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

July 15, 1992

Docket No. 50-286

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M76970)

The Commission has issued the enclosed Amendment No. 124 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 in response to your application transmitted by letter dated June 11, 1990, as supplemented June 18, 1991, February 11, 1992, and May 13, 1992.

The amendment extends the expiration date of the facility operating license from August 13, 2009, to December 12, 2015.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

N. F. Conicella

Nicola F. Conicella, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.124 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

Mr. Ralph E. Beedle
Power Authority of the State
of New York

Indian Point Nuclear Generating
Station Unit No. 3

cc:

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Mayor, Village of Buchanan
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Buchanan, New York 10511

DATED: July 15, 1992

AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. DPR-59-INDIAN POINT 3

Docket File
NRC & Local PDRs
PDI-1 Reading
SAVarga
JACalvo
RACapra
CSVogan
DOudinot
NConicella
OGC
GHill (4), P1 37
OPA
DHagan, MNBB 3302
WJones, P130A
CGrimes, 11E23
ACRS (10)
JWiggins
OC/LFMB
Plant File
CCheng
BElliot
CCowgill, RI

cc: Plant Service List

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
License No. DPR-64

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

   A. The application for amendment by the Power Authority of the State of New York (the licensee) dated June 11, 1990, as supplemented June 18, 1991, February 11, 1992, and May 13, 1992, 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 operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that 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.

2. Accordingly, Paragraph 3, page 7a, of Facility Operating License No. DPR-64 is hereby amended to read as follows:

"This amended license is effective at 12:01 a.m., March 10, 1978, and shall expire at midnight, December 12, 2015."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Page 7a of license

Date of Issuance: July 15, 1992

Page 7a is attached for convenience, for the composite license to reflect this change.

3. This amended license is effective at 12:01 a.m., March 10, 1978, and shall expire at midnight, December 12, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment: Changes to the Technical Specifications

Date of Issuance: March 8, 1978

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated June 11, 1990, as supplemented June 18, 1991, February 11, 1992, and May 13, 1992, the Power Authority of the State of New York (the licensee) requested an amendment to Facility Operating License No. DPR-64 for Indian Point Nuclear Generating Unit No. 3. The proposed amendment would extend the expiration date of the facility operating license from August 13, 2009, to December 12, 2015. The June 18, 1991, February 11, 1992, and May 13, 1992, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 DISCUSSION

Title 10 CFR 50.51 specifies that each license will be issued for a fixed period of time not to exceed 40 years from the date of issuance. The currently licensed term for Indian Point Nuclear Generating Unit No. 3 is 40 years commencing with issuance of the construction permit on August 13, 1969. Accounting for the time that was required for plant construction, this represents an effective operating license term of approximately 33 years and 8 months. Consistent with Section 50.51 of the Commission's regulations, the licensee, by its application of June 11, 1990, as supplemented June 18, 1991, February 11, 1992, and May 13, 1992, seeks extension of the operating license term for Indian Point 3 such that the fixed period of the license would be 40 years from the date of issuance of the operating license.

The licensee's request for extension of the facility operating license is based on the fact that a 40-year service life was considered during the design and construction of the plants. Although this does not mean that some equipment and components will not wear out and require replacement during the life of the plant, design features were included in the construction and subsequent modifications were made which ensure the ability to test, inspect, and perform preventive and corrective maintenance of the plant structures systems, and components. Surveillance and maintenance practices which are implemented in accordance with the American Society of Mechanical Engineers (ASME) Code and the facility Technical Specifications (TS) provide reasonable assurance that any unexpected degradation in safety-related plant equipment will be identified and corrected.

3.0 EVALUATION

The NRC staff has evaluated the safety issues associated with issuance of the proposed license amendment which would allow an additional 6 years and 4 months of plant operation. The issues addressed consist of additional radiation exposure to the facility operating staff, impacts on the offsite population, nonradiological impacts, and the general aging of plant structures and equipment. The impact of additional radiation exposure to the facility operating staff and the impact on the general population in the vicinity of Indian Point 3 and the nonradiological impacts are addressed in the environmental assessment dated June 25, 1992.

Indian Point 3 was designed, constructed, and licensed for a 40-year service life as discussed in the Final Safety Analysis Report (FSAR). This 40-year design life is based upon operation at a thermal power level of 3025 MWT with a cumulative lifetime capacity factor of 80% or 32 effective full power years. To date, Indian Point 3 has attained a cumulative capacity factor of approximately 54% and is projecting a capacity factor of 75% for the next few cycles.

Reactor Pressure Vessel

The reactor vessel was initially designed and licensed based on an assumed 40-year service life with an 80% capacity factor. A comprehensive vessel material surveillance program in accordance with 10 CFR Part 50, Appendix H, provides the means for continuous monitoring of the reactor vessel throughout the life of the plant. Three (3) capsule specimens (Capsules T, Y, and Z) have been removed and analyzed to date. The licensee completed an update of reference temperatures (RT_{pts}) values in accordance with 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." However, on May 15, 1991, the NRC amended 10 CFR 50.61 (56 FR 22300), making the procedure for calculating the amount of embrittlement consistent with the methodology of Regulatory Guide 1.99, Revision 2, published in May 1988. The amended rule did not change the screening criteria of 270 °F for plates, forgings, and axial weld materials or 300 °F for circumferential weld materials. Consequently, by letter dated December 13, 1991, the NRC staff requested that the licensee provide the most limiting RT_{pts} calculated in accordance with the methodology of Regulatory Guide 1.99, Revision 2, for the proposed license expiration date.

In the supplemental submittal dated February 11, 1992, the licensee provided the most limiting RT_{pts} of 262 °F for the proposed expiration date of December 12, 2015. This temperature, which is for base plate B2803-3, was calculated using the methodology of Regulatory Guide, 1.99, Revision 2. The neutron fluence value used was the most limiting possible. The flux values were obtained from reactor vessel surveillance capsules and the data reflects the high neutron leakage from early core loading patterns, not the currently utilized low leakage core designs. Additionally, the licensee assumed a

conservative plant capacity of 80%. Hence, the screening criteria is not exceeded as a result of revising the facility operating license expiration date. The licensee is continuing to investigate low leakage core loadings to further improve the flux reduction. The NRC staff is currently reviewing generic neutron fluence issues which indicate that the calculated fluence values may be unconservatively low. However, even if the licensee assumed the same plant capacity factor of 80% and fluence values 20% higher than those submitted, the screening criteria would still not be exceeded at the end of the proposed license expiration date of December 12, 2015. In accordance with 10 CFR 50.61, the licensee will update the RT_{pts} values whenever changes in core loadings, surveillance measurements, or other information indicates a significant change in the projected values.

Periodic reactor vessel inservice inspection and testing requirements provide further assurance that any degradation will be identified in a timely manner. The structural integrity of the reactor vessel is assured because it was originally designed assuming a 40-year lifetime. It is monitored, inspected, and tested to detect degradation processes at an early stage of development, and it is operated with procedures to assure that design conditions are not exceeded.

Structures

The design and construction of structures and supports was in accordance with various codes and standards applicable at the time of plant construction. The design basis, fabrication, construction and quality assurance criteria for the plant were reviewed and found acceptable by the NRC staff in Safety Evaluation Report (SER) dated September 21, 1973. Industrial experience with such structures and supports confirms that a service life in excess of 40 years can be anticipated.

The design of the concrete containment was based on the ultimate strength provisions of the American Concrete Institute (ACI) 318-63 Code. Section 3.8 of the SER states that the class I structures were designed in accordance with ACI 318-63 for reinforced concrete with the use of specific loading combination applicable to nuclear power plant design conditions. For the structural steel, the American Institute of Steel Construction (AISC) specifications were utilized.

The containment structure is subject to periodic inspection and testing which provide reasonable assurance that containment structural integrity remains adequate throughout the operating life of the plant, including the proposed extension period.

Mechanical Equipment

With regard to equipment lifetime, some components are expected to require replacement during the life of the plant. Some of the major components already replaced at Indian Point 3 include the steam generators, the main condensers, feedwater heaters, and the service water pumps. These upgrades were performed to increase the efficiency and reliability of plant systems, and ultimately to increase the useful life of Indian Point 3. Periodic inservice inspection and inservice testing requirements have been incorporated into the plant TS and procedures. Any degradation in safety-related plant equipment is identified and corrected based on the surveillance and maintenance programs which are implemented in accordance with ASME Codes and the plant TS. In addition, it should be noted that improvements in existing maintenance and surveillance programs, as well as the development of new programs, are constantly underway at Indian Point 3. Such programs enable early detection of degradation which could affect operability of the component and provide reasonable assurance of continuous operating integrity for the entire operating life of the plant including the proposed extension period.

The primary side pressure boundary components within the Nuclear Steam Supply System scope were designed and constructed for a 40-year design life. The equipment design life is based on the time period of exposure to an operating environment. The 40-year design is based on 32 effective full power years of operation. During the plant construction, materials were not exposed to the operating environment except for system functional tests. The system components were not subjected to a radiation environment until after the operating license became effective. The licensee is committed to a periodic inservice inspection program for the reactor coolant system per TS Section 4.3. Additional monitoring programs are currently in effect or are being developed to evaluate thermal/pressure cycles and to monitor fatigue effects on key components.

Environmental Qualification Of Electrical Equipment

Aging analyses have been performed for all safety-related electrical equipment in accordance with 10 CFR 50.49, "Environmental qualification of electrical equipment important to safety for nuclear power plants." Designated lifetimes have been identified for this equipment. These lifetimes have been incorporated into plant equipment maintenance and surveillance procedures to ensure that all safety-related electrical equipment remains qualified and available to perform its safety function regardless of the overall age of the plant. The Environmental Qualification (EQ) program for electrical equipment operating in a harsh environment is described in Appendix 6F of Indian Point 3 FSAR. The program ensures that EQ is maintained for required electrical equipment within the scope of 10 CFR 50.49.

4.0 SUMMARY

The NRC staff concluded in the environmental assessment dated June 25, 1992, that there would be no significant radiological or nonradiological impact associated with the proposed license amendment. The NRC staff concludes from its considerations of the design, operation, testing, and monitoring of the reactor pressure vessel, structures and mechanical equipment, that an extension of the facility operating license for Indian Point 3 to a 40-year service life is consistent with the facility's licensing basis and there is reasonable assurance that the plant will be able to continue to operate safely for the additional period authorized by this amendment. The NRC staff also concludes that issues associated with plant degradation, to date, have been adequately addressed. Accordingly, the NRC staff finds the proposed change to the expiration date of the facility operating license for Indian Point 3 to be acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on July 7, 1992, (57 FR 29904). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of the amendment will not have a significant effect on the quality of the human environment.

7.0 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 operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
Daniele Oudinot

Date: July 15, 1992

Docket No. 50-286

July 15, 1992

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M76970)

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A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original Signed By:
Nicola F. Conicella, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.124 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PDI-1	PE:PDI-1 for	PM:PDI-1	EMCB	OGC
NAME	CSVogan	DOudinot:pc	NConicella	JWiggins*	CBarth*
DATE	7/9/92	7/9/92	7/9/92	04/16/92	04/16/92

OFFICE	D:PDI-1				
NAME	RACapra RAC				
DATE	7/15/92	/92	/92	/92	/92

*See previous concurrence

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