Docket File

July 7, 1981

RECEIVED JUL 09 1981 U.S. NUCLEAR REGULATORY COMMISSION

Docket No. 50-333

Mr. George T. Berry
President & Chief Operating Officer
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 57 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Plant. The amendment consists of changes to the license and the Technical Specifications in response to your application dated June 24, 1981.

These changes to the Technical Specifications, as agreed to by members of your staff, involve incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. These requirements are related to: (1) Emergency Power Supply/Inadequate Core Cooling, (2) Valve Position Indication, (3) Containment Isolation, (4) Shift Technical Advisors, (5) Integrity of Systems Outside Containment, and (6) Iodine Monitoring.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 57 to DPR-59
2. Safety Evaluation
3. Notice

cc w/enclosures
See next page

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NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO 1980—329-824

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Power Authority of the State
of New York

cc:

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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. 57
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 June 24, 1981, 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, Facility Operating License No. DPR-59 is hereby amended by revising paragraph 2.C.(2) and adding paragraphs 2.C.(4) and 2.C.(5) to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 57 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(4) Systems Integrity

The licensee shall implement a program to reduce leakage from the systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing maintenance and periodic visual inspection requirements, and

2. Leak test requirements for the systems at a frequency not to exceed operating cycle intervals.

(5) Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in areas vital to the mitigation of or recovery from an accident. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 7, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 57

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove	Insert
76	76
76a	76a
76b	76b
84	84
85	85
248	248
248a	248a
249	249
260	260

TABLE 3.2-6

SURVEILLANCE INSTRUMENTATION

Miminum No. of Operable Instrument Channels	Instrument	Type Indication and Range	No. of Channels Provided by Design	Action
2	(Reactor Level ((Note 3) ((Reactor Level ((Note 4)	Indicator) 0 - +60)) Recorder) 0 - +60)	5	(13)(2)
1	Reactor Level	Indicator -150 - +60	2	(2)
2	(Reactor Pressure ((Note 5) ((Reactor Pressure ((Note 6)	Indicator) 0-1200 psig)) Recorder) 0-1200 psig)	5	(1)(2)
1	(Drywell Pressure ((Narrow Range) ((((Drywell Pressure ((Wide Range) ((	(Narrow Range) Indicator) Recorder) 10 - 19 psia)) (Wide Range)) Indicator) Recorder) 0 - 100 psia)	2	(2)
2	(Drywell Temperature (((Drywell Temperature	Indicator) 50 - 250° F)) Recorder) 50 - 350° F)	4	(1)(2)
2	(Suppression Chamber (Temperature ((Suppression Chamber (Temperature	Indicator) 50 - 250° F)) Recorder) 50 - 350° F)	4	(1)(2)

TABLE 3.2-6

SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	No. of Channels Provided by Design	Action
1	(Suppression Chamber (Water Level ((Wide Range) ((Suppression Chamber (Water Level ((Narrow Range)	Indicator) Recorder) -72 to + 72 inches)) Indicator) Recorder) -6 to +6 inches)	2	(2)
N/A	Control Rod Position Indication	Indicator Position 00 to 48	1	(7)
2	Source Range Monitors	Indicator Recorder 1 to 10^6 cps	4	(8)
3	Intermediate Range Monitor	Indicator Recorder 10^{-4} to 40% Rated Power	8	(8)(9)
2	Average Power Range Monitor	Indicator Recorder 0-125% Rated Power	6	(8)(9)
1	Drywell-Suppression Chamber Differential Pressure	Recorder 0 to 5 psi Computer 0 to 5 psi	2	(2)
1	Safety/Relief Valve Position Indicator (Note 10)	Indicator Open/Closed	2	(12)(11)

NOTES FOR TABLE 3.2-6

1. From and after the date that the minimum number of operable instrument channels is one less than the minimum number specified for each parameter, continued operation is permissible during the succeeding 30 days unless the minimum number specified is made operable sooner.

NOTES FOR TABLE 3.2-6 (CONTINUED)

2. In the event that all indications of this parameter is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

3. Three (3) indicators from level instrument channel A, B, & C. Channel A or B are utilized for feedwater control, reactor water high and low level alarms, recirculation pump runback. High level trip of main turbine and feedwater pump turbine utilizes channel A, B, & C.

4. One (1) recorder utilized the same level instrument channel as selected for feedwater control.

5. Three (3) indicators from reactor pressure instrument channel A, B, & C. Channel A or B are utilized for feedwater control and reactor pressure high alarm.

6. One (1) recorder. Utilizes the same reactor pressure instrument channel as selected for feedwater control.

7. The position of each of the 137 control rods is monitored by the Rod Position Information System. For control rods in which the position is unknown, refer to Paragraph 3.3.A.

8. Neutron monitoring operability requirements are specified by Table 3.1-1 and Paragraph 3.3.B.4.

9. A minimum of 3 IRM or 2 APRM channels respectively must be operable (or tripped) in each safety system.

10. Each Safety Relief Valve is equipped with two primary acoustical detectors (of which one is in service). A thermocouple detector serves as a secondary indicator.

11. From and after the date that none of the acoustical detectors is operable but the thermocouple is operable, continued operation is permissible until the next outage in which a primary containment entry is made. Both acoustical detectors shall be made operable prior to restart.

12. In the event that both primary and secondary indications of this parameter for any one valve are disabled and neither indication can be restored in forty-eight (48) hours, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in twelve (12) hours and in a Cold Shutdown within the next twenty-four (24) hours.

13. From and after the date that the minimum number of operable instrument channels is one less than the minimum number specified for each parameter, continued operation is permissable during the succeeding 7 days unless the minimum number specified is made operable sooner.

TABLE 4.2-6

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLIANCE INSTRUMENTATION

INSTRUMENT CHANNEL	CALIBRATION FREQUENCY	INSTRUMENT CHECK
1.) Reactor Water Level	Once/6 months	Once Each Shift
2.) Reactor Pressure	Once/6 months	Once Each Shift
3.) Drywell Pressure	Once/6 months	Once Each Shift
4.) Drywell Temperature	Once/6 months	Once Each Shift
5.) Suppression Chamber Temperature	Once/6 months	Once Each Shift
6.) Suppression Chamber Water Level	Once/6 months	Once Each Shift
7.) Control Rod Position Indication	N/A	Once Each Shift
8.) Neutron Monitoring (APRM)	Five/week	Once Each Shift
9.) Neutron Monitoring (IRM and SRM)	Note 10	Note 10
10.) Drywell-Suppression Chamber Differential Pressure	Once/6 months	Once Each Shift
11.) Safety/Relief Valve Position Indicator (Primary)	Note 11	Once/Month
12.) Safety/Relief Valve Position Indication (Secondary)	Note 11	Once/Month

NOTES FOR TABLES 4.2-1 THROUGH 4.2-6

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 instrument operate in a environment similar to that of JAFNPP.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed prior to each startup or prior to preplanned shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

3. 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.

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

5. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2-1 since they are tested on Table 4.1-2.

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

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

8. Uses same instrumentation as Main Steam Line High Radiation. See Table 4.1-2.

9. See Technical Specification 1.0.F.4, Definitions, for meaning of term, "Instrument Check".

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

11. Functional test is performed once each operating cycle.

6. In additic to items 1, 2 & 3 above, 'o additional operators shall be readily available on site whenever the reactor is in other than cold shutdown. During cold shutdown, an additional operator shall be readily available on site.

7. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

8. In the event of illness or absenteeism up to two (2) hours is allowed to restore the shift crew or fire-brigade to normal complement.

9. A Fire Brigade of five (5) or more members shall be maintained on site at all times. This excludes two (2) members of the minimum shift crew necessary for safe shutdown and any personnel required for other essential functions during a fire emergency.

10. A Shift Technical Advisor shall be on site and readily available to the control room except during the cold shutdown or refuel mode.

6.3 PLANT STAFF QUALIFICATIONS

The minimum qualifications with regard to educational background and experience for plant staff positions shown in Fig. 6.2-1 shall meet or exceed the minimum qualifications of ANSI NI8.1-1971 for comparable positions; except for the Radiation and Environmental Services Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Coordinator to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI NI8. 1-1971.

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55, Appendix A. In addition fire brigade training shall meet or exceed the requirements of NFPA 27-1975, except for Fire Brigade training sessions which shall be held at least quarterly. The effective date for implementation of fire brigade training is March 17, 1978.

6.5 REVIEW AND AUDIT

Two seperate review groups for the review and audit of plant operations have been constituted. One of these, the Plant Operating Review Committee (PORC), is an onsite group. The other is an independent review and audit group, the offsite Safety Review Committee (SRC).

6.5.1 PLANT OPERATING REVIEW COMMITTEE (PORC)

(A) Membership

The PORC is comprised of the Resident Manager (Chairman), Superintendent of Power (Vice Chairman), Operations Superintendent, Maintenance Superintendent, Technical Services Superintendent, Instrument and Control Superintendent, Radiological and Environmental Services Superintendent and Reactor Analyst. Special consultant to provide expert advice may be utilized when the nature of a particular problem dictates.

(B) Alternates

Alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in PORC activities at any one time.

(C) Meeting Frequency

Meetings will be called by the Chairman as the occasions for review or investigation arise. Meetings will be no less frequent than once a month.

(D) Quorum

The Chairman or Vice Chairman and four members, including designated alternates, shall constitute a quorum.

(E) Responsibilities

1. Review plant procedures, and changes thereto, required by Specification 6.8.

2. Review proposed tests and experiments that affect nuclear safety.

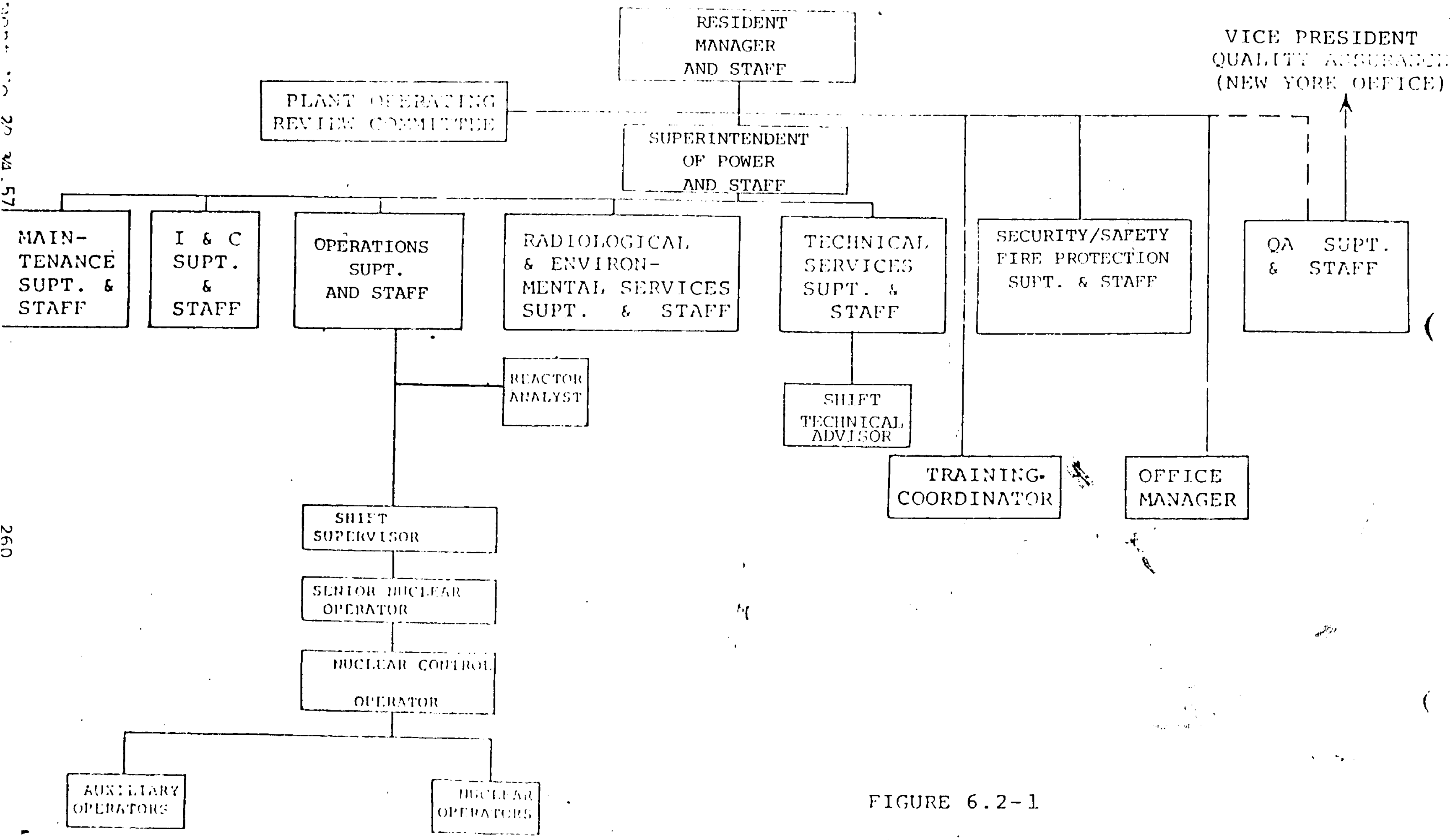
3. Review proposed changes to the Operating License and Technical Specifications.

4. Review proposed changes or modifications to plant systems or equipment that affect nuclear safety

5. Investigate violations of the Technical Specifications and prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Resident Manager, who will forward the report to the Manager - Nuclear Operations and to the Chairman of the Safety Review Committee.

6. Review plant operations to detect potential safety hazards.

7. Review the Security Plan and implementing procedures annually.



*SRO - SENIOR REACTOR OPERATOR
**RO - REACTOR OPERATOR

FIGURE 6.2-1

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
PLANT STAFF ORGANIZATION

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 57 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 June 24, 1981, the Power Authority of the State of New York (licensee) proposed changes to the Technical Specifications (TS) appended to Facility Operating License No. DPR-59. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in response to the NRC's letter dated July 2, 1980. These changes are discussed in Sections 2 and 3 of this evaluation.

2.0 Background Information

By our letter dated September 13, 1979 we issued to all operating nuclear power plants requirements established as a result of our review of the Three Mile Island (Unit 2) accident. Certain of these requirements, designated Lessons Learned Category "A", were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to the licensee dated March 25, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Category "A" requirements, we requested that licensees amend their Technical Specifications to incorporate additional Surveillance Requirements and Limiting Conditions of Operation, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application, dated June 24, 1981, is in response to our request. Each of the issues identified by the NRC and the licensee's response is discussed in the following evaluation.

3.0 Evaluation

1) Emergency Power Supply/Inadequate Core Cooling

As applicable to Boiling Water Reactors (BWR's), we indicated that water level instrumentation is important to post-accident monitoring and that surveillance of this instrumentation should be performed.

The licensee's response did not address the requirement for returning an inoperable level instrument to operable. Subsequently, the licensee has agreed to a seven day corrective action requirement if one level instrument is inoperable, and an immediate shutdown (cold shutdown in 30 hours) requirement in the event both level instruments are found to be inoperable. We agree that these corrective action statements are adequate and therefore satisfy our guidelines.

We have reviewed the proposed Technical Specifications for water level instrumentation. The surveillance requirements for instrument checks (once per shift) and calibration (once per six months) meet our guidelines. Based on this, we conclude that the licensee's response satisfies our request.

2) Valve Position Indication

Our requirements for installation of a reliable position indicating system for relief and safety valves was based on the need to provide the operator with a diagnostic aid to reduce the ambiguity between indications that might indicate either an open relief/safety valve or a small line break. Such a system did not need to be safety grade provided that backup methods of determining valve position are available.

The required indication was to be provided to plant operators located in the control room. The staff has found acceptable two methods of satisfying this requirement: (1) Separate audible and visual indication in the control room for each valve, or (2) Use of the control room computer to obtain information for each specific valve.

We have reviewed the proposed Technical Specifications for safety relief valve position indication. We agree that an extended period of inoperability for one of the two indicators for each valve is acceptable. However, if both indicators for a particular valve are inoperable, repairs should be accomplished within 48 hours or an orderly shutdown should be initiated. The licensee has agreed to this requirement which is reflected in the attached Technical Specifications. Based on this, we conclude that the licensee has satisfied this requirement.

3) Containment Isolation

Our request indicated that the specifications should include a Table of Containment Isolation Valves which reflect the diverse isolation signal requirement of this Lessons Learned issue. The licensee response indicated that this requirement is presently covered in the FitzPatrick Technical Specifications 3.2 and 4.2 and Bases. We have reviewed these Technical Specifications and Bases and conclude that the licensee has adequately responded to this requirement.

4) Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The STA function includes both accident and operating experience assessment. The licensee response proposed TS changes which provide for the Shift Technical Advisor. We have reviewed these changes and conclude that the licensee has satisfied this requirement.

5) Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment.

By letter dated June 24, 1981, the licensee agreed to a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program includes (1) provisions establishing preventive maintenance and periodic visual inspection requirements, and (2) leak test requirements for each system at a frequency not to exceed refueling cycle intervals. We have reviewed this program and conclude that the licensee has satisfied this requirement. The proposed change to Operating License DPR-59 will ensure compliance.

6) Iodine Monitoring

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions.

By letter dated June 24, 1981, the licensee agreed to a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions.

This program includes (1) Training of personnel, (2) Procedures for monitoring, and (3) Provisions for maintenance of sampling and analysis equipment. We have reviewed this program and conclude that the licensee has satisfied this requirement. The proposed change to Operating License DPR-59 will ensure compliance.

4.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: July 7, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 57 to Facility Operating License No. DPR-59 issued to the Power Authority of the State of New York (the licensee) which revised the License and Technical Specifications for operation of the James A. FitzPatrick Nuclear Power Plant (the facility) located in Oswego County, New York. The amendment is effective as of its date of issuance.

The revisions to the license and Technical Specifications involve incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. These requirements are related to (1) Emergency Power Supply/Inadequate Core Cooling, (2) Valve Position Indication, (3) Containment Isolation, (4) Shift Technical Advisors, (5) Integrity of Systems Outside Containment, and (6) Iodine Monitoring.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated June 24, 1981, (2) Amendment No. 57 to License No. DPR-59, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D. C. and at the Penfield Library, State University College at Oswego, Oswego, New York 13126. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 7th day of July 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing