7/28/78

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Docket No. 50-333

Power Authority of the State
of New York
ATTN: Mr. George T. Berry
General Manager and
Chief Engineer
10 Columbus Circle
New York, New York 10019

Gentlemen:

The Commission has issued the enclosed Amendment No. 38 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 partial response to your application submitted by letter dated January 13, 1978.

This amendment revises the Technical Specifications by (1) revision of the specification relative to radiological protection to explicitly define the control for access to High Radiation Areas, (2) reduction of the squib charge batch size for surveillance testing of the Standby Liquid Control System and (3) miscellaneous, minor changes to correct errors in the current specifications.

Your letter dated January 13, 1978 included a proposal to reduce the frequency of the control rod scram time surveillance requirements. As indicated by our June 1, 1978 request for additional information, we have not yet completed our review of this item.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

/S/

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

GD
Const. 1

Enclosures:
1. Amendment No. 38 to License No. DPR-59
2. Safety Evaluation
3. Notice

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7/13/78

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DATE	7/11/78	7/11/78	7/21/78	7/27/78		7-13-78

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26 Federal Plaza
New York, New York 10007

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. 38
License No. DPR-59

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

   A. The application for amendment by the Power Authority of the State of New York (the licensee) dated January 13, 1978 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. 38, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 28, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. There are no changes on those pages marked with an asterisk (*).

Remove	Replace
11	11
33	33
34	34*
55	55*
56	56
57	57
105	105*
106	106
109	109
110	110*
246	246

Add page 256

2.1 (cont'd)

2. Reactor Water Low Level Scram Trip Setting (LL1)

Reactor low water level scram setting shall be $\geq$177 in. (+12.5 in. indicated level) above the top of the active fuel (TAF) at normal operating conditions.

3. Turbine Stop Valve Closure Scram Trip Setting

Turbine stop valve scram shall be $\leq$10 percent valve closure from full open when above 217 psig turbine first stage pressure.

4. Turbine Control Valve Fast Closure Scram Trip Setting

Turbine control valve fast closure scram on control oil pressure shall be set at 500 $<P<$850 psig.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

Main steam line isolation valve closure scram shall be $\leq$10 percent valve closure from full open.

6. Main Steam Line Isolation Valve Closure on Low Pressure

When in the run mode main steam line low presure initiation of main steam line isolation valve closure shall be $\geq$825 psig.

Amendment No. ~~14~~, ~~37~~, 38

subchannel. APRM's B, D and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, main steam isolation valve (MSIV) closure, and generator load rejection, turbine stop valve closure are discussed in Sections 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the Core and Containment Cooling Systems (ECCS) initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above normal levels that due to the nitrogen and oxygen radioactivity are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector offgas monitors which cause an isolation of the main condenser offgas line.

A Reactor Mode Switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (high flux in startup or refuel) System provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The IRM System provides protection against short reactor periods in these ranges.

The Control Rod Drive Scram system is designed so that all of the water which

is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 36 gal. of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 36 gal. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A Source Range Monitor (SRM) System is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph 7.5.4 FSAR). Thus, the IRM and APRM are required in the refuel and startup/hot standby modes. In the power range the APRM System provides required protection (reference paragraph 7.5.7 FSAR). Thus the IRM System is not required in the run mode. The APRM's cover only the power range. The IRM's and APRM's provide adequate coverage in the startup and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for startup and run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1-1 operable in the refuel mode assures that shifting to the refuel mode during reactor power operation does not diminish the protection provided by the Reactor Protection System.

Turbine stop valve closure occurs at 10 percent of valve closure. Below 217 psig turbine first stage pressure (30 percent of rated), the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the Core Cooling Systems, Control Rod Block and Standby Gas Treatment Systems. The objectives of the specifications are to assure the effectiveness of the protective instrumentation when required, even during periods when portions of such systems are out of service for maintenance, and to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2-1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177 in. above the top of the active fuel closes all isolation valves except those in Group 1. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level, this trip setting is adequate to prevent uncovering the core in the case of a break in the largest line assuming a 60 sec valve closing time. Required closing times are less than this.

The low-low reactor water level instrumentation is set to trip when reactor water level is 126.5 in. above the top of the active fuel (-38 in. on the instrument). This trip closes main

steam line isolation valves, main steam drain valves, recirc. sample valves (Group 1), initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18 in. above the top of the active fuel. This trip activates the remainder of the ECCS subsystems, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate ECCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10CFR100 will not be exceeded. For large breaks up to the complete circumferential break of a 24 in. recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating ECCS, it causes isolation of Groups B and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Specification 3.7 for isolation valve closure group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures peak at approximately 1,000° F and release of radioactivity to the environs is below 10CFR100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel exhaust duct and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Specification 3.7 for valve group. The setting is 40° F above maximum ambient for the main steam line tunnel detector. For large breaks, the high steam flow instrumentation is a

backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10CFR100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR.

Pressure instrumentation is provided to close the main steam isolation valves in the run mode when the main steam line pressure drops below 825 psig. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the run mode is less severe than the loss of feedwater analyzed in Section 14.5 of the FSAR, therefore, closure of the main steam isolation valves for thermal transient protection when not in the run mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

The trip settings of $\leq$300 percent of design flow for high flow and 40 °F above maximum ambient for high temperature are such that uncovering the core is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of $\leq$ 300 percent for high flow and 40° F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high flow temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that uncovering the core is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not de-

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without control rods.

Specification:

A. Normal Operation

During periods when fuel is in the reactor and prior to startup from a cold condition, the Standby Liquid Control System shall be operable except as specified in 3.4.B below. This system need not be operable when the reactor is in the cold condition, all rods are fully inserted and Specification 3.3.A is met.

4.4 SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The operability of the Standby Liquid Control System shall be verified by performance of the following tests:

1. At least once per month –

   Demineralized water shall be recycled to the test tank. Pump minimum flow rate of 39 gpm shall be verified against a system head of ≥1,275 psig.

2. At least once during each operating cycle –

   Manually initiate the system, except the explosive valves and

pump solution in the recirculation path.

Explode one of three primer assemblies manufactured in same batch to verify proper function. Then install the two remaining primer assemblies of the same batch in the explosive valves.

Demineralized water shall be injected into the reactor vessel to test that valves (except explosive valves) not checked by the recirculation test are not clogged.

Test that the setting of the system pressure relief valves is between 1,400 and 1,490 psig.

3. Disassemble and inspect one explosive valve so that it can be established that the valve is not clogged. Both valves shall be inspected in the course of two operating cycles.

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, and continued operation permitted, provided that:

1. The component is returned to an operable condition within 7 days.

B. Operation with Inoperable Components

When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

from overpressure. The pressure relief valves discharge back to the standby liquid control pump suction line.

B. Operation with Inoperable Components

Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the remaining system will perform its intended function and that the reliability of the system is good is obtained by demonstrating operation of the pump in the operable circuit at least once daily.

C. Sodium Pentaborate Solution

The solution saturation temperature of 17 percent sodium pentaborate, by weight, is 75°F. To guard against boron precipitation, the solution including that in the pump suction piping is kept at least 10°F above the saturation temperature by a tank heater and by heat tracing in the pump suction piping. The 10°F margin is included in Fig. 3.4-2. Temperature and liquid level alarms for the system are annunciated in the control room.

Pump operability is checked on a frequency to assure a high reliability of operation of the system should it ever be required.

Once the solution has been made up, boron concentration will not vary unless more boron or more water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

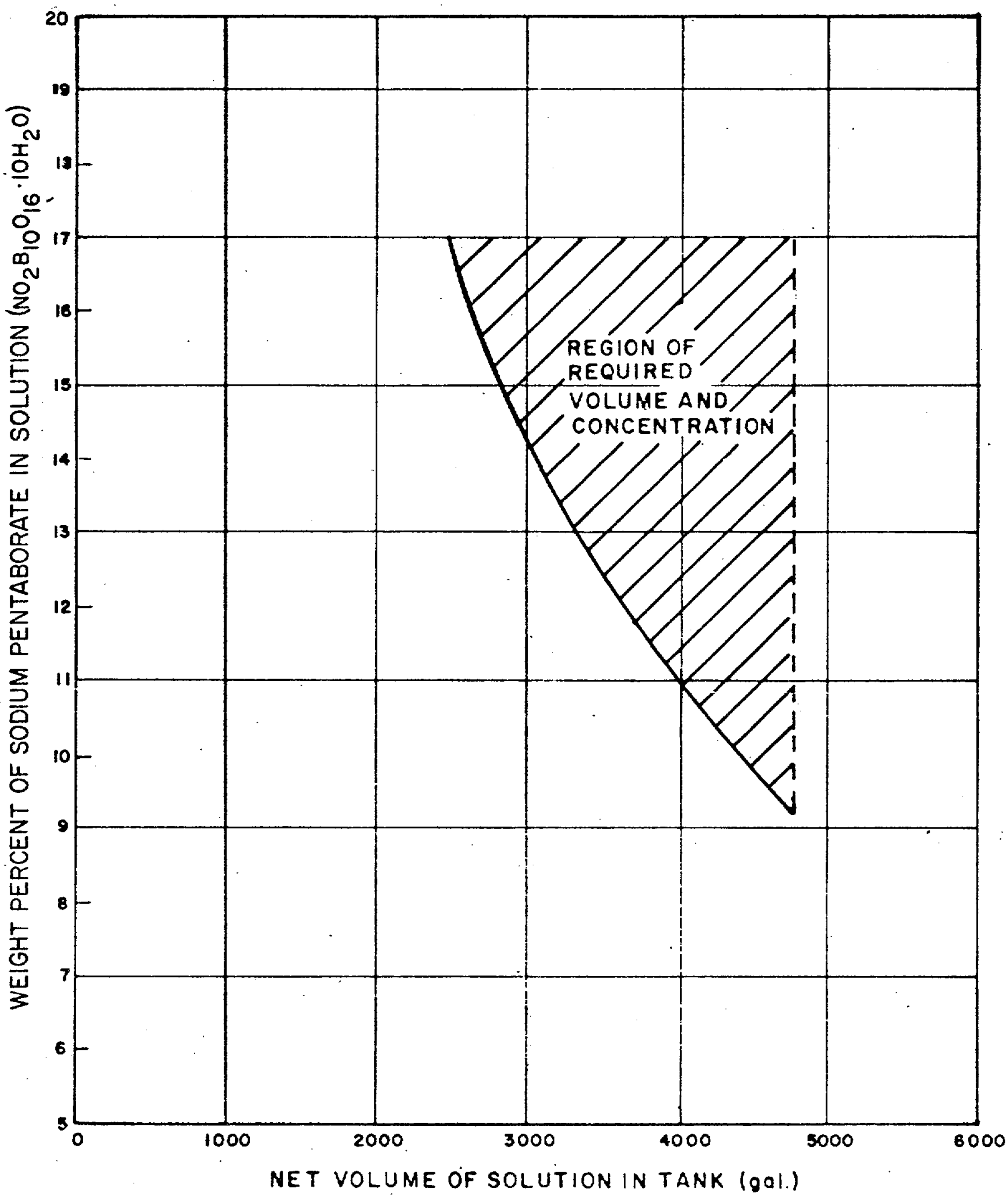

FIG. 3.4-1

SODIUM PENTABORATE SOLUTION VOLUME-CONCENTRATION REQUIREMENTS

B. The Keff of the spent fuel storage pool is ≤0.90 under normal conditions, and <0.95 during abnormal conditions as described in Section 9.3 of the FSAR.

5.6 SEISMIC DESIGN

The reactor building and all engineered safeguards are designed on basis of dynamic analysis using acceleration response spectrum curves which are normalized to a ground motion of 0.08 g, for the Operating Basis Earthquake, and 0.15 g, for the Design Basis Earthquake.

6.11 (A) High Radiation Area

1. In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c) (2) of 10 CFR 20, each High Radiation Area (i.e., $\geq$100 mrem/hr) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility in the Radiation Work Permit.

2. The requirements of 6.11.A.1 above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Radiation Protection and Radiochemistry Supervisor.

*Radiation Protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

INTRODUCTION

By letter dated January 13, 1978, the Power Authority of the State of New York proposed changes to the Technical Specifications appended to Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. Among the licensee's requested changes were: (1) revision of the specification relative to radiological protection to explicitly define the control for access to High Radiation Areas, (2) reduction of the squib charge batch size for surveillance testing of the Standby Liquid Control System and (3) miscellaneous, minor changes to correct errors in the current specifications.

EVALUATION

High Radiation Area Control

The licensee proposed to incorporate into the Technical Specifications an alternative to the control device or alarm signal specified in paragraph 20.203(c)(2) of 10 CFR 20. The revision requires that each High Radiation Area which is 1000 mrem/hr or less, be barricaded and conspicuously posted and that entrance thereto be controlled by issuance of a Radiation Work Permit. The licensee's proposal is consistent with staff guidance as specified in paragraph 6.12 of Standard Technical Specifications for General Electric Boiling Water Reactors, NUREG-0123, Rev. 1 and is therefore acceptable.

Standby Liquid Control System

The current specifications for surveillance of the Standby Liquid Control System (SLCS) requires that at least once during each operating cycle, two of six charges manufactured in the same batch will be exploded and the untested charges will be installed in the

system. The licensee's proposal would revise the requirement such that one of three charges would be exploded to verify proper function. We have reviewed the licensee's submittal and determined that the proposed technique provides an acceptable assurance of operability.

The licensee also proposed a minor revision to the Bases section of the SLCS to reflect the as-installed configuration of the relief line piping. As originally built, the relief line discharged to the test tank whcch contains demineralized water. To preclude inadvertent addition of borated water to the test tank, the licensee rerouted this line to the pump suction line such that borated water would be vented back to the standby liquid control solution tank. This revision is minor in nature, clarifies the as-installed configuration and is acceptable.

Miscellaneous Technical Specification Changes

The licensee proposed a number of changes regarding safety limit settings which are corrections of typographical errors. Each of these are identified below.

(1) The limiting safety system setting for reactor water low level scram trip is incorrectly designated as "less than" rather than "greater than" 177 in. (+12.5 in., indicated level) above the top of the active fuel ... .

(2) The Bases section for the reactor scram due to main steam line high radiation level incorrectly states that the setting is six times normal background rather than the specified ≤ 3 X normal full power background.

(3) The Bases section for the main steamline high flow setpoint for initiating primary containment isolation incorrectly states that the setpoint is $\leq 120\%$ of rated steam flow rather than $\leq 140\%$ of rated steam flow as specified in Table 3.2-1.

(4) The Design Features portion of the Technical Specifications incorrectly states that the design value for the Operating Basis Earthquake is 0.80g instead of 0.08g as supported by the FSAR, Section 2.6 and the Staff's Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant, dated November 20, 1972.

We have reviewed the licensee's submittal and determined that these changes constitute corrections of typographical errors, do not involve revision to limiting safety system settings, limiting conditions for operation or to surveillance requirements and are therefore acceptable as proposed.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 28, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 38 to Facility Operating License No. DPR-59, issued to Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the James A. FitzPatrick Nuclear Power Plant (the facility) located in Oswego County, New York. The amendment is effective as of its date of issuance.

This amendment revises the Technical Specifications by (1) revision of the specification relative to radiological protection to explicitly define the control for access to High Radiation Areas, (2) reduction of the squib charge batch size for surveillance testing of the Standby Liquid Control System and (3) miscellaneous minor changes to correct errors in the current specifications.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated January 13, 1978, (2) Amendment No. 38 to License No. DPR-59, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oswego County Office Building, 46 East Bridge Street, Oswego, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 28 day of July 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors