JUNE 14 1978

Docket No. 50-333

Distribution
Docket
ORB #3
Local PDR
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VStello
BGrimes
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SSheppard
TIppolito
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DEisenhut
ACRS (16)
OPA (CMiles)
DRoss
TBAbernathy
JRBuchanan
RDiggs

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. 37 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 applications submitted by letters dated July 20, 1977 and May 18, 1978.

This amendment revises the Technical Specifications by (1) revising the Table of Minimum Test and Calibration Frequency for Primary Containment Isolation System, (2) revising the "keep-full" instrumentation sensors on core spray and residual heat removal system high point vents, (3) increasing the LPCI Pump Discharge Pressure Interlock setpoint, (4) lowering the setpoint for Main Steamline Isolation Valve Closure, and (5) deleting the Respiratory Protection Program.

Your letter dated July 20, 1977 included a proposal to change the failure mode of the reactor building/suppression chamber vacuum breaker air operated valves. We have not yet completed our review of this item.

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

Sincerely,

Original signed by

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

Const. 1
GD

Enclosures and ccs:
See page 2

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	ORB #3	PSB	RSB	OELD	ORB #3	
SURNAME	*SSheppard/Verrelli	*	*	*	TIppolito	
DATE	6/6/78	6/13/78	6/13/78	6/ 7/78	6/14/78	

Distribution
Docket JRBuchanan
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Docket No. 50-333

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George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

OELD
WOlmstead
6/7/78

Enclosures and ccs:
See page 2

OFFICE			ORB #3	PSB	RSB	ORB #3
SURNAME			Sheppard/Verrelli		FDCoffman	GLear
DATE			6/6/78	6/13/78	6/13/78	6/ /78

Enclosures:
1. Amendment No. 37
2. Safety Evaluation
3. Notice

cc w/enclosures:

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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. 37
License No. DPR-59

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

    A. The applications for amendment by the Power Authority of the State of New York (the licensee) dated July 20, 1977 and May 18, 1978 comply 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. 37, 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

T A Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 14, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 37

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
iii	iii
11	11
20	20
57	57
64	64
68	68
77	77*
78	78
123	123
255	255
256	(deleted)
257	(deleted)
258	258
262	(deleted)
263	(deleted)
264	(deleted)

TABLE OF CONTENTS (CONT'D)

2.1 (cont'd)

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

Reactor low water level scram setting shall be $\leq$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.

2.1 BASES (Cont'd)

The low pressure isolation of the main steam lines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the Reactor Mode Switch be in the Startup position where protection of the fuel cladding integrity safety limit is provided by the APRM high neutron flux scram and the IRM. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at ≤10 percent valve closure, there is no increase in neutron flux.

6. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation minimum limit at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitute an unsafe condition.

C. References

1. Linford, R.B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor, "NEDO-10802, Feb., 1973.

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 6 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-

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 Channels	Action (2)
2 (6)	Reactor Low Water Level	≥12.5 Indicated Level (3)	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 (4)	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 Tunnel Exhaust Duct High Temperature	≤ 40° F above max ambient	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
1	Reactor Cleanup System High Temperature	≤ 140°F	1 Inst. Channel	C
2	Low Condenser Vacuum closes MSIV's	≥8" Hg. Vac (8)	4 Inst. Channels	B

Amendment No. ~~19~~, 37

TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

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 Channels	Remarks
1	Auto Blowdown Timer	120 sec ± 5 sec	2 Timers	In conjunction with Low Reactor Water Level High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown.
2	RHR (LPCI) Pump Discharge Pressure Interlock	125 psig±20 psig	4 Channels	Defers ADS actuation pending confirmation of low pressure core cooling system operation.
2	Core Spray Pump Discharge Pressure Interlock	100 psig±10 psig	4 Channels	LPCI or Core Spray Pump running interlock.
1	RHR (LPCI) Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.
1	HPCI Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.

Amendment No. ~~3~~, 37

TABLE 3.2-7

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum 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	≥ 1120 psig	4	(2)
1	Reactor Low-low Water Level	≥ -38 in. indicated level	4	(2)

Notes for Table 3.2-7

1. Whenever the reactor is in the run mode, there shall be one operable trip system 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.

TABLE 4.2-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

Instrument Channel (5)	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2) Reactor Low-Low Water Level	(1)	Once/3 months	Once/day
3) Main Steam High Temp.	(1)	Once/operating cycle	Once/day
4) Main Steam High Flow	(1)	Once/3 months	None
5) Main Steam Low Pressure	(1)	Once/3 months	None
6) Reactor Water Cleanup High Temp.	(1)	Once/3 months	None
7) Condenser Low Vacuum	(1)	Once/operating cycle	None

Logic System Functional Test (4) (6)	Frequency
1) Main Steam Line Isolation Valves Main Steam Line Drain Valves Reactor Water Sample Valves	Once/6 months
2) RHR - Isolation Valve Control Shutdown Cooling Valves Head Spray	Once/6 months
3) Reactor Water Cleanup Isolation	Once/6 months
4) Drywell Isolation Valves TIP Withdrawal Atmospheric Control Valves Sump Drain Valves	Once/6 months
5) Standby Gas Treatment System Reactor Building Isolation	Once/6 months

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.5.1, 3.5.2 and 3.5.3. If at any time it is determined that the limiting value for APLHGR is being exceeded action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI, RCIC, or Core Spray System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI, RCIC, and Core Spray shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested every month.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

(B) The fol.. ving records shall be reta ed for the duration of the Facility Operating License:

1. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

2. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

3. Records of facility radiation and contamination surveys.

4. Records of radiation exposure for all individuals entering radiation control areas.

5. Records of gaseous and liquid radioactive material released to the environs.

6. Records of transient or operational cycles for those facility components identified in Table 6.10-1.

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of Quality Assurance activities required by the Quality Assurance Manual.

10. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

11. Records of meetings of the PORC and the SRC.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared and adhered to for all plant operations. These procedures shall be formulated to maintain radiation exposures received during operation and maintenance as far below the limits specified in 10 CFR 20 as practicable. The procedures shall include planning, preparation, and training for operation and maintenance activities. They shall also include exposure allocation, radiation and contamination control techniques, and final debriefing.

6.12 INDUSTRIAL SECURITY PROGRAM

(A) An industrial security program shall be maintained throughout the life of the plant in accordance with the provisions of the Plant Security Plan. Annual review of the Plant Security Plan shall be performed by the Plant Operating Review Committee and the Safety Review Committee.

6.13 EMERGENCY PLAN

(A) A Site Emergency Plan shall be maintained throughout the life of the plant in accordance with the provisions of 10 CFR 50, Appendix E.

(B) Site evacuation exercises will be conducted annually utilizing applicable provisions contained within the Emergency Plan. The exercise shall involve coordination with offsite support groups and include communication checks.

(C) The Emergency Plan and implementing procedures shall be reviewed on an annual basis by the PORC and SRC.

6.14 FIRE PROTECTION PROGRAM

6.14.A An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified licensee personnel or an outside fire prtoection firm.

6.14.B An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 37 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

I. INTRODUCTION

By letters dated July 20, 1977 and May 18, 1978, the Power Authority of the State of New York (licensee) 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 requests, which are supported by this Safety Evaluation are (1) corrections of errors in the Table of Minimum Test and Calibration Frequency for Primary Containment Isolation System (PCIS), by adding condenser low vacuum and deleting reactor high water level, (2) revise the surveillance associated with "keep-full" instrumentation for core spray and residual heat removal systems (RHR) by indicating that sensors are level switches instead of pressure switches, (3) increase the Low Pressure Coolant Injection (LPCI) discharge pressure set point from 50 to 125 psig, (4) lower the set point for the Main Steamline Isolation Valve (MSIV) pressure closure from 850 psig to 825 psig. In the amendment the Commission has deleted from the Technical Specifications, the respiratory protection program in accordance with the revocation provisions of the current specifications.

II. EVALUATION

PCIS Instrumentation

Tables 3.2-1 and 4.2-1 of the current Technical Specifications list the Instrumentation that Initiates Primary Containment Isolation and the Minimum Test and Calibration Frequency in PCIS respectively. Condenser Low Vacuum is a PCIS function (as listed in Table 3.2-1) but was inadvertently omitted from Table 4.2-1 at the time of original issuance of the specifications. Reactor High Water Level is not a PCIS function but is erroneously listed in Table 4.2-1. Thus the addition of Condenser Low Vacuum and the deletion of Reactor High Water level from Table 4.2-1 is acceptable based on the staff's determination that this change is an editorial correction to the Technical Specifications as currently written.

Keep-Full Instrumentation

The current specifications for FitzPatrick require that the discharge piping from the pump discharge for the Core Spray, Low Pressure Coolant Injection (LPCI) mode of operation of the RHR, High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems be maintained filled with water. If the discharge piping of these systems is not filled, a water hammer could develop in this piping when the pump or pumps are started.

The current surveillance requirement associated with this specification identifies pressure switches as the type of instrumentation monitoring the Core Spray and LPCI (RHR) discharge lines. The licensee stated that level switches had been installed on the Core Spray and LPCI systems high point vents and proposed changes in the text to indicate that level switches instead of pressure switches are the type of instrumentation monitoring these lines. We conclude that the licensee's proposed change is acceptable. We have previously determined (Amendment 14 to DPR-57 for Hatch Unit 1) that the use of float switches, located at high points in the discharge piping provides a method for monitoring the condition of the piping which is just as effective as the alternate method which makes use of pressure switches.

LPCI Pump Discharge Pressure Interlock

The LPCI pump discharge pressure interlock is provided to inhibit actuation of the Automatic Depressurization System (ADS) in the event of a malfunction of the LPCI system to develop sufficient pressure to inject coolant during a postulated loss of coolant accident. A parallel logic interlock is provided in the Core Spray System. The licensee proposed to increase the interlock set point for the LPCI system from 50 ± 9 psig to 125 ± 20 psig to remove a superfluous indication when the RHR System is being used in the shutdown cooling mode of operation. We have reviewed the licensee's submittal and agree with his conclusion that the increase in set point is of no consequence to the function of the interlock. A revised set point of 125 $\pm$ 20 psig is still substantially less than the RHR pump design pressure of 450 psig such that the inhibit to ADS actuation in the event of LPCI failure is retained. Thus, we conclude that the proposed change is acceptable.

MSIV Pressure Set Point

The licensee's application dated May 18, 1978 proposed a reduction in the MSIV low main steamline set point closure from 850 psig to 825 psig. The licensee stated that the reduction of the set point was desirable to prevent unnecessary isolation of the main steamlines during pressure transients which he has experienced following a reactor scram from power. Main steamline isolation in such a situation results in the loss of normal plant pressure control and forces decay heat removal by means of relief valves.

The purpose of the automatic closure of the MSIVs due to a low main steamline pressure is to provide protection against rapid depressurization of the reactor vessel through the steam lines and the resultant fast cooldown of the reactor vessel. Such an event could occur if the turbine pressure regulator, which normally controls steam pressure at the inlet to the turbine between 920 psig and 950 psig, failed in such a manner that the turbine bypass valves stuck in the open position.

In support of the proposed change, the licensee provided an evaluation of the effect of the lower set point on rapid depressurization and on steamline break accident analysis. The licensee also referenced Edwin I. Hatch Nuclear Plant Unit 1 (50-321) submittal, dated October 9, 1975 which provided a bounding analysis for a reduction in the main steamline low pressure setpoint from 880 psig to 825 psig.

The proposed change in the set point to 825 psig, a reduction of 25 psig, would result in an increase of only 3°F in the reactor vessel (coolant) thermal transient in the event of an abnormal operational transient caused by a turbine pressure regulator failure.

The value of 3°F is conservatively based on the difference in the saturation temperature of steam at 850 psig and 825 psig and does not take credit for the effects of the additional heat energy stored in the reactor vessel metal; the stored heat energy in the vessel would act to reduce the magnitude of the thermal transient (cooldown). The potential thermal shock effects on the reactor vessel which could occur as a result of the proposed change are insignificant when compared to the effects of reactor vessel coolant cooldown (or heatup) at the maximum allowable rate of 100°F in any one hour period as specified in the FitzPatrick Technical Specifications.

The proposed reduction in the set point for MSIV closure on low steamline pressure would not significantly affect the maximum pressure differential experienced across the core internals as a result of a turbine pressure regulator failure. The maximum pressure differential experienced across the core internals during such an abnormal operational transient is primarily a function of the turbine bypass steam flow capacity, which is not affected by the proposed change, rather than the duration of the transient which could be slightly increased (approximately 3 seconds) as a result of the proposed change.

Since the MSIV low pressure closure trip setting is not relied upon for protection against a postulated steamline break, the proposed change would not result in a significant increase in the consequences of such an event. Mitigation of postulated steamline breaks at FitzPatrick is provided by the following instrumentation systems:

a. In the event of a large steamline break outside of primary containment, differential pressure switches would detect high main steamline flow and would initiate closure of the MSIVs.

b. In the event of a small steamline break outside of primary containment, temperature detectors located in the steam tunnel surrounding the main steamline from the drywell to the turbine building would sense high air temperature and would initiate closure of the MSIV's.

Based on the staff's review of the licensee's submittal, the Hatch 1 analysis and other previously approved reductions in MSIV pressure set points (Amendments 29, 26, and 4 to DPR-33, DPR-52 and DPR-68 for Browns Ferry Nuclear Generating Plant, Units 1, 2 and 3) we conclude that the lower set point would not have significant effects on previously analyzed transients and is acceptable.

Respiratory Protection Program

By our letter dated July 28, 1977, we advised the licensee of an amended Section 20.103 of 10 CFR 20 which became effective on December 29, 1976. The licensee was further advised that one effect of the revision is that in order to receive credit for use of respiratory protection equipment at their facility after December 28, 1977, such use must be as stipulated in Regulatory Guide 8.15 rather than as specified in the current Technical Specifications for FitzPatrick. Since the license is deemed to contain and is subject to the conditions specified in 10 CFR 20 and in the absence of written objection from the licensee, this amendment executes the revocation provision of the current specifications on respiratory protection by deleting Section 6.11.B.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in authorized 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, negative declaration, or 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: June 14, 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. 37 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.

The amendment revised the Technical Specifications by (1) revising the Table of Minimum Test and Calibration Frequency for Primary Containment Isolation System, (2) revising the "keep-full" instrumentation sensors on core spray and residual heat removal system high point vents, (3) increasing the LPCI pump discharge pressure interlock setpoint, (4) lowering the set point for Main Steamline Isolation Valve Closure, and (5) deletion of the Respiratory Protection Program.

The applications for the amendment comply 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 applications for amendment dated July 20, 1977 and May 18, 1978, (2) Amendment No. 37 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 14th day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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