October 29, 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. 95 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 April 26, 1985.

The amendment revises the Technical Specifications to reduce testing requirements for the Emergency Diesel Generators in response to NRC Generic Letter 84-15.

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. 95 to License No. DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. John C. Brons
Power Authority of the State of New York

James A. FitzPatrick Nuclear
Power Plant

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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. 95
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 April 26, 1985, 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. 95 , 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: October 29, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 95

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

114

115

115a

116

117

126

217

3.5 (cont'd)

2. From and after the date that one of the Core Spray Systems is made or found inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the system is made operable earlier, provided that during the 7 days all active components of the other Core Spray System and the LPCI System shall be operable.

3. The LPCI mode of the RHR System shall be operable whenever irradiated fuel is in the reactor and prior to reactor startup from a cold condition, except as specified below.

a. From the time that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the pump is made operable earlier provided that during such 7 days the remaining active components of the LPCI, containment spray mode, and all active components of both Core Spray Systems are operable.

4.5 (cont'd)

2. When it is determined that one Core Spray System is inoperable, the operable Core Spray System, and the LPCI System, shall be demonstrated to be operable immediately. The remaining Core Spray System shall be demonstrated to be operable daily thereafter.

3. LPCI System testing shall be as specified in 4.5.A.1a, b, c, d, f and g except that three RHR pumps shall deliver at least 23,100 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

a. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI, containment spray subsystem and both Core Spray Systems required for operation shall be demonstrated to be operable immediately, and the remaining RHR pumps shall be demonstrated to be operable daily thereafter.

3.5 (Cont'd)

b. From the time that the LPCI mode is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the LPCI mode is made operable earlier provided that during these 7 days all active components of both Core Spray Systems and the containment spray subsystem (including two RHR pumps) shall be operable.

c. When the reactor water temperature is greater than 212°F, the motor operator for the RHR cross-tie valve (MOV20) shall be maintained disconnected from its electric power source. It shall be maintained chain-locked in the closed position. The manually operated gate valve (10-RHR-09) in the cross-tie line, in series with the motor operated valve, shall be maintained locked in the closed position.

4.a. The reactor shall not be started up with the RHR System supplying cooling to the fuel pool.

b. The RHR System shall not supply cooling to the spent fuel pool when the reactor coolant temperature is above 212°F.

4.5 (Cont'd)

b. When it is determined that the LPCI mode is inoperable, both Core Spray Systems, and the containment spray subsystem shall be demonstrated to be operable immediately and daily thereafter.

c. The power source disconnect and chain lock to motor operated RHR cross-tie valve, and lock on manually operated gate valve shall be inspected once each operating cycle to verify that both valves are closed and locked.

3.5 (Cont'd)

5. All recirculation pump discharge valves and bypass valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

6. If the requirements of 3.5.A cannot be met, the reactor shall be placed in the cold condition within 24 hrs.

B. CONTAINMENT COOLING SUBSYSTEM MODE (OF THE RHR SYSTEM)

1. Both subsystems of the containment cooling mode, each including two RHR, one ESW pump and two RHRSW pumps shall be operable whenever there is irradiated fuel in the reactor vessel, prior to startup from a cold condition, and reactor coolant temperature $\geq$ 212°F except as specified below:

2. Continued reactor operation is permissible for 30 days with one spray loop inoperable and with reactor water temperature greater than 212°F.

4.5 (Cont'd)

5. All recirculation pump discharge and bypass valves shall be 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.

B. CONTAINMENT COOLING SUBSYSTEM MODE (OF THE RHR SYSTEM)

1. Subsystems of the containment cooling mode are tested in conjunction with the test performed on the LPCI System and given in 4.5.A.1.a, b, c, and d. Residual heat removal service water pumps, each loop consisting of two pumps operating in parallel, will be included in testing, supplying 8,000 gpm. The Emergency Service Water System, each loop of which consists of a single operating emergency service water pump of 3,700 gpm will be tested in accordance with Section 4.11D.

During each five-year period, an air test shall be performed on the containment spray headers and nozzles.

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

3.5 (Cont'd)

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

4. Should one of the containment cooling subsystems become inoperable, continued reactor operation is permissible for a period not to exceed 7 days, unless such subsystem is sooner made operable provided that during such 7 days all active components of the other containment cooling subsystem are operable.

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

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

4.5 (Cont'd)

3. When one containment cooling subsystem loop becomes inoperable, the operable loop shall be demonstrated to be operable immediately and daily thereafter.

3.5 (cont'd)

DELETED

C. HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)

1. The HPCI System shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition, except as specified below:

4.5 (cont'd)

C. HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)

Surveillance of HPCI System shall be performed as follows provided reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within 10 days of continuous operation from the time steam becomes available.

1. HPCI System testing shall be as specified in 4.5.A.1.a, b, c, d, f, and g except that the HPCI pump shall deliver at least 4,250 gpm against a system head correspon-ding to a reactor vessel pressure of 1120 psig to 150 psig.

3.5 BASES (cont'd)

the RHR System in conjunction with the Core Spray System provides adequate cooling for break areas of approximately 0.2 sq. ft. up to and including the double-ended reactor recirculation line break without assistance from the high pressure Emergency Core Cooling Systems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference 8. Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the Core Spray and LPCI Systems constitute a 1-out-of-2 systems; however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in Reference 8 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgement.

Should one Core Spray System become inoperable, the remaining Core Spray and the entire LPCI System are available should the need for core cooling arise. To assure that the remaining Core Spray and LPCI Systems are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves. Based on judgements of the reliability of the remaining systems, i.e., the Core Spray and LPCI, a seven-day repair period was obtained.

Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment are available. Three RHR pumps in conjunction with the Core Spray System will perform the core cooling function. Because of the availability of the majority of the

3.9 (cont'd)

3. From and after the time that one of the Emergency Diesel Generator Systems is made or found to be inoperable, continued reactor operation is permissible for a period not to exceed 7 days provided that the two incoming power sources are available and connected to the emergency bus associated with the inoperable Emergency Diesel Generator System and that the remaining Diesel Generator System is operable. At the end of the 7-day period, the reactor shall be placed in a cold condition within 24 hours, unless one or both diesel generator systems are made operable sooner.

4. When both Emergency Diesel Generator Systems are made or found to be inoperable, a reactor shutdown shall be initiated within two hours and the reactor placed in a cold condition within 24 hours after initiation of shutdown.

3. The emergency diesel generator system instrumentation shall be checked during the monthly generator test.

4. Once each operating cycle, the conditions under which the Emergency Diesel Generator System is required will be simulated to demonstrate that the pair of diesel generators will start, accelerate, force parallel, and accept the emergency loads in the prescribed sequence.

5. Once within one hour and at least once per twenty-four hours thereafter while the reactor is being operated in accordance with Specifications 3.9.B.1, 3.9.B.2, and 3.9.B.3, the availability of the operable Emergency Diesel Generators shall be demonstrated by manual starting and force paralleling where applicable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 95 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

The reliability of emergency diesel generators (DG) is one of the main factors affecting the risk from station blackout. The improvement of DG reliability can therefore reduce the risk of core damage from station blackout events. The NRC staff has concluded that excessive testing results in degradation of diesel engines and the potential for reduced reliability. The staff is concerned with the number of additional DG tests required for earlier-licensed operating plants, under their current technical specifications, compared with more recently licensed plants using the Standard Technical Specifications. In an effort to reduce excessive testing of DGs in these older plants and to make their technical specifications comparable with the Standard Technical Specifications in this regard, Generic Letter (GL) 84-15 (D. Eisenhut to All Licensees, dated July 2, 1984) recommended that the requirement for testing DGs when subsystems of the emergency core cooling system (ECCS) are declared inoperable, be deleted from plant unique technical specifications. Subsequently, the affected licensees were invited to submit revised technical specifications to reflect this change.

2.0 EVALUATION

By letter dated April 26, 1985, PASNY submitted proposed revisions to the DG technical specifications (TS). The proposed TS and associated bases would eliminate the requirement for DG testing when subsystems of the ECCS are declared inoperable. The specific changes are as follows:

Surveillance/Bases	Page	Inoperable ECCS Equipment
1. 4.5.A.2	114	one core spray system
2. 4.5.A.3(a)	114	one of the RHR pumps
3. 4.5.B.3(b)	115	LPCI mode
4. 4.5.B.3	116	one containment cooling subsystem

We have reviewed the above revisions to the DG TS and associated bases and find the changes consistent with the intent of Generic Letter 84-15 to improve and maintain DG reliability by reducing excessive DG testing. Therefore, we find the proposed technical specification changes acceptable.

To further reduce the number of DG starts, the licensee's letter dated April 26, 1985 also proposed to revise TS section 4.9.B.5 on page 217. In the current FitzPatrick TS, the DG are required to be tested once every 8 hours when either one or both offsite power sources or when one of the DG is declared inoperable. The proposed revision would change the requirement for DG testing from once every 8 hours to once every 24 hours. We find this change consistent with the intent of Generic Letter 84-15 to improve and maintain DG reliability by reducing excessive DG testing. Therefore, we find the proposed change acceptable.

The licensee has also proposed changes that would delete the requirement that emergency diesel generators shall be operable from the Limiting Conditions for Operation (LCO) which are applicable when the following systems are declared inoperable: Core Spray (CS); Low Pressure Coolant Injection (LPCI) mode of Residual Heat Removal (RHR); and Containment Cooling. This deletion has been proposed because operability requirements and the LCO for the emergency diesel generators are already specified in Section 3.9.B of the TS under "Emergency A-C Power System." In addition Section 3.0.E of the FitzPatrick TS states that reactor operation is governed by the time limits of the Action Statement of the LCO for the emergency power source; and, not by the Action Statement of the individual system that is determined to be inoperable due to the inoperability of its emergency power source. We have reviewed the proposed changes and find that they will eliminate redundancy and create consistency within the TS. On this basis, we find the proposed changes acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

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 Contributor: O. Chopra

Dated: October 29, 1985