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

July 23, 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. M85832)

The Commission has issued the enclosed Amendment No. 193 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated December 18, 1992.

The amendment revises TS 4.3.B.1 and associated Bases to require verification of control rod coupling integrity each time a control rod is withdrawn to the "full out" position and prior to declaring a control rod operable after work on a control rod or the control rod drive system that could affect coupling. These changes make the James A. FitzPartrick TS consistent with the guidance provided by the NRC's revised Standard Technical Specifications for General Electric plants (NUREG-1433).

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. 193 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Ralph E. Beedle
Power Authority of the State of New York

James A. FitzPatrick Nuclear
Power Plant

cc:

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DATED: July 23, 1993

AMENDMENT NO. 193 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
J. Menning
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
R. Jones, 8/E/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I
D. Brinkman

cc: Plant Service list

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. 193
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated December 18, 1992, 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. 193, 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

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: July 23, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 193

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
91	91
99	99
100	100

3.3 (cont'd)

B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

4.3 (cont'd)

B. Control Rods

1. Demonstrate that each control rod drive does not go to the overtravel position:

   a. Each time a control rod is withdrawn to the "full out" position.

   b. Prior to declaring a control rod OPERABLE, after work on a control rod or the CRD System that could affect coupling.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3.3 and 4.3 BASES (cont'd)

the control cell geometry and local k_∞. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the analytically strongest) is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38% Δk. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity Margin - Inoperable Control Rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted, it is in a safe position of maximum contribution to shutdown reactivity. If it is in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest operable control rod does not insert.

Inoperable bypassed rods will be limited within any group to not more than one control rod of a (5x5) twenty-five control rod array.

Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rods

1. Coupling verification is performed to ensure the control rod is connected to the Control Rod Drive (CRD). The Surveillance requires demonstrating a CRD does not go to the overtravel position when it is fully withdrawn. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" (notch position 48) position or prior to declaring the control rod to be OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the

3.3 and 4.3 BASES (cont'd)

"full out" position during the performance of SR 4.3.A.2.a. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved, and operating experience related to uncoupling events.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 in. in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the Primary Coolant System. The design basis is given in subsection 3.8.2 of the FSAR, and the safety evaluation is given in subsection 3.8.4. This support is not required if the Reactor Coolant System is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restricts the order of control rod withdrawal and insertion to be equivalent to the Banked Position Withdrawal Sequence (BPWS). These sequences are established such that the drop of any in-sequence control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal/gm. An enthalpy of 280 cal/gm is well below the level at which rapid fuel dispersal could occur (i.e. 425 cal/gm.). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections 3.6.6, 7.7.4.3 and 14.6.1.2 of the FSAR, NEDE-24011 and NEDO-10527 including supplements 1 and 2 to NEDO-10527.

In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 10%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193 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 December 18, 1992, 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 (TS). The requested changes would revise TS 4.3.B.1 and associated Bases to require verification of control rod coupling integrity each time a control rod is withdrawn to the "full out" position and prior to declaring a control rod operable after work on a control rod or the control rod drive system that could affect coupling. The proposed changes would make the James A. FitzPartrick TS consistent with the guidance provided by the NRC's revised Standard Technical Specifications for General Electric plants (NUREG-1433).

2.0 EVALUATION

TS 4.3.B.1 currently requires confirmation of control rod coupling integrity during the initial withdrawal following a refueling outage or after control rod or control rod drive (CRD) maintenance. This confirmation is accomplished by verifying that the CRD does not go to the overtravel position when the control rod is fully withdrawn. TS 4.3.B.1 also currently requires verification of discernible response on the nuclear instrumentation during initial control rod withdrawal after each refueling outage or after maintenance on the control rods or CRDs.

The proposed change to TS 4.3.B.1 would require confirmation of control rod coupling integrity each time a control rod is withdrawn to the "full out" position and prior to declaring a control rod operable after work on a control rod or the CRD system that could affect coupling. Confirmation of coupling is to be accomplished by demonstrating that each control rod drive does not go to the overtravel position when a control rod is fully withdrawn. The proposed change provides additional assurance of control rod to CRD coupling through the additional surveillance testing each time a control rod is fully withdrawn. This additional surveillance is consistent with the guidance provided in the NRC's revised Standard Technical Specifications for General Electric plants (NUREG-1433) and is therefore acceptable.

The requirement to verify discernible response on the nuclear instrumentation during control rod withdrawal does not demonstrate control rod coupling integrity but does demonstrate control rod movement and that the control rods are not stuck. However, TS 4.3.A.2.a. provides adequate demonstration of control rod movement and that the control rods are not stuck on a weekly basis and therefore, deletion of the requirement for discernible response on the nuclear instrumentation from TS 4.3.B.1 is also acceptable. Therefore, we find the proposed technical specification changes 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 16229). 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:
Donald S. Brinkman

Date: July 23, 1993

July 23, 1993

Docket No. 50-333

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
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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. 193 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

PDI-1:LA	PDI-1:PM	PDI-1:PM	SRXB*	OGC*	PDI-1:D
CVogan	DBrinkman:avl	JMenning	RJones	JHull	RACapra
7/6/93	7/6/93	7/6/93	06/23/93	06/23/93	6/23/93

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