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

September 28, 1994

Docket No. 50-333

Mr. William J. Cahill, Jr.
Executive Vice President, Nuclear
Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Cahill:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M89368)

The Commission has issued the enclosed Amendment No. 217 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 20, 1989, as supplemented January 16, 1990, January 3, 1992, January 30, 1992, May 5, 1993, and May 26, 1993, and superseded March 2, 1994.

The amendment modifies the Safety/Relief Valve (SRV) performance limits. Specifically, the requested changes: (1) modify TS 2.2.1.B, and its associated Bases, to establish a single nominal SRV setpoint of 1110 psig; (2) modify TS 4.6.E, and its associated Bases, to increase the SRV setpoint tolerance to 3%; and (3) modify TSs 3.5.D, 3.6.E and 4.5.D, and their associated Bases, to allow for two SRVs (or Automatic Depressurization System (ADS) valves) to be inoperable during continuous power operation.

In addition, the amendment clarifies terminology, corrects typographical errors, removes a surveillance requirement which should have been deleted as part of Amendment No. 130, and deletes a duplicate specification. Specifically, the requested changes: (1) modify TS 1.2.1, and the Bases sections for TSs 3.6.E and 4.6.E, to clarify terminology; (2) modify TS 3.5.D.2, and the Bases sections for TSs 1.2 and 2.2, to correct typographical errors; (3) modify TS 4.2.B, Table 4.2-2, to correct an error made in Amendment No. 130 that failed to delete the requirement to perform logic functional testing on the ADS bellows pressure switch; (4) modify TS 4.5.D.1.b, to move the TS to 4.6.E.4, a new section, and clarify the requirements associated with SRV manual actuation testing; (5) modify Bases sections for TSs 3.6.E and 4.6.E, to move the Bases for the SRV manual actuation testing to the applicable sections; and (6) modify TS 4.6.E.4, to delete a duplicate specification pertaining to the annual report of SRV failures.

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

Sincerely,

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. 217 to DPR-59
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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DATED: September 28, 1994

AMENDMENT NO. 217 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
C. Miller, 14/A/4
M. Case
C. Vogan
J. Menning
K. Shembarger, 11/H/3
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
Wanda Jones, P-370
C. Grimes, 11/F/23
C. Hammer, 7/E/23
M. Razzaque, 8/E/23
A. D'Angelo, 8/D/1
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OPA
OC/LFMB
PD plant-specific file
C. Cowgill, Region I
R. Wessman, 7/E/23
T. Collins, 8/E/23
R. Barrett, 8/H/7

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. 217
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 20, 1989, as supplemented January 16, 1990, January 3, 1992, January 30, 1992, May 5, 1993, and May 26, 1993, and superseded March 2, 1994, 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. 217 , 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 prior to startup from the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael J. Case

Michael J. Case, Acting 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: September 28, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 217

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
27	27
29	29
80	80
119	119
120	120
128	128
142a	142a
143	143
152	152

1.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to limits on reactor coolant system pressure.

OBJECTIVE:

To establish a limit below which the integrity of the Reactor Coolant System is not threatened due to an overpressure condition.

SPECIFICATION:

1. The reactor vessel dome pressure shall not exceed 1,325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor coolant system safety limits from being exceeded.

OBJECTIVE:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATION:

1. The Limiting Safety System setting shall be specified below:

   A. Reactor coolant high pressure scram shall be $\leq$1,045 psig.

   B. At least 9 of the 11 reactor coolant system safety/relief valves shall have a nominal setting of 1110 psig with an allowable setpoint error of $\pm$3 percent.

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575 °F for the reactor vessel, 1148 psig at 568 °F for the recirculation suction piping and 1274 psig at 575 °F for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% x 1,250 = 1,375 psig) and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% x 1,150 = 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The limiting vessel overpressure transient event is a main steam isolation valve closure with flux scram. This event was analyzed within NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP," assuming 9 of the 11 SRVs were operable with opening pressures less than or equal to 1195 psig. The resultant peak vessel pressure for the event was shown to be less than the vessel pressure code limit of 1375 psig. (See current reload analysis for the reactor response to the main steam isolation valve closure with flux scram event). The value of 1195 psig is the SRV opening pressure up to which plant performance has been analyzed, assuming 2 SRVs are inoperable. Therefore, SRV opening pressures below 1195 psig ensure that the ASME Code limit on peak reactor pressure is satisfied.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

TABLE 4.2-2 (Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

	Logic System Functional Test		Frequency
1)	Core Spray Subsystem	(7) (9)	Once/6 months
2)	Low Pressure Coolant Injection Subsystem	(7) (9)	Once/6 months
3)	Containment Cooling Subsystem		Once/6 months
4)	HPCI Subsystem	(7) (9)	Once/6 months
5)	HPCI Subsystem Auto Isolation	(7)	Once/6 months
6)	ADS Subsystem	(7) (9)	Once/6 months
7)	RCIC Subsystem Auto Isolation	(7)	Once/6 months

NOTE: See notes following Table 4.2-5.

3.5 (cont'd)

D. Automatic Depressurization System (ADS)

1. The ADS shall be operable with at least 5 of the 7 ADS valves operable:

   a. whenever the reactor pressure is greater than 100 psig and irradiated fuel is in the reactor vessel, and

   b. prior to reactor startup from a cold condition.

4.5 (cont'd)

D. Automatic Depressurization System (ADS)

1. Surveillance of the Automatic Depressurization System shall be performed during each operating cycle as follows:

   a. A simulated automatic initiation which opens all pilot valves.

   b. A simulated automatic initiation which is inhibited by the override switches.

3.5 (cont'd)

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 ADS components, 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 below 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 two valves of the ADS are 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 two relief/safety valves of the ADS are inoperable, the HPCI System shall be verified to be operable immediately.

3.5 BASES (cont'd)

C. High Pressure Coolant Injection (HPCI) System

The High Pressure Coolant Injection System is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or Core Spray Systems can protect the core.

The HPCI meets this requirement without the use of a-c electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled and thus no clad damage occurs. Refer to Section 6.5.3 of the FSAR.

Low power physics testing and reactor operator training with inoperable component(s) will be conducted only when the HPCI System is not required, (reactor coolant temperature ≤212 °F andcoolant pressure ≤150 psig). If the plant parameters are below the point where the HPCI System is required, physics testing and operator training will not place the plant in an unsafe condition.

Operability of the HPCI System is required only when reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212 °F because core spray and low pressure coolant injection can protect the core for any size pipe break at low pressure.

D. Automatic Depressurization System (ADS)

The relief valves of the ADS are a backup to the HPCI subsystem. They enable the Core Spray or LPCI Systems to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the Core Spray or LPCI Systems. The core spray and/or LPCI provide sufficient flow of coolant to limit fuel clad temperatures to well below clad fragmentation and to assure that core geometry remains intact.

The ADS has sufficient excess capacity such that only five of the seven valves are required operable during power operation (see NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP").

Loss of three ADS valves reduces the pressure relieving capacity, and, thus, a 24 hour action to a cold condition with reactor pressures less than 100 psig is specified.

Low power physics testing and reactor operator training with inoperable components will be conducted only when that component or system is not required, (reactor coolant temperature ≤212 °F and reactor vessel vented or the reactor vessel head removed). With the reactor coolant temperature ≤212 °F and the Reactor vessel vented or the

3.6 (cont'd)

E. 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 at least 9 of 11 safety/relief valves shall be operable. The Automatic Depressurization System valves shall be operable as required by specification 3.5.D.

4.6 (cont'd)

E. Safety/Relief Valves

1. At least 5 of the 11 safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. All valves shall be tested every two operating cycles.* The testing shall demonstrate that the 11 safety/relief valves actuate at 1110 psig ±3%.

* 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. If Specification 3.6.E.1 is not met, the reactor shall be placed in a cold condition within 24 hours.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Specification 3.6.E.1 above, provided that reactor coolant temperature is $\leq$212 °F and the reactor vessel is vented or the reactor vessel head is removed.

4. The provisions of Specification 3.0.D are not applicable.

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. Manually open each safety/relief valve while bypassing steam to the condenser and observe a $\geq$10% closure of the turbine bypass valves, to verify that the safety/relief valve has opened. This test shall be performed at least once each operating cycle within the first 12 hours of continuous power operation at a reactor steam dome pressure of $\geq$940 psig.

* 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.

3.6 and 4.6 BASES (cont'd)

E. Safety/Relief Valves

The safety/relief valves (SRVs) have two modes of operation; the safety mode or the relief mode. In the safety mode (or spring mode of operation) the spring loaded pilot valve opens when the steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. The safety mode of operation is required during pressurization transients to ensure vessel pressures do not exceed the reactor coolant pressure safety limit of 1,375 psig.

In the relief mode the spring loaded pilot valve opens when the spring force is overcome by nitrogen pressure which is provided to the valve through a solenoid operated valve. The solenoid operated valve is actuated by the ADS logic system (for those SRVs which are included in the ADS) or manually by the operator from a control switch in the main control room or at the remote ADS panel. Operation of the SRVs in the relief mode for the ADS is discussed in the Bases for Specification 3.5.D.

Experiences in safety/relief valve testing have shown that failure or deterioration of safety/relief valves can be adequately detected if at least 5 of the 11 valves are bench tested once per operating cycle so that all valves are tested every two operating cycles. Furthermore, safety/relief valve testing experience has demonstrated that safety/relief valves which actuate within ±3% of the design pressure setpoint are considered operable (see ANSI/ASME OM-1-1981). The safety bases for a single nominal valve opening pressure of 1110 psig are described in NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP." The single nominal setpoint is set below the reactor vessel design pressure (1250 psig) per the requirements of Article 9 of the ASME Code - Section III, Nuclear Vessels. The setting of 1110 psig preserves the safety margins associated with the HPCI and RCIC turbine overspeed systems and the Mark I torus loading analyses. Based on safety/relief valve testing experience and the analysis referenced above, the safety/relief valves are bench tested to demonstrate that in-service opening pressures are within the nominal pressure setpoints ±3% and then the valves are returned to service with opening pressures at the nominal setpoints ±1%. In this manner, valve integrity is maintained from cycle to cycle.

The analyses with NEDC-31697P also provide the safety basis for which 2 SRVs are permitted inoperable during continuous power operation. With more than 2 SRVs inoperable, the margin to the reactor vessel pressure safety limit is significantly reduced, therefore, the plant must enter a cold condition within 24 hours once more than 2 SRVs are determined to be inoperable. (See reload evaluation for the current cycle).

A manual actuation of each SRV is performed to verify that the valves are mechanically functional and that no blockage exists in the valve discharge line. Adequate reactor steam dome pressure must be available to perform this test, in accordance with the manufacturer's recommendations, to avoid damaging the valve. Therefore, plant start-up is allowed and sufficient time is provided after the required pressure is achieved (940 psig) to perform this test.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the safety/relief and safety valves are

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 217 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 20, 1989, as supplemented by letters dated January 16, 1990, January 3, 1992, January 30, 1992, May 5, 1993, and May 26, 1993, and superseded March 2, 1994, 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 modify the Safety/Relief Valve (SRV) performance limits. Specifically, the requested changes would: (1) modify TS 2.2.1.B, and its associated Bases, to establish a single nominal SRV setpoint of 1110 psig; (2) modify TS 4.6.E, and its associated Bases, to increase the SRV setpoint tolerance to 3%; and (3) modify TSs 3.5.D, 3.6.E, and 4.5.D, and their associated Bases, to allow for two SRVs (or Automatic Depressurization System (ADS) valves) to be inoperable during continuous power operation.

In addition, miscellaneous changes were requested to several TSs to clarify terminology, correct typographical errors, remove a surveillance requirement which should have been deleted as part of Amendment No. 130, and delete a duplicate TS. Specifically, the requested changes would: (1) modify TS 1.2.1, and the Bases sections for TSs 3.6.E and 4.6.E, to clarify terminology; (2) modify TS 3.5.D.2, and the Bases sections for TSs 1.2 and 2.2, to correct typographical errors; (3) modify TS 4.2.B, Table 4.2-2, to correct an error made in Amendment No. 130 that failed to delete the requirement to perform logic functional testing on the ADS bellows pressure switch; (4) modify TS 4.5.D.1.b, to move the TS to 4.6.E.4, a new section, and clarify the requirements associated with SRV manual actuation testing; (5) modify Bases sections for TSs 3.6.E and 4.6.E, to move the bases for the SRV manual actuation testing to the applicable sections; and (6) modify TS 4.6.E.4, to delete a duplicate TS pertaining to the annual report of SRV failures. The miscellaneous proposed changes are administrative in nature.

2.0 DISCUSSION

The James A. FitzPatrick Nuclear Power Plant is designed with eleven SRVs, seven of which are also ADS valves.

Technical Specification 2.2.1.B currently requires the SRV nominal settings to be 1090 psig for two SRVs, 1105 psig for two SRVs and 1140 psig for the remaining seven SRVs. TS 4.6.E currently allows a setpoint error of +/- 1% for each SRV. The TSs assure that the structural acceptance criteria set forth in the Mark I Containment Short Term Program are satisfied.

TS 3.5.D currently requires all seven ADS valves to be operable during plant operation, with a contingency that plant operation can continue for 30 days with one ADS valve inoperable. The redundancy requirement assures adequate core spray and low pressure coolant injection flow in the event of a small pipe break coincident with a failure of the high pressure coolant injection system.

To reduce the number of forced outages and decrease maintenance and surveillance testing costs, the licensee has proposed: (1) to modify TS 2.2.1.B to establish a single nominal SRV setpoint of 1110 psig; (2) to modify TS 4.6.E, to increase the SRV setpoint tolerance to 3%; and (3) to modify TS 3.5.D to permit two ADS valves or SRVs to be inoperable whenever the reactor pressure is greater than 100 psig and irradiated fuel is in the reactor vessel, and prior to reactor startup from a cold shutdown. In addition, modifications to TS 3.6.E and to Bases Sections 2.2.1.B, 3.6.E, and 4.6.E, were proposed to support the changes proposed to TSs 2.2.1.B, 4.6.E, and 3.5.D.

The licensee has also proposed modifications, all administrative in nature, to several TSs associated with the SRVs. The following is a list of all the proposed changes:

1. TS 1.2.1: change the phrase "reactor coolant system pressure" to "reactor vessel dome pressure."

2. TS 2.2.1.B: change

   "Reactor coolant system safety/relief valve nominal settings shall be as follows:

   Safety/Relief Valves

   2 valves at 1090 psig
   2 valves at 1105 psig
   7 valves at 1140 psig

The allowable setpoint error for each safety/relief valve shall be +/- 1 percent."

to read:

"At least 9 of the 11 reactor coolant system safety/relief valves shall have a nominal setting of 1110 psig with an allowable setpoint error of ± 3 percent."

3. TS 3.5.D.1: replace with the following:

   "The ADS shall be operable with at least 5 of 7 ADS valves operable:

   a. whenever the reactor pressure is greater than 100 psig and irradiated fuel is in the reactor vessel, and

   b. prior to reactor startup from a cold condition."

4. TS 3.5.D.2: delete the "," after "100 psig."

5. TS 3.5.D.3: delete the cross-reference to action statements 3.5.D.1.a and 3.5.D.1.b and add "ADS." The revised TS reads:

   "Low power physics testing and reactor operator training shall be permitted with inoperable ADS components, provided that reactor coolant temperature is ≤ 212 °F and the reactor vessel is vented or reactor vessel head is removed."

6. TS 3.6.E.1: delete the words "Safety and" from the title, change the word "all" to "at least 9 of 11," and delete the phrase "except as specified by Specification 3.6.E.2." The revised TS shall read as follows:

   "During reactor power operation conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212 °F, the safety mode of at least 9 of 11 safety/relief valves shall be operable. The Automatic Depressurization System valves shall be operable as required by specification 3.5.D."

7. TS 3.6.E.2: delete specification.

8. TS 3.6.E.3: delete the cross-reference to TS 3.6.E.2 and renumber the TS to be 3.6.E.2.

9. TS 3.6.E.4: change the cross-reference from "Item B.2" to "Technical Specification 3.6.E.1" and renumber this specification to be 3.6.E.3.

10. TS 3.6.E.4 (New): add "The provisions of Technical Specification 3.0.D are not applicable."

11. TS 4.2.B, Table 4.2-2: delete item 8, "ADS Relief Valve Bellow Pressure Switch."

12. TS 4.5.D.1.b: move this specification to new Section 4.6.E.4 and add "This test shall be performed at least once each operating cycle within the first 12 hours of continuous operation at a reactor steam dome pressure of ≥ 940 psig."

13. TS 4.5.D.2: revise to read as follows:

    "A logic system functional test.

    a. When it is determined that two valves of the ADS are 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 two relief/safety valves of the ADS are inoperable, the HPCI System shall be verified to be operable immediately."

14. TS 4.6.E.1: delete the words "Safety and" from the title, change "one half of all" to "5 of the 11," delete cross-reference to TS 2.2.B, and add the revised valve actuation setpoints. The revised TS shall read as follows:

    "At least 5 of 11 safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. All valves shall be tested every two operating cycles. The testing shall demonstrate that the 11 safety/relief valves actuate at 1110 psig +/- 3%."

15. TS 4.6.E.4: delete the following:

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

In addition to the proposed modifications to the TS associated with the SRVs, the licensee has proposed changes to the associated the Bases sections.

3.0 EVALUATION

The licensee has proposed several changes, both technical and administrative in nature, to the TSs associated with the SRVs.

First, the licensee proposed technical changes to establish: (1) a single nominal SRV setpoint of 1110 psig; (2) a SRV setpoint tolerance of 3%; and (3) an allowance for two SRVs to be inoperable during continuous power operation. To justify the change, the licensee submitted two reports prepared by General Electric (GE): NEDC-31967P-Revision 1 and OPE 10-379. The reports present the results of licensing bases calculations showing that with the proposed modifications, vessel overpressurization limits and loss-of-coolant accident/emergency core cooling system (LOCA/ECCS) performance requirements are satisfied. The reports also show that the proposed changes do not have a significant impact on plant piping and containment structure performance. At the NRC staff's request, additional information on plant response to Anticipated Transient Without Scram (ATWS) events (NEDE-24222, "Assessment of BWR Mitigation of ATWS", Volume II), was provided by the licensee. The report shows that in the event of the most limiting ATWS event, the peak reactor vessel pressure will be within acceptable limits.

The following provides the details of the staff's review of the proposed changes:

Effects Of Proposed Changes On Reactor Vessel And ECCS

- The licensee's proposed SRV nominal setpoint (1110 psig) is set below the reactor vessel design pressure of 1250 psig, satisfying the requirements of Article 9 of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code-Section III, Nuclear Vessels. The setpoint is low enough to ensure high-pressure coolant injection and reactor core isolation coolant rated flow is still achievable and turbine overspeed margins are maintained. Also, since the proposed setpoint is within the range of the current staggered setpoints, the overall likelihood of inadvertent valve opening (from either downward setpoint drift or simmer problems) is not expected to change significantly. The staff finds the proposed SRV nominal setpoint and tolerance change has no significant impact on the reactor vessel and ECCS operation.

- NEDE-24222 concluded that for the most limiting ATWS event (main steam isolation valve closure, with all SRVs operable and Alternate Rod Insertion system failure) a peak reactor vessel pressure of 1296 psig will be reached. With two SRVs out-of-service, the nominal relief valve capacity will be reduced by approximately 18 percent. As a result, peak vessel pressure will increase by 137 psi to approximately 1433 psig. Since the peak vessel pressure is less than the peak pressure expected to occur under the most limiting ATWS event (1500 psig), the staff finds operation of the plant with two SRVs inoperable to be acceptable.

Effects Of Proposed Changes On Plant Piping And Containment Structures

- The staff reviewed the licensee's supporting technical evaluations which considered the effects of increased loads that will result from the simultaneous actuation of all SRVs. These evaluations analyzed the following loads:

  (a) SRV thrust loads on the main steam piping.
  (b) Loads on SRV discharge piping due to increased motion of main steam lines.
  (c) Water jet loads on submerged structures.
  (d) Air bubble drag loads.
  (e) Torus shell loads.
  (f) Torus support loads.
  (g) Torus attached piping loads.

  The plant main steam lines were analyzed in OPE 10-379 for the simultaneous actuation of all SRVs at 1140 psig. The analysis considered the increased motion of these lines and the attached SRV discharge piping by analyzing the steam line which the licensee determined is the most highly stressed.

  The licensee stated that the FitzPatrick Mark I Plant Unique Analysis Report (PUAR) calculated the effects of the simultaneous actuation of all SRVs at 1140 psig on the submerged structures, torus shell, torus supports and attached piping. This analysis demonstrates that the resulting loads will not cause the stresses in these components to exceed the allowable values. These stresses were determined by combining the SRV discharge loads with other appropriate loads including the safe shutdown earthquake loads.

  Thrust loads on SRV piping and T-quenchers were determined using the relief valve forced outage rate (RVFOR) computer program. After the analysis was performed, the licensee discovered an error in an equation which determines the water clearing thrust loads and associated stresses in the submerged SRV discharge piping. The effects of this error have been studied by the licensee, and the results of the tests at the Monticello plant were compared with loads predicted by a version of RVFOR with the error and a corrected version of RVFOR. These comparisons show that Monticello's test results are consistent with the loads predicted by the corrected version of RVFOR and that the version with the error overestimates these loads by 40% to 50%. A letter from GE to the licensee, dated May 25, 1984, indicates that the reduction of these loads is not constant and should be quantified by plant unique calculations if credit for their reduction is to be taken. Tests were also conducted at FitzPatrick and the resulting loads were compared to the RVFOR model results. As a result, it was determined that the loads predicted by RVFOR with the error are conservative for

FitzPatrick, and there is sufficient margin in these components to withstand the water clearing thrust loads at FitzPatrick as analyzed in the PUAR.

Based on the analysis performed for the plant piping and containment structures for the simultaneous discharge of all 11 SRVs at 1110 psig, the licensee has determined that the allowable stresses in these plant components will not be exceeded for the limiting combination of loads.

Based on the evaluation, the staff agrees that the analysis which the licensee has provided demonstrates the adequacy of the plant piping and containment structures for the proposed SRV setpoint and tolerance change. The licensee has shown that for the simultaneous discharge of all 11 SRVs at a single nominal setpoint of 1110 psig, the allowable stresses in these plant components will not be exceeded. The staff finds the proposed TS SRV setpoint change has no significant safety impact on plant piping and containment structures.

- A review of the original analysis on the impact of the increased SRV pressure setpoint on containment response shows that the setpoint used in the analysis was slightly higher than the proposed value. In addition, the most limiting drywell pressure transient was found to be the design basis LOCA rather than a SRV event, and the most limiting drywell temperature transient was found to be the main steam line break event. The licensee has determined that neither of these transients is affected by the increased SRV setpoint.

  The licensee's containment response analysis shows that the increased SRV setpoint is bounded by the original analyses. However, the proposed increase in setpoint tolerance will cause the maximum possible relief pressure to exceed the analyzed worst case by approximately 3 psi. Based on the assumptions used in determining the blowdown mass and energy release rate, the slight overshoot is considered to be insignificant relative to the containment temperature and pressure response. The staff concludes that the licensee's analysis of containment temperature and pressure response remains as the bounding analyses and is acceptable given the extremely small change in blowdown mass and energy input into the drywell via the SRVs. Therefore, the staff finds that the proposed TS SRV setpoint change has no significant impact on the containment.

- The licensee evaluated the effects of the increased SRV setpoint and tolerance on the containment hydrodynamic loads. The loads, documented in the PUAR, were applied to the containment and were found to be most affected by the fluid mass rate from the SRV. The PUAR assumed a mass flow rate of 291 lb/sec for each SRV at a setpoint of 1140 psig. The PUAR assumed that SRV load case A 1.2 and C 3.2 were the bounding cases with steam flow rates of 291 lbs/sec. The resultant hydrodynamic loads, which occur from an SRV discharge, are essentially proportional to the mass flow rate from the SRV.

The licensee has addressed the effect of increased SRV setpoints and the tolerance change upon the analyzed SRV flows and determined that the maximum flow rate is bounded by the assumed value of 291 lbs/sec. The staff finds the licensee's analysis acceptable based on the determination that the bounding case steam flow rates used in the PUAR are not exceeded by the proposed SRV setpoint change.

In addition to the proposed technical changes, the licensee proposed several administrative changes to the TSs, unrelated to SRV performance, to correct typographical errors, to remove a surveillance requirement which should have been deleted as part of Amendment No. 130, to clarify when SRV manual actuation is performed, and to delete a duplicate TS. The staff finds the changes: (1) do not involve a plant modification, (2) do not impact any procedural or administrative controls, and (3) do not involve a reduction in safety margin. Therefore, the staff finds the changes to be acceptable.

Based on the above evaluation, the staff concludes that there is no significant safety impact on vessel overpressure margin, ECCS/LOCA performance, plant piping, or containment structures due to operation with: (1) two SRVs out of service, (2) all 11 SRV setpoints set at a single nominal setpoint of 1110 psig, and (3) a setpoint tolerance of +/- 3%. Therefore, the staff finds these proposed changes, in addition to the miscellaneous changes which are administrative in nature, to be acceptable. The staff has no objections to the proposed changes to the TS Bases.

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

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on July 14, 1993 (58 FR 37972). By letter dated March 2, 1994, the licensee subsequently superseded the original application for amendment as supplemented. The application dated March 2, 1994, differed from the original application in that it eliminated a provision in the original application that would have allowed the use of an SRV setpoint upper limit as an acceptable means to reduce the number of License Event Reports. The environmental assessment and finding of no significant impact published on July 14, 1993, therefore, addressed all of the provisions of the application dated March 2, 1994.

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

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 Contributors:
C. Hammer
M. Razzaque
A. D'Angelo

Date: September 28, 1994

UNITED STATES NUCLEAR REGULATORY COMMISSION

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No.217 to Facility Operating License No. DPR-59 issued to Power Authority of the State of New York, which revised the Technical Specifications (TSs) for operation of the FitzPatrick Nuclear Power Plant located in Oswego County, New York.

The amendment modifies the Safety/Relief Valve (SRV) performance limits. Specifically, the requested changes: (1) modify TS 2.2.1.B, and its associated Bases, to establish a single nominal SRV setpoint of 1110 psig; (2) modify TS 4.6.E, and its associated Bases, to increase the SRV setpoint tolerance to 3%; and (3) modify TSs 3.5.D, 3.6.E and 4.5.D, and their associated Bases, to allow for two SRVs (or automatic depressurization system (ADS) valves) to be inoperable during continuous power operation.

In addition, the amendment clarifies terminology, corrects typographical errors, removes a surveillance requirement which should have been deleted as part of Amendment No. 130, and deletes a duplicate TS. Specifically, the requested changes: (1) modify TS 1.2.1, and the Bases sections for TS 3.6.E and 4.6.E, to clarify terminology; (2) modify TS 3.5.D.2, and the Bases sections for TSs 1.2 and 2.2, to correct typographical errors; (3) modify

TS 4.2.B, Table 4.2-2, to correct an error made in Amendment No. 130 that failed to delete the requirement to perform logic functional testing on the ADS bellows pressure switch; (4) modify TS 4.5.D.1.b, to move the TS to 4.6.E.4, a new section, and clarify the requirements associated with SRV manual actuation testing; (5) modify Bases sections for TSs 3.6.E and 4.6.E, to move the Bases for the SRV manual actuation testing to the applicable sections; and (6) modify TS 4.6.E.4, to delete a duplicate specification pertaining to the annual report of SRV failures.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on May 15, 1990 (55 FR 20228) and on May 25, 1994 (59 FR 27064). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment (EA) related to the action and has determined not to prepare an environmental impact statement. Based upon the EA, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action, see: (1) the application for amendment dated December 20, 1989, as supplemented by letters dated

January 16, 1990, January 3, 1992, January 30, 1992, May 5, 1993, and May 26, 1993 and superseded March 2, 1994, (2) Amendment No.217 to License No. DPR-59, and (3) the Commission's related Safety Evaluation and EA. These letters are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, D.C. and at the Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York.

Dated at Rockville, Maryland, this 28th day of September 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael J. Case

Michael J. Case, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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:

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

Enclosures:
1. Amendment No. 217 to DPR-59
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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*See previous concurrence

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