50-333

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

February 27, 1998

Mr. James Knubel
Chief Nuclear Officer
Power Authority of the State of
New York
123 Main Street
White Plains, NY 10601

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M94209)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 242 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated December 14, 1995, as supplemented September 26, 1997. The amendment supplements Amendment No. 241 to Facility Operating License No. DPR-59 dated December 2, 1997. In implementing Amendment No. 241, your staff identified that the NRC staff had omitted seven TS pages from the issuance of the amendment. Your staff also identified two administrative errors in the staff's safety evaluation (SE). The seven TS pages are provided in this amendment. The corrected SE is enclosed. We regret any inconvenience these errors have caused.

The amendment changes the James A. FitzPatrick TSs to incorporate the inservice testing requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The proposed change adds a new surveillance requirement, 4.0.E, which refers to the requirements of Section XI of the ASME Code and Addenda established by 10 CFR 50.55a(f). Ancillary changes are also required since the proposed Specification 4.0.E replaces the surveillance testing requirements of safety related pump and motor-operated valves and extends the surveillance testing frequency of other components from once every month, to coincide with the ASME Section XI requirements. As discussed in the September 26, 1997 letter, there are three issues within this amendment proposal that will be addressed in separate amendments:

1) Changing the verification that a containment penetration with an inoperable isolation valve is properly isolated from once per day to once per 31 days.

2) The wording used in specifications 3.5.G, 4.5.G.1, 4.5.G.2 and 4.5.G.3 has been changed to make it consistent with Standard Technical Specifications, SR 3.5.1.2 and 3.5.3.1.

3) Changing SR 4.7.A.5.a from monthly to quarterly.

A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Daniel H. Dorman, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures: 1. Amendment No. 242 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

February 27, 1998

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Sincerely,

Original Signed by:

Daniel H. Dorman, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-333

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February 27, 1998

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Sincerely,

Original Signed by:

Daniel H. Dorman, Project Manager
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DATE	02/26/98	02/25/98	02/26/98	02/26/98	

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DATED: February 27, 1998

AMENDMENT NO. 242 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

Docket File
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James Knubel
Power Authority of the State
of New York

James A. FitzPatrick Nuclear
Power Plant

cc:

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DATED: February 27, 1998

AMENDMENT NO. 242 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

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

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. 242
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 December 14, 1995, as supplemented September 26, 1997, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 242, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 27, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 242

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
132	132
133	133
177	177
178	178
185	185
186	186
197	197

4.5 BASES

The testing interval for the Core and Containment Cooling Systems is based on a quantitative reliability analysis, industry practice, judgement, and practicality. The Emergency Core Cooling Systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 450 psig; thus, during operation even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems will be automatically actuated once per 24 months. In the case of the Core Spray System, condensate storage tank water will be pumped to the vessel to verify the operability of the core spray header. On a monthly basis, correct alignment shall be verified for manual, power operated, or automatic valves in ECCS and RCIC system flow paths to provide assurance that proper flow paths will exist for system operation. For the HPCI and RCIC Systems, this requirement also includes the steam flowpath for the turbines and the flow controller position. This surveillance requirement does not apply to valves that cannot be inadvertently misaligned such as check valves, or to valves that are locked, sealed, or otherwise secured in position. A valve that receives an initiation signal is allowed to be in a non-accident position provided the valve will automatically reposition in the proper stroke time upon receipt of the initiation signal. The monthly frequency of this requirement is based upon engineering judgement and is supported by procedural controls governing valve operation that ensure correct valve positions. This frequency is further supported by the Inservice Testing Program, which demonstrates system pump and power operated valve operability. This combination of automatic actuation tests, periodic pump and valve testing, and monthly flow path verification is adequate to demonstrate operability of these systems.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining cooling equipment. Consistent with the definition of operable in Section 4.0.C, demonstrate means conduct a test to show; verify means that the associated surveillance activities have been satisfactorily performed within the specified time interval.

The RCIC flow rate is described in the UFSAR. The flow rates to be delivered to the reactor core for HPCI, the LPCI mode of RHR, and CS are based on the SAFER/GESTR LOCA analysis. The flow rates for the LPCI mode of RHR and CS are modified by a 10 percent reduction from the SAFER/GESTR LOCA analysis. The reductions are based on a sensitivity analysis (General Electric MDE-93-0786) performed for the parameters used in the SAFER/GESTR analysis.

The CS surveillance requirement includes an allowance for system leakage in addition to the flow rate required to be delivered to the reactor core. The leak rate from the core spray piping inside the reactor but outside the core shroud is assumed in the UFSAR and includes a known loss of less than 20 gpm from the 1/4 inch diameter vent hole in the core spray T-box connection in each of the loops, and in the B loop, a potential additional loss of less than 40 gpm from a clamshell repair whose structural weld covers only 5/6 of the circumference of the pipe. Both of these identified sources of leakage occur in the space between the reactor vessel wall and the core shroud. Therefore flow lost through these leak sources does not contribute to core cooling.

4.5 BASES (cont'd)

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC Systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Instrumentation has been provided in the Core Spray System and LPCI System to monitor the presence of water in the discharge piping. This instrumentation is functionally tested monthly to ensure that during the interval between the monthly checks the status of the discharge piping is monitored on a continuous basis.

Normally the low pressure ECCS subsystems required by Specification 3.5.F.1 are demonstrated operable by the surveillance tests in Specifications 4.5.A.1 and 4.5.A.3. Section 4.5.F specifies periodic surveillance tests for the low pressure ECCS subsystems which are applicable when the reactor is in the cold condition. These tests in conjunction with the requirements on filled discharge piping (Specification 3.5.G), and the requirements on ECCS actuation instrumentation (Specification 3.2.B), assure adequate ECCS capability in the cold condition. The water level in the suppression pool, or the Condensate Storage Tanks (CST) when the suppression pool is inoperable, is checked once each shift to ensure that sufficient water is available for core cooling.

3.7 (cont'd)

4. Pressure Suppression Chamber Reactor Building Vacuum Breakers

   a. Except as specified in 3.7.A.4.b below, two Pressure Suppression Chamber Reactor Building Vacuum Breakers shall be operable at all times when the primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber reactor building vacuum breakers shall be $\leq$0.5 psi below reactor building pressure.

   b. From and after the date that one of the pressure suppression chamber reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days, unless such vacuum

4.7 (cont'd)

4. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

   a. The pressure suppression chamber-reactor building vacuum breakers shall be checked for proper operation in accordance with the Inservice Testing Program.

   b. Instrumentation associated with pressure suppression chamber-reactor building vacuum breakers shall be functionally tested once per 92 days.

3.7 (cont'd)

breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. When primary containment integrity is required, all drywell suppression chamber vacuum breakers shall be operable and positioned in the fully closed position except during testing and as specified in 3.7.A.5.b below.

b. One drywell suppression chamber vacuum breaker may be non-fully closed so long as it is determined to be not more than 1° open as indicated by the position lights.

c. One drywell suppression chamber vacuum breaker may be determined to be inoperable for opening.

d. Deleted

4.7 (cont'd)

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. Each drywell suppression chamber vacuum breaker shall be exercised through an opening - closing cycle monthly.

b. When it is determined that one vacuum breaker is inoperable for fully closing when operability is required, the operable breakers shall be exercised immediately, and every 15 days thereafter until the inoperable valve has been returned to normal service.

c. Each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation in accordance with the Inservice Testing Program.

d. A leak test of the drywell to suppression chamber structure shall be conducted once per 24 months; the acceptable leak rate is $\leq$0.25 in. water/min, over a 10 min period, with the drywell at 1 psid.

3.7 (cont'd)

D. Primary Containment Isolation Valves

1. Whenever primary containment integrity is required per 3.7.A.2, containment isolation valves and all instrument line excess flow check valves shall be operable, except as specified in 3.7.D.2. The containment vent and purge valves shall be limited to opening angles less than or equal to that specified below:

Valve Number	Maximum Opening Angle
27AOV-111	40°
27AOV-112	40°
27AOV-113	40°
27AOV-114	50°
27AOV-115	50°
27AOV-116	50°
27AOV-117	50°
27AOV-118	50°

4.7 (cont'd)

c. Secondary containment capability to maintain a 1/4 in. of water vacuum under calm wind conditions with a filter train flow rate of not more than 6,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

	Item	Frequency
a.	The operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and for closure time.	In accordance with the Inservice Testing Program
b.	Instrument line excess flow check valves shall be tested for proper operation.	In accordance with the Inservice Testing Program
c.	All normally open power-operated isolation valves (except for the main steam isolation valves) shall be fully closed and reopened.	In accordance with the Inservice Testing Program

3.7 (cont'd)

2. With one or more of the containment isolation valves inoperable, maintain at least one isolation valve operable in each affected penetration that is open and:

   a. Restore the inoperable valve(s) to operable status within 4 hours; or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the closed position. Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control; or

   c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or a blind flange.

3. If Specifications 3.7.D.1 or 3.7.D.2 cannot be met the reactor shall be in the cold condition within 24 hrs.

4.7 (cont'd)

	Item	Frequency
d.	With the reactor at a reduced power level, fast close each main steam isolation valve, one at a time, and verify closure time.	In accordance with the Inservice Testing Program
e.	Main steam isolation valves shall be exercised by partial closure and subsequent reopening.	Twice per Week

2. Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

4.7 BASES (cont'd)

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 in. restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

A list of containment isolation valves, including a brief description of each valve is included in Section 7.3 of the updated FSAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 242 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

1.0 INTRODUCTION

By letter dated December 14, 1995, as supplemented September 26, 1997, the Power Authority of the State of New York (the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant Technical Specifications (TSs). The requested changes would incorporate the inservice testing (IST) requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). This proposed amendment replaces the existing Technical Specifications Surveillance test requirements for pumps and valves with the requirements and criteria of ASME Section XI. This revises the testing frequency of pumps and valves to be consistent with ASME Section XI. The change replaces multiple individual test requirements with a single requirement. Reactor Core Isolation Cooling (RCIC) system pump and valve testing (which is not part of the Inservice Testing Program) is also changed from monthly to once per 92 days. This makes RCIC testing consistent with ASME Section XI testing of emergency core cooling system (ECCS). The September 26, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The proposed amendment incorporates the requirements of the NRC-approved version of the Section XI ASME Code for inspecting and testing of ASME Code Class 1, 2, and 3 components. The proposed testing program for pumps and valves follows the requirements of ASME Section XI, Paragraph IWP-3400, "Frequency of Inservice Tests," and Paragraph IWV-3400, "Inservice Tests, category A and B Valves" and the NRC Standard Review Plan 3.9.6, "Inservice Testing of Pumps and Valves," which states in part, "The pump and valve test programs are acceptable if they meet the requirements for establishing reference values and the periodic testing schedule of IWP-3000 and IWV-3000, respectively, of Section XI of the ASME Code. The allowable ranges of inservice test quantities, corrective actions, and bearing temperature tests for pumps are established by IWP-3000 and IWP-4000. The pump test schedule in the plant technical specification is required to comply with these rules." The licensee has proposed adding TS 4.0.E to provide that the licensee shall follow and perform all required pump and valve testing in accordance with ASME Section XI, except when written relief has been granted by the NRC pursuant to 10 CFR Part 50, Section 50.55a(f)(6)(I). Only those changes to the testing frequency of pumps and valves allowed under ASME section XI can be made without requesting NRC approval through a relief request. The FitzPatrick TSs contain, in part, pump and valve surveillance test requirements for the following systems:

- Standby Liquid Control System (4.4.A)
- Core Spray System and Residual Heat Removal System (4.5.A)
- Containment Cooling (4.5.B)
- High-Pressure Coolant Injection (4.5.C)
- Reactor Core Isolation Cooling System (4.5.E)
- Emergency Core Cooling System - Cold Condition (4.5.F)
- Primary Containment Isolation Valves (4.7.D)
- Pressure Suppression Chamber-Reactor Building and Pressure Suppression Chamber-Drywell Vacuum Breakers (4.7.A.4 & 5)
- Emergency Service Water System (4.11.D)

The proposed amendment would modify the TS to follow the requirements of ASME Section XI, Paragraph IWP-3400, "Frequency of Inservice Tests," and Paragraph IWV-3400, "Inservice Test for Category A and B Valves."

The RCIC System is not part of the IST program or the ECCS. However, the RCIC System is included with the ECCS section of the TS because of their similar functions. The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss-of-coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the high-pressure coolant injection (HPCI) and RCIC systems perform similar functions. The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. A 92-day Frequency for Surveillance Requirement 4.5.E.1.c is consistent with the IST program requirements and will ensure that the system will limit the release of radioactive materials to the environment following a loss-of-coolant accident (LOCA).

The changing of the testing requirements of the pump and valves for multiple individual test requirements to a single requirement in the TS listed above is consistent with the BWR4 Standard Technical Specifications, NUREG-1433 and does not change the ability of the system to limit the release of radioactive materials to the environment following a LOCA. NRC staff finds the above proposed changes to the James A. FitzPatrick Nuclear Power Plant TS acceptable.

There are three issues within this amendment proposal that will be addressed in separate amendments:

1) Changing the verification that a containment penetration with an inoperable isolation valve is properly isolated from once per day to once per 31 days.

2) The wording used in specifications 3.5.G, 4.5.G.1, 4.5.G.2 and 4.5.G.3 has been changed to make it consistent with Standard Technical Specifications, SR 3.5.1.2 and 3.5.3.1.

3) Changing SR 4.7.A.5.a from monthly to quarterly.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 1635). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 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: Karen R. Cotton

Date: February 27, 1998