July 10, 198

Docket No. 50-333

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Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 66123)

The Commission has issued the enclosed Amendment No. 133 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the operating license in response to your application transmitted by letter dated August 19, 1987.

The amendment extends the expiration date of the operating license from May 20, 2010 to October 17, 2014.

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

Sincerely,

Original signed by

David E. LaBarge, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 133 to DPR-59
2. Safety Evaluation

cc: w/enclosures
See next page

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[AMEND 333 TAC 66123] * See previous concurrence

OFC	:PDI-1	:PDI-1	:EMTB	:OGC	:PDI-1	:	:
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123 Main Street
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The Commission has issued the enclosed Amendment No. 133 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated August 19, 1987.

The amendment extends the expiration date of the operating license from May 20, 2010 to October 17, 2014.

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

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Original signed by

David E. LaBarge, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

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See next page

[AMEND 333 TAC 66123]

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NAME	:CVogan	:DLaBarge:vr	:CCheng	:	:RCapra	:	:
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White Plains, New York 10601

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 133
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated August 19, 1987, 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, Facility Operating License No. DPR-59 is hereby amended to read as follows:

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A. Change paragraph 2.F. to read as follows:

This amended license is effective at 11:59 p.m., EDST, June 4, 1977, and shall expire at midnight October 17, 2014.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

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

Date of Issuance: July 10, 1989

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 133 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

INTRODUCTION

By letter dated August 19, 1987, the Power Authority of the State of New York (PASNY or the licensee) requested an amendment to the Facility Operating License DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The proposed amendment would extend the expiration date for the license from May 20, 2010 to October 17, 2014.

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 the FitzPatrick plant is 40 years, commencing with the issuance of the construction permit on May 20, 1970. However, since the license expiration date is May 20, 2010, if the construction time is taken into account, this represents an effective operating license term of about 35.5 years from the date of issuance of the operating license (October 17, 1974). Consistent with Section 50.51 of the Commission's regulations, the licensee has requested an extension of the operating license so that the fixed period of the license would end 40 years after the date of issuance of the operating license.

EVALUATION

The NRC staff has evaluated the safety issues associated with issuance of the proposed license amendment which would allow approximately four and one-half additional years of plant operation. The issues addressed consist of additional radiation exposure to the licensee's operating staff, impacts on the off-site population, 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 the FitzPatrick nuclear plant are addressed in the NRC staff's Environmental Assessment dated April 27, 1989.

The components of the reactor coolant pressure boundary were designed, built and tested to ANSI B31.1.0 (1967) and the appropriate ASME Boiler and Pressure Vessel Codes, Regulatory standards, and supplemental criteria in compliance

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with the requirements of 10 CFR Part 50, Section 50.55a, "Codes and Standards." The initial inservice inspection program was described in the FSAR and Technical Specifications and comply with the requirements of 10 CFR 50.55a(g). Upon completion of the first ten-year inspection interval on July 28, 1985, the licensee evaluated the inservice inspection program and implemented enhancements. Also, by letter dated September 30, 1985, the licensee submitted a revision to the program for the second ten-year interval as required by 10 CFR 50.55a(g) and submitted relief requests by letter dated August 6, 1987. Both of these submittals are currently undergoing NRC review. Additionally, the licensee is evaluating the inservice testing program in accordance with Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," dated April 3, 1989. Therefore, the program is either current or provisions are in place which will address or identify any concerns. Present Inservice Inspection is done in accordance with ASME Section XI 1980 Edition through Summer of 1981 Addenda.

Inspections conducted at several boiling water reactors (BWRs) indicated intergranular stress corrosion cracking (IGSCC) in large-diameter stainless steel pipe. Since the NRC staff considered this a generic problem, the Commission issued Generic Letter 84-11 which required a reinspection program at all BWRs. This program involved welds in stainless steel pipes greater than four inches in diameter, in systems that are part of or connected to the reactor coolant pressure boundary, out to the second isolation valve. If IGSCC was discovered, repair, analysis and additional surveillance were required to ensure the continued integrity of the affected pipe.

Generic Letter 88-01, issued on January 25, 1988, superseded Generic Letter 84-11, and included a copy of NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." NUREG-0313, Revision 2, describes methods acceptable to the staff to control the susceptibility of BWR ASME Boiler and Pressure Vessel Code Class 1, 2, and 3 pressure boundary piping and safe ends to intergranular stress corrosion cracking. The revision describes the technical bases for the staff's positions on the following items: materials of construction; processes to minimize or control IGSCC; water chemistry; reinforcement by weld overlay; replacement of piping; stress improvements; clamping devices; crack characterization and repair criteria; inspection methods, schedules, and personnel; and limits on number of cracked weldments in piping. For piping that does not conform to the staff positions, varying degrees of inservice inspection are required to ensure structural integrity of the pressure boundary piping system, pursuant to paragraph 50.55a(g)(6)(ii) of 10 CFR Part 50.

By letter dated August 16, 1988, and supported by letter dated August 19, 1988, and others, the licensee responded to Generic Letter 88-01, describing the licensee's plans and program for implementation of the NRC staff's positions specified in the generic letter. The licensee response still is under review. However, the staff will ensure that the licensee's program meets the staff requirements.

In their responses to Generic Letter 88-01, the licensee has outlined a comprehensive and aggressive IGSCC detection and mitigation program. This program includes: (1) inspection of susceptible welds; (2) Induction Heating Stress Improvement (IHSI) of welds in the Reactor Recirculation and the Residual Heat Removal Systems applied during two maintenance outages in 1984 and Resistance Heating Stress Improvement (RHSI) applied to two Reactor Recirculation System welds during the 1987 refueling outage; (3) planned selective replacement or removal of piping such as the RHR suction piping and Reactor Water Cleanup suction piping which were replaced and the Reactor Recirculation System bypass lines which were removed during the 1987 refueling outage; and (4) hydrogen water chemistry and crack arrest verification system. This program, as well as application of weld overlays when IGSCC is detected and other requirements of NUREG-0313, have been incorporated to address the IGSCC concerns rather than replacement of the austentic stainless steel piping.

From our evaluation we conclude that compliance with the codes, standards, and regulatory requirements to which the mechanical equipment for the FitzPatrick Plant was originally analyzed, constructed, tested and inspected, including the inservice inspection programs being in compliance with Section XI of the ASME Boiler and Pressure Vessel Code and the other augmented inspections of austenitic stainless steel piping, provide adequate assurance that the structural integrity of components important to safety will be maintained during the additional periods authorized by this amendment. Any significant degradation by an active mechanism would be discovered and the mechanical equipment or component restored to an acceptable condition. Therefore, the age of the mechanical equipment or component should not be a consideration in the extension of the operating license.

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 bases, fabrication, construction, and quality assurance criteria for the plant were reviewed by the NRC staff and presented in the Safety Evaluation Report for the James A. FitzPatrick Nuclear Power Plant dated November 20, 1972. Industrial experience with such structures and supports confirms that a service life in excess of 40 years can be anticipated.

Support structures were constructed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Section 9.3 of the staff's SER states that the Class 1 structures and equipment were designed and analyzed in accordance with ACI 318-63 code allowable values for reinforced concrete and the AISC code allowable values for structural steel.

The use of the indicated codes, standards, and specifications in the design, analyses, and construction; Appendix B of 10 CFR Part 50 for quality assurance; and the identified testing and inservice surveillance requirements; provide reasonable assurance that the concrete and steel structures would withstand continued service for an extended period of at least four and one-half years without impairment of structural integrity.

The Final Safety Analysis Report (FSAR) states that the reactor vessel was designed and fabricated for a service life of 40 years, based on the specified design and operating conditions. The vessel was designed, fabricated and inspected in accordance with the requirements of Section III of the 1965 ASME Boiler and Pressure Code edition, winter 1966 addenda, and the applicable interpretations and Code Cases for a Class A vessel. Operating limitations on temperature and pressure were established using Section III of the ASME Boiler and Pressure Vessel Code and Appendix G of 10 CFR Part 50.

The integrity and performance capability of the ferritic materials in the reactor vessel is assured because the fracture toughness is monitored with a surveillance program in conformance, to the extent practical, with the recommendations of Appendix H, 10 CFR Part 50, "Reactor Vessel Materials Surveillance Program Requirements." The ferritic materials must meet the fracture toughness properties of Section III of the ASME Boiler and Pressure Vessel Code and Appendix G, 10 CFR Part 50, "Fracture Toughness Properties."

In accordance with the reactor vessel material surveillance test program, specimens of the vessel base metal, the weld heat affected zone metal, and the weld metal from a reactor steel joint which simulates a welded joint in the reactor vessel, along with neutron monitor wires, were placed in capsules near the core mid-height prior to initial reactor startup. They were placed on the reactor vessel wall where neutron exposure is similar to that of the vessel wall. Selected groups of specimens are removed at intervals over the lifetime of the reactor and tested to compare mechanical properties with the properties of control specimens which are not irradiated.

The first surveillance capsule was removed from the reactor vessel in 1985 during the Reload 6/cycle 7 refueling outage, which corresponds to 5.98 effective full power years. The analysis of the surveillance capsule was reported in General Electric Company Report MDE-49-0386 and submitted to the NRC by the licensee by letter dated April 30, 1986.

The analysis of the surveillance capsule projected the end of life adjusted reference temperature of the limiting beltline material to be 104°F. Also the upper shelf energy for plate and weld material were determined to be 58 ft-lb transverse and 72 ft lb, respectively. Paragraph IV. B of Appendix G, 10 CFR Part 50 requires that the reactor vessel beltline materials maintain an end-of-life upper-shelf energy of no less than 50 ft-lbs and the predicted adjusted reference temperature must not exceed 200°F. Therefore, the projected fracture toughness properties for the reactor vessel are within the limits set by Appendix G.

Another temperature limitation in the older boiling water reactors pertains to feedwater nozzles, which are prone to cracking. In the 1970's, the feedwater nozzles in BWR plants were found to have cracks due to feedwater leaking through the thermal sleeve and sparger. The nozzle cracking problem has since been resolved. Nevertheless, a pressure-temperature limit has been incorporated into the TS for heatup, cooldown, and critical operation to eliminate future cracking.

Section 50.55a(g) of 10 CFR Part 50, which incorporates Section XI of the ASME Boiler and Pressure Vessel Code, addresses the ongoing inservice inspection and testing requirements for ASME Code Class 1, Class 2 and Class 3 components, pumps and valves. The license has described their current inservice inspection and testing program for welds and supports in a submittal dated July 28, 1985, and for pumps and valves in a submittal dated November 15, 1985. The program details are still under NRC review. However, the current program meets the staff's requirements and the staff will ensure that any issues generated as a result of the present review are properly resolved.

We conclude that there are no special considerations regarding reactor vessel, pump or valve degradation due to the proposed operating lifetime extension. 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. Additionally, the testing program for pumps and valves will enable early detection of degredation which could affect operability of the component.

The NRC staff concluded in the Environmental Assessment that the annual radiological effects during the additional years of operation that would be authorized by the proposed license amendments are not more than were previously estimated in the Final Environmental Statements, and are acceptable.

The staff concludes from its considerations of the design, operation, testing and monitoring of the mechanical equipment, structures, and the reactor vessels that an extension of the operating licenses for the FitzPatrick Nuclear Power Plant to a 40-year service life is consistent with the FSAR, SER, and submittals made by the licensee, and that there is reasonable assurance that the plant will be able to continue to operate safely for the additional period authorized by this amendment. We also conclude that the plant is operated in compliance with the Commission's regulations, and issues associated with plant degradation have been adequately addressed.

In summary, we find that extension of the operating license for the FitzPatrick Nuclear Power Plant to allow 40-year service life is consistent with the Final Environmental Statement and Safety Evaluation Report for the plant and that the Commission's previous findings are not changed.

ENVIRONMENTAL CONSIDERATION

A Notice of Issuance of Environmental Assessment and Finding of No Significant Impact relating to the proposed extension of the Facility Operating License termination dates for the FitzPatrick Nuclear Power Plant was published in the Federal Register on May 4, 1989 (54 FR 19265).

CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register on November 18, 1987 (52 FR 44247) with a correction published on December 28, 1987 (52 FR 48891), and consulted with the State of New York. No public comments were received, and the State of New York had no comments.

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

Dated: July 10, 1989

PRINCIPAL CONTRIBUTORS:

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L. P. Crocker