May 31, 1989

Docket No. 50-333

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Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NOS. 54533, 61632, AND 51678)

The Commission has issued the enclosed Amendment No. 130 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Specifications in response to your application transmitted by letter dated December 6, 1984, as supplemented and superseded (in part) by letters dated October 18, 1985 and October 20, 1986.

The amendment incorporates changes in response to the acceptance criteria and guidance of Generic Letters 83-02 and 83-36, "NUREG-0737 Technical Specifications." The following items are addressed by this amendment: Limit Overtime (I.A.1.3), Radiation Signal on Purge Valves (II.E.4.2.7), RCIC Restart and RCIC Suction (II.K.3.13 and II.K.3.22), Report Safety and Relief Valve Failures and Challenges (II.K.3.3), Post Accident Sampling (II.B.3), Noble Gas Effluent Monitors (II.F.1.1), Sampling and Analysis of Plant Effluents (II.F.1.2), Containment High-Range Monitor (II.F.1.3), Containment Pressure Monitor (II.F.1.4), Containment Water Level Monitor (II.F.1.5), and Containment Hydrogen Monitor (II.F.1.6).

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

Sincerely,

Original signed by

David E. LaBarge, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 130 to DPR-59
2. Safety Evaluation

cc: w/enclosures
See next page

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[TACS 54533, 61632, 51678 AMEND]

						SRXB	PRPB	
						WMHodges	JCunningham	
						5/3/89	5/26/89	
OFC	:PDI-1	:PDI-1	:OGC	:PDI-1	:SPLB	: ECEB	: SICB	
NAME	:CVogan	:DLaBarge:vr	: w/ minor changes	:RCapra	for JCraig	: CMcCracken	: SNewberry	
DATE	: 4/24/89	: 4/24/89	: 5/30/89	: 5/31/89	: 4/27/89	: 5/1/89	: / /89	

OFFICIAL RECORD COPY

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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. 130
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated December 6, 1984, as supplemented and superseded (in part) by letters dated October 18, 1985 and October 20, 1986, 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.130 , 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 and shall be implemented within 30 days thereafter.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects, I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 31, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 130

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

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iv	iv
v	v
vi	vi
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54	54
60	60
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---	77a
---	77b
---	86a
121	121
142a	142a
142b	142b
143	143
143a	143a
143b	143b
177	177
211	211
---	211a
212	212
213	213
247	247
---	247a
254b	254b
---	258e

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3.2 LIMITING CONDITIONS FOR OPERATION

3.2 INSTRUMENTATION

Applicability:

Applies to the plant instrumentation which either (1) initiates and controls a protective function, or (2) provides information to aid the operator in monitoring and assessing plant status during normal and accident conditions.

Objective:

To assure the operability of the aforementioned instrumentation.

Specifications:

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2-1.

B. Core and Containment Cooling Systems - Initiation and Control

The limiting conditions for operation for the instrumentation that initiates or controls the Core and Containment Cooling Systems are given in Table 3.2-2. This instrumentation must be oper- able when the system(s) it initiates or

4.2 SURVEILLANCE REQUIREMENTS

4.2 INSTRUMENTATION

Applicability:

Applies to the surveillance requirement of the instrumentation which either (1) initiates and controls protective function, or (2) provides information to aid the operator in monitoring and assessing plant status during normal and accident conditions.

Objective:

To specify the type and frequency of surveillance to be applied to the aforementioned instrumentation.

Specifications:

A. Primary Containment Isolation Functions

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-1.

System logic shall be functionally tested as indicated in Table 4.2-1.

B. Core and Containment Cooling Systems - Initiation and Control

Instrumentation shall be functionally tested, calibrated, and checked as indicated in Table 4.2-2.

3.2 (cont'd)

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-5.

F. Surveillance Information Readouts

The limiting conditions for the instrumentation that provide(s) surveillance information readouts are given in Table 3.2-6.

G. Recirculation Pump Trip

The limiting conditions for operation for the instrumentation that trip(s) the recirculation pumps as a means of limiting the consequences of a failure to scram during an anticipated transient are given in Table 3.2-7.

H. Accident Monitoring Instrumentation

The limiting conditions for operation of the instrumentation that provides accident monitoring are given in Table 3.2-8.

I. 4 kv Emergency Bus Undervoltage Trip

The limiting conditions for operation for the instrumentation that prevents damage to electrical equipment or circuits as a result of either a degraded or loss-of-voltage condition on the emergency electrical buses are given in Table 3.2-2.

4.2 (cont'd)

E. Drywell Leak Detection

Instrumentation shall be calibrated and checked as indicated in Table 4.2-5

F. Surveillance Information Readouts

Instrumentation shall be calibrated and checked as indicated in Table 4.2-6

G. Recirculation Pump Trip

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-7.

System logic shall be functionally tested as indicated in Table 4.2-7.

H. Accident Monitoring Instrumentation

Instrumentation shall be demonstrated operable by performance of a channel check and channel calibration as indicated in Table 4.2-8.

3.2 BASES (cont'd)

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report, NEDO-10349, dated March, 1971.

Accident monitoring instrumentation provides additional information which is helpful to the operator in assessing plant conditions following an accident by (1) providing information needed to permit the operators to take preplanned manual actions to accomplish safe plant shutdown; (2) determining whether systems are performing their intended functions; (3) providing information to the operators that will enable them to determine the potential for a breach of the barrier to radioactivity release and if a barrier has been breached; (4) furnishing data for deciding on the need to take unplanned action if an automatic or manually initiated safety system is not functioning properly or the plant is not responding properly to the safety systems in operation; and (5) allowing for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any problem. This instrumentation has been upgraded to conform with the acceptance criteria of NUREG-0737 and NRC Generic Letter 83-36.

The Emergency Bus Undervoltage Trip System transfers the 4 kv emergency electrical buses to the Emergency Diesel Generators in the event an undervoltage condition is detected. The system has two levels of protection: (1) degraded voltage protection, and (2) loss-of-voltage protection. Degraded voltage protection prevents a sustained low voltage condition from damaging safety-related equipment. The degraded voltage protection has two time delays. A short time delay coincident with a loss-of-coolant accident (LOCA) and a longer time delay to allow normal plant evolutions without unnecessarily starting the Emergency Diesel Generators. * The loss-of-voltage protection prevents a more severe voltage drop from causing a long term interruption of power. Time delays are included in the system to prevent inadvertent transfers due to spurious voltage decreases. Therefore, both the duration and severity of the voltage drop are sensed by the Emergency Bus Undervoltage Trip System.

* Modification approved for cycle 9 only

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
22	2	Condensate Storage Tank Low Level	≥ 59.5 inches above tank bottom (= 15,600 gal. avail)	2 Inst. Channels	Transfers RCIC pump suction to suppression chamber
23					
24					
25	1	Core Spray Sparger to Reactor Pressure vessel d/p	≤ 0.5 psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break
26	2	Condensate Storage Tank Low Level	≥ 59.5 in. above tank bottom (= 15,600 gal avail)	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.
27	2	Suppression Chamber High Level	≤ 6 in. above normal level	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.
28	1	RCIC Turbine Steam Line High Flow	≤ 282 in. H_2O psid	2 Inst. Channels	Close Isolation Valves in RCIC Subsystem.

TABLE 3.2-8

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	NO. OF CHANNELS PROVIDED BY DESIGN	MINIMUM NO. OF OPERABLE CHANNELS REQUIRED	ACTION	MEASUREMENT RANGE
1. Stack High Range Effluent Monitor (17RM-53A&B)	2	1	B	10^{-1} to 10^{7} mR/hr.
2. Turbine Building Vent High Range Effluent Monitor (17RM-434A&B)	2	1	B	10^{-1} to 10^{7} mR/hr.
3. Radwaste Building Vent High Range Effluent Monitor (17RM-463A&B)	2	1	B	10^{-1} to 10^{7} mR/hr.
4. Containment High Range Radiation Monitor* (17RM-104A&B)	2	1	A	1 to 10^{8} R/hr.
5. Containment Pressure (wide range 27PT-115A2&B2) (narrow range 27PT-115A1&B1)	2 wide range 2 narrow range	1 1	A A	0 to 250 psig -5 to +5 psig
6. Drywell Level (23LT-203A1/A2 & 23LT-203B1/B2)	2	1	A	22 to 100 ft. (H_2O)
7. Suppression Pool Level (23LT-202A&B)	2	1	A	1.7 to 27.5 ft. (H_2O)
8. Reactor Vessel Pressure (06PT-61A&B)	2	1	A	0 to 1500 psig
9. Drywell Hydrogen Concentration Monitor (27PCA-101A&B)	2	1	A	0 to 30% H_2

* At less than or equal to 450 R/hr, closes vent and purge valves

TABLE 3.2-8 (cont'd)

ACCIDENT MONITORING INSTRUMENTATION

NOTES FOR TABLE 3.2-8

A. With the number of operable channels less than the required minimum, either restore the inoperable channels to operable status within 30 days, or: (1) initiate an alternate method of monitoring the appropriate parameter(s), or (2) be in a cold condition within the next 24 hours.

B. With the number of OPERABLE channels less than required by the minimum channels OPERABLE requirements, initiate the alternate method of monitoring the appropriate parameter(s) within 72 hours and: 1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or 2) prepare and submit a Special Report to the Commission within 14 days following the event outlining the cause of the inoperability, the action taken, and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.2-8

MINIMUM TEST AND CALIBRATION FREQUENCY FOR ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	INSTRUMENT FUNCTIONAL TEST	CALIBRATION FREQUENCY	INSTRUMENT CHECK
1. Stack High Range Effluent Monitor 17RM-53A, B	Once/Operating Cycle	Once/Operating Cycle	Once/day
2. Turbine Building Vent High Range Effluent Monitor 17RM-434A, B	Once/Operating Cycle	Once/Operating Cycle	Once/day
3. Radwaste Building Vent High Range Effluent Monitor 17RM-463A, B	Once/Operating Cycle	Once/Operating Cycle	Once/day
4. Containment High Range Radiation Monitor 17RM-104A, B	Once/Operating Cycle	Once/Operating Cycle	Once/day
5. Containment Pressure Transmitter Wide Range 27PT-115A2, B2 Narrow Range 27PT-115A1, B1	N/A	Once/Operating Cycle	Once/day
6. Drywell Level Transmitter 23LT-203A1/A2 and 23LT-203B1/B2	N/A	Once/Operating Cycle	Once/day
7. Suppression Pool Level Transmitter 23LT-202A, B	N/A	Once/Operating Cycle	Once/day
8. Reactor Vessel Pressure Channel 06PT-61A, B	N/A	Once/Operating Cycle	Once/day
9. Drywell Hydrogen Concentration Analyzer 27PCA-101A, B	N/A	Once/Operating Cycle	Once/day

3.5 (Cont'd)

E. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212^{o}F except from the time that the RCIC System is made or found to be inoperable for any reason, continued reactor power operation is permissible during the succeeding 7 days unless the system is made operable earlier provided that during these 7 days the HPCI System is operable.

2. If the requirements of 3.5.E cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hours.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.5.E.2 above, provided that reactor coolant temperature is $\leq$ 212°F.

4.5 (Cont'd)

E. Reactor Core Isolation Cooling (RCIC) System

1. RCIC System testing shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within ten days of continuous operation from the time steam becomes available.

Item	Frequency
a. Simulated Automatic Actuation (and Restart*) Test	Once/operating cycle
b. Pump Operability	Once/month
c. Motor Operated Valve Operability	Once/month
d. Flow Rate	Once/3 months
e. Testable Check Valves	Tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.
f. Logic System Functional Test	Once/operating cycle

* Automatic restart on a low water level signal which is subsequent to a high water level trip.

3.6 (cont'd)

E. <u>Safety and Safety/Relief Valves</u>

1. During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212°F, the safety mode of all safety/relief valves shall be operable, except as specified by Specification 3.6.E.2. The Automatic Depressurization System Valves shall be operable as required by Specification 3.5.D.

4.6 (cont'd)

E. <u>Safety and Safety/Relief Valves</u>

1. At least one half of all safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. The safety/relief valve settings shall be set as required in Specification 2.2.B. All valves shall be tested every two operating cycles.

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2. a. From and after the date that the safety valve function of one safety/relief valve is made or found to be inoperable, continued operation is permissible only during the succeeding 30 days unless such valve is made operable sooner.

   b. From and after the time that the safety valve function on two safety/relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 7 days unless such valves are sooner made operable.

3. If Specification 3.6.B.1 and 3.6.B.2 are not met, the reactor shall be placed in a cold condition within 24 hours.

4. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Item B.2 above, provided that reactor coolant temperature is $\leq 212°F$ and the reactor vessel is vented or the reactor vessel head is removed.

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

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

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3.7 (cont'd)

4. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

   a. Except as specified in 3.7.A.4.b below, two Pressure Suppression Chamber-Reactor Building Vacuum Breakers shall be operable at all times when the primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber reactor building vacuum breakers shall be $\leq$ 0.5 psi below reactor building pressure.

   b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days, unless

4.7 (cont'd)

When the primary containment is inerted, it shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. The monitoring system may be taken out of service for maintenance, but shall be returned to service as soon as possible.

4. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

   a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentations including setpoint shall be checked for proper operation every three months.

TABLE 4.7-2
EXCEPTION TO TYPE C TESTS

Certain Type C tests will be performed or omitted as follows:

Penetration	System	Valve	Local Leak Rate Test Performed
X-7A, B, C and D	Main Steam	29-AOV-80A, B C, and D 29-AOV-86A, B, C, and D	These valves are air-operated globe valves - pressurized in reverse direction and measurement of leakage will be equivalent to results from pressure applied in the same direction as when the valves would be required to perform its safety function. Therefore, pressure will be applied between the isolation valves and leakage measured. A water seal of 25 psig will be used on the inboard valve to determine the outboard valve's leak rate. (limit 11.5 SCFH at 25 psig)
X-10	RCIC	13-MOV-15	See X-25 (27-AOV-131A, B)
X-11	HPCI	23-MOV-15	See X-25 (27-AOV-131A, B)
X-25	Dry Well Inerting CAD and Purge	27-AOV-112	This valve is a butterfly valve - pressurization in reverse direction and measurement of leakage will be equivalent to results from pressure applied in the same direction as that when the valve would be required to perform its safety function.

TABLE 4.7-2
EXCEPTION TO TYPE C TESTS

(CONTINUED)

Penetration	System	Valve	Local Leak Rate Test Performed
X-25	Dry Well Inerting	27-AOV-131A 27 AOV-131B	These valves will be tested in the reverse direction, since the system was not designed for test pressure to be applied in the same direction as that when the valve would be required to perform its safety function. Basis - The pressurization direction was not a requirement at the time of plant design.
X-26 A/B	Dry Well Inerting CAD and Purge	27-AOV-113 27-MOV-122	See X-25 (27-AOV-112) This globe valve will be tested in the reverse direction. See X-25 (27-AOV-131A, B)

TABLE 4.7-2 (CONT'D)

Penetration	System	Valve	Local Leak Rate Test Performed
		27-SOV-120B 27-SOV-121B 27-SOV-122B	See X-25 (27-AOV-131A, B)
X-31 Bd	Dry Well Inerting CAD and Purge	27-SOV-125B	See X-25 (27-AOV-131A)
X-39A	Cont. Spray	10-MOV-31A	This valve will be pressurized in the reverse direction and leakage measured. See X-25 (27-SOV-131A, B)
X-39B	Cont. Spray	10-MOV-31A	See X-39A
X-45	ILRT	VSM-100T	See X-25 (27-AOV-131A, B)
X-59	Dry Well Inerting Cad and Purge	27-SOV-123A	See X-25 (27-AOV-131A, B)
X-202	Torus Vacuum Breakers	AOV-101A/B	See X-25 (27-AOV-112)
X-203A	Dry Well Inerting CAD and Purge	27-SOV-119B	See X-25 (27-AOV-131A, B)
X-203B	Dry Well Inerting CAD and Purge	27-SOV-124A	See X-25 (27-AOV-131A)
X-205	Dry Well Inerting CAD and Purge	27-AOV-117 27-MOV-117	See X-25 (27-AOV-112) See X-25 (27-MOV-113)
X-210 A/B	RCIC, RHR		Will not be tested as lines are water sealed by suppression chamber water See X-25 (27-AOV-131A, B)
X-211A	RHR	10-MOV-38A	This valve will be tested in the reverse direction. See X-25 (27-AOV-131A, B)
X-211B	RHR	10-MOV-38B	This valve will be tested in the reverse direction.
X-212	RCIC	13-MOV-130	See X-25 (27-AOV-131A/B)
X-218	ILRT	VSM-100T	See X-25 (27-AOV-131A/B)
X-220	Dry Well Inerting CAD and Purge	27-AOV-116 27-SOV-132A 27-SOV-132B	See X-25 (27-AOV-112) See X-25 (27-AOV-131A/B)
X-222	HPCI		See X-210 A/B
X-224	RHR		See X-210 A/B
X-225	RHR		See X-210 A/B

TABLE 4.7-2 (CONT'D)

Penetration	System	Valve	Local Leak Rate Test Performed
X-226	HPCI		See X-210 A/B
X-227	Core Spray		See X-210 A/B
X-228	Condensate		See X-210 A/B

6.0 ADMINISTRATIVE CONTROLS

Administrative Controls are the means by which plant operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, plant organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of these measures are necessary to ensure safe and efficient facility operation.

6.1 RESPONSIBILITY

The Resident Manager is responsible for safe operation of the plant. During periods when the Resident Manager is unavailable, the Superintendent of Power will assume his responsibilities. In the event both are unavailable, the Resident Manager may delegate this responsibility to other qualified supervisory personnel. The Resident Manager reports directly to the Executive Vice President-Nuclear Generation, as shown in Fig. 6.1-1.

6.2 PLANT STAFF ORGANIZATION

The plant staff organization shall be as shown in Figure 6.2-1 and:

1. Each shift crew shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

2. An SRO or an SRO with a license limited to fuel handling shall directly supervise all Core Alterations. This person shall have no other duties during this time;

3. A fire brigade of five (5) or more members shall be maintained on site at all times. This excludes two (2) members of the minimum shift crew necessary for safe shutdown and any personnel required for other essential functions during a fire emergency;

4. In the event of illness or unexpected absence, up to two (2) hours is allowed to restore the shift crew or fire brigade to the minimum complement.

5. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and maintenance personnel who are working on safety-related systems.

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating.

However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Resident Manager or the Superintendent of Power, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Resident Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

Amendment No. ~~46~~, ~~111~~ 130

(A) ROUTINE REPORTS (Continued)

1. STARTUP REPORT (Continued)

b. Startup Reports shall be submitted within (1) 90 days following completion of the startup test program, or (2) 90 days following resumption or commencement of commercial power operation, or whichever is earliest. If the Start-up Report does not cover both events, i.e., completion of startup test program and resumption or commencement of commercial power operation, supplementary reports shall be submitted at least every three months until both events are completed.

2. ANNUAL REPORTS

a. Annual Occupational Exposure Tabulation

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. Annual Report of S/RV Failures and Challenges

An annual report of safety/relief valve failures and challenges will be submitted prior to March 1 of each year.

3. MONTHLY OPERATING REPORT

A report providing a narrative summary of facility operating experience, major safety-related maintenance, and other pertinent information, should be submitted no later than the 15th of each month following the calendar month covered to the USNRC Director, Office of Management Information and Program Control.

1/ This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.

6.19 <u>POSTACCIDENT SAMPLING PROGRAM</u>

A program shall be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

A) Training of personnel,
B) Procedures for sampling and analysis,
C) Provisions for maintenance of sampling and analysis

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letters dated January 10, 1983 and November 1, 1983, Generic Letters 83-02 and 83-36, respectively, (NUREG-0737 Technical Specification) were transmitted by the Director, Division of Licensing to all boiling water reactor licensees. The generic letters provided staff guidance on the content of Technical Specifications (TS) associated with certain NUREG-0737 items. The Power Authority of the State of New York (PASNY), licensee for the James A. FitzPatrick Nuclear Power Plant, responded to the generic letters (GL) by letter dated December 6, 1984. This response was supplemented and superseded (in part) by letters dated October 18, 1985 and October 20, 1986.

This Safety Evaluation addresses TS proposed by the licensee for the following NUREG-0737 items:

GL 83-02

Limit Overtime (I.A.1.3)
Radiation Signal on Purge Valves (II.E.4.2.7)
Report Safety and Relief Valve Failures and Challenges (II.K.3.3)
RCIC Restart and RCIC Suction (II.K.3.13, II.K.3.22)

GL 83-36

Post-Accident Sampling (II.B.3)
Noble Gas Effluent Monitors (II.F.1.1)
Sampling and Analysis of Plant Effluents (II.F.1.2)
Containment High-Range Monitor (II.F.1.3)
Containment Pressure Monitor (II.F.1.4)
Containment Water Level Monitor (II.F.1.5)
Containment Hydrogen Monitor (II.F.1.6)

The proposed TS for Control Room Habitability Requirements (III.D.3.4) are being reviewed as a separate matter (TAC 61632). Certain other administrative and non-technical changes have also been incorporated into this amendment for clarity and consistency of the TS.

2.0 EVALUATION

2.1 Limit Overtime (Item I.A.1.3)

GL 83-02 provided the following overtime guidance:

> "Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.
>
> Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:
>
> a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.
>
> b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.
>
> c. A break of at least eight hours should be allowed between work periods, including shift turnover time.
>
> d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.
>
> Any deviation from the above guidelines shall be authorized by the (Plant Superintendent) or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the (Plant Superintendent) or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized."

The licensee's submittal dated October 20, 1986 incorporated the guidance into TS Section 6.2 (pages 247 and 247a). Therefore, the staff finds this response acceptable.

2.2 Dedicated Hydrogen Penetrations (Item II.E.4.1)

GL 83-02 provided the following guidance concerning isolation of containment purge and vent isolation valves on high radiation:

> "NUREG-0737 requires that containment purge and vent isolation valves must close on a high radiation signal to reduce the amount of radiation released outside containment following a release of radioactive materials to containment. The BWR Owners' Group has taken exception to this requirement and submitted their evaluation to NRC. NRC is currently reviewing the latest submittals of the Owners' Group. Technical Specifications for this item will be established after the technical resolution of this issue is completed."

In their letter dated October 20, 1986, the licensee proposed changes to the TS which added the closure signal for the containment vent and purge valves on high radiation (page 77a). Since the response is in accordance with the GL criteria, the response is satisfactory.

2.3 Report Safety and Relief Valve Failures and Challenges (Item II.K.3.3)

GL 83-02 guidance requires licensees to report all challenges to safety and safety/relief valves in their annual report to the Commission. Also, the GL guidance requires prompt notification (within 24 hours) with a written follow-up report within 14 days if a safety or safety/relief valve failure or malfunction occurs.

The TS submittal dated October 20, 1986 included the requirement to report safety/relief valve challenges in the annual report. Since reporting requirements concerning ECCS components is addressed by 10 CFR Parts 50.72 and 50.73, it has been determined that it is not necessary to address the GL reporting requirements (24-hour and 14-day) in the TS. Therefore, the staff has determined that the licensee's response to this item of including the information in the annual report is satisfactory.

2.4 RCIC Restart and RCIC Suction (II.K.3.13, II.K.3.22)

GL 83-02 provided the following guidance concerning the RCIC System:

> "The design of RCIC should be modified such that:
>
> 1) The system will restart on subsequent low water level after it has been terminated by a high water level signal;
>
> 2) RCIC system suction will automatically switchover from the condensate storage tank to the suppression pool when the condensate storage tank level is low.
>
> Provide technical specifications for both of the above modifications. It could be included with other technical specifications for the RCIC

system. Typical acceptable limiting conditions for operation (LCO) and surveillance requirements, for instrumentation and system operational capability, are given in Enclosure 2."

Enclosure 2 to GL 83-02 contains guidance related to operability and surveillance requirements.

In the licensee's submittal of December 6, 1984, a change to page 121 was made to the RCIC system requirements to include the automatic restart requirement and a change to page 70a was made to include the condensate storage tank suction valve automatic switchover instrumentation. The limiting conditions for operation and surveillance requirements specified for these changes are consistent with existing TS requirements for similar instrumentation.

Based on the method used to incorporate of the requirements and their consistency with similar requirements, the staff finds the changes acceptable.

2.5 Post-Accident Sampling (Item II.B.3)

GL 83-36 provided the following guidance concerning post-accident sampling:

"Licensees should ensure that their plant has the capability to obtain and analyze reactor coolant and containment atmosphere samples under accident conditions. An administrative program should be established, implemented, and maintained to ensure this capability. The program should include:

a. training of personnel
b. procedures of sampling and analysis, and
c. provisions for maintenance of sampling and analysis equipment

It is acceptable to the staff, if the licensee elects to reference this program in the administrative controls section of the Technical Specifications and include a detailed description of the program in the plant operation manuals. A copy of the program should be easily available to the operating staff during accident and transient conditions".

In the licensee's submittal dated October 20, 1986, a new TS page (page 258e) was added to administrative controls Section 6.12 to establish, implement, and maintain a program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and the containment atmosphere under accident conditions. Since the program includes the provisions given above, the staff finds the response satisfactory.

2.6 Noble Gas Effluent Monitors (Item II.F.1.1)

GL 83-36 provided the following guidance concerning the limiting conditions for operation (LCO) for effluent monitors:

> "With the number of OPERABLE channels less than required by the minimum channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:
>
> 1) either restore the inoperable channel(s) to OPERABLE status within seven days of the event, or
>
> 2) prepare and submit a Special Report to the Commission (pursuant to Specification 6.9.2) within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status."

The licensee's submittal dated October 20, 1986 proposed a similar LCO (Note B on page 77b). The licensee's submittal of December 6, 1984 provided the surveillance requirements (page 86a). GL 83-36 requires one noble gas monitor (one channel) and one preplanned alternative method of monitoring which would be initiated within 72 hours of monitor failure. The licensee's design provides two channels, however, with the second channel serving as the installed alternate monitoring method. The licensee has proposed that the second channel will be placed into operation as the preplanned alternate monitor within 72 hours of failure of the required monitor. Also, if one channel is not restored within seven days, a special report will be submitted to the NRC. The staff finds that this is more conservative than the requirements in the generic letter and that it is, therefore, acceptable.

2.7 Sampling and Analysis of Plant Effluents (Item II.F.1.2)

GL 83-36 requires that an administrative program be established, implemented, and maintained to ensure the collection and analysis of representative samples of radioactive iodine and particulates in plant gaseous effluents during and following an accident. The generic letter also stipulates that the program should include:

a) training of personnel
b) procedures for sampling and analysis, and
c) provisions for maintenance of sampling and analysis equipment.

The licensee, in their October 20, 1986 submittal, proposed to incorporate this program, including the above three items, into the administrative controls Section 6.12 (page 258e) of the TS which also addressed the Post-Accident Sampling program. The staff finds this to be acceptable.

2.8 Containment High-Range Radiation Monitor (Item II.F.1.3)

The following guidance was provided in GL 83-36 regarding containment high-range radiation monitors:

> "A minimum of two in-containment radiation-level monitors with a maximum range of 10^8 rad/hr (10^7 R/hr for photon only) should be operable at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible. If the monitor is not restored to operable condition within seven days after the failure, a special report should be submitted to the NRC within 14 days following the event, outlining the cause of inoperability, actions taken and the planned schedule for restoring the equipment to operable status."

In the submittal dated December 6, 1984 and October 20, 1986, the licensee proposed addition of two high range containment radiation monitors with a range of 1 rad/hr to 10^8 rad/hr. to TS Tables 3.2-8 and 4.2-8, "Accident Monitoring Instrumentation" (pages 77a and 86a). The licensee further proposed that with the number of operable channels less than the minimum required (one channel), either the inoperable channel will be restored to operable status within 30 days, or: (1) initiate an alternate method of monitoring the containment radiation, or (2) be in a cold condition within the next 24 hours.

In its October 18, 1985 letter, the licensee explained the basis for the following differences between the guidance provided in the generic letter and the proposed TS changes: (1) two of two channels required in the generic letter vs one of two required by the TS proposal, and (2) incorporation of a 30-day instrument restoration TS requirement rather than the 14 day generic letter reporting requirement. The licensee stated that the generic letter guidance concerning channel operability would be more restrictive than the TS proposal for some accident mitigating systems and that their approach is consistent with the fact that the plant equipment can be divided into three categories: (1) equipment necessary to prevent an accident; (2) equipment used to mitigate an accident; and, (3) equipment used for monitoring during and following an accident. Thus, the TS proposal is consistent with a "defense in depth" approach and is appropriate for the third category of equipment (monitoring) which would be used following a low probability accident.

The staff has reviewed the licensee's position, in light of the information provided above, and finds it acceptable. Imposing the guidance provided in GL 83-36 in this case will not result in a significant increase in the level of protection of public health and safety. Additionally, in accordance with 10 CFR Part 50.72, the licensee would be required to notify the NRC within one hour if a shutdown is required in accordance with a TS Limiting Condition for Operation. Also, the licensee would be required by 10 CFR Part 50.73 to submit a written report within 30 days for such an event. This would satisfy the reporting requirements of the GL. Therefore, the TS proposed by the licensee for Item II.F.1.3 is acceptable.

2.9 Containment Pressure Monitor (Item II.F.1.4)

GL 83-36 provided the following guidance concerning containment pressure monitoring:

> "Containment pressure should be continuously indicated in the control room of each operating reactor during Power Operation and Startup Modes. Two channels should be operable at all times when the reactor is operating in any of the above mentioned modes. Technical Specifications for these monitors should be included with other accident monitoring instrumentation in the present Technical Specifications. Limiting conditions for operation (including the required Actions) for the containment pressure monitor should be similar to other accident monitoring instrumentation included in the present Technical Specifications."

In their submittal of October 20, 1986, the licensee proposed changes to the TS for the containment pressure monitor (pages 77a, 77b). The proposed LCO is, as requested by the staff, similar to other accident monitoring instrumentation in the present TS. In their submittal of December 6, 1984, the licensee proposed changes to the surveillance requirements for the instrumentation (page 86a).

In response to staff concerns that the proposed TS differed from the generic letter guidance, the licensee's October 18, 1985 letter provided the following basis for their position:

> "The Authority proposed TS requires that the inoperable channel be restored within 30 days. Otherwise, either an alternate monitoring method would be initiated, or the plant would be shut down. According to the Staff's proposal, the plant would have to shut down in 48 hours.
>
> As for Item II.F.1.3, the Authority's approach is consistent with the fact that there are three levels of importance for plant equipment which could be divided into three categories: (1) equipment necessary to prevent an accident; (2) equipment necessary to mitigate an accident; and (3) equipment used for monitoring an accident.
>
> The Authority's proposed Action Statement for this item is consistent with those for other accident monitoring instrumentation and adequate for a monitoring system."

Because the approach of Items II.F.1.3 and II.F.1.4 are similar, the staff finds that the proposed TS is acceptable for the reasons stated in the discussion of Item II.F.1.3 above.

2.10 Containment Water Level Monitor (Item II.F.1.5)

GL 83-36 provided the following guidance concerning containment water level monitoring:

> "A continuous indication of suppression pool water level should be provided in the control room of each reactor during Power Operation and Startup Modes. Two channels should be operable at all times when the reactor is operating in any of the above mentioned modes. Technical Specifications for suppression pool water level monitors should be included with other accident monitoring instrumentation in the present Technical Specifications. Limiting conditions for operation (LCO) for these monitors should be similar to other accident monitoring instrumentation included in the present Technical Specifications. Typical acceptable LCO and surveillance requirements for accident monitoring instrumentation are included in Enclosure 3.
>
> The BWRs with dry containment should have at least two channels for wide range instruments and one channel of narrow range instrument operable at all times during above mentioned modes. LCOs for wide range monitor should include the requirement that the inoperable channel will be restored to operable status within 30 days or the reactor will be brought to hot shutdown condition as required by other accident monitoring instrumentation."

In their submittal of October 20, 1986, the licensee proposed changes to the TS for the containment water level monitor (pages 77a, 77b). The proposed LCO is, as requested by the staff, similar to other accident monitoring instrumentation in the present TS. In their submittal of December 6, 1984, the licensee proposed changes to the surveillance requirements for the instrumentation (page 86a).

In response to staff concerns that the proposed TS differed from the generic letter guidance, the licensee stated in their October 18, 1985 letter that "Our response is the same as for Item II.F.1.4." For the reasons stated in the discussion of Items II.F.1.3 and II.F.1.4 above, we find this proposed TS acceptable.

2.11 Containment Hydrogen Monitor (Item II.F.1.6)

GL 83-36 provided the following guidance concerning Item II.F.1.6:

> "Two independent containment hydrogen monitors should be operable (should be capable of performing the required function) at all times when the reactor is operating in Power Operation and Startup modes. Technical Specifications for hydrogen monitors should be included with other accident monitoring instrumentation in the present Technical Specification. Typical acceptable LCO and surveillance requirements are included in Enclosure 3."

Enclosure 3 stated:

> "a. With the number of OPERABLE channels less than two restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.
>
> b. With the number of OPERABLE channels less than one restore at least one channel to OPERABLE status within seven days or be in at least HOT SHUTDOWN within the next 12 hours."

In their submittal of October 20, 1986, the licensee proposed changes to the TS for the containment hydrogen monitor (pages 77a, 77b). The LCO permits a 30 day period for initiation of alternate monitoring or 31 days to reach cold shutdown if both monitors fail. In their submittal of December 6, 1984, the licensee proposed changes to the surveillance requirements for the instrumentation (page 86a).

In response to staff concerns that the proposed TS differed from the generic letter guidance, the licensee stated in their October 18, 1985 letter that:

> "The NYPA proposed TS requires that the inoperable channel be restored within 30 days. Otherwise, either an alternate monitoring method would be initiated, or the plant would be shut down. According to the Staff's proposal, the plant would have to be shut down within seven days.
>
> Our proposed Action Statement for this item is consistent with those provided for other accident monitoring instrumentation.
>
> Furthermore, it should be noted that in addition to hydrogen concentration monitors FitzPatrick has redundant oxygen monitoring equipment, and a TS requirement that oxygen concentration in containment be less than 4%."

The staff has reviewed the licensee's position and finds that the proposed TS provide a sufficiently restrictive set of requirements which represent an acceptable alternative to the guidance of Generic Letter 83-36.

2.12 Summary

Based on our review of the licensee's submittals and letters dated December 6, 1984, October 18, 1985, and October 20, 1986, and the evaluations provided above, we find the proposed TS for the following NUREG-0737 items to be acceptable:

Limit Overtime (I.A.1.3)
Radiation Signal on Purge Valves (II.E.4.2.7)
Report Safety and Relief Valve Failures and Challenges (II.K.3.3)
RCIC Restart and RCIC Suction (II.K.3.13, II.K.3.22)
Post-Accident Sampling (II.B.3)
Noble Gas Effluent Monitors (II.F.1.1)
Sampling and Analysis of Plant Effluents (II.F.1.2)
Containment High-Range Monitor (II.F.1.3)
Containment Pressure Monitor (II.F.1.4)
Containment Water Level Monitor (II.F.1.5)
Containment Hydrogen Monitor (II.F.1.6)

Additionally, we find that the administrative changes proposed on the enclosed pages are necessary to support the above TS changes and are, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: May 31, 1989

PRINCIPAL CONTRIBUTORS:

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