.24E-04	1.84E-05
INSULATION	9.65E-06	1.00E+00	1.00E+00	9.65E-06	2.10E+06	ALUMINUM	2.51E-10	6.02E-21	0.00E+00
NST I.D. WALL	9.00E-06	1.00E+00	5.00E+00	4.50E-05	7.85E+05	CARBON STEEL	1.82E-09	5.83E-04	3.32E-05
NST O.D. WALL	2.68E-13	1.00E+00	1.00E+01	2.68E-12	1.34E+06	CARBON STEEL	1.85E-16	5.93E-11	3.38E-12
BIOSHIELD	2.68E-05	1.00E+00	1.00E+01	2.68E-04	1.00E+08	CONCRETE	3.70E-06	1.98E-09	1.26E-10
						TOTALS	1.32E-03	4.20E+01	6.70E-01

TABLE 2
(continued)

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME= 1988

COMPONENT	AR-39 (CURIES)	CA-41 (CURIES)	MN-54 (CURIES)	FE-55 (CURIES)	CO-60 (CURIES)	NI-59 (CURIES)	NI-63 (CURIES)	ZN-65 (CURIES)	NB-94 (CURIES)	MO-93 (CURIES)
SHIM RODS CORE A ONLY	0.00E+00	0.00E+00	2.80E-09	3.55E+00	1.77E+02	6.85E+00	6.64E+02	0.00E+00	1.94E-02	1.95E-02
CRB CORES A&B	0.00E+00	0.00E+00	7.59E-01	9.74E+03	3.34E+04	1.11E+02	1.37E+04	0.00E+00	3.47E-01	3.70E-02
CRB FOLLOWERS CORES A&B	0.00E+00	0.00E+00	1.32E-03	1.72E+01	1.94E+02	4.56E-01	5.64E+01	2.55E-07	3.78E-01	0.00E+00
FILLER RODS CORE A ONLY	0.00E+00	0.00E+00	1.93E-14	2.15E-03	1.08E+00	8.64E-02	7.87E+00	6.31E-22	6.53E-02	0.00E+00
FILLER RODS CORES A&B	0.00E+00	0.00E+00	5.25E-04	6.84E+00	7.69E+01	1.81E-01	2.24E+01	1.01E-07	1.50E-01	0.00E+00
HOLDOWN COLUMNS CORES A&B	0.00E+00	0.00E+00	2.18E-01	2.80E+03	9.60E+03	3.19E+01	3.94E+03	0.00E+00	9.96E-02	1.06E-02
FILLER BARS CORES A&B	0.00E+00	0.00E+00	4.36E-03	5.60E+01	1.92E+02	6.38E-01	7.90E+01	0.00E+00	2.00E-03	2.13E-04
CORE SHROUD	0.00E+00	0.00E+00	7.64E-02	9.81E+02	3.37E+03	1.12E+01	1.38E+03	0.00E+00	3.49E-02	3.72E-03
UPPER INTERNALS	0.00E+00	0.00E+00	1.69E-01	2.17E+03	7.44E+03	2.47E+01	3.06E+03	0.00E+00	7.72E-02	8.23E-03
LOWER INTERNALS	0.00E+00	0.00E+00	1.69E-01	2.17E+03	7.44E+03	2.47E+01	3.06E+03	0.00E+00	7.72E-02	8.23E-03
THERM. SHLD. #1	0.00E+00	0.00E+00	2.81E-02	3.60E+02	1.24E+03	4.10E+00	5.08E+02	0.00E+00	1.28E-02	1.37E-03
THERM. SHLD. #2	0.00E+00	0.00E+00	1.72E-02	2.21E+02	7.60E+02	2.52E+00	3.12E+02	0.00E+00	7.88E-03	8.40E-04
RX VESSEL CLAD	0.00E+00	0.00E+00	2.94E-04	3.78E+00	1.30E+01	4.30E-02	5.33E+00	0.00E+00	1.35E-04	1.43E-05
RX VESSEL CORE HEIGHT	0.00E+00	0.00E+00	1.02E-03	1.30E+01	2.94E+00	7.14E-03	8.89E-01	0.00E+00	7.49E-05	0.00E+00
RX VESSEL 3' BELOW CORE	0.00E+00	0.00E+00	7.27E-05	9.26E-01	2.09E-01	5.07E-04	6.31E-02	0.00E+00	5.31E-06	0.00E+00
RX VESSEL >3' BELOW CORE	0.00E+00	0.00E+00	3.15E-04	4.02E+00	9.06E-01	2.20E-03	2.74E-01	0.00E+00	2.30E-05	0.00E+00
RX VESSEL 3' ABOVE CORE	0.00E+00	0.00E+00	3.64E-05	4.63E-01	1.04E-01	2.53E-04	3.15E-02	0.00E+00	2.66E-06	0.00E+00
RX VESSEL >3' ABOVE CORE	0.00E+00	0.00E+00	9.91E-05	1.26E+00	2.84E-01	6.91E-04	8.59E-02	0.00E+00	7.24E-06	0.00E+00
INSULATION	0.00E+00	0.00E+00	3.22E-07	4.11E-03	1.05E-04	0.00E+00	3.47E-04	6.59E-08	0.00E+00	0.00E+00
NST I.D. WALL	0.00E+00	0.00E+00	1.78E-04	2.27E+00	5.12E-01	1.24E-03	1.55E-01	0.00E+00	1.30E-05	0.00E+00
NST O.D. WALL	0.00E+00	0.00E+00	1.81E-11	2.31E-07	5.21E-08	1.26E-10	1.57E-08	0.00E+00	1.33E-12	0.00E+00
BIOSHIELD	1.08E-07	1.48E-08	1.62E-11	2.06E-07	2.20E-11	2.08E-11	2.29E-09	0.00E+00	0.00E+00	0.00E+00
	1.08E-07	1.48E-08	1.44E+00	1.85E+04	6.39E+04	2.18E+02	2.68E+04	4.22E-07	1.27E+00	8.98E-02

TABLE 2
(continued)

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME= 1988

COMPONENT	TC-99 (CURIES)	SB-125 (CURIES)	I-129 (CURIES)	BA-133 (CURIES)	SM-151 (CURIES)	EU-152 (CURIES)	EU-153 (CURIES)	EU-154 (CURIES)	TOTAL CURIES
SHIM RODS CORE A ONLY	8.84E-04	3.63E-06	2.10E-17	2.70E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.53E+02
CRB CORES A&B	1.50E-02	8.74E-03	6.98E-15	1.59E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.70E+04
CRB FOLLOWERS CORES A&B	1.32E-05	0.00E+00	3.39E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.71E+02
FILLER RODS CORE A ONLY	1.07E-07	0.00E+00	1.16E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.55E+00
FILLER RODS CORES A&B	5.23E-06	0.00E+00	1.34E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.08E+02
HOLDOWN COLUMNS CORES A&B	4.30E-03	2.51E-03	2.00E-15	4.57E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.64E+04
FILLER BARS CORES A&B	8.62E-05	5.03E-05	4.02E-17	9.15E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.28E+02
CORE SHROUD	1.51E-03	8.80E-04	7.03E-16	1.60E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.74E+03
UPPER INTERNALS	3.33E-03	1.94E-03	1.55E-15	3.54E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.27E+04
LOWER INTERNALS	3.33E-03	1.94E-03	1.55E-15	3.54E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.27E+04
THERM. SHLD. #1	5.54E-04	3.23E-04	2.58E-16	5.88E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.11E+03
THERM. SHLD. #2	3.40E-04	1.98E-04	1.59E-16	3.61E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.30E+03
RX VESSEL CLAD	5.81E-06	3.39E-06	2.71E-18	6.17E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.21E+01
RX VESSEL CORE HEIGHT	8.65E-17	8.50E-06	3.71E-18	8.38E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.69E+01
RX VESSEL 3' BELOW CORE	6.13E-18	6.03E-07	2.64E-19	5.95E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.20E+00
RX VESSEL >3' BELOW CORE	2.66E-17	2.62E-06	1.14E-18	2.58E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.20E+00
RX VESSEL 3' ABOVE CORE	3.07E-18	3.02E-07	1.32E-19	2.97E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.99E-01
RX VESSEL >3' ABOVE CORE	8.36E-18	8.22E-07	3.59E-19	8.10E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.63E+00
INSULATION	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.57E-03
NST I.D. WALL	1.50E-17	1.48E-06	6.46E-19	1.46E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.94E+00
NST O.D. WALL	1.53E-24	1.50E-13	6.57E-26	1.48E-11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.99E-07
BIOSHIELD	0.00E+00	0.00E+00	0.00E+00	1.18E-09	1.06E-09	2.64E-07	2.54E-08	5.09E-10	4.32E-06
	2.94E-02	1.66E-02	4.86E-13	3.05E+00	1.06E-09	2.64E-07	2.54E-08	5.09E-10	1.10E+05

TABLE 3

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME= 2013

COMPONENT	RADIAL FLUX REDUCTION	AXIAL FLUX REDUCTION	SPECTRAL FLUX FACTOR	TOTAL FLUX REDUCTION	VOLUME (CM**3)	MATERIAL	H-3 (CURIES)	C-14 (CURIES)	CL-36 (CURIES)
SHIM RODS CORE A ONLY	1.00E+00	1.00E+00	1.00E+00	1.00E+00	5.11E+04	STAINLESS STEEL	2.59E-09	1.02E+00	1.92E-02
CRB CORES A&B	1.00E+00	1.00E+00	1.00E+00	1.00E+00	2.09E+05	STAINLESS STEEL	3.53E-06	1.92E+01	3.43E-01
CRB FOLLOWERS CORES A&B	1.00E+00	1.00E+00	1.00E+00	1.00E+00	2.09E+05	ZIRCALOY-2	2.26E-04	2.80E+00	3.43E-06
FILLER RODS CORE A ONLY	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.57E+05	ZIRCALOY-2	1.51E-06	4.55E-01	1.05E-07
FILLER RODS CORES A&B	5.00E-01	1.00E+00	1.00E+00	5.00E-01	1.66E+05	ZIRCALOY-2	8.96E-05	1.11E+00	1.36E-06
HOLDOWN COLUMNS CORES A&B	2.40E-02	1.00E+00	1.00E+00	2.40E-02	2.50E+06	STAINLESS STEEL	1.01E-06	5.52E+00	9.84E-02
FILLER BARS CORES A&B	2.40E-02	1.00E+00	1.00E+00	2.40E-02	5.07E+04	STAINLESS STEEL	2.06E-08	1.12E-01	2.00E-03
CORE SHROUD	6.30E-03	1.00E+00	5.00E+00	3.15E-02	6.68E+05	STAINLESS STEEL	3.56E-07	1.94E+00	3.45E-02
UPPER INTERNALS	1.00E+00	1.00E-02	5.00E+00	5.00E-02	9.30E+05	STAINLESS STEEL	7.86E-07	4.28E+00	7.63E-02
LOWER INTERNALS	1.00E+00	1.00E-02	5.00E+00	5.00E-02	9.30E+05	STAINLESS STEEL	7.86E-07	4.28E+00	7.63E-02
THERM. SHLD. #1	1.38E-03	1.00E+00	5.00E+00	6.90E-03	1.12E+06	STAINLESS STEEL	1.31E-07	7.11E-01	1.27E-02
THERM. SHLD. #2	5.05E-04	1.00E+00	5.00E+00	2.53E-03	1.88E+06	STAINLESS STEEL	8.02E-08	4.37E-01	7.79E-03
RX VESSEL CLAD	2.15E-04	1.00E+00	5.00E+00	1.08E-03	7.54E+04	STAINLESS STEEL	1.37E-09	7.46E-03	1.33E-04
RX VESSEL CORE HEIGHT	4.50E-05	1.00E+00	1.00E+00	4.50E-05	4.51E+06	CARBON STEEL	2.58E-09	3.33E-03	1.91E-04
RX VESSEL 3' BELOW CORE	4.50E-05	2.00E-01	1.00E+00	9.00E-06	1.60E+06	CARBON STEEL	1.83E-10	2.36E-04	1.35E-05
RX VESSEL >3' BELOW CORE	4.50E-05	2.00E-01	1.00E+00	9.00E-06	6.94E+06	CARBON STEEL	7.93E-10	1.02E-03	5.87E-05
RX VESSEL 3' ABOVE CORE	4.50E-05	1.00E-01	1.00E+00	4.50E-06	1.60E+06	CARBON STEEL	9.14E-11	1.18E-04	6.77E-06
RX VESSEL >3' ABOVE CORE	4.50E-05	1.00E-01	1.00E+00	4.50E-06	4.36E+06	CARBON STEEL	2.49E-10	3.22E-04	1.84E-05
INSULATION	9.65E-06	1.00E+00	1.00E+00	9.65E-06	2.10E+06	ALUMINUM	6.20E-11	6.02E-21	0.00E+00
NST I.D. WALL	9.00E-06	1.00E+00	5.00E+00	4.50E-05	7.85E+05	CARBON STEEL	4.49E-10	5.79E-04	3.32E-05
NST O.D. WALL	2.68E-13	1.00E+00	1.00E+01	2.68E-12	1.34E+06	CARBON STEEL	4.56E-17	5.89E-11	3.38E-12
BIOSHIELD	2.68E-05	1.00E+00	1.00E+01	2.68E-04	1.00E+08	CONCRETE	9.09E-07	1.98E-09	1.26E-10
						TOTALS	3.24E-04	4.19E+01	6.70E-01

TABLE 3
(continued)

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME= 2013

COMPONENT	AR-39 (CURIES)	CA-41 (CURIES)	MN-54 (CURIES)	FE-55 (CURIES)	CO-60 (CURIES)	NI-59 (CURIES)	NI-63 (CURIES)	ZN-65 (CURIES)	NB-94 (CURIES)	MO-93 (CURIES)
SHIM RODS CORE A ONLY	0.00E+00	0.00E+00	1.30E-15	4.52E-03	6.59E+00	6.85E+00	5.47E+02	0.00E+00	1.94E-02	1.94E-02
CRB CORES A&B	0.00E+00	0.00E+00	1.21E-09	1.24E+01	1.25E+03	1.11E+02	1.14E+04	0.00E+00	3.47E-01	3.70E-01
CRB FOLLOWERS CORES A&B	0.00E+00	0.00E+00	2.11E-12	2.19E-02	7.23E+00	4.56E-01	4.68E+01	1.37E-18	3.78E-01	0.00E+00
FILLER RODS CORE A ONLY	0.00E+00	0.00E+00	6.97E-18	2.45E-05	1.18E-01	8.64E-02	6.94E+00	1.70E-29	6.53E-02	0.00E+00
FILLER RODS CORES A&B	0.00E+00	0.00E+00	8.38E-13	8.72E-03	2.87E+00	1.81E-01	1.86E+01	5.43E-19	1.50E-01	0.00E+00
HOLDOWN COLUMNS CORES A&B	0.00E+00	0.00E+00	3.47E-10	3.56E+00	3.58E+02	3.19E+01	3.26E+03	0.00E+00	9.96E-02	1.06E-01
FILLER BARS CORES A&B	0.00E+00	0.00E+00	7.03E-12	7.23E-02	7.26E+00	6.46E-01	6.62E+01	0.00E+00	2.02E-03	2.15E-03
CORE SHROUD	0.00E+00	0.00E+00	1.22E-10	1.25E+00	1.26E+02	1.12E+01	1.14E+03	0.00E+00	3.49E-02	3.72E-02
UPPER INTERNALS	0.00E+00	0.00E+00	2.69E-10	2.76E+00	2.78E+02	2.47E+01	2.53E+03	0.00E+00	7.72E-02	8.23E-02
LOWER INTERNALS	0.00E+00	0.00E+00	2.69E-10	2.76E+00	2.78E+02	2.47E+01	2.53E+03	0.00E+00	7.72E-02	8.23E-02
THERM. SHLD. #1	0.00E+00	0.00E+00	4.47E-11	4.59E-01	4.61E+01	4.10E+00	4.20E+02	0.00E+00	1.28E-02	1.37E-02
THERM. SHLD. #2	0.00E+00	0.00E+00	2.74E-11	2.82E-01	2.83E+01	2.52E+00	2.58E+02	0.00E+00	7.88E-03	8.40E-03
RX VESSEL CLAD	0.00E+00	0.00E+00	4.68E-13	4.81E-03	4.84E-01	4.30E-02	4.41E+00	0.00E+00	1.35E-04	1.43E-04
RX VESSEL CORE HEIGHT	0.00E+00	0.00E+00	1.64E-12	1.66E-02	1.09E-01	7.14E-03	7.37E-01	0.00E+00	7.49E-05	0.00E+00
RX VESSEL 3' BELOW CORE	0.00E+00	0.00E+00	1.16E-13	1.18E-03	7.76E-03	5.07E-04	5.23E-02	0.00E+00	5.31E-06	0.00E+00
RX VESSEL >3' BELOW CORE	0.00E+00	0.00E+00	5.03E-13	5.12E-03	3.37E-02	2.20E-03	2.27E-01	0.00E+00	2.30E-05	0.00E+00
RX VESSEL 3' ABOVE CORE	0.00E+00	0.00E+00	5.80E-14	5.90E-04	3.88E-03	2.53E-04	2.61E-02	0.00E+00	2.66E-06	0.00E+00
RX VESSEL >3' ABOVE CORE	0.00E+00	0.00E+00	1.58E-13	1.61E-03	1.06E-02	6.91E-04	7.12E-02	0.00E+00	7.24E-06	0.00E+00
INSULATION	0.00E+00	0.00E+00	5.17E-16	5.23E-06	3.93E-06	0.00E+00	2.86E-04	3.53E-19	0.00E+00	0.00E+00
NST I.D. WALL	0.00E+00	0.00E+00	2.85E-13	2.89E-03	1.90E-02	1.24E-03	1.28E-01	0.00E+00	1.30E-05	0.00E+00
NST O.D. WALL	0.00E+00	0.00E+00	2.89E-20	2.94E-10	1.94E-09	1.26E-10	1.30E-08	0.00E+00	1.33E-12	0.00E+00
BIOSHIELD	1.02E-07	1.48E-08	7.53E-18	2.62E-10	8.20E-13	2.08E-11	1.89E-09	0.00E+00	0.00E+00	0.00E+00
	1.02E-07	1.48E-08	2.30E-09	2.36E+01	2.39E+03	2.18E+02	2.22E+04	2.26E-18	1.27E+00	7.22E-01

TABLE 3
(continued)

ACTIVATION ANALYSIS OF
INDIAN POINT UNIT #1
TIME= 2013

COMPONENT	TC-99 (CURIES)	SB-125 (CURIES)	I-129 (CURIES)	BA-133 (CURIES)	SM-151 (CURIES)	EU-152 (CURIES)	EU-153 (CURIES)	EU-154 (CURIES)	TOTAL CURIES
SHIM RODS CORE A ONLY	8.84E-04	6.95E-09	2.10E-17	5.37E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.61E+02
CRB CORES A&B	1.50E-02	1.67E-05	6.98E-15	3.18E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.28E+04
CRB FOLLOWERS CORES A&B	1.32E-05	0.00E+00	3.39E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.77E+01
FILLER RODS CORE A ONLY	1.07E-07	0.00E+00	1.16E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.66E+00
FILLER RODS CORES A&B	5.23E-06	0.00E+00	1.34E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.29E+01
HOLDOWN COLUMNS CORES A&B	4.30E-03	4.81E-06	2.00E-15	9.12E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.66E+03
FILLER BARS CORES A&B	8.72E-05	9.75E-08	4.06E-17	1.85E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.43E+01
CORE SHROUD	1.51E-03	1.69E-06	7.03E-16	3.20E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.28E+03
UPPER INTERNALS	3.33E-03	3.72E-06	1.55E-15	7.07E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.84E+03
LOWER INTERNALS	3.33E-03	3.72E-06	1.55E-15	7.07E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.84E+03
THERM. SHLD. #1	5.54E-04	6.19E-07	2.58E-15	1.17E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.72E+02
THERM. SHLD. #2	3.40E-04	3.80E-07	1.59E-16	7.22E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.90E+02
RX VESSEL CLAD	5.81E-06	6.49E-09	2.71E-18	1.23E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.95E+00
RX VESSEL CORE HEIGHT	8.65E-17	1.63E-08	3.71E-18	1.67E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.74E-01
RX VESSEL 3' BELOW CORE	6.13E-18	1.16E-09	2.64E-19	1.18E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.20E-02
RX VESSEL >3' BELOW CORE	2.66E-17	5.02E-09	1.14E-18	5.13E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.69E-01
RX VESSEL 3' ABOVE CORE	3.07E-18	5.79E-10	1.32E-19	5.92E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.10E-02
RX VESSEL >3' ABOVE CORE	8.36E-18	1.58E-09	3.59E-19	1.61E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.45E-02
INSULATION	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.95E-04
NST I.D. WALL	1.50E-17	2.84E-09	6.46E-19	2.90E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.52E-01
NST O.D. WALL	1.53E-24	2.89E-16	6.57E-26	2.95E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.55E-08
BIOSHIELD	0.00E+00	0.00E+00	0.00E+00	2.35E-10	8.76E-10	7.37E-08	3.38E-09	1.54E-11	1.11E-06
	2.94E-02	3.18E-05	4.86E-13	6.09E-01	8.76E-10	7.37E-08	3.38E-09	1.54E-11	2.49E+04

TABLE 4

TOTAL RESIDUAL RADIONUCLIDE INVENTORY IN PRIMARY SYSTEM
INDIAN POINT STATION, UNIT ONE
(CURIES)

NUCLIDE	HALF-LIFE (YR.)	SAMPLE* DATE 2 NOV 1987	CALCULATED JAN 1988	CALCULATED 2013
* FE-55	2.70E+00	1.75E+02	1.68E+02	2.75E-01
CO-60	5.27E+00	2.02E+02	1.98E+02	7.38E+00
* NI-63	1.00E+02	1.46E+02	1.46E+02	1.22E+02
CS-137	3.02E+01	2.33E+01	2.32E+01	1.31E+01
* CS-134	2.06E+00	4.19E-01	3.97E-01	8.84E-05
* NI-59	7.50E+04	1.24E+00	1.24E+00	1.24E+00
* MN-54	8.54E-01	7.83E-03	6.87E-03	1.06E-11
TOTALS		5.48E+02	5.36E+02	1.44E+02

* NOTE: CURIE CONTENTS FOR THESE ISOTOPES HAVE BEEN ESTIMATED AT JANUARY 1, 1988 AND DECAY-CORRECTED AS APPROPRIATE. THE ESTIMATES HAVE BEEN OBTAINED BY USING THE DATA IN TABLE 6.1a FOR JANUARY 1, 1988, TO CALCULATE A RATIO BETWEEN THE ISOTOPE OF INTEREST AND THE CO-60 ACTIVITY DETERMINED BY ACTUAL SAMPLING.

TABLE 5
TOTAL RESIDUAL RADIONUCLIDE INVENTORY IN AUXILIARY SYSTEMS
INDIAN POINT STATION, UNIT ONE
(CURIES)

NUCLIDE	HALF-LIFE (YR.)	SAMPLE* DATE 25 MAY 1982	CALCULATED JAN 1988	CALCULATED 2013
FE-55	2.70E+00	1.03E+02	2.45E+01	4.00E-02
CO-60	5.27E+00	6.00E+01	2.87E+01	1.07E+00
NI-63	1.00E+02	2.20E+01	2.12E+01	1.78E+01
CS-137	3.02E+01	4.40E+00	3.87E+00	2.18E+00
CS-134	2.06E+00	3.80E-01	5.78E-02	1.29E-05
NI-59	7.50E+04	1.80E-01	1.80E-01	1.80E-01
MN-54	8.54E-01	9.40E-02	9.99E-04	1.55E-12
TOTALS		1.90E+02	7.85E+01	2.13E+01

*NOTE: EXTRACTED FROM REFERENCE 5, TABLE S.1.

UNITED STATES NUCLEAR REGULATORY COMMISSION

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC

INDIAN POINT UNIT NO. 1

DOCKET NO. 50-3

NOTICE OF ISSUANCE OF ENVIRONMENTAL ASSESSMENT

AND FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is considering the issuance of an order authorizing the decommissioning of Indian Point Unit No. 1 (IP-1) that is licensed to Consolidated Edison Company of New York, Inc. (Con Edison). The proposed Decommissioning Plan involves safe storage (SAFSTOR) of IP-1 until after IP-2 is permanently shut down, at which time both units would be decontaminated and dismantled. The staff has evaluated the proposed SAFSTOR of IP-1 to 2013, consistent with the licensee's amended Decommissioning Plan. The IP-1 license is being renewed only to October 14, 2006, to be consistent with a Notice of Consideration of Issuance of Amendment and Opportunity for Hearing which was published in the FEDERAL REGISTER on December 31, 1985 in order to put new Technical Specifications for the current shutdown condition in place.

Description of Proposed Action

IP-1 has been shut down since October 31, 1974, and all spent fuel has been removed from the reactor and transferred to the IP-1 spent fuel storage pools. Approval of the Decommissioning Plan will allow Con Edison to retain IP-1 in a SAFSTOR status in accordance with an approved Decommissioning Plan. SAFSTOR of IP-1 will allow continued use of the site for electric power production by IP-2. A significant portion of IP-1 equipment is being used to support IP-2 operations.

Finding of No Significant impact

The staff has reviewed the proposed decommissioning relative to the requirements given in 10 CFR Part 51. In the SAFSTOR alternative, IP-1 will be safely stored and subsequently decontaminated to levels that permit release of the property to unrestricted use. IP-1 data on radionuclide inventories for activation and contamination shows that cobalt-60 is the dominant gamma-emitting radionuclide and that an initial 378,000 curies of cobalt-60 in the reactor vessel and its internals at reactor shutdown will decrease the 2,390 curies by 2013. Data on primary system contamination shows that the inventory of cobalt-60 will decrease from 198 curies in 1988 to 7 curies in 2013 and that cesium-137 will decrease from 23 curies to 13 curies over the same period of time. Data on auxiliary systems contamination also shows a decrease during the SAFSTOR period. These reductions in radioactivity will reduce potential exposures to personnel during final dismantling and also may reduce waste volume for disposal.

Based upon its Environmental Assessment, the staff concluded that there are no significant environmental impacts associated with the proposed decommissioning and that the proposed decommissioning will not have a significant effect on the quality of the human environment. Therefore, the Commission has determined, pursuant to 10 CFR 51.31, not to prepare an environmental impact statement for the proposed decommissioning of IP-1.

For further details with respect to this action, see: (1) the licensee's application for authorization to decommission IP-1, dated October 17, 1980, as revised October 13, 1981; July 31, 1986; March 28, 1988; August 10, 1989;

March 28 and July 17, 1990; February 5, April 2, July 31, September 20, and October 12, 1993; May 13 and August 11, 1994; and July 19, 1995; (2) the NRC's Environmental Assessment and Finding of No Significant Impact; and 3) the NRC's Safety Evaluation. These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC, and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York.

Dated at Rockville, Maryland, this 25th day of January 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

Seymour H. Weiss

Seymour H. Weiss, Director
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation