May 16, 1985

Docket No. 50-333

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

Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. 90 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 request dated October 2, 1984 as supplemented October 22, 1984.

This amendment revises the Technical Specifications to permit a temporary increase in the main steam line high radiation scram and isolation setpoints to facilitate the testing of hydrogen addition to coolant water as a potential inhibitor of intergranular stress corrosion cracking. This amendment is in effect only during Operating Cycle 7.

A copy of our Safety Evaluation is enclosed.

Sincerely,

Original signed by/

Harvey I. Abelson, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 90 to License No. DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION	FWitt		
Docket File	SNorris	BGrimes	OPA, CMiles
NRC PDR	HAbelson	TBarnhart (4)	RDiggs
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ORB#2 Reading	LJHarmon	EButcher	Extra - 5
HThompson	ELJordan	ACRS (10)	BGrimes

*Please see previous concurrence page.

DL:ORB#2	DL:ORB#2	DL:ORB#2	OELD	DSI:CPB	DL:AD-OR
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04/10/85	04/10/85	05/02/85	05/07/85	05/02/85	05/15/85

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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. 90
License No. DPR-59

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

   A. The application for amendment by the Power Authority of the State of New York (the licensee) dated October 2, 1984 as supplemented October 22, 1984 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. 90 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 16, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 90

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise the Appendix "A" Technical Specifications as follows:

Remove	Insert
33	33
41a	41a
43a	43a
57	57
64	64
65	65

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 (pressure switches) for the drywell are 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.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector offgas monitors which cause an isolation of the main condenser offgas line. During the Hydrogen Addition Test, the normal background Main Steam Line Radiation Level is expected to increase by a factor of approximately 5 at the maximum hydrogen addition rate as indicated in note 16, Table 3.1-1. The scram setpoint will be reset to three times the projected background radiation level prior to performance of the test. The setpoint will be restored to normal following completion of the hydrogen addition test.

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.

The Control Rod Drive Scram System is designed so that all of the water which

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	$\geq$ 2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	$\leq$ 1045 psig	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	$\leq$ 2.7 psig	X(7)	X (7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	$\geq$ 12.5 in. indicated level ($\geq$ 177 in. above the top of active fuel)	X	X	X	4 Instrument Channels	A
3	High Water Level in Scram Discharge Volume	$\leq$ 34.5 gallons per Instrument Volume	X(2)	X	X	8 Instrument Channels	A
2	Main Steam Line High Radiation	$\leq$ 3x normal full power background (16)	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	$\leq$ 10% valve closure	X(3)(5)	X(3)(5)	X(5)	8 Instrument Channels	A

Table 3.1-1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (Cont'd)

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 placed in the Run position.

16.* During the proposed Hydrogen Addition Test, the normal background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to ≤ three times the increased radiation levels. The test will be conducted at power levels > 80% of normal rated power. During controlled power reduction, the setpoint will be readjusted prior to going below 20% rated power. If due to a recirculation pump trip or other unanticipated power reduction event, the reactor drops below 20% rated power without the setpoint change, control rod withdrawal will be prohibited until the necessary trip setpoint adjustment is made.

*This specification is in effect only during Operating Cycle 7.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.1.2 FSAR. During the Hydrogen Addition Test, the normal background Main Steam Line Radiation Level is expected to increase by approximately a factor of 5 at the peak hydrogen concentration as indicated in note 16, Table 3.1-1. With the hydrogen addition, the fission product release would still be well within the 10 CFR 100 guidelines in the event of a control rod drop accident.

Pressure instrumentation is provided to close the main steam isolation valves in the run mode when the main steam line pressure drop below 825 psig. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the run mode is less severe than the loss of feedwater analyzed in Section 14.5 of the FSAR, therefore, closure of the main steam isolation valves for thermal transient protection when not in the run mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

The trip settings of $\leq$ 300 percent of design flow for this high flow of 40°F above maximum ambient for high temperature are such that uncovering the core is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of $\leq$ 300 percent for high flow and 40°F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high flow temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that uncovering the core is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not de-

TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (2)
2 (6)	Reactor Low Water Level	≥ 12.5 in Indicated Level (≥ 177 in. above the top of active fuel)	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low Water Level	≥ -38 in. indicated level (≥ 126.5 in. above the top of active fuel)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	≤ 2.7 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	≤ 3 x Normal Rated Full Power Background (9)	4 Inst. Channels	B
2	Low Pressure Main Steam Line	≥ 825 psig (7)	4 Inst. Channels	B
2	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Leak Detection High Temperature	≤ 40°F above max ambient	4 Inst. Channels	B
3	Reactor Cleanup System Equipment Area High Temperature	≤ 40°F above max ambient	6 Inst. Channels	C
2	Low Condenser Vacuum Closes MSIV's	≥ 8" Hg. Vac (8)	4 Inst. Channels	B

Table 3.2-1 (Cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

NOTES FOR TABLE 3.2-1

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

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

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

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

   C. Isolate Reactor Water Cleanup System.

   D. Isolate shutdown cooling.

3. Deleted

4. Deleted

5. Two required for each steam line.

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

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

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

9. The trip level setpoint will be maintained at $\leq$3 times normal rated full power background. See note 16 to Table 3.1-1 for re-setting trip level setpoint just prior to the Hydrogen Addition Test, and re-setting of the Main Steam Line Radiation Monitor for power levels below 20%.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 90 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 submittal dated October 2, 1984, as supplemented October 22, 1984, the Power Authority of the State of New York (PASNY/licensee) proposed a Technical Specification (TS) change to permit a temporary increase in the FitzPatrick main steam line high radiation scram and isolation setpoints to facilitate the testing of hydrogen addition water chemistry at FitzPatrick. This proposed change is necessary to the test, since it is anticipated that main steam line radiation levels may increase by a factor of five over the routinely experienced dose rates during maximum hydrogen addition rates. The purpose of this test is to study the feasibility of hydrogen addition to the coolant water to inhibit intergranular stress corrosion cracking (IGSCC) at FitzPatrick. PASNY has evaluated all other aspects of the proposed test under 10 CFR 50.59.

2.0 Evaluation

The main steam line radiation monitors (MSLRM) are used to detect gross failure of fuel cladding during normal operation that may be caused by any number of mechanisms (e.g., pellet-cladding mechanical interaction, manufacturing defects, etc.). When high radiation in the main steam lines is detected by the MSLRM, a reactor trip is initiated to reduce the possibility of additional failure of fuel cladding and, at the same time, the main steam line isolation valves (MSIV) are closed to limit the release of fission products. To perform this reactor trip and main steam line isolation function, the MSLRM trip setting is set high enough above background radiation levels to prevent spurious trips yet low enough to detect gross failures in fuel cladding. For abnormal operational occurrences (transients), the analyses that are performed determine limiting conditions of operation (LCO) and Reactor Protection System trip settings to preclude any fuel failures by meeting specified acceptable fuel design limits (SAFDL) as required by GDC 10. Therefore, credit is not required or taken for an MSLRM-initiated trip in the analysis of transients. In addition, no credit is taken for an MSLRM trip in accident analyses (see below for its use in the control rod drop radiological dose calculation); however, the MSLRM trip will provide a backup trip to other primary Reactor Protection System trips if an accident were to result in significant fuel failures.

In the calculation of the radiological consequences of the control rod drop accident (CRDA), credit is taken for the MSLRM to provide a signal to close the MSIV upon the detection of high radiation in the main steam lines. The total time required to isolate the main steam lines, together with other assumptions, determines the amount of fission product activity transported to the condenser before the main steam lines are isolated. The CRDA analysis, however, does not take credit for a reactor trip from the MSLRM in assuring that the fuel dispersal (enthalpy) criterion is met.

For a CRDA occurring at power levels above 20% of rated power, there is a significant margin to the fuel cladding failure threshold. The CRDA becomes a concern only at power levels below 10%. The licensee has stated that the hydrogen addition test will be conducted at power levels above 80% of rated power and that MSLRM setpoint readjustments will be performed only above 20% of rated power. In addition, the licensee has stated that, during controlled power reduction, restoration to pre-test setpoint values will be performed prior to going below 20% of rated power. If, due to a recirculation pump trip or other unanticipated power reduction event, the reactor power drops below 20% rated power with the MSLRM setpoint at its test value, control rod withdrawal will be prohibited until the necessary setpoint readjustment can be made.

The capability for monitoring fuel defects and failures will be maintained through continued operability of the main steam radiation monitoring scram and isolation system, routine radiation surveys, the performance of daily primary coolant water analyses, and the continued operability of the Steam Jet-Air Ejector Off-Gas Monitor. Furthermore, the licensee's existing quality assurance program and operating procedures, as applied to instrument adjustments, will minimize the potential for error associated with readjusting the MSLRM setpoint.

Based on the licensee's commitment to increase the MSLRM setpoint to its test value only when the plant is operating at power levels greater than 20% rated power, to restore the setpoint to its pre-test value prior to reducing power below 20% of rated power and to prohibit control rod withdrawal in the event of an uncontrolled power reduction below 20% with the MSLRM setpoint at its test value, as well as the licensee's capability to monitor fuel defects and failures during the test, we conclude that the proposed TS changes are acceptable.

We have also reviewed the proposed changes to assure that the licensee considered the radiological implications of the dose rate increase associated with nitrogen 16 (N-16) equilibrium changes during hydrogen addition at boiling water reactors (BWRs). In addition, we have evaluated the submittals to determine whether the licensee adequately considered radiation protection/ALARA measures for the course of the test in accordance with 10 CFR 20.1(c) and Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposure At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

The licensee has indicated that normal radiation protection/ALARA practices and procedures for FitzPatrick will be continued through the test. Additionally, main steam system dose rates will be monitored by surveys on a routine basis, particularly in accessible areas. An overall objective of the test is to determine general in-plant and site boundary dose rate increases as a result of hydrogen addition. Additionally, specific locations where temporary shielding may be needed for long term implementation of hydrogen injection will be identified.

The licensee has taken additional measures to ensure that personnel exposure during the hydrogen addition testing are ALARA. These measures are:

(1) In-plant surveys will be taken at various hydrogen flow rates (i.e., radiation levels),

(2) Area radiation monitors will be logged at specific increments of hydrogen addition,

(3) Site boundary surveys will be conducted with Reuter-Stokes high pressure ionization chambers for measuring N-16 gamma;

(4) Gamma isotopic surveys will be conducted in the environment during the CERT (Constant Extension Rate Test) test by an outside vendor. Surveys will be compared with normal operating data before/after the test.

The staff has discussed details of dose control measures and surveillance efforts planned for the test with licensee representatives. A similar test was conducted for the Dresden 2 and Peach Bottom facilities following a staff review and approval of a similar Technical Specification change. The measures proposed for radiation protection/ALARA at FitzPatrick are consistent with those utilized at Dresden 2 during the successful tests at that unit, where no significant unanticipated radiological problems occurred.

The licensee has a radiation protection/ALARA program which has been recognized as adequate in overall NRC appraisals and includes the capability to conduct special tests and maintenance in accordance with 10 CFR Part 20 and is consistent with the criteria of Regulatory Guide 8.8. An ALARA review of this program will be performed and submitted to NRC within 90 days of completion of testing/implementation.

Based on the adequacy of the licensee's radiation protection/ALARA program, utilization of special surveys to monitor dose rate changes at the site boundary, the success of the initial effort at Dresden 2 and the consistency of that effort with anticipated results, and the licensee's discussion of specific radiation protective/ALARA measures to be utilized, we find that the licensee has the capability to assure worker radiological protection and keep doses as low as is reasonably achievable. Based on these capabilities and the licensee's planned actions, we conclude that the proposed TS changes are acceptable.

3.0 Environmental Consideration

This amendment involves a change in the installation or use of a facility component 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 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.

Principal Contributors: D. Fieno, F. Witt

Dated: May 16, 1985