DCD-016

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Docket No. 50-333

September 8, 1982

Mr. Leroy W. Sinclair
President and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10010

Dear Mr. Sinclair:

The Commission has issued the enclosed Amendment No. 70 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 application dated August 12, 1982. This amendment was authorized by telephone on August 13, 1981, and was confirmed by letter dated August 13, 1981.

The amendment allows continued plant operation with one inoperable relief valve until the next cold shutdown, scheduled for October 1982 outage. With respect to the operable valves the Technical Specifications required that: (1) A prompt report be made in the event that there is an indication of valve leakage, and (2) An engineering evaluation be forwarded justifying continued operation. NRC approval of this evaluation is necessary prior to continuing power operation for more than 90 days after initial discovery of valve inoperability.

During the next shutdown you have agreed to repair or replace the inoperable relief valve and all inoperable thermocouples. In addition, as mutually agreed to by members of your staff the inoperable valve shall be pressure tested and inspected in the as-found condition and test results shall be forwarded to the NRC within one month of plant restart.

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

Sincerely,

Original signed by

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

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PDR ADOCK 05000333
P PDR

Amdt + FRN only

Enclosures:
1. Amendment No. 70 to DPR-59
2. Safety Evaluation
3. Notice

cc w/enclosures
See next page

Previous concurrence sheet concurred on by:

DL:ORB#2	DL:ORB#2	DL:ORB#2
SNorris	PPolk:pob:MC	DVassallo
8/24/82	8/24/82	8/25/82

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD		
SURNAME	SNorris	PPolk:pob:MC	DVassallo	GLainas	Goddard		
DATE	~~8/ /82~~ 9/7/82	~~8/ /82~~ 9/7/82	8/26/82	8/26/82	8/30/82		

NRC FORM 318 (10-80) NRCM 0240

OFFICIAL RECORD COPY

USGPO: 1981—335-960

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Power Authority of the State
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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. 70
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 dated August 12, 1982 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. 70 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment became effective August 13, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 8, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 70

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A Technical Specifications as follows:

Remove	Insert
76c	76c
142a	142a
-	142b
143	143
-	143a
-	143b

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 acoustical detectors of which one is in service and a backup thermocouple detector. In the event that a thermocouple is inoperable SRV performance shall be monitored daily with the associated acoustical detector.

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.

E. <u>Safety and Safety/Relief Valves</u>

1. During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212°F, the safety mode of all safety/relief valves shall be operable, except as specified by Specification 3.6.E.2. The Automatic Depressurization System valves shall be operable as required by Specification 3.5.D.

E. <u>Safety and Safety/Relief Valves</u>

1. At least one half of all safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. The safety/relief valve settings shall be set as required in Specification 2.2.B. All valves shall be tested every two operating cycles.

This page is effective after the October 1982 outage.

E. Safety and Safety/Relief Valves

1. During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212°F, the safety mode of all safety/relief valves shall be operable, except as specified by Specification 3.6.E.2. The Automatic Depressurization System valves shall be operable as required by Specification 3.5.D.

2. Reactor operation may continue with one safety/relief valve inoperable. From and after the date that two safety/relief valves are made or found inoperable, continued reactor operation is permissible only during the succeeding 7 days, unless one valve is made operable.

E. Safety and Safety/Relief Valves

1. At least one half of all safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. The safety/relief valve settings shall be set as required in Specification 2.2.B. All valves shall be tested every two operating cycles.

This page is effective until the October 1982 outage.

(cont'd)

2. a. From and after the date that the safety valve function of one safety/relief valve is made or found to be inoperable, continued operation is permissible only during the succeeding 30 days unless such valve is sooner made operable.

   b. From and after the time that the safety valve function on two safety/relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 7 days unless such valves are sooner made operable.

3. If Specification 3.6.B.1 and 3.6.B.2 are not met, the reactor shall be placed in a cold condition within 24 hr.

4. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Item B.2 above, provided that reactor coolant temperature is $\leq 212^{o}F$ and the reactor vessel is vented or the reactor vessel head is removed.

IPP

4.6 (cont'd)

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. The integrity of the safety/relief valve bellows shall be continuously monitored.

   a. The bellows monitoring pressure switches shall be removed and bench checked once/operating cycle. Modified safety/relief valves with two-stage assemblies do not have a bellows arrangement and are, therefore, not subject to this requirement.

4. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

3.6 (cont'd)

3. If Specification 3.6.E.1 and 3.6.E.2 are not met the reactor shall be placed in a cold condition within 24 hr.

4. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.6.E.2, and provided that reactor coolant temperature is $\leq 212^{o}F$ and the reactor vessel is vented or the vessel head is removed.

4.6 (cont'd)

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. Deleted

4. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

This page is effective until the October 1982 outage.

3.6 (cont'd)

5. If, for a period of longer than 24 hours, the temperature of any safety/relief discharge pipe is more than 40°F above its steady state value, or the acoustical monitor reading of any safety/relief valve discharge pipe is more than 3 times greater than its steady state value, the following actions shall be taken:

   a. a report shall be issued in accordance with 6.9.A.4.1 which addresses the actions that have been taken or a schedule of actions to be taken.

   b. an engineering evaluation shall be performed justifying continued operation for the corresponding increase in temperature or acoustical monitor reading.

   c. the affected safety/relief valve shall be removed at the next cold shutdown of 72 hours or more, tested in the as-found condition, and recalibrated as necessary prior to reinstallation.

   d. NRC approval of the engineering evaluation specified in 3.6.E.5.b above shall be obtained prior to continuing power operation for more than 90 days after the initial discovery of the 40°F increase in temperature or the factor of 3 increase in acoustical monitor reading.

   The steady state values of temperature and acoustical monitor readings shall be as measured after 5 days of steady state power operation.

This page is effective until the October 1982 outage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 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

Author: P. Polk
G. Holahan

1.0 Introduction

By letter dated August 12, 1982 the Power Authority of the State of New York (the licensee) forwarded a proposed Technical Specification change that would allow continued plant operation with one safety relief valve (SRV) inoperable. Prior to this letter the licensee had declared the subject valve inoperable with high temperature on the discharge side of the valve a major consideration in making the inoperability determination. Such temperature readings had been increasing since plant restart after the Spring 1982 refueling outage. At this time the temperature has stabilized at approximately 295 F.

2.0 Background

The safety relief valves employed at FitzPatrick are two stage Target Rock valves. The setpoints for these eleven valves are grouped with 2 valves set at 1090 psig, 2 valves set at 1105 psig and 7 valves set at 1140 psig.

The license amendment proposed by the licensee provides revised limiting conditions for operation when an SRV is inoperable for any reason. It is also intended to address concerns that SRV leakage as indicated by elevated SRV tailpipe temperatures, renders the affected valve inoperable. These concerns have been expressed as a result of a recent increase in one SRV tailpipe temperature. Setpoint drift experienced during as-received testing of SRV's at Wyle Laboratories has been attributed, at least in part, to excessive valve leakage.

SRV leakage is not monitored directly but is indicated by increased tailpipe thermocouple and acoustical monitor readings. The possibility of proposing SRV tailpipe temperatures at which the valve would be declared inoperable, because of leakage, has been investigated. General Electric (the NSSS vendor) and Target Rock (the SRV vendor) have both stated that they have not been able to identify or develop a definitive correlation between leakage and tailpipe temperature, by calculation or by using empirical data. However, the licensee has indicated that as-found testing conducted by Target Rock showed that a 40° to $60^{\circ}F$ increase in tailpipe temperature indicates a leakage rate from negligible up to 200 pounds per hour. Further testing by Target Rock has indicated that a leakage rate of 200 pounds per hour should not affect SRV setpoint or response time. Target Rock also indicated that a 200 pound per hour leakage rate would not render a valve inoperable.

3.0 Evaluation

We have reviewed the licensee's proposed amendment which will allow continued plant operation with one two-stage Target Rock relief valve inoperable. The FitzPatrick plant presently has eleven relief valves seven of which perform the Automatic Depressurization System (ADS) function. The remaining four valves fulfill the safety relief function. The subject inoperable valve is a safety relief valve set at 1140 psig.

The licensee's submittal addressed the following analyses: (1) failure of the safety relief valve to open post accident; (2) failure of the valve in the open position during normal plant operation; and (3) change in valve setpoint when called upon post accident. The analyses are addressed in the following paragraphs.

With respect to valve failure to open post accident, a plant specific analyses was performed for the FitzPatrick plant. The worst case transient was evaluated assuming: (1) the lowest setpoint SRV inoperable, and (2) the Main Steam Isolation Valves (MSIV) fail to close and the reactor trips on high neutron flux. (Note that MSIV failure to close is considered the single failure in addition to the initial event. The accident assuming that MSIV's function to close and a single failure of an SRV in addition to the inoperable SRV is less severe than the analyzed MSIV failure). The evaluation results in an increase of peak vessel pressure of 15 psig with a margin of 85 psig to the ASME Boiler and Pressure Vessel Code upset limit of 1375 psig.

Regarding SRV failure in the open position during normal plant operation, such a single SRV failure was evaluated in the original licensing review for FitzPatrick. In order to determine the affect of a second simultaneous SRV failure in the open position a 10 CFR 50, Appendix K evaluation was performed. In essence, this is a small break LOCA and the analyses determined that fuel peak centerline temperature (PCT) does not change since this accident does not become limiting. Fuel PCT is less than 1300°F for this accident, which is significantly below the Appendix K limit of 2200°F.

With respect to reduced valve setpoint, an analyses was conducted to support license amendment No. 54 to Operating License No. DPR-59 dated April 13, 1981. This amendment concluded that a reduction in valve setpoint of 50 psig increases torus loadings and would reduce torus safety margins. However, since Mark 1 Containment system modifications had been completed, the FitzPatrick plant margins satisfy the criteria for the Mark 1 interim period. (The reduced valve setpoints will result in reduced reactor vessel peak pressures.)

Regarding an increased SRV setpoint, the worst case is assumed to be a total valve failure to open; i.e., opening at a increased value is a less severe transient than not opening at all. As previously discussed above, this results in a 15 psig increase in peak reactor vessel pressure.

With respect to operation of the FitzPatrick Plant with a single inoperable SRV, we conclude, based upon the foregoing, that such operation will not have a significant adverse inpact on plant safety. In the eventuality that a second SRV becomes inoperable the proposed LCO/surveillance requirements associated with SRV monitoring provide assurance that valve leakage will be identified when valve leakage is minimal, thereby minimizing the potential for valve setpoint drift. The additional testing, reporting, and engineering evaluations required by these Technical Specifications assures timely identification and resolution of any problems. Consequently, we find the licensee's proposed Technical Specifications acceptable.

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 Section 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 an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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: September 8, 1982

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. 70 to Facility Operating License No. DPR-59 issued to the Power Authority of the State of New York (the licensee), which revises the Technical Specifications for operation of the James A. FitzPatrick Nuclear Power Plant (the facility), located in Oswego County, New York. The amendment was authorized by telephone on August 13, 1982, and was confirmed by letter dated August 13, 1982.

The amendment allows continued plant operation with one relief valve inoperable until the next FitzPatrick plant outage.

The application for 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 the amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the 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 the amendment.

For further details with respect to this action, see (1) the application for amendment dated August 12, 1982, (2) Amendment No. 70 to License No. DPR-59, ad (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, N. W., Washington, D.C. and at the Penfield Library, State University College of Oswego, Oswego, New York. A copy of 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 8th day of September 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DVassallo

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