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

August 11, 1993

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. M86766)

The Commission has issued the enclosed Amendment No. 195 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated June 16, 1993, as supplemented by letter dated July 30, 1993.

Those portions of your June 16, 1993, application not contained in this amendment are still under NRC staff review.

The amendment extends the current intervals for certain surveillances that are required to be performed once each operating cycle until the end of the next refueling outage, scheduled to start in January 1995. Specifically, the amendment extends the current intervals for bench checking and disassembling safety/relief valves in accordance with TSs 4.6.E.1 and 4.6.E.2, and for testing excess flow check valves in accordance with TS 4.7.D.1.b.

A copy of the related Safety Evaluation (SE) is enclosed. The staff's SE recommends that the condition of excess flow check valve 29EFC-34B be visually examined during the 3-week maintenance outage scheduled for September 1993.

Mr. Jack Gray of your staff advised me during a telephone discussion on August 10, 1993, that this condition of the SE was acceptable. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

John E. Menning

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 195 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

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DATED: August 11, 1993

AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
J. Menning
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
C. Hammer, 7/E/23
P. Campbell, 7/E/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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. 195
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 June 16, 1993, as supplemented July 30, 1993, 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. 195 , 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: August 11, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 195

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
142a	142a
143	143
185	185

3.6 (cont'd)

E. Safety and Safety/Relief Valves

1. During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212 °F, the safety mode of all safety/relief valves shall be operable, except as specified by Specification 3.6.E.2. The Automatic Depressurization System valves shall be operable as required by specification 3.5.D.

4.6 (cont'd)

E. Safety and Safety/Relief Valves

1. At least one half of all safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. The safety/relief valve settings shall be set as required in Specification 2.2.B. All valves shall be tested every two operating cycles.*

* The current surveillance interval for bench checking safety/relief valves is extended until the end of R11/C12 refueling outage scheduled for January, 1995. This is a one-time extension, effective only for this surveillance interval. The next surveillance interval will begin after the completion of the bench check testing and after the safety/relief valves are declared operable.

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 Specifications 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 ≤ 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.

* The current surveillance interval for disassembling and inspecting at least one safety/relief valve is extended until the end of R11/C12 refueling outage scheduled for January, 1995. This is a one-time extension, effective only for this surveillance interval. The next surveillance interval will begin upon completion of this surveillance.

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:

a. At least once per operating cycle, the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and for closure time.

b. At least once per operating cycle, the instrument line excess flow check valves shall be tested for proper operation.*

c. At least once per quarter:

(1.) All normally open power-operated isolation valves (except for the main stream line and Reactor Building Closed Loop Cooling Water System (RBCLCWS) power-operated isolation valves) shall be fully closed and reopened.

* The current surveillance interval for testing instrument line excess flow check valves is extended until the end of the R11/C12 refueling outage scheduled for January, 1995. This is a one-time extension, effective only for this surveillance interval. The next surveillance interval will begin upon completion of this surveillance.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195 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 June 16, 1993, as supplemented July 30, 1993, 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 provide a one-time extension of the current intervals for various surveillances that are required to be performed once each operating cycle. Specifically, the requested changes would extend the current intervals for bench checking and disassembling safety/relief valves in accordance with TSs 4.6.E.1 and 4.6.E.2, and for functionally testing 10-percent of each snubber type in accordance with TS 4.6.1.3. The requested changes would also extend the current interval for testing instrument line excess flow check valves in accordance with TS 4.7.D.1.b. The licensee requested a one-time extension until the end of the next refueling outage, currently scheduled to start in January 1995. The only planned plant shutdowns between now and the next refueling outage are two, 3-week maintenance outages scheduled for September 1993 and April 1994. Following the January 1995 refueling outage, the regular schedules specified in the TSs for these surveillances would be resumed. This safety evaluation only addresses the changes proposed for the current safety/relief valve and excess flow check valve surveillance intervals. The proposed change to the snubber surveillance interval remains under review by the NRC staff and will be the subject of future correspondence. The July 30, 1993, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

One-time extensions of the normal surveillance intervals were requested by the licensee to allow related testing to be performed during the next scheduled refueling outage. Typically, an 18-month surveillance interval, with a 25 percent extension allowed by TS 4.0.B.1, coincides with an operating cycle from one refueling outage to the next. However, the previous refueling outage lasted for 14 months, from November 1991 to January 1993. For surveillance requirements that can only be accomplished during a refueling outage, the intervals will exceed the normal 18-months for an operating cycle, even with the 25 percent extension. The first such surveillance will be due in August 1993. To meet the currently required schedules, a lengthy plant shutdown after one-third of the operating cycle would be necessary.

2.1 Safety/Relief Valves

The safety/relief valves (SRVs) provide overpressure protection for the reactor vessel and main steam lines. TS 4.6.E.1 requires that at least half of the SRVs be setpoint tested or replaced each operating cycle. A commitment in Licensee Event Report (LER) 92-016 further requires that all SRVs, rather than half, be setpoint tested each operating cycle. The current scheduled surveillance due dates for the eleven SRVs are between April 3, 1993, and July 18, 1993. When the 25 percent interval extension as provided for in TS 4.0.B.1 is applied, the due dates are extended to between August 18, 1993, and December 2, 1993. Similarly, TS 4.6.E.2 requires that at least one SRV be disassembled and inspected once each operating cycle. Applying the 25 percent interval extension, this surveillance is due on August 30, 1993. The requested one-time TS change would extend the current interval for these SRV surveillance requirements until the refueling outage that is scheduled to begin in January 1995.

The licensee provided test data information which indicates several occurrences of significant setpoint drift in the high direction. Similar industry information indicates that for the plant model Target Rock 2-stage SRV, there can be significant setpoint drift high. The SRV setpoint affects the self-actuation of the valves on high pressure; however, the setpoint does not affect the Automatic Depressurization System or the manual actuation capability. To support the increased surveillance interval described above, the licensee performed an analysis to demonstrate that the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) pressure limit of 1375 psig (or 110% of design pressure) would not be exceeded for a condition where nine SRVs drifted high to 1195 psig and two SRVs did not open at all. This would reasonably bound the observed setpoint drift observed in the plant data. The analysis also confirmed that the actuation of the SRVs at the higher setpoints would not adversely affect the High Pressure Coolant Injection System, the Reactor Core Isolation Cooling System, the Primary Containment Integrity, the fuel thermal limits, or Emergency Core Cooling System/Loss of Coolant Accident performance.

The staff requested additional information in a letter dated July 6, 1993, to determine the effects which the additional time period between surveillances would have on the SRV setpoint drift and to determine the licensee's action plan for reducing the severity of the setpoint drift. The licensee's response dated July 30, 1993, provided SRV setpoint test data for the period of 1983 to 1993. A review of this data indicates that there would not be any increased tendency toward additional drift for longer than normal surveillance periods in the range of the period being requested. Also, during the 14-month period of the last refueling outage, the plant was not operating and no degradation is expected to have occurred. The corrosion induced bonding which causes most of the setpoint drift takes place in an operating environment and SRV setpoint drift during the extended period between surveillances should be similar to that experienced in normal surveillance intervals. In order to reduce pilot disk-to-seat sticking, the licensee has committed to install new pilot disks containing a catalyst material in half of the valves during the 1995 outage. The new disks with the catalyst material will reduce the large concentrations of oxygen in the pilot valve area. If this method fails to reduce the drift problem, the licensee has agreed to consider implementing the addition of additional pressure-sensing power actuation equipment.

During the current operating cycle, the licensee has determined that one SRV is leaking. However, leakage of either the main disk or the pilot disk is not expected to significantly affect the valve setpoint.

The staff has determined that the licensee has performed an adequate analysis and evaluation to demonstrate the acceptability of the requested one-time extension of the SRV surveillance intervals. Therefore, it is acceptable for the licensee to perform the SRV setpoint testing, disassembly and inspection required by TS 4.6.E.1, TS 4.6.E.2, and the LER 92-016 commitments during the refueling outage scheduled to begin January 1995. Thereafter, these surveillances will be required at the regular TS intervals.

2.3 Instrument Line Excess Flow Check Valves

Instrument piping which is connected to the reactor coolant system, and which exits the primary containment, is designed with excess flow check valves. The instrument piping ends at the instrument connection. Also, each line contains a 0.25-inch restricting orifice inside primary containment and a manual isolation valve upstream of the excess flow check valves. The end of each line is inside the reactor building, but outside primary containment. The excess flow check valves in these lines isolate reactor coolant in the event an instrument leaks or breaks. The valves isolate by closing on forward flow and thus prevent reactor coolant leakage outside the containment. There are 81 instrument line excess flow check valves installed in the Fitzpatrick plant.

Instrument line excess flow check valves are tested once each operating cycle to verify proper operation as required by current TS 4.7.D.1.b. The following table lists the test dates and results of this testing for the last 12 years:

Test Date	Test Results
08/02/80	100% passed
02/09/82	100% passed
08/23/83	100% passed
05/08/85	100% passed
04/02/87	100% passed
10/28/88	100% passed
06/02/90	97.5% passed
10/05/92	96.3% passed

Of the two valves that failed the acceptance criteria in 1990, one failed again in 1992. The repeated failure was experienced by valve 29EFV-34B. The licensee did not identify any industry data for these valves, but the plant data indicates that the valves have a high reliability. Even though a degrading trend may be developing, the low likelihood of failure of an instrument line and its associated excess flow check valve during the 14-to-16- month extension does not warrant shutdown to perform the overdue surveillance.

Excess flow check valve testing is performed in conjunction with the reactor pressure vessel (RPV) system leakage test near the end of each refueling outage to meet inservice inspection requirements of ASME Code, Section XI. The RPV system leakage test is performed at a pressure of approximately 1000 psig and demonstrates the integrity of the pressure vessel and the Class 1 primary piping. When the pressure is above 600 psig, the excess flow check valves testing can be completed in approximately 24 hours. Preparation time to perform valve lineups and to pressurize the reactor vessel is approximately 2 days. Depressurization and valve lineup restoration requires an additional day. Therefore, if the plant was shutdown to solely perform the overdue surveillances, testing the excess flow check valves would require approximately 4 days. The effort would be essentially the same to test only a percentage of the valves. RPV system leakage tests are neither planned nor required during the 3-week maintenance outages scheduled for September 1993 and April 1994. Without a one-time change to the TSs, the 3-week outages would have to be extended by approximately 4 days to complete the excess flow check valve testing.

Based on (1) the low failure rate of the valves, (2) the capability to isolate the lines with manual valves upstream of the excess flow check valves in the event an instrument experiences a problem, (3) the extensions to the maintenance outages that would be required to perform testing, and (4) the resumption of the regular schedule following the January 1995 refueling outage, the requested one-time extension of the current interval for the surveillance requirement of TS 4.7.D.1.b is acceptable. However, the licensee should, as a minimum, visually examine the condition of valve 29EFV-34B during the 3-week maintenance outage in September 1993.

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 (58 FR 36444). 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 Contributors:
P. Campbell
C. Hammer

Date: August 11, 1993

Mr. Jack Gray of your staff advised me during a telephone discussion on August 10, 1993, that this condition of the SE was acceptable. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 195 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

Distribution:
See attached sheet

*See previous concurrence

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