UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 29, 1993

Docket No. 50-333

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M87857)

The Commission has issued the enclosed Amendment No. 203 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated September 24, 1993.

The amendment makes miscellaneous administrative changes including typographical and editorial corrections to the Appendix A TSs and Appendix B Radiological Effluent TSs. The amendment does not make any substantive changes to the TSs.

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

Sincerely,

John E. Menning

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 203 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

DF01

cc:

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DATED: December 29, 1993

AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
M. Griggs
J. Menning
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

050056

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 203
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 September 24, 1993, 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. 203, 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian C. McCabe for

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 29, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 203

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
55	55
56	56
64	64
96	96
116	116
142	142
144	144
150	150
162a	162a
186	186
192	192
197	197
247	247

Revise Appendix B as follows:

Remove Pages	Insert Pages
23	23
33	33
37	37
66	66

3.2 BASES

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

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

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

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

The low water level instrumentation set to trip at 177 in. above the top of the active fuel closes all isolation valves except those in Group 1. Details of the isolation valve grouping are given in Section 7.3 of the updated FSAR. For valves which isolate at this level, this trip setting is adequate to prevent uncovering the core in the case of a break in the largest line.

The low-low reactor water level instrumentation is set to trip when reactor water level is 126.5 in. above the top of active fuel. This trip

3.2 BASES (cont'd)

initiates the HPCI and RCIC systems and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18 in. above the top of active fuel. This trip activates the remainder of the ECCS subsystems, closes the main steam isolation valves, main steam line drain valves and reactor water sample line isolation valves, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate ECCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 24 in. recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 of the updated FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating ECCS, it causes isolation of Groups B and C isolation valves. For the breaks discussed above, this instrumentation will generally initiate ECCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. Details of the isolation valve closure group are given in Section 7.3 of the updated FSAR. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperature peak at approximately 1,000°F and release of radioactivity to the environs is below 10 CFR 100 guidelines. Reference Section 14.6.5 of the updated FSAR.

Table 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. 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	≥ 177 in. above TAF	4	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2	D
2	Reactor Low-Low-Low Water Level	≥ 18 in. above the TAF	4	A
2 (6)	High Drywell Pressure	≤ 2.7 psig	4	A
2	High Radiation Main Steam Line Tunnel	≤ 3 x Normal Rated Full Power Background (9)	4	B
2	Low Pressure Main Steam Line	≥ 825 psig (7)	4	B
2	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4	B
2	Main Steam Line Leak Detection High Temperature	≤ 40°F above max ambient	4	B
4	Reactor Cleanup System Equipment Area High Temperature	≤ 40°F above max ambient	8	C
2	Low Condenser Vacuum Closes MSIV's	≥ 8" Hg. Vac (7)(8)	4	B

3.3.C (cont'd)

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Control Rod Notch Position Observed	Average Scram Insertion Time (Seconds)
46	0.361
38	0.977
24	2.112
04	3.764

3. The maximum scram insertion time for 90 percent insertion of any operable control rod shall not exceed 7.00 sec.

4.3.C (cont'd)

2. At 16-week intervals, 10 percent of the operable control rod drives shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. All control rods shall be determined operable once each operating cycle by demonstrating the scram discharge volume drain and vent valves operable when the scram test initiated by placing the mode switch in the SHUTDOWN position is performed as required by Table 4.1-1 and by verifying that the drain and vent valves:

   a. Close in less that 30 seconds after receipt of a signal for control rods to scram, and

   b. Open when the scram signal is reset.

3.5 (cont'd)

2. Should one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining components of the containment cooling mode subsystems are operable.

3. Should one of the containment cooling subsystems become inoperable or should one RHRSW pump in each subsystem become inoperable, continued reactor operation is permissible for a period not to exceed 7 days.

4. If the requirements of 3.5.B.2 or 3.5.B.3 cannot be met, the reactor shall be placed in a cold condition within 24 hr.

5. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature <212°F with an inoperable component(s) as specified in 3.5.B above.

4.5 (cont'd)

2. When it is determined that one RHRSW pump of the components required in 3.5.B.1 above is inoperable, the remaining components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.

3. When one containment cooling subsystem becomes inoperable, the redundant containment cooling subsystem shall be verified to be operable immediately and daily thereafter. When one RHRSW pump in each subsystem becomes inoperable, the remaining components of the containment cooling subsystems shall be verified to be operable immediately and daily thereafter.

3.6 (cont'd)

5. With the Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring or Floor Drain Sump Monitoring) inoperable, restore the system to operable status within 24 hours or be in at least hot shutdown within the next 12 hours and in the cold condition within the following 24 hours.

6. With the Primary Containment Atmosphere Radioactivity Monitoring System (gaseous) or the Primary Containment Atmosphere Radioactivity Monitoring System (particulate) inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours. Otherwise be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

4.6 (cont'd)

3. Drywell Continuous Atmosphere Radioactivity Monitoring System instrumentation shall be functionally tested and calibrated as specified in Table 4.6-2.

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

G. Jet Pumps

Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

F. Structural Integrity

1. Nondestructive inspections shall be performed on the ASME Boiler and Pressure Vessel Code Class 1, 2 and 3 components and supports in accordance with the requirements of the weld and support inservice inspection program. This inservice inspection program is based on an NRC approved edition of, and addenda to, Section XI of the ASME Boiler and Pressure Vessel Code which is in effect 12 months or less prior to the beginning of the inspection interval.

2. An augmented inservice inspection program is required for those high stressed circumferential piping joints in the main steam and feedwater lines larger than 4 inches in diameter, where no restraint against pipe whip is provided. The augmented in-service inspection program shall consist of 100 percent inspection of these welds per inspection interval.

3. An Inservice Inspection Program for piping identified in the NRC Generic Letter 88-01 shall be implemented in accordance with NRC staff positions on schedules, methods, personnel, and sample expansion included in this Generic Letter, or in accordance with alternate measures approved by the NRC staff.

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

3.6 and 4.6 BASES (cont'd)

than 100,000 lb/hr, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Fig. 4.6-1 are not exceeded. At steaming rates of at least 100,000 lb/hr, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal ranges. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This is not necessarily the case. Conductivity could be high due to the presence of a neutral salt; e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are, in fact, high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the condition include operation of the Reactor Cleanup System, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the Reactor Water Cleanup System to reestablish the purity of the reactor coolant.

During startup periods, which are in the category of less than 100,000 lb/hr, conductivity may exceed 2 μmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μmho/cm (other than short-term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses of the reactor coolant required by Specification 4.6.C.1 may be performed by a gamma scan.

D. Coolant Leakage

Allowable leakage rates of coolant from the Reactor Coolant System have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to make up Reactor Coolant System leakage in the event of loss of off-site a-c power. The normally expected background leakage due to equipment design and the detection capability for determining system

Table 4.6-2

Minimum Test and Calibration Frequency for Drywell Continuous Atmosphere Radioactivity Monitoring System

Inst. Channel	Inst. Functional Test	Calibration	Sensor Check
1. Air Particulate Analyzer	None	Once / 3 mos.	once / day
2. Gaseous Activity Analyzer	None	Once / 3 mos.	once / day
3. Iodine Analyzer	None	Once / 3 mos.	once / day

3.7 (cont'd)

2. With one or more of the containment isolation valves inoperable, maintain at least one isolation valve operable in each affected penetration that is open and:

   a. Restore the inoperable valve(s) to operable status within 4 hours; or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the closed position. Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control; or

   c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or a blind flange.

3. If Specifications 3.7.D.1 or 3.7.D.2 cannot be met the reactor shall be in the cold condition within 24 hrs.

4.7 (cont'd)

(2.) With the reactor at a reduced power level, fast close each main steam isolation valve, one at a time, and verify closure time.

d. At least twice per week, the main steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

e. The RBCLCWS isolation valves shall be fully closed and reopened any time the reactor is in the cold condition exceeding 48 hours, if the valves have not been fully closed and reopened during the preceding 92 days.

2. Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

3. Not Used

3.7 BASES (cont'd)

of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the Pressure Suppression System. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

The containment isolation valves on the containment vent and purge lines may be open for safety related reasons. Safety related reasons include, but are not limited to, the following: inerting or de-inerting primary containment; maintaining containment oxygen concentration; maintaining drywell and suppression pool atmospheric pressures; and maintaining the differential pressure between the drywell and suppression pool. These valves have been modified to limit the maximum angle of opening as shown in 3.7.D.1.

Nine remote manual isolation valves have been added to the Reactor Building Closed Loop Cooling Water System (RBCLCWS) in order to comply with 10 CFR 50 Appendix A GDC 57; These valves are air operated (with solenoid pilot valves), normally open, and are designed to fail "open"on loss of electrical power or "as is" upon loss of instrument air. Each AOV is provided with a Seismic Class I accumulator tank to allow operation of the valves upon loss of instrument air up to 2 full valve cycles. The fail-open design permits continued operation of the system to supply water to the recirculation pump-motor coolers and drywell coolers during normal operation and as necessary under accident conditions. If there is a postulated accident, and indications of leakage from RBCLCWS appear, the operator will selectively close the AOV's affected to provide containment isolation.

A list of containment isolation valves, including a brief description of each valve is included in Section 7.3 of the updated FSAR.

4.7 BASES (cont'd)

operability results in a more reliable system.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 in. restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

The RBCLCWS valves are excluded from the quarterly surveillance requirements because closure of these valves will eliminate the coolant flow to the drywell air and recirculation pump-motor coolers. Without cooling water, the drywell air and equipment temperature will increase and may cause damage to the equipment during normal plant operations. Therefore, testing of these valves would only be conducted in the cold condition.

A list of containment isolation valves, including a brief description of each valve is included in Section 7.3 of the updated FSAR.

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, one of the three General Managers will assume this responsibility. In the event all four 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.

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities that affect the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR.

2. The Resident Manager shall be responsible for overall plant operation, and shall have control over those onsite activities that are necessary for safe operation and maintenance of the plant.

3. The Executive Vice President - Nuclear Generation shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Plant Staff

The plant staff organization shall be as follows:

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

NOTES FOR TABLE 3.2-1 (continued)

(d) Main stack gaseous sampling and analysis shall also be performed following shutdown, startup, or a thermal power change exceeding 20% of rated thermal power in one hour.

1. This requirement applies only if:

   - Analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3; and

   - The noble gas monitor shows that effluent activity has increased more than a factor of 3; and

   - Correction for increases due to changes in thermal power level have been made in both cases.

(e) Main stack iodine and particulate sampling shall also be performed daily following each shutdown, startup or thermal power change exceeding 20% of rated thermal power in one hour.

1. Daily sampling is not required for thermal power changes if the off gas charcoal filters are in service.

2. In addition, this requirement applies only if:

   - Analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3; and

   - The noble gas monitor shows that effluent activity has increased more than a factor of 3; and

   - Corrections for increases due to changes in thermal power level have been made in both cases.

3. Daily sampling shall be performed until two consecutive samples show no increase in concentration but not to exceed 7 consecutive days.

4. LLDs may be increased by a factor of 10 for analysis of daily samples.

5. Analysis of daily and weekly samples shall be completed within 48 hours of changing.

(f) Incinerated oil may be discharged via points other than the main stack and building vents (i.e., auxiliary boiler). Release shall be accounted for based on pre-release grab sample data.

(g) Samples of incinerated oil releases shall be collected from and representative of filtered oil in liquid form. Whenever oil samples cannot be filtered such as No. 6 bunker fuel oil, raw oil samples shall be collected and analyzed.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

treatment system under the following conditions:

1. The offgas dilution steam flow instrumentation shall alarm and automatically isolate the offgas recombiner system at a low flow setpoint greater than or equal to 6300 pounds per hour and at a high flow setpoint less than or equal to 7900 pounds per hour.

2. The offgas recombiner inlet temperature sensor shall alarm and automatically isolate the offgas recombiner system at a temperature setpoint of greater than or equal to 125°C.

3. The offgas recombiner outlet temperature sensor shall alarm and automatically isolate the offgas treatment system at a temperature setpoint of greater than or equal to 150°C.

c. In lieu of continuous hydrogen or oxygen monitoring, the condenser offgas treatment system recombiner effluent shall be analyzed to verify that it contains less than or equal to 4% hydrogen by volume.

d. With the requirements of the above specifications not satisfied, restore the recombiner system to within operating specifications or suspend use of the charcoal treatment system within 48 hours.

1. An instrument check shall be performed daily when the offgas treatment system is in operation.

2. An instrument channel functional test shall be performed once per operating cycle.

3. An instrument channel calibration shall be performed once per operating cycle.

c. With condenser offgas treatment system recombiner in service, in lieu of continuous hydrogen or oxygen monitoring, the hydrogen content shall be verified weekly to be less than or equal to 4 % by volume.

In the event that the hydrogen content cannot be verified, operation of this system may continue for up to 14 days.

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels	Trip Function	Trip Level Settings	Total Number of Instrument Channels Provided by Design	Actions
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)
1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)
1(a)	SJAE Radiation Monitors	$\leq$ 500,000 μCi/sec	2	(e)
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)
1(a)	Main Control Room Ventilation	$\leq 4 \times 10^3$ cpm[i]	1	(g)
(h)	Mechanical Vacuum Pump Isolation	$\leq$3 x Normal Full Power Background	4	(h)

NOTES FOR TABLE 3.10-1

(a) Whenever the systems are required to be operable, there shall be one operable or tripped instrument channel per system. From and after the time it is found that this cannot be met, the indicated action shall be taken.

(b) Trip level setting is in accordance with the methods and procedures of the ODCM.

(c) Cease operation of the refueling equipment.

(d) Isolate secondary containment and start the SBGTS.

(e) Bring the SJAE release rate within the limit within 72 hours or be in hot standby within the next 12 hours.

(f) Refer to Appendix B LCO 3.1.d.

(g) Control room isolation is manually initiated.

(h) Uses same sensors as primary containment isolation on high main steam line radiation. Refer to Appendix A Table 3.2-1 for minimum number of operable instrument channels and action required.

(i) Conversion factor is 8.15×10^7 cpm - 1μCi/cc.

Amendment No. ~~93~~, ~~127~~, 203

7.0 ADMINISTRATIVE CONTROLS

7.1 RESPONSIBILITY

a. The Resident Manager shall have direct responsibility for assuring the operation of the James A. FitzPatrick Plant is conducted in such a manner as to provide continuing protection to the environment. During periods when the Resident Manager is unavailable, one of the three General Managers will assume this responsibility. In the event all four are unavailable, the Resident Manager may delegate this responsibility to other qualified supervisory personnel.

b. Implementation of the Radiological Effluent Technical Specifications is the responsibility of the General Manager - Operations, with the assistance of the plant staff organization.

7.2 PROCEDURES

Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 "Facility Administrative Policies and Procedures" of ANSI 18.7-1972 and Regulatory Guide 1.33, November 1972, Appendix A. In addition, procedures shall be established, implemented and maintained for the PCP, ODCM, and Quality Control Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1.

7.3 REPORTING REQUIREMENTS

a. Planned Liquid and Gaseous Releases

The limits for radioactive materials contained in liquid and gaseous effluents are contained in Specifications 2.3, 3.3 and 3.4.

b. Environmental Samples Exceeding Limits of Table 6.1-2

When the limits of Table 6.1-2 are exceeded, refer to Specification 6.1.b for reporting requirements.

c. Semiannual Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

1. The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit using as guidance Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203 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 letter dated September 24, 1993, the Power Authority of the State of New York (the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant Technical Specifications (TSs). The requested changes make miscellaneous administrative changes including typographical and editorial corrections to the Appendix A TSs and Appendix B Radiological Effluent TSs (RETS).

2.0 EVALUATION

The licensee has proposed miscellaneous typographical and editorial corrections to the Appendix A TSs and Appendix B RETS. The staff has reviewed these proposed changes and determined that they are acceptable since they do not involve any substantive changes to requirements.

In addition to the above noted typographical and editorial corrections, the licensee also proposed to delete an erroneous surveillance requirement from Appendix A TS 4.3.C.3.b. TS 4.3.C.3.b currently requires verification that after placing the mode switch in the SHUTDOWN position the scram discharge instrument volume (SDIV) drain and vent valves open when the scram signal is reset or the SDIV trip is bypassed. The proposed change would require verification that the drain and vent valves open when the scram signal is reset, and delete reference to bypassing of the SDIV trip. When the mode switch is placed in the SHUTDOWN position as part of this testing, the scram discharge volume isolation valve solenoids de-energize to close the SDIV drain and vent valves. At the end of the testing, the operator clears all signals to reset the reactor protection system including the SDIV high-level signal, if necessary. Once all signals are cleared, the operator can reset the scram which will reopen the drain and vent valves. It is the scram reset and not the SDIV high-level trip bypass that causes the SDIV vent and drain valves to open. The requirement in TS 4.3.C.3.b to verify that the SDIV vent and drain valves open when the SDIV trip is bypassed is, therefore, inconsistent with the design of the plant.

The staff has reviewed the above change to the surveillance requirement in Appendix A TS 4.3.C.3.b and found it to be consistent with the design of the plant and Section 3.1.8.3 of NUREG-1433, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/4)." The proposed change is, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 62155). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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 the amendment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
M. Griggs

Date: December 29, 1993

December 29, 1993

Docket No. 50-333

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M87857)

The Commission has issued the enclosed Amendment No. 203 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated September 24, 1993.

The amendment makes miscellaneous administrative changes including typographical and editorial corrections to the Appendix A TSs and Appendix B Radiological Effluent TSs. The amendment does not make any substantive changes to the TSs.

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

Sincerely,
Original signed by:
John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 203 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

Distribution:
See attached sheet

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