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

March 9, 1992

Docket No. 50-333

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 JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M82630)

The Commission has issued the enclosed Amendment No. 179 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 January 9, 1992.

The amendment revises the technical specifications to permit hydrostatic pressure and leakage testing of the Reactor Coolant System (RCS) as required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code at RCS temperatures exceeding 212 degrees F. During this testing, the High Pressure Coolant Injection, Reactor Core Isolation Cooling, and the Automatic Depressurization System/Safety Relief Valves are not required to be operable.

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,

Brian C. McCabe

Brian C. McCabe, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 179 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

cp-1

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Power Authority of the State of New York

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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. 179
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 January 9, 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, 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. 179, 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 30 days.

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:
Changes to the Technical
Specifications

Date of Issuance: March 9, 1992

3.5 (cont'd)

during such time, the HPCI System is operable.

2. If the requirements of 3.5.D.1 cannot be met, the reactor shall be placed in the cold condition and pressure less than 100 psig, within 24 hr.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.5.1.a and 3.5.1.b above, provided that reactor coolant temperature is $\leq$212°F and the reactor vessel is vented or reactor vessel head is removed.

4. The ADS is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted.

4.5 (cont'd)

2. A logic system functional test.

a. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the operable ADS valves and the HPCI subsystem shall be verified to be operable immediately and at least weekly thereafter.

b. When it is determined that more than one relief/safety valve of the ADS is inoperable, the HPCI System shall be verified to be operable immediately.

3.5 (Cont'd)

E. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212°F except from the time that the RCIC System is made or found to be inoperable for any reason, continued reactor power operation is permissible during the succeeding 7 days unless the system is made operable earlier provided that during these 7 days the HPCI System is operable.

2. If the requirements of 3.5.E cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hours.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.5.E.2 above, provided that reactor coolant temperature is $\leq$212°F.

4. The RCIC system is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted.

4.5 (Cont'd)

E. Reactor Core Isolation Cooling (RCIC) System

1. RCIC System testing shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within ten days of continuous operation from the time steam becomes available.

	Item	Frequency
a.	Simulated Automatic Actuation (and Restart*) Test	Once/operating cycle
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month
d.	Flow Rate	Once/3 months
e.	Testable Check Valves	Tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.
f.	Logic System Functional Test	Once/operating cycle

* Automatic restart on a low water level signal which is subsequent to a high water level trip.

3.6 (cont'd)

a. $\leq$20°F when to the left of curve C.

b. $\leq$100°F when on or to the right of curve C.

Specifications 3.5.C, 3.5.D, 3.5.E and 3.6.E which would become effective because of an increase in reactor coolant temperature above 212°F or pressures above 100 and 150 psig are not required while conducting the RCS hydrostatic pressure and leakage tests between 212°F and 300°F provided all control rods are fully inserted.

3. Non-Nuclear Heatup and Cooldown

During heatup by non-nuclear means (mechanical), cooldown following nuclear shutdown and low power physics tests the Reactor Coolant System pressure and temperature shall be on or to the right of the curve B shown in Figure 3.6-1 Part 1, 2, or 3 and the maximum temperature change during any one hour shall be $\leq$100°F.

4. Core Critical Operation

During all modes of operation with a critical core (except for low power physics tests) the reactor Coolant System pressure and temperature shall be at or to the right of the curve C shown in Figure 3.6-1 Part 1, 2, or 3 and the maximum temperature change during any one hour shall be $\leq$100°F.

4.6 (cont'd)

3. Non-Nuclear Heatup and Cooldown

During heatup by Non-Nuclear means, cooldown following nuclear shutdown and low power physics tests, the reactor coolant system pressure and temperature shall be recorded every 30 minutes until two consecutive temperature readings are within 5°F of each other.

4. Core Critical Operation

During all modes of operation with a critical core (except for low power physics tests) the reactor Coolant System pressure and temperature shall be recorded within 30 minutes prior to withdrawal of control rods to bring the reactor critical and every 30 minutes during heatup until two consecutive temperature readings are within 5°F of each other.

3.6 (cont'd)

5. With any of the limits of 3.6.A.1 through 3.6.A.4 above exceeded, either

   a. restore the temperature and/or pressure to within the limits within 30 minutes, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system, and determine that the reactor coolant system remains acceptable for continued operations; or

   b. be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. Idle Recirculation Loop Startup

   When Reactor Coolant System temperature is > 140°F an idle recirculation loop shall not be started unless:

   a. The temperature differential between the reactor coolant system and the reactor vessel bottom head drain line is ≤ 145°F, and

   b. When both loops are idle, the temperature difference between the reactor coolant system and the idle loop to be started is ≤ 50°F, or

   c. When only one loop is idle, the temperature difference between the idle loop and the operating loop is ≤ 50°F.

4.6 (cont'd)

5. Not Used

6. Idle Recirculation Loop Startup

   Within 30 minutes prior to startup of an idle loop:

   a. The differential temperature between the reactor coolant system and the reactor vessel bottom head drain line shall be recorded, and

   b. When both loops are idle, the differential temperature between the reactor coolant system and the idle loop to be started shall be recorded, or

   c. When only one loop is idle, the temperature differential between the idle loop and the operating loop shall be recorded.

3.6 (cont'd)

4.6 (cont'd)

7. Reactor Vessel Flux Monitoring

The reactor vessel Flux Monitoring Surveillance Program complies with the intent of the May, 1983 revision to 10 CFR 50, Appendices G and H. The next flux monitoring surveillance capsule shall be removed after 15 effective full power years (EFPYs) and the test procedures and reporting requirements shall meet the requirements of ASTM E 185-82.

B. Deleted

B. Deleted

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed the equilibrium value of 3.1 μCi/gm of dose equivalent I-131. This limit may be exceeded, following a power transient, for a maximum of 48 hr. During this iodine activity transient the iodine concentrations shall not exceed the equilibrium limits by more than a factor of 10 whenever the main steamline isolation valves are open. The reactor shall not be operated more than 5 percent of its annual power operation under this exception to the equilibrium limits. If the iodine concentration exceeds the equilibrium limit by more than a factor of 10, the reactor shall be placed in a cold condition within 24 hr.

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hr and analyzed for gross gamma activity.

   b. Isotopic analysis of a sample of reactor coolant shall be made at least once/month.

   c. A sample of reactor coolant shall be taken prior to startup and at 4 hr intervals during startup and analyzed for gross gamma activity.

   d. During plant steady state operation and following an offgas activity increase (at the Steam Jet Air Ejectors) of 10,000 μCi/sec within a 48 hr. period or a power level change of $\geq$20 percent of full rated power/hr reactor coolant samples shall be taken and analyzed for gross gamma activity. At least three samples will be taken at 4 hr intervals. These sampling requirements may be omitted whenever the equilibrium I-131 concentration in the reactor coolant is less than 0.007 μCi/ml.

3.6 (cont'd)

2.

a. From and after the date that the safety valve function of one safety/relief valve is made or found to be inoperable, continued operation is permissible only during the succeeding 30 days unless such valve is made operable sooner.

b. From and after the time that the safety valve function on two safety/relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 7 days unless such valves are sooner made operable.

3. If Specification 3.6.E.1 and 3.6.E.2 are not met, the reactor shall be placed in a cold condition within 24 hours.

4. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Item B.2 above, provided that reactor coolant temperature is $\leq$212°F and the reactor vessel is vented or the reactor vessel head is removed.

5. The Safety and Safety/Relief Valves are not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted.

4.6 (cont'd)

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

4. An annual report of safety/refief vave failures and challenges will be sent to the NRC in accordance with Section 6.9.A.2.b.

Amendment No. ~~43~~, ~~70~~, ~~130~~, ~~134~~, 179

3.6 and 4.6 BASES (cont'd)

Fig. 3.6-1, curve B, provides limitations for plant heatup and cooldown when the reactor is not critical or during low power physics tests. The thermal limitation is based on maximum heatup and cooldown rates of 100°F/hr in any one-hour period.

Fig. 3.6-1, curve C, establishes operating limits when core is critical. These limits include a margin of 40°F as required by 10 CFR 50 Appendix G.

The requirements for cold boltup of the reactor vessel closure are based on NDT temperature plus a 60°F factor of safety. This factor is based on the requirements of the ASME Code to which the vessel was built. For Fig. 3.6-1, curves A, B and C, margins are only added to the low temperature portion of the curve where non-ductile failure is a concern. The closure flanges have an NDT temperature not greater than 30°F and are not subject to any appreciable neutron radiation exposure. Therefore, the minimum temperature of the flanges when the studs are in tension is 30°F plus 60°F, or 90°F.

Specification 3.6.A.2 identifies four LCOs that become effective with increased reactor coolant temperature or pressure but are not in effect during the hydrostatic and leakage tests. This is necessary because, as reactor fluence increases, the minimum test temperature and pressure rises into ranges normally associated with startup or hot shutdown. RCS pressure and temperature are used throughout the Technical Specifications as a basis for establishing plant mode and system operability requirements. Some LCOs and restrictions cannot be satisfied during the test at elevated temperatures. For example, Specifications 3.5.C.1 and 3.5.E.1 require that HPCI and RCIC be operable when reactor pressure exceeds 150 psig and 212°F. HPCI and RCIC cannot be made operable during the test because piping normally filled with steam is filled with water during the test.

Hydrostatic and leakage tests shall be terminated before the reactor coolant temperature exceeds 300°F. This temperature limit is based on providing a 50°F band for operating flexibility between the 300°F limit and the highest estimated minimum testing temperature at 32 EFPY (approximately 250°F).

The protection provided by LCOs applicable during cold shutdown plus the requirement that all control rods be fully inserted are adequate to ensure protection of public health and safety. The hydrostatic test is performed once every 10 years while the leakage test is performed after each refueling when conditions are similar to cold shutdown (i.e., after the reactor has been shutdown and decay heat and the energy stored in the core is very low). The consequences of accidents (small and large break LOCAs, MSLB, etc.) are bounded by analyses that assume full power operation. Specification 3.5.A requires the low pressure ECCS systems to be operable. Specifications 3.7.A, 3.7.B and 3.7.C require the containment, SGTS and secondary containment to be operable. Specifications 3.2.A, 3.2.B and Appendix B, Specification 3.8 require instrumentation that initiate containment, low pressure ECCS, SBGT and secondary containment be operable. Emergency power is required by Specification 3.9.B.

3.6 and 4.6 BASES (cont'd)

B. Deleted

C. Coolant Chemistry

A radioactivity concentration limit of 20 μCi/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Radiological Effluent Technical Specification Section 3.2.a if there is a failure or a prolonged shutdown of the cleanup demineralizer.

In the event of a steam line rupture outside the drywell, with this coolant activity level, the resultant radiological dose at the site boundary would be 33 rem to the thyroid, under adverse meteorological conditions assuming no more than 3.1 μCi/gm of dose equivalent I-131. The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hr. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant. Also during reactor startups and large power changes which could affect iodine levels, samples of reactor coolant shall be analyzed to insure iodine concentrations are below allowable levels. Analysis is required whenever the I-131 concentration is within a factor of 100 of its allowable equilibrium value. The necessity for continued sampling following power and offgas transients will be reviewed within 2 years of initial plant startup.

The surveillance requirements 4.6.C.1 may be satisfied by a continuous monitoring system capable of determining the total iodine concentration in the coolant on a real time basis, and annunciating at appropriate concentration levels such that sampling for isotopic analysis can be initiated. The design details of such a system must be submitted for evaluation and accepted by the Commission prior to its implementation and incorporation in these Technical Specifications.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

Materials in the Reactor Coolant System are primarily 304 stainless steel and Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. The attached graph, Fig. 4.6-1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup, and hot standby. During these periods with steaming rates less

3.6 and 4.6 BASES (cont'd)

than 100,000 lb/hr, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Fig. 4.6-1 are not exceeded. At steaming rates of at least 100,000 lb/hr, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal ranges. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This is not necessarily the case. Conductivity could be high due to the presence of a neutral salt; e.g., $Na_2\ SO_4$, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are, in fact, high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the condition include operation of the Reactor Cleanup System, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the Reactor Water Cleanup System to reestablish the purity of the reactor coolant. During startup periods, which are in the category of less than 100,000 lb/hr, conductivity may exceed 2 μmho/cm because of the initial evolution of gases and the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μmho/cm (other than short-term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuousl(monitored. The samples of the coolant which are taken every 96 hr will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses of the reactor coolant required by Specification 4.6.C.1 may be performed by a gamma scan.

D. Coolant Leakage

Allowable leakage rates of coolant from the Reactor Coolan System have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to make up Reactor Coolant System leakage in the event of loss of off-site a-c power. The normally expected background leakage due to equipment design and the detection capability for determining system

UNITED STATES
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WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 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 January 9, 1992, 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 (TS). The requested changes would permit hydrostatic pressure and leakage testing of the Reactor Coolant System (RCS) as required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code at RCS temperatures exceeding 212 degrees F. During this testing, the High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), and the Automatic Depressurization System (ADS)/Safety Relief Valves (SRV) are not required to be operable.

2.0 BACKGROUND

Hydrostatic testing and system leakage testing of the Reactor Coolant System is required by Section XI of the ASME B&PV code. NRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," is used to calculate the reactor pressure vessel pressure and temperature (P-T) limits required for this test. The P-T curves defining these limits are periodically recalculated to consider the results of analyses of irradiated surveillance specimens to account for accumulated reactor fluence.

The current curves require that these tests be conducted at RCS temperatures approaching 190 degrees F. Because decay heat and mechanical heat used to heat the reactor coolant do not allow exact control, the operators require margin to maintain the test temperature between the minimum temperature limit and the maximum temperature limit of 212 degrees F. Furthermore, in the future, these curves will be revised to require temperatures that exceed 212 degrees F as the accumulated fluence increases. An extrapolation from the minimum test temperature at 16 effective full power years (EFPY) indicates that minimum testing temperature will peak at about 250 degrees F at 32 EFPY.

ATTACHMENT TO LICENSE AMENDMENT NO. 179

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
118	118
120	120
121	121
137	137
138	138
139	139
143	143
148	148
149	149
150	150

3.5 (cont'd)

a. From and after the date that the HPCI System is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the Automatic Depressurization System, the Core Spray System, LPCI System, and Reactor Core Isolation Cooling System are operable.

b. If the requirements of 3.5.C.1 cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hrs.

2. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature $\leq$212°F with an inoperable component(s) as specified in 3.5.C.1 above.

3. The HPCI system is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted.

4.5 (cont'd)

a. When it is determined that the HPCI subsystem is inoperable the RCIC, the LPCI subsystem, both core spray subsystems, and the ADS subsystem actuation logic shall be verified to be operable immediately. The RCIC system and ADS subsystem logic shall be verified to be operable daily thereafter.

As reactor fluence increases, the minimum test temperature and pressure rises into ranges normally associated with startup or hot shutdown. RCS pressure and temperature are used throughout the TS as a basis for establishing system operability requirements. However, some Limiting Conditions for Operation (LCO) cannot be satisfied during hydrostatic and leak tests at elevated temperatures. Specifically, certain LCOs for HPCI, RCIC, ADS, and the SRVs cannot be satisfied during these tests for reasons discussed below:

1. TS 3.5.C: Requires that the HPCI system be operable when irradiated fuel is in the vessel, the reactor pressure is greater than 150 psig, and the reactor coolant temperature is greater than 212 degrees F. HPCI cannot be made operable during the test because piping normally filled with steam is filled with water during the test.

2. TS 3.5.D: Requires that the ADS system be operable when irradiated fuel is in the vessel, reactor pressure is greater than 100 psig, and prior to startup from the cold condition. The ADS has not been evaluated for operability in the water-solid condition and may not be operable.

3. TS 3.5.E: Requires that the RCIC System be operable when irradiated fuel is in the vessel, the reactor pressure is greater than 150 psig, and the reactor coolant temperature is greater than 212 degrees F. RCIC cannot be made operable during the test because piping normally filled with steam is filled with water during the test.

4. TS 3.6.E: Requires the SRVs to be operable when the reactor coolant system exceeds atmospheric pressure and temperature is greater than 212 degrees F. The SRVs will have to be gagged closed when test pressures exceed the SRV setpoints thus rendering them inoperable.

As stated above, the required hydrostatic pressure and inservice leak testing cannot be conducted without making HPCI, RCIC, ADS, and SRVs inoperable. The proposed changes to the TS will allow testing to be conducted at elevated temperatures with these systems inoperable.

3.0 EVALUATION

As outlined in Chapter 6 of the Updated Final Safety Analysis Report (UFSAR), "Emergency Core Cooling System (ECCS)," in the event of a Loss of Coolant Accident (LOCA), the ECCS is designed to remove residual heat including stored heat and heat generated due to radioactive decay, such that excessive fuel clad temperature is prevented. The objective of the ECCS is to limit, in conjunction with primary and secondary containments, the release of radioactive materials to the environs following a LOCA so that resulting radiation exposures are kept within the guideline values given in 10 CFR Part 100. In order to satisfy the Safety Design Bases, four systems are provided for emergency core cooling:

1. HPCI System
2. Automatic Depressurization System (ADS)
3. Core Spray System
4. LPCI, an operating mode of the RHR System

These are, in addition to the other systems which supply core coolant, feedwater, control rod drive (CRD) hydraulic pumps, and RCIC.

The manner in which the ECCS operate to protect the core is a function of the rate at which coolant is lost from the break in the Reactor Coolant Pressure Boundary. The HPCI System is designed to operate while the Reactor Coolant System is at high pressure. The Core Spray System and LPCI are designed for operation at low pressures. If the break in the Reactor Coolant Pressure Boundary is of such a size that the loss-of-coolant exceeds the capacity of the HPCI System, Reactor Coolant System pressure drops at a rate fast enough to allow the Core Spray System and LPCI to pump additional coolant into the reactor vessel in time to cool the core.

Automatic depressurization is provided to automatically reduce Reactor Coolant System pressure if a break has occurred and vessel water level is not maintained by the HPCI System and other water addition systems. Rapid depressurization of the Reactor Coolant System is desirable to permit flow from the Core Spray and LPCI Systems to enter the vessel, so that the temperature rise in the core is limited.

During hydrostatic testing and system leakage testing of the RCS, the Recirculation pumps are in operation and a water-solid condition is maintained to control the necessary pressure and temperature. Reactor water makeup, pressure, and level is controlled using the Control Rod Drive and Reactor Water Cleanup systems. During the tests, all control rods are inserted to ensure the core remains subcritical and adequate subcriticality margins are maintained. Furthermore, the decay heat level is minimized following the refueling or maintenance activities, and the reactor is maintained at or near cold shutdown conditions.

During the hydrostatic pressure and leak test conditions, the postulated worst case accident is a LOCA. The effects of a small or large break LOCA are bounded by the existing plant analyses. This is assured by the following test conditions: the control rods are maintained fully inserted to maintain subcriticality margins, the reactor coolant inventory is large, the reactor coolant energy (enthalpy) is significantly less than that during power operation, and the decay heat is low. With a small break LOCA, the RCS will depressurize while the operator terminates the test and initiates RHR cooling and/or low pressure ECCS, as necessary. With a large break LOCA, the reactor will rapidly depressurize and all low pressure ECCS with their initiating instrumentation will be available. The operability of these low pressure ECCS Systems is assured by the requirements of TS 3.5, "Core and Containment Cooling Systems."

Primary containment integrity will be maintained during the hydrostatic testing and system leakage testing of the RCS. Furthermore, other systems designed to restrict radiological release (i.e., Secondary Containment and the Standby Gas Treatment System) will also be available. The availability of these systems will assure that offsite releases remain within the guideline values of 10 CFR Part 100.

For the above reasons, the NRC staff finds that the proposed amendment is acceptable.

4.0 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.

5.0 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. 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 (57 FR 4494). 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.

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 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:
Brian McCabe

Date: March 9, 1992

March 9, 1992

Docket No. 50-333

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 JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M82630)

The Commission has issued the enclosed Amendment No. 179 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 January 9, 1992.

The amendment revises the technical specifications to permit hydrostatic pressure and leakage testing of the Reactor Coolant System (RCS) as required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code at RCS temperatures exceeding 212 degrees F. During this testing, the High Pressure Coolant Injection, Reactor Core Isolation Cooling, and the Automatic Depressurization System/Safety Relief Valves are not required to be operable.

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
Brian C. McCabe, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 179to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDI-1:LA	PDI-1PM	SRXB	OGC	PDI-1:D
NAME	CSVogan	BCMcCabe:pc	RJones		RACapra
DATE	02/18/92	02/18/92	02/19/92	02/19/92/	03/09/92

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FILENAME: C:\WP\WPDOCS\M82630.AMD

DATED: March 9, 1992

AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

Docket File
NRC & Local PDRs
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PD Plant-specific file

cc: Plant Service list