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

March 9, 1994

Docket No. 50-333

Mr. William A. Josiger, Acting
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Josiger:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M86978)

The Commission has issued the enclosed Amendment No. 207 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 July 15, 1993.

The amendment revises the TSs to eliminate the reactor scram and Main Steam Line Isolation Valve closure requirements associated with the Main Steam Line Radiation Monitors. The changes are consistent with Licensing Topical Report NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," dated May 1987.

This license amendment is effective as of the date of its issuance to be implemented prior to startup following the next FitzPatrick refueling outage. Please notify the NRC, in writing, within 30 days of implementing this amendment.

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,

Brian C. McCabe

Brian C. McCabe, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 207 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

DF01

cc:

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DATED: March 9, 1994

AMENDMENT NO. 207 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
B. McCabe
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
R. Jones, 8/E/23
L. Cunningham, 10/D/4
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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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. 207
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 July 15, 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. 207, 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 prior to startup following the next FitzPatrick refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

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: March 9, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
33	33
34	34
41a	41a
43	43
45	45
46	46
47	47
57	57
64	64
65	65
78	78
84	84

Revise Appendix B as follows:

39	39

3.1 BASES (cont'd)

subchannel. APRM's B, D and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, main steam isolation valve (MSIV) closure, and generator load rejection, turbine stop valve closure are discussed in Sections 2.1 and 2.2.

Instrumentation for the drywell is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the Core and Containment Cooling Systems (ECCS) initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A Reactor Mode Switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (high flux in startup or refuel) System provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The IRM System provides protection against short reactor periods in these ranges.

3.1 BASES (cont'd)

The Control Rod Drive Scram System is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. Each scram discharge instrument volume accommodates in excess of 34 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level detection instruments have been provided in each instrument volume which alarm and scram the reactor when the volume of water reaches 34.5 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A Source Range Monitor (SRM) System is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph 7.5.4 FSAR).

The IRM high flux and APRM $\leq$15% power scrams provide adequate coverage in the startup and intermediate range. Thus, the IRM and APRM systems are required to be operable in the refuel and startup/hot standby modes. The APRM $\leq$120% power and flow referenced scrams provide required protection in the power range (reference FSAR Section 7.5.7). The power range is covered only by the APRMs. Thus, the IRM system is not required in the run mode.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for startup and run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1-1 operable in the refuel mode assures that shifting to the refuel mode during reactor power operation does not diminish the protection provided by the Reactor Protection System.

Turbine stop valve closure occurs at 10 percent of valve closure. Below 217 psig turbine first stage pressure (30 percent of rated), the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting[1]	Modes in Which Function Must be Operable: Refuel (6)	Startup	Run	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
2	APRM Downscale	≥ 2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	≤ 1045 psig	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤ 2.7 psig	X(7)	X(7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥ 177 in. above TAF	X	X	X	4 Instrument Channels	A
3	High Water Level in Scram Discharge Volume	≤ 34.5 gallons per Instrument Volume	X(2)	X	X	8 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤ 10% valve closure			X(5)	8 Instrument Channels	A

Amendment No. ~~10~~, ~~43~~, ~~67~~, ~~78~~, ~~87~~, ~~90~~, ~~119~~, ~~122~~ 207

TABLE 3.1-1 (cont'd)
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1- (cont'd)

C. High Flux IRM.

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

E. APRM 15% Power Trip.

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

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

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

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

11. See Section 2.1.A.1.

12. The APRM Flow Referenced Neutron Flux Scram setting shall be less than or equal to the limit specified in the Core Operating Limits Report.

13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained as specified in the Core Operating Limits Report.

14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is place in the Run position.

Amendment No. 49, 62, 64, 67, 69, 72, 74, 109, 117, 159, 162 207

TABLE 4.1-1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TEST
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

Instrument Channel	Group (2)	Functional Test	Minimum Frequency (3)
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/month.(1)
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Once/month.
Turbine First Stage Pressure Permissive	B	Trip Channel and Alarm (4)	Once/month.(1)(8)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/month.(1)

NOTES FOR TABLE 4.1-1

1. Initially once every month until acceptable failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of JAFNPP.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required on the part of the system that is not required to be operable or are tripped.

   If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the instrument channels.

TABLE 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Maximum frequency once/week
APRM High Flux Output Signal	B	Heat Balance	Daily
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source	Every refueling outage
LPRM Signal	B	TIP System Traverse	Every 1000 effective full power hours
High Reactor Pressure	B	Standard Pressure Source	Note (6)
High Drywell Pressure	B	Standard Pressure Source	Note (6)
Reactor Low Water Level	B	Standard Pressure Source	Note (6)
High Water Level in Scram Discharge Instrument Volume	A	Water Column, Note (5)	Once/operating cycle, Note (5)
High Water Level in Scram Discharge Instrument Volume	B	Standard Pressure Source	Every 3 months
Main Steam Line Isolation Valve Closure	A	Note (4)	Note (4)
Turbine First Stage Pressure Permissive	B	Standard Pressure Source	Note (6)

TABLE 4.1-2 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
Turbine Control Valve Fast Closure Oil Pressure Trip	A	Standard Pressure Source	Once/operating cycle
Turbine Stop Valve Closure	A	Note (4)	Note (4)

NOTES FOR TABLE 4.1-2

1. A description of three groups is included in the Bases of this Specification.
2. Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.
3. Deleted
4. Actuation of these switches by normal means will be performed during the refueling outages.
5. Calibration shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.
6. Sensor calibration once per operating cycle. Master/slave trip unit calibration once per 6 months.

3.2 BASES (cont'd)

High radiation monitors in the area of the main steam lines have been provided to detect gross fuel failure as in the control rod drop accident. A trip setting of 3 times normal full-power background is established to close the main steam line drain valves, the recirculation loop sample valves, the mechanical vacuum pump isolation valves, and trip the pumps, to limit fission product release. For changes in the Hydrogen Water Chemistry hydrogen injection rate, the trip setpoint may be adjusted based on a calculated value of the expected radiation level. Hydrogen addition will result in an increase in the N-16 carryover in the main steam.

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 approximately 300 percent of design flow for this high flow or 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 settings of approximately 300 percent for high flow or 40°F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high 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 decrease to the Safety Limit. The trip

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	4	E
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

Amendment No. 10, 37, 48, 67, 90, 103, 119, 122, 185, 203 207

TABLE 3.2-1 (Cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

NOTES FOR TABLE 3.2-1

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

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

   A. Place the reactor in the cold condition within 24 hours.
   B. Isolate the main steam lines within eight hours.
   C. Isolate Reactor Water Cleanup System within four hours.
   D. Isolate shutdown cooling within four hours.
   E. Isolate the main steam line drain valves, the recirculation loop sample valves, and the mechanical vacuum pumps, within eight hours.

3. Deleted

4. Deleted

5. Two required for each steam line.

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

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

8. Bypassed when mode switch is not in run mode and turbine stop valves are closed.

TABLE 4.2-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

	Instrument Channel (8)	Instrument Functional Test	Calibration Frequency	Instrument Check (4)
1)	Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2)	Reactor Low-Low-Low Water Level	(1)(5)	(15)	Once/day
3)	Main Steam High Temp.	(1)(5)	(15)	Once/day
4)	Main Steam High Flow	(1)(5)	(15)	Once/day
5)	Main Steam Low Pressure	(1)(5)	(15)	Once/day
6)	Reactor Water Cleanup High Temp.	(1)	Once/3 months	None
7)	Condenser Low Vacuum	(1)(5)	(15)	Once/day
8)	Main Steam Line High Radiation	(1)(5)	(11)	Once/day

	Logic System Functional Test (7) (9)	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	Once/6 months
3)	Reactor Water Cleanup Isolation	Once/6 months
4)	Drywell Isolation Valves TIP Withdrawal Atmospheric Control Valves	Once/6 months
5)	Standby Gas Treatment System Reactor Building Isolation	Once/6 months

NOTE: See notes following Table 4.2-5.

Amendment No. ~~37~~, ~~89~~, ~~136~~, ~~181~~, ~~182~~, ~~190~~ 207

NOTES FOR TABLES 4.2-1 THROUGH 4.2-5

1. Initially once every month until acceptance failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instruments operate in a environment similar to that of JAFNPP.

2. Functional tests are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed within seven (7) days prior to each startup.

3. Calibrations are not required when these instruments are not required to be operable or are tripped. Calibration tests shall be performed within seven (7) days prior to each startup or prior to a pre-planned shutdown.

4. Instrument checks are not required when these instruments are not required to be operable or are tripped.

5. This instrumentation is exempt from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

6. These instrument channels will be calibrated using simulated electrical signals once every three months.

7. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

8. Reactor low water level, and high drywell pressure are not included on Table 4.2-1 since they are listed on Table 4.1-2.

9. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

10. At least one (1) Main Stack Dilution Fan is required to be in operation in order to isokinetically sample the Main Stack.

11. Perform a calibration once per operating cycle using a radiation source. Perform an instrument channel alignment once every 3 months using the built-in current source.

12. (Deleted)

13. Calibration and instrument check surveillance for SRM and IRM Instruments are as specified in Tables 4.1-1, 4.1-2, 4.2-3.

14. Functional test is performed once each operating cycle.

15. Sensor calibration once per operating cycle. Master/slave trip unit calibration once per 6 months.

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

(f) Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 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 July 15, 1993, the Power Authority of the State of New York (PASNY, the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant Technical Specifications. The requested changes would eliminate the reactor scram and Main Steam Isolation Valve (MSIV) closure requirements associated with the Main Steam Line Radiation Monitors (MSLRM). The licensee has experienced spurious trips of the MSLRM channels on numerous occasions in the past. Failure of one of the MSLRM was a contributing factor to an automatic reactor scram on May 25, 1993.

The licensee states that elimination of the trip functions will reduce scram frequency, maintain availability of condenser heat sink, eliminate the potential for trips due to hydrogen water chemistry, and increase operator control over radioactive releases. The licensee's proposed change is based on the May 1987 Boiling-Water Reactor (BWR) Owners Group Licensing Topical Report NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," and the Standard Review Plan (SRP) 15.4.9, Revision 2, July 1981.

As stated in NEDO-31400, and indicated in FitzPatrick Final Safety Analysis Report (FSAR), the automatic reactor shutdown on the MSLRM trip is not given credit in the analysis of any design basis event for BWRs. The FSAR assumes that only in a control rod drop accident (CRDA) do MSIVs close on the MSLRM trip. However, the SRP 15.4.9 recommends that the radioactive source term is assumed to already have been transferred to the condenser and turbine prior to MSIV closure. The SRP 15.4.9 also states that the plant site and dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated CRDA if the calculated whole-body and thyroid doses at the exclusion area boundaries are within 25 percent of the exposure guideline values of 10 CFR Part 100.

In NEDO-31400, a reevaluation of the role of the MSLRM in the CRDA analysis was performed, confirming that removal of the MSLRM scram/isolation features would not compromise CRDA consequences. The topical report also evaluated the potential effect on occupational exposure in the event of a sudden release of radioactivity from the fuel and concluded that the elimination of the scram/isolation features would have no adverse effect.

2.0 EVALUATION

In a May 15, 1991, Safety Evaluation (SE), the NRC staff accepted the referencing of NEDO-31400 for the elimination of the MSIV closure function and scram function of the MSLRM, as long as the following three conditions were met:

1. The applicant demonstrates that the assumption with regard to input values, including power per assembly, Chi/Q, and decay times, that were made in the generic analysis, bound those for the plant.

2. The applicant includes sufficient evidence, implemented or proposed operating procedures or equivalent commitments, to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases.

3. The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint to 1.5 times the nominal ^{16}N background dose rate at the monitor locations and commit to promptly sample the reactor coolant to determine possible contamination levels in the reactor coolant and the need for additional corrective action, if the MSLRM or offgas radiation monitors or both exceed their alarm setpoints.

In response to Condition 1, the licensee stated that the assumptions made in the generic analysis bound those for FitzPatrick. The NRC staff has reviewed the licensee's assumptions for the values such as Chi/Q and power level per assembly and has concluded that the generic analysis assumptions bound those presented in the FitzPatrick analysis.

In response to Condition 2, the licensee's submittal indicated that they have procedures in place which provide reasonable assurance that the plant is capable of responding to increased radiation levels as detected by the offgas monitor; the Annunciator Response Procedures (ARPs) and an Abnormal Operating Procedure (AOP) currently control the plant response. The licensee states that procedures are in place which ensure that actions are taken to limit occupational doses and environmental releases. The licensee has stated that the Fitzpatrick Operating License permits bypassing of the Offgas Treatment System during plant startup. NEDO-31400 states that this condition is acceptable provided the offgas radiation monitors are being utilized to automatically isolate the offgas process line. The Steam Jet Air Ejector isolation feature conforms with the NEDO-31400 criteria, and precludes a direct release to the environment. In the event the offgas system is isolated, the offgas dose is equivalent to the FSAR design basis scenario (with MSIV isolation) since in this case the activity is assumed to be transferred to the main condenser, followed by a ground level release. The NRC conditions stipulated in the safety evaluation for the offgas radiation monitor will be implemented as described for Condition 3. The existing procedures will be revised to remain consistent with the commitments and requirements of the requested change.

The NRC staff has reviewed the licensee's commitment and has determined it is acceptable and responsive to Condition 2, which was required to be addressed per Topical Report NEDO-31400.

In response to Condition 3, the licensee has stated that concurrent with the modification of the MSLRM trip, the alarm setpoints on the MSLRM and offgas radiation monitor will be adjusted to less than or equal to 1.5 times the normal full power N-16 background dose rate. This does account for the increased N-16 carryover due to hydrogen water chemistry. The licensee has also stated that prior to modification of the MSLRM trip, the plant procedures will be revised to require prompt sampling of the reactor coolant to determine the need for corrective actions, if the MSLRM or offgas radiation monitors, or both, exceed their alarm setpoints.

In review of the licensee's commitment, the staff has determined that the Condition 3 has been satisfied.

3.0 SUMMARY

The NRC staff has reviewed the licensee's submittal and safety analysis and concludes that there are no adverse safety implications associated with removal of the MSLRM scram and MSIV closure function since the licensee has provided reasonable assurance that the offsite radiation exposure levels are within the guidelines of 10 CFR Part 100 and SRP 15.4.9. Therefore, the NRC staff finds that TS changes proposed by the licensee are acceptable.

4.0 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.

5.0 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. 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 41513). 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.

6.0 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 Contributors:
R. Frahm
R. Emch

Date: March 9, 1994

Docket No. 50-333 March 9, 1994

Mr. William A. Josiger, Acting
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Josiger:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M86978)

The Commission has issued the enclosed Amendment No. 207 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 July 15, 1993.

The amendment revises the TSs to eliminate the reactor scram and Main Steam Line Isolation Valve closure requirements associated with the Main Steam Line Radiation Monitors. The changes are consistent with Licensing Topical Report NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," dated May 1987.

This license amendment is effective as of the date of its issuance to be implemented prior to startup following the next FitzPatrick refueling outage. Please notify the NRC, in writing, within 30 days of implementing this amendment.

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:

Brian C. McCabe, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 207 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

Distribution:
See attached sheet

*See previous concurrence

LA:PDI-1	PM:PDI-1	*OGC	D:PDI-1		
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