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

February 8, 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. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M84621)

The Commission has issued the enclosed Amendment No. 204 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 28, 1992.

The amendment revises the flow requirement for the Core Spray (CS) pumps and the associated Bases. The change reduces the CS pump minimum flow acceptance criteria by 10% and addresses an inconsistency between the system leakage rates in the Updated Final Safety Analysis Report and the Technical Specifications. Specifically, the surveillance testing required by the TSs is intended to verify the capability of a core spray pump to deliver acceptable flow to the core. The new CS pump minimum flow acceptance criteria now accounts for system leakage that is not delivered to the core.

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. 204 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

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James A. FitzPatrick Nuclear
Power Plant

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DATED: February 8, 1994

AMENDMENT NO. 204 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
R. Jones, 8/E/23
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

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. 204
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 28, 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. 204 , 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: February 8, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 204

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
113	113
122	122
132	132

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 204 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 28, 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 (TSs). The requested changes would revise the flow requirement for the Core Spray (CS) pumps and the associated Bases. The change reduces the CS pump minimum flow acceptance criteria by 10% and addresses an inconsistency between the system leakage rates in the Updated Final Safety Analysis Report (UFSAR) and the TSs. Specifically, the surveillance testing required by the TSs is intended to verify the capability of a core spray pump to deliver acceptable flow to the core. The new CS pump minimum flow acceptance criteria now accounts for system leakage that is not delivered to the core.

2.0 DISCUSSION

The Emergency Core Cooling System (ECCS) for the James A. FitzPatrick Nuclear Power Plant consists of the following systems:

(a) High Pressure Coolant Injection (HPCI) system.
(b) Automatic Depressurization System (ADS).
(c) Core Spray System (consisting of two loops, one pump per loop, each pump powered from separate diesel generators).
(d) Low Pressure Coolant Injection (LPCI) mode of the RHR system (consisting of two loops, two pumps per loop, with each pump in a loop powered from a different diesel generator).

The CS System is one of several ECCSs used to mitigate the consequences of loss-of-coolant accidents (LOCAs). Core spray is comprised of two subsystems (independent loops) with each subsystem consisting of a 100 percent capacity motor driven pump, piping, valves and a sparger to transfer water from the suppression pool to the reactor vessel. The A and B core spray lines enter the reactor vessel through two nozzles located 180° apart to provide physical

separation. Each nozzle has a thermal sleeve that is welded into a T box. Two pipes are run from the T box to form a semicircular header with a downcomer at each end. The downcomer has an elbow where the spray lines pass through the upper part of the shroud and into the spray sparger.

The core spray pumps are tested in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and Technical Specifications 4.5.A.1.b and 4.5.F.1 to ensure that adequate emergency core cooling capacity is available. The current requirement in the TSs is that core spray pumps deliver at least 4625 gpm against a system head corresponding to a reactor vessel pressure greater than or equal to 113 psi above primary containment pressure. The surveillance test should also account for system leakage that is not delivered to the core. Surveillance testing is conducted in accordance with the In-Service Testing (IST) program.

The purpose of this change request is to reduce the flow requirement for core spray surveillance testing. The change will also clarify the testing requirements for system leakage.

3.0 EVALUATION

The CS system is an emergency core cooling system used to mitigate the consequences of loss-of-coolant accidents and to provide inventory makeup in the alternate shutdown cooling mode in the event that the suction path from the reactor becomes unavailable for shutdown cooling or reactor inventory is lost. The surveillance testing required by Technical Specifications 4.5.A.1.b and 4.5.F.1 is intended to verify the capability of the core spray pump to deliver to the core the flow assumed in the SAFER/GESTR LOCA analysis ("JAFNPP SAFER/GESTR - LOCA Analysis," General Electric Company, NEDC-31317P, October 1986).

A sensitivity analysis (GE "Sensitivity of the JAFNPP Safety Systems Performance to Fundamental System Parameters," July 1986) was performed by the licensee based on the SAFER/GESTR LOCA analysis to assess the conservatism in current and proposed TS requirements for ECCS components. The sensitivity analysis varied component performance requirements to determine the sensitivity of the SAFER/GESTR LOCA analysis results for the design basis accident (i.e., recirculation line break). The flow rates for CS, LPCI, and HPCI were reduced by 10% over their entire range in the analysis. For CS, the reduction was equivalent to a minimum rated flow of 4,163 gpm to the spray nozzles at a reactor vessel pressure equal to 113 psi above containment pressure.

Reducing the CS pump minimum flow acceptance criteria by 10% effectively reduces the criteria from 4625 gpm to 4163 gpm. However, the CS flowrate used in the LOCA analysis is the CS flowrate inside the core shroud. System leakage (i.e., the difference between CS pump flowrate and CS flowrate inside

the core shroud) must also be accounted for when establishing CS pump minimum flow acceptance criteria.

When the FitzPatrick plant was being designed, leakage was postulated to occur from the thermal sleeve between the T box and vessel nozzle and a quarter inch vent hole in the T box that allowed for release of noncondensable gases. The leakage requirement included in this proposed TS change is based on an assessment of the actual system leakage. The assessment was part of the analysis used to validate CS flowrate after repair of a crack in the core spray piping outside the shroud on the "B" loop. The assessment identifies the elimination of thermal sleeve leakage before plant operation and calculates the upper bound leakage from the upper T box vent hole (0.25 plus or minus .05 inch) as less than 20 gpm. The crack in the "B" loop core spray piping was repaired by welding a clam shell on the upper riser outside the shroud. The weld covers only 5/6 of the circumference of the pipe and calculations conservatively conclude that leakage from the unwelded sector is less than 40 gpm.

Based on the above, the required CS flowrate must allow for leakage of 20 gpm and 60 gpm to the "A" and "B" loops, respectively. Since 4163 gpm is required for delivery to the core, the new acceptance criteria (4265 gpm) conservatively bounds the calculated maximum leak rate.

The LOCA analysis performed using the approved SAFER/GESTR evaluation models per Appendix K to 10 CFR Part 50, demonstrates that, for a 10% reduction in all ECCS flow rates, the peak cladding temperature (PCT) will increase by 88 °F. Since the current limiting licensing PCT is more than 600 °F below the allowable 2200 °F, the plant will continue to meet the requirements of Appendix K to 10 CFR Part 50 and 10 CFR 50.46 with over 500 °F margin. The statistical upper bound PCT remains at least 150 °F less than the Appendix K case and will meet the 1600 °F limit. The metal water requirements of 10 CFR 50.46 were also evaluated using the reduced CS flow rate. This evaluation indicated an increase in temperature will result in a small increase in the metal water reaction for the limiting break accidents. However, even with this small increase, the metal water requirements of 10 CFR 50.46 are still met. Therefore, the proposed reduced CS flow rate of 4265 gpm is acceptable based on LOCA considerations.

The requirements for inventory makeup to mitigate the consequences of inadvertent draindown while the unit is shutdown were also evaluated. This evaluation concluded that the limiting double ended guillotine break of the recirculation line is larger than any opening associated with draindown and, therefore, the requirements for makeup to mitigate the consequences of an inadvertent draindown while shutdown are bounded by the LOCA. Analysis indicates that a single CS system is capable of long-term cooling for a LOCA and a single CS system is suitable for the cold condition. Therefore, the analysis concludes that there is no adverse safety impact associated with this change to the flow criteria.

The LPCI system is relied on to supply makeup water to the reactor during postulated fire events in accordance with Appendix R of 10 CFR Part 50. These are not pipe break events but are postulated fire events which can threaten the ability of the plant to maintain reactor vessel water inventory, depleted by decay heat and sensible heat boiloff. The core spray pumps were assumed to be inoperable. Therefore, there is no increase in the PCT for the worst case Appendix R fire due to a reduction in core spray flow.

In conclusion, the analysis demonstrates that there is no safety impact associated with this change to the flow criteria. The conclusions of the plant's accident analysis as documented in the UFSAR and the NRC staff's SER at operating license stage are not altered by these changes to the TSs. Therefore, based on the above evaluation, the proposed change to the CS pump flow acceptance criteria is 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 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 (57 FR 58250). 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: Brian C. McCabe

Date: February 8, 1994

3.5 (cont'd)

4.5 (cont'd)

b.	Flow Rate Test - Core spray pumps shall deliver at least 4,265 gpm against a system head corresponding to a reactor vessel pressure greater than or equal to 113 psi above primary containment pressure.	Once/3 Months
c.	Pump Operability	Once/month
d.	Motor Operated Valve	Once/month
e.	Core Spray Header Δp Instrumentation Check Calibrate Test	 Once/day Once/3 months Once/3 months
f.	Logic System Functional Test	Once/each operating cycle
g.	Testable Check Valves	Tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

3.5 (cont'd)

F. ECCS-Cold Condition

1. A minimum of two low pressure Emergency Core Cooling subsystems shall be operable whenever irradiated fuel is in the reactor, the reactor is in the cold condition, and work is being performed with the potential for draining the reactor vessel.

2. A minimum of one low pressure Emergency Core Cooling subsystem shall be operable whenever irradiated fuel is in the reactor, the reactor is in the cold condition, and no work is being performed with the potential for draining the reactor vessel.

3. Emergency Core Cooling subsystems are not required to be operable provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and the water level above the fuel is in accordance with Specification 3.10.C.

4. With the requirements of 3.5.F.1, 3.5.F.2, or 3.5.F.3 not satisfied, suspend core alterations and all operations with the potential for draining the reactor vessel. Restore at least one system to operable status within 4 hours or establish Secondary Containment Integrity within the next 8 hours.

4.5 (cont'd)

F. ECCS-Cold Condition

Surveillance of the low pressure ECCS systems required by 3.5.F.1 and 3.5.F.2 shall be as follows:

1. Perform a flowrate test at least once every 3 months on the required Core Spray pump(s) and/or the RHR pump(s). Each Core Spray pump shall deliver at least 4,265 gpm against a system head corresponding to a reactor vessel pressure greater than or equal to 113 psi above primary containment pressure. Each RHR pump shall deliver at least 8910 gpm against a system head corresponding to a reactor vessel to primary containment differential pressure of $\geq$ 20 psid.
2. Perform a monthly operability test on the required Core Spray and/or LPCI motor operated valves.
3. Once each shift verify the suppression pool water level is greater than or equal to 10.33 ft. whenever the low pressure ECCS subsystems are aligned to the suppression pool.
4. Once each shift verify a minimum of 324 inches of water is available in the Condensate Storage Tanks (CST) whenever the Core Spray System(s) is aligned to the tanks.

4.5 BASES

The testing interval for the Core and Containment Cooling Systems is based on a quantitative reliability analysis, industry practice, judgement, and practicality. The Emergency Core Cooling Systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 450 psig; thus, during operation even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems will be automatically actuated during a refueling outage. In the case of the Core Spray System, condensate storage tank water will be pumped to the vessel to verify the operability of the core spray header. To increase the availability of the individual components of the Core and Containment Cooling Systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise, the pumps and motor-operated valves are also tested each month to assure their operability. The combination automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining cooling equipment. Consistent with the definition of operable in Section 4.0.C, demonstrate means conduct a test to show; verify means that the associated surveillance activities have been satisfactorily performed within the specified time interval.

The RCIC flow rate is described in the UFSAR. The flow rates to be delivered to the reactor core for HPCI, the LPCI mode of RHR, and CS are based on the SAFER/GESTR LOCA analysis. The flow rates for the LPCI mode of RHR and CS are modified by a 10 percent reduction from the SAFER/GESTR LOCA analysis. The reductions are based on a sensitivity analysis (General Electric MDE-83-0786) performed for the parameters used in the SAFER/GESTR analysis.

The CS surveillance requirement includes an allowance for system leakage in addition to the flow rate required to be delivered to the reactor core. The leak rate from the core spray piping inside the reactor but outside the core shroud is assumed in the UFSAR and includes a known loss of less than 20 gpm from the 1/4 inch diameter vent hole in the core spray T-box connection in each of the loops, and in the B loop, a potential additional loss of less than 40 gpm from a clamshell repair whose structural weld covers only 5/6 of the circumference of the pipe. Both of these identified sources of leakage occur in the space between the reactor vessel wall and the core shroud. Therefore flow lost through these leak sources does not contribute to core cooling.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC Systems are filled provides for a visual observation that water flows from a high point vent. This ensures that

February 8, 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. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M84621)

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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:
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. 204 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page
*See previous concurrence

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