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February 26, 1982

Docket No. 50-333

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

Dear Mr. Sinclair:

The Commission has issued the enclosed Amendment No. 67 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application dated February 8, 1982.

The amendment establishes new vessel level setpoints that are consistent with the installation of a common reference level required by TMI Action Item II.K.3.27 in NUREG-0737.

By this action, we consider Action Item II.K.3.27 to be complete for your facility.

We have not, however, performed an evaluation of your water level instrumentation with respect to the human factors aspects. That evaluation will be performed as part of the detailed Control Room review (Action Item I.D.1) that you are expected to conduct per NUREG-0737.

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

Sincerely,

Philip J. Polk, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 67 to DPR-59
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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SURNAME	SNorris	PPolk	DVassallo	TNovak		JVanVliet
DATE	2/23/82	2/24/82	2/24/82	2/24/82	2/25/82	2/24/82

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Power Authority of the State
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cc:

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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. 67
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 dated February 8, 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 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. 67 , 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

for

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 26, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 67

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A Technical Specifications as follows:

Remove	Insert
41a	41a
-	41b
43	43
64	64
65	65
66	66
67	67
71	71
76	76
76a	76a
77	77
102	102

TABLE 3.1-1 (Cont'd)

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		
2	APRM Downscale	≥2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	≤ 1045 psig	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤2.7 psig	X(7)	X(7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥12.5 in. indicated level (≥ 177 in. above the top of active fuel)	X	X	X	4 Instrument Channels	A
2	High Water Level in Scram Discharge Volume	≤36 gal	X(2)	X	X	4 Instrument Channels	A
2	Main Steam Line High Radiation	≤3 x normal full power background	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤10% valve closure	X(3)(5)	X(3)(5)	X(5)	8 Instrument Channels	A

TABLE 3.1-1 (Cont'd)

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		
2	Turbine Control Valve Fast Closure	500<P<850 psig Control oil pressure between fast closure solenoid and disc dump valve			X(4)	4 Instrument Channels	A or C

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (cont'd)

C. High Flux IRM

D. Scram Discharge Instrument Volume High Level when any control rod in a control cell containing fuel is not fully inserted

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 MW/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 - Loop Recirculation flow in percent of rated (rated is 34.2×10^6 lb/hr)

S_n - Scram setting in percent of initial

13. The Average Power Range Monitor scram function is varied (Figure 1.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.

TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (2)
2 (6)	Reactor Low Water Level	≥12.5 in Indicated Level (≥177 in. above the top of active fuel)	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low Water Level	≥ -38 in. indicated level (≥ 126.5 in. above the top of active fuel)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	≤2.7 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	≤3 x Normal Rated Full Power Background	4 Inst. Channels	B
2	Low Pressure Main Steam Line	≥ 825 psig (7)	4 Inst. Channels	B
2	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Leak Detection High Temperature	≤ 40°F above max ambient	4 Inst. Channels	B
3	Reactor Cleanup System Equipment Area High Temperature	≤ 40°F above max ambient	6 Inst. Channels	C
2	Low Condenser Vacuum closes MSIV's	≥ 8" Hg. Vac (8)	4 Inst. Channels	B

TABLE 3.2-1 (Cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

NOTES FOR TABLE 3.2-1

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. From and after the time it is found that the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have main steam lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate shutdown cooling.

3. Deleted

4. Deleted

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in run mode (interlocked with Mode Switch).

8. Bypassed when reactor pressure is less than 1005 psig and turbine stop valves are closed.

TABLE 3.2-2

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Item No.	Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design For Both Trip Systems	Remarks
1	2	Reactor Low-Low Water Level	$\geq$-38 in. indicated level ($\geq$126.5 in. above the top of active fuel)	4 HPCI & RCIC Inst. Channels	Initiates HPCI, RCIC & SGTS.
2	2	Reactor Low-Low-Low Water level	$\geq$ -146.5 in. indicated level ($\geq$ 18 in. above the top of active fuel)	4 Core Spray & RHR Instrument Channels	Initiates Core Spray, LPCI, and Emergency Diesel Generators.
				4 ADS Instrument Channels	Initiates ADS in conjunction with confirmatory low level, High Drywell Pressure, 120 second time delay and LPIC or Core Spray pump discharge pressure interlock.
3	2	Reactor High Water Level	$\leq$+58 in. indicated level ($\leq$222.5 in. above the top of active fuel)	2 Inst. Channels	Trips HPCI and RCIC Turbines
4	1	Reactor Low Level (inside shroud)	$\geq$ +352 in. above vessel zero($\geq$ 0 in above the top of active fuel)	2 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition

TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Item No.	Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Remarks
5	2	Containment High Pressure	1< P< 2.7 psig	4 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
6	1	Confirmatory Low Level	≥ 12.5 in. indicated level (≥ 177 in. above the top of active fuel)	2 Inst. Channels	ADS Permissive.
7	2	High Drywell Pressure	≤ 2.7 psig	HPCI Inst. Channels	Initiates Core Spray LPCI, HPCI & SGTS.
				4 RHR & Core Spray Inst. Channels	Initiates starting of Diesel Generators
8	2	Reactor Low Pressure	≥ 450 psig	4 Inst. Channels	Permissive for opening Core Spray and LPCI Admission valves.

TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT

COOLING SYSTEMS

NOTES FOR TABLE 3.2-2

1. Whenever any ECCS subsystem is required by specification 3.5 to be operable, there shall be two operable trip systems. From and after the time it is found that the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the cold condition within 24 hours.

2. Deleted

3. Refer to Technical Specification 3.5.A for limiting conditions for operation, failure of one (1) instrument channel disables one (1) pump.

TABLE 3.2-6

SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	No. of Channels Provided by Design
2	Reactor Level (Note 3)	Indicator 0 - +60 (164.5 to 224.5 in. above the top of active fuel)	5
	Reactor Level (Note 4)	Recorder 0 - +60 (164.5 to 224.5 in. above the top of active fuel)	
1	Reactor Level	Indicator -150 - +60 (14.5 to 224.5 in. above the top of active fuel)	2
2	Reactor Pressure (Note 5)	Indicator 0-1200 psig	5
	Reactor Pressure (Note 6)	Recorder 0-1200 psig	
1	Drywell Pressure (Narrow Range)	(Narrow Range) Indicator Recorder 10 - 19 psia	2
	Drywell Pressure (Wide Range)	(Wide Range) Indicator Recorder 0 - 100 psia	

TABLE 3.2-6 (Cont'd)

SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	No. of Channels Provided by Design
2	Drywell Temperature	Indicator 50 - 250° F	4
	Drywell Temperature	Recorder 50 - 350° F	
2	Suppression Chamber Temperature	Indicator 50 - 250° F	4
	Suppression Chamber Temperature	Recorder 50 - 350° F	

TABLE 3.2-7

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minumum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action
1	Reactor High Pressure	$\geq$ 1120 psig	4	(2)
1	Reactor Low-Low Water Level	$\geq$ -38 in. indicated level ($\geq$126.5 in. above the top of active fuel	4	(2)

Notes for Table 3.2-7

1. Whenever the reactor is in the run mode, there shall be one operable trip sytem for each parameter for each operating recirculation pump. From and after the time it is found that this cannot be met, the indicated action shall be taken.

2. Reduce power and place the Mode Selector Switch in a Mode other than the Run Mode within 24 hours.

3.3 and 4.3 BASES (cont'd)

rods have been withdrawn (e.g., groups A_{12} and A_{34}, it is demonstrated that the Group Notch made for the control drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 20% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCB occurs and subsequently at appropriate stages of the control rod insertion.

4. The Source Range Monitor (SRM) System performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per sec. assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transient cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of squattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR limits as shown in specification 3.1.B). During use of such patterns, it is judged that testing of the RBM System prior to withdrawal of such rods to assure its operability will assure that improper withdraw does not occur. It is the responsibility of the Reactor Analyst to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other qualified personnel may perform this function.

C. Scram Insertion Times

The Control Rod System is designated to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit. Scram insertion time test criteria of Section 3.3.C.1 were used to generate the generic scram reactivity curve shown in NEDE-24011-P-A. This generic curve was used in analysis of non-pressurization transients to determine MCPR limits. Therefore, the required protection is provided.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 67 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

Author: J. Van Vliet
P. Polk

1.0 Introduction

By letter dated February 8, 1982 the Power Authority of the State of New York (the licensee) forwarded a proposed Technical Specification change that establishes revised vessel level setpoints that are consistent with a new common instrument zero level. The proposed common reference level which is the top of the enriched fuel is 352" above the vessel zero level. By an earlier submittal dated November 25, 1981 the licensee described the method by which the reference level would be changed. Establishment of the common zero level for all reactor vessel level instrumentation is called for in accordance with TMI Action Item II.K.3.27 in NUREG-0737.

2.0 Evaluation

We have reviewed each of the proposed revised setpoints and find them to be consistent with the previously established safety settings. The licensee has proposed operating with dual, color coded water level scales for fuel Cycle 5. During Cycle 5, the operating and emergency procedures, which refer to reactor vessel water level, will be systematically revised to include the level above the top of active fuel. As the procedures are revised, the operators will receive appropriate training. During the next refueling outage (reload 5), when all procedure revisions and operator training are complete, the old scales will be removed. In sum, the licensee has committed to revise procedures as appropriate and to conduct operator training. The required changes to operating and emergency procedures will be entered prior to operating with the new setpoints installed, and operator retraining will be performed.

Since no change in actual water level for any function is involved in the proposed Technical Specification revisions, and since no instrumentation is being changed, we find the proposed Technical Specification revisions acceptable for use.

3.0 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 Section 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.

4.0 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 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 26, 1982

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. 67 to Operating License No. DPR-59 issued to the Power Authority of the State of New York (the licensee), which revises 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 establishes new vessel level setpoints that are consistent with the installation of a common reference level required by TMI Action Item II.K.3.27 in NUREG-0737.

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 8, 1982, (2) Amendment No. 67 to License No. DPR-59, and (3) the Commission's related Safety Evaluation. 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 26th day of February 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Vernon L. Rooney

Vernon L. Rooney, Acting Chief
Operating Reactors Branch #2
Division of Licensing