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January 6, 1983

Docket No. 50-333

Mr. Leroy W. Sinclair
President and Chief Operating Officer
Power Authority of the States of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Sinclair:

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to Technical Specifications in response to your request dated February 20, 1981, as supplemented by letters dated November 18, 1981 and February 19, 1982.

The amendment revises the definition of rated loop recirculation flow and extends the power/flow operating envelope within previously analyzed limits to provide more operating flexibility during power ascension and reduction operations.

A copy of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosure: as stated
1. Amendment No. 72
2. Safety Evaluation
3. Notice

cc:
See next page

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DATE	12/15/82	12/17/82	12/20/82	12/20/82	12/ /82		

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
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 February 20, 1981, as supplamented by letters dated November 18, 1981 and February 19, 1982, 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 1:

   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. 72 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 6, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 72

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise the Appendix "A" Technical Specifications as follows:

Remove	Replace
vii	vii
6	6
8	8
10	10
12	12
15	15
23	23
41	41
43	43
73	73

LIST OF FIGURES

surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted $\pm$ 25 percent. The interval as pertaining to instrument and electric surveillance shall never exceed one operating cycle. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

U. Thermal Parameters

1. Minimum critical power ratio (MCPR)-Ratio of that power in a fuel assembly which is calculated to cause some point in that fuel assembly to experience boiling transition to the actual assembly operating power as calculated by application of the GEXL correlation (Reference NEDE-10958).

2. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. The design LHGRis 13.4 KW/ft for 8X8, 8X8R and P8X8R bundles.

3. Maximum Fraction of Limiting Power Density- The Maximum Fraction of Limiting Power Density (MFLPD) is the hightest value existing in the core of the Fraction of Limiting Power Density (FLPD).

4. Transition Boiling - Transition boiling means the boiling region between nucleate and film boiling. Transition boiling is the region in which both nucleate and film boiling occur intermittently with neither type being completely stable.

V. Electrically Disarmed Control Rod

To disarm a rod driven electrically, the four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred. Electrical disarming does not eliminate position indication.

W. High Pressure Water Fire Protection System

The High Pressure Water Fire Protection System consists of: a water source and pumps; and distribution system piping with associated post indicator valves (isolation valves). Such valves include the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler or water spray subsystem.

X. Staggered Test Basis

A Staggered Test Basis shall consist of:

a. A test schedule for a system, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

Y. Rated Recirculation Flow

That drive flow which produces a core flow of 77.0×10^6 lb/hr.

1.1 (cont'd)

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 785 psig)

When the reactor pressure is $\leq$ 785 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 (cont'd)

A.1.b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

APRM - The APRM flux scram setting shall be $\leq$ 15 percent of rated neutron flux with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

c. APRM Flux Scram Trip Settings (Run Mode)

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be:

$$S \leq 0.66\ W + 54\%$$

where:

S = Setting in percent of rated thermal power (2436 MWT)

W = Recirculation flow in percent of rated

For no combination of recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

A.1.d. APRM Rod Trip Setting

The APRM Rod block trip setting shall be:

$$S \leq 0.66\ W + 42\%$$

where:

S = Rod block setting in percent of thermal power (2436 MWT)

W = Recirculation flow rate in percent of rated

In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 42\%)\left(\frac{FRP}{MFLPD}\right)$$

where:

FRP = fraction of rated thermal power (2436 MWT)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8X8, 8X8R and P8X8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

1.1 BASES

1.1 FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR> 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

A. Reactor Pressure >785 psig and Core Flow> 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variable, i.e., the operating domain. The current load line limit analysis contains the current operating domain map. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the MCPR operating limits specified for the normal operating conditions in specification 3.1.B, more than 99.9 % of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are

BASES

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the FitzPatrick Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition 2535 MWt. The analyses were based upon plant operation in accordance with the operating map given in the current load line limit analysis. In addition, 2436 is the licensed maximum power level of FitzPatrick, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Fuel cladding integrity is assured by the operating limit MCPR's for steady state conditions given in Specification 3.1.B. These operating limit MCPR's are derived from the established fuel cladding integrity Safety Limit, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient.

The most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO. The type of transients evaluated were increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit, the required operating limit MCPR of Specification 3.1.B is obtained.

The evaluation of a given transient begins with the system initial parameters shown in the current reload analysis and reference 2 that are input to a core dynamic behavior transient computer program described in references 1 and 3. The output of these programs along with the initial MCPR form the input for the further analyses of the thermally limited bundle with a single channel transient thermal hydraulic code. The principal result of the evaluation is the reduction in MCPR caused by the transient.

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TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
3	IRM High Flux	≤120/125 of full scale	X	X		8 Instrument Channels	A
3	IRM Inoperative		X	X		8 Instrument Channels	A
2	APRM Neutron Flux-Startup(15)	≤15% Power	X	X		6 Instrument Channels	A
2	APRM Flow Referenced Neutron Flux (12) (13) (14) (Not to exceed 117%)	$S \leq (0.66W+54\%) \times \left[\frac{FRP}{MFLPD}\right]$			X	6 Instrument Channels	A or B
2	APRM Fixed High Neutron Flux (14)	≤120% Power			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B

Amendment No. ~~10~~, ~~30~~, ~~43~~ 72

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (cont'd)

C. High Flux IRM

D. Scram Discharge Volume High Level

E. APRM 15% Power Trip

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

9. The APRM downscale trip is automatically bypassed when the IRM Instrumentation is operable and not high.

10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.

11. See Section 2.1.A.1

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP).

where:

FRP = Fraction of rated thermal power (2436 MWT)

MFLPD = Maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8X8, 8X8R and P8X8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

W = Recirculation flow in percent of rated

S = Scram setting in percent of initial

13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3 (cont'd)

The APRM and RRM rod blocks need not be operable in start-up mode. From and after the time it is found that the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. From and after the time it is found that the first column cannot be met for both trip systems, the systems shall be tripped.

2. IRM downscale is bypassed when it is on its lowest range.

3. This function is bypassed when the count is $\geq$ 100 cps.

4. One of the four SRM inputs may be bypassed.

5. This SRM Function is bypassed when the IRM range switches are on range 8 or above.

6. The trip is bypassed when the reactor power is $\leq$ 30%.

7. This function is bypassed when the Mode Switch is placed in Run.

8. S = Rod Block Monitor Setting in percent of initial.

   W = Recirculation flow in percent of rated

   K = Intercept values of 39%, 40%, 41% and 42% can be used with appropriate MCPR Limits from Section 3.1.B.

9. When the reactor is subcritical and the reactor water temperature is less than 212°F, the control rod block is required to be operable only if any control rod in a control cell containing fuel is is not fully inserted.

10. When the control rod block function associated with scram discharge instrument volume high water level is not operable when required to be operable, the trip system shall be tripped.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 72 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

In a letter dated February 20, 1981, the Power Authority of the State of New York (the licensee) requested an amendment to Appendix A of Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant (Reference 1). The proposed changes were supplemented by letters of November 18, 1981 (Reference 4) and February 19, 1982 (Reference 5). The amendment would allow use of redefined rated recirculation flow which corresponds to the core flow used in the analysis of safety limits. Also the changes would permit the licensee to operate within an extended power/flow operating envelope bounded by a new operating line (References 2 and 3) above the present Average Power Range Monitor (APRM) rod blockline. However, any rod movement would still be limited to the present APRM rod blockline. This amendment would provide more operating flexibility during power ascension and reduction operations while maintaining the plant within design basis and previously established safety limits.

2.0 Discussion

2.1 Refinition of Recirculation Flow

The present Technical Specifications (TS) define rated recirculation flow as 34.2×10^6 lb/hr. However, the safety analyses of transients and accidents employ rated recirculation flow defined as that value which produces a core flow of 77.0×10^6 lb/hr. Consequently, the present TS definition of 34.2×10^6 lb/hr rated recirculation flow is more conservative than that employed in the safety analysis, since the recirculation flow rate, as currently defined, is greater than that necessary to produce a rated core flow of 77.0×10^6 lb/hr. Therefore, the licensee has proposed to redefine recirculation flow as that recirculation flow which produces a rated core flow of 77.0×10^6 lb/hr. The licensee would use the conservative recirculation flow value of 34.2×10^6 lb/hr at the beginning of the operating cycle, and then adjust the value based on actual measurement as appropriate during the cycle.

2.2 Extended Power/Flow Operating Line

Operating flexibility during power ascension in proceeding from a low power/low core-flow condition to a high power/high core-flow condition is affected by many factors, including the power/flow ratios defined by the power/flow curve. The power/flow curve is the locus of power from a fixed rod pattern as a function of flow from which the occurrence of postulated transients will yield results within analyzed and acceptable limits. Operation of the FitzPatrick Plant utilizing the power flow curves is described in Reference 6. The restrictions imposed by the power/flow operator envelope assure acceptable pressure and thermal margins during postulated transients. This requires an analysis of abnormal operating transients with degraded scram reactivity characteristics which are dependent on fuel exposure.

The licensee has proposed to extend the upper bound of the power/flow operating envelope as follows: the proposed power/flow line would follow the new APRM Rod Blockline (0.58W + 50%) up to an intercept point of 85% power/61% flow, and then a linear path to 100% power/94% flow (100/94), followed by constant 100% power to the 100/100 point.

As referenced to in the licensee's analysis, the "new APRM Rod Blockline" (0.58W + 50%) defines the new upper bound of the operating region and the rod block intercept point (85/61). However, the actual APRM Rod Blockline remains the same as in the present TS. The new upper bound and present APRM rod blocks are defined as follows:

For the new upper bound of the power/flow operating region and rod block intercept point,

$$S = 0.58W + 50\%$$

For the APRM Rod Blockline,

$$S \leq 0.66W + 42\%$$

where,

S = Setting in percent of rated power (2435 MWT)

W = Recirculation flow in percent of rated.

The intercept point of 85% power/61% flow establishes the highest power level permitted when operating on the new upper bound. This will be sufficiently high to provide the desired operational flexibility during power ascension, but low enough to ensure adequate safety margin from the analysis limit of the 104% power/100% flow point. Abnormal operational transients have been analyzed for the planned operational conditions up to the thermal power conditions of 104% rated thermal power (2533 MWT).

As an added conservatism, the actual APRM Rod Blockline is below the proposed new upper bound.

The licensee has performed an evaluation in support of the proposed change which is summarized below:

1. The scram reactivity insertion characteristics were determined for end-of-cycle (EOC) conditions, at 104/100, 85/61, and 100/94 power flow points (References 2 and 3). These values were used in analyzing the most limiting abnormal operational transients: load rejection without bypass, turbine trip without bypass, feedwater controller failure, loss of feedwater heating, rod withdrawal error, and high pressure coolant injection. Each transient analysis for Reload 3/Cycle 4 (References 2 and 3) and Reload 4/Cycle 5 (Reference 5) shows that the limiting transient was the load-rejection-without-bypass at the 104/100 point and that the existing operating limits for minimum critical power ratio (MCPR) were applicable to the new power/flow line and are therefore acceptable within the new operating envelope. Thus, the licensing basis values still remain the most limiting values.

2. Compliance with the ASME pressure vessel code was verified (Reference 2) for all main steamline isolation valve (MSIV) closures with flux scram. Again, the limiting condition occurs at the 104/100 point with the peak vessel bottom pressure of 1264 psig. The ASME Boiler and Pressure Vessel code limit is 1375 psig.

3. A reanalysis of the rod withdrawal accident based on the proposed new upper bound was conducted with acceptable results. The actual rod block during operations will occur with less rod withdrawal (i.e., at 0.66 W + 42%), which is conservative.

4. ECCS analysis verified the applicability of the extended region for Cycles 3, 4, and 5 (References 2, 3, and 5).

5. The Minimum Critical Power Ratio (MCPR) requirements along the proposed new upper bound (Reference 2) show small increases in the MCPR requirements at lower load conditions during abnormal transients and are therefore acceptable. This trend is consistently demonstrated and the increments are approximately same for 7 x 7, 8 x 8, 8x 8R fuels.

6. Thermal hydraulic analysis was performed for the proposed new limiting power/flow line (References 2, 3). The decay ratios determined from the limiting reactor core stability conditions show that, at the intersection of natural circulation and extrapolated rod blocklines, the ratios were 0.76, 0.85, and 0.87 for Reload 2/Cycle 3, Reload 3/Cycle 4, and Reload 4/Cycle 5 respectively, well within the bound of the ultimate performance criteria of 1.0.

At the most responsive intersection of natural circulation and extrapolated rod blocklines, the channel performance calculations for Cycle 3 and Cycle 4 yielded decay ratios of 0.23 and 0.22 respectively for 7 x 7 fuel channels, 0.39 and 0.38 respectively for 8 x 8 fuel channels, and 0.34 and 0.30 respectively for 8 x 8R fuel channels. Reload Core 4 is composed of only 8 x 8, 8 x 8R and P8 x 8R fuel channels, and their respective decay ratios are 0.37, 0.30, and 0.30. Again, they are well below the ultimate performance criteria of 1.0. In addition, operation in the natural circulation

mode above 60% power level (as the most responsive conditions) is not a normal mode of operation. Furthermore, the intersection of extrapolated rod blocklines and the natural circulation line is outside of the operating bound. The reactor core stability conditions are therefore acceptable.

3.0 Evaluation

3.1 Redefinition of Recirculation Flow

We have reviewed the information provided by the licensee in support of the proposed modification. We find that the proposed modification is consistent with the analysis employed in the JAFNPP Final Safety Analysis Report. We also find that the proposed change has been evaluated for its impact on the flow-biased APRM rod block and trip setpoints. For a nominal rated recirculation value of 33 x 10^6 lb/hr which produces 77 x 10^6 lb/hr, APRM Flux Scram and Rod Block Trip Settings are approximately 3% lower than by using the present TS rated flow of 34.2 x 10^6 lb/hr. Therefore, the proposed change would be within the bound of the analysis. In order to maintain an equal margin of safety as given in the previous analysis, the rated recirculation flow would be calibrated conservatively against the rated core flow of 77 x 10^6 lb/hr.

In addition, conservatism was incorporated in the transient analysis in estimating the controlling factors such that the void coefficient used was about 25% greater than the nominal maximum value expected during the core lifetime, and analyzed active core flow was 88% of total core flow.

On the basis of the foregoing considerations, the change proposed by the licensee is acceptable.

3.2 Extended Power/Flow Operating Line

We have reviewed the information provided by the licensee in support of this proposed modification. We find that the licensee has conducted appropriate load-line limit analyses to verify:

1. Reactor Core Stability at the highest power/lowest flow point:

2. That the highest power/lowest flow point is a more limiting condition than any other condition within the expanded operating envelope;

3. The impact on Emergency Core Cooling System (ECCS) performance;

4. Abnormal operational transients for points along the proposed power/flow curve;

5. Scram Reactivity Insertion capabilities.

The licensee has also provided a safety analysis demonstrating that transients and accidents more severe than those analyzed at the 104/100 point will not occur during operation within or along the proposed power/flow line.

Based on our review, we find that the change proposed by the licensee will allow reactor power ascension to proceed safely along the modified power/flow line, and is therefore, acceptable.

4.0 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.

5.0 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 an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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: January 6, 1982

Principal Contributors: J. Chung; J. Hegner

6.0 References

1. PASNY letter to USNRC dated February 20, 1981, proposed changes to Technical Specifications.

2. NEDO-24243 "General Electric Boiling Water Reactor Load Line Limit Analysis for James A. FitzPatrick Nuclear Power Plant, February 1980.

3. NEDO-24243, Supplement 1, "General Electric Boiling Water Reactor Load Line Limit Analysis for James A. Fitzpatrick Nuclear Power Plant Cycle 4", July 1980.

4. PASNY letter to USNRC dated November 18, 1981, Changes to the Proposed Technical Specifications for Reload 4/Cycle 5.

5. PASNY letter to USNRC dated February 19, 1982, Proposed Changes to the Technical Specifications for Reload 4/Cycle 5 - Revision 1.

6. Final Safety Analysis Report for the James A. FitzPatrick Nuclear Power Plant; Section 3.7; Docket No. 50-333.

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. 72 to Facility Operating License No. DPR-59 issued to the Power Authority of the State of New York (the licensee), which revised the 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 the date of issuance.

The amendment revises the definition of rated loop recirculation flow and extends the power/flow operating envelope within previously analyzed limits to provide more operating flexibility during power ascension and reduction operations.

The application for 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 the amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the 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 the amendment.

For further details with respect to this action, see (1) the application for amendment dated February 20, 1981, as supplemented by letters dated November 18, 1981 and February 19, 1982 (2) Amendment No. 72 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 Penfield Library, State University College of Oswego, 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 Licensing.

Dated at Bethesda, Maryland this 6th day of January, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing