December 19, 1986

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. 103 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 June 25, 1986.

The amendment changes the Technical Specifications to reflect a lowering of the reactor water level setpoint of the Main Steam Isolation Valves from Level 2 to Level 1.

A copy of the Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original signed by

Harvey I. Abelson, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 103 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. 103
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 25, 1986, 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. 103 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 19, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 103

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
55
56
64
206

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the Core Cooling Systems, Control Rod Block and Standby Gas Treatment Systems. The objectives of the specifications are to assure the effectiveness of the protective instrumentation when required, even during periods when portions of such systems are out of service for maintenance, and to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2-1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177 in. above the top of the active fuel closes all isolation valves except those in Group 1. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level, this trip setting is adequate to prevent uncovering the core in the case of a break in the largest line assuming a 60 sec valve closing time. Required closing times are less than this.

The low-low reactor water level instrumentation is set to trip when reactor water level is 126.5 in. above the top of active fuel (-38 in. on the instrument). This trip

initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18 in. above the top of active fuel. This trip activates the remainder of the ECCS subsystems, closes the main steam isolation valves, main steam line drain valves and reactor water sample line isolation valves, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate ECCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10CFR100 will not be exceeded. For large breaks up to the complete circumferential break of a 24 in. recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating ECCS, it causes isolation of Groups B and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate ECCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Specification 3.7 for isolation valve closure group. The water level instrumentation initiates protection for the full spectrum of loss- of-coolant accidents.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperature peak at approximately 1,000°F and release of radioactivity to the environs is below 10CFR100 guidelines. Reference Section 14.6.5 FSAR.

JAFNPP
TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (2)
2 (6)	Reactor Low Water Level	≥12.5 in. Indicated Level (≥177 in. above the top of active fuel)	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤75 psig	2 Inst. Channels	D
2	Reactor Low-Low-Low Water Level	≥18 in. above the top of active fuel.	4 Inst. Channels	A
2 (6)	High Drywell Pressure	≤2.7 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	≤3 x Normal Rated Full Power Background (9)	4 Inst. Channels	B
2	Low Pressure Main Steam Line	≥825 psig (7)	4 Inst. Channels	B
2	High Flow Main Steam Line	≤140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Leak Detection High Temperature	≤40°F above max ambient	4 Inst. Channels	B
3	Reactor Cleanup System Equipment Area High Temperature	≤40°F above max ambient	6 Inst. Channels	C
2	Low Condenser Vacuum Closes MSIV's	≥8" Hg. Vac (8)	4 Inst. Channels	B

NOTES FOR TABLE 3.7-1

ISOLATION SIGNAL CODES

Signal	Description
A*	Reactor vessel low water level - (A scram occurs at this level also. This is the highest of the three low water level signals.
B*	Reactor vessel low-low-low water level - (This is the lowest of the three low water level signals.
C*	High radiation - main steam line
D*	Line break - main steam line (steam line high steam flow)
E*	Line break - main steam line (steam line high temperature)
F*	High drywell pressure
G	Reactor vessel low water level or high drywell pressure (Emergency Core Cooling Systems are started)
H	
J*	Line break in Reactor Water Cleanup System - high space temperature
K*	Line break in RCIC System steam line to turbine (high steam line space temperature, high steam flow, low steam line pressure, or high turbine exhaust pressure)
L*	Line break in HPCI System steam line to turbine (high steam line space temperature, high steam flow, low steam line pressure, or high turbine exhaust pressure)
M	
P*	Low main steam line pressure at inlet to main turbine (RUN mode only)
S	Low drywell pressure
T	Low reactor pressure permissive to open core spray and RHR-LPCI valves

* These are the isolation functions of the Primary Containment and Reactor Vessel Isolation Control System; other functions are given for information only.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 103 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 25, 1986, the Power Authority of the State of New York, licensee for the James A. Fitzpatrick Nuclear Power Plant, proposed a change to the Technical Specifications (TS). This change would lower the reactor water level setpoint for closure of the main steam isolation valves (MSIVs), main steam line drain valves (MSLDVs) and reactor water sample line isolation valves (RWSVs) from Level 2 to Level 1. The main purpose of this change is to reduce challenges to the safety relief valves (SRVs). The probability of closing the MSIVs due to variation of water level following a scram would be reduced as a result of this change and reactor pressure would be maintained below the SRV setpoint pressure by the turbine bypass valves for a longer period of time. The main condenser would then continue to act as a heat sink until Level 1 is reached. With the MSIVs open and the main condenser available, the relief valves would not be challenged. Should Level 1 be reached, MSIV closure would cause reactor pressure to rise causing SRV actuation and discharge to the suppression pool. The resulting heat load on the suppression pool produced by the delayed SRV discharge would therefore be reduced.

The change in MSIV water level setpoint from Level 2 to Level 1 will not affect the initiation point of the High Pressure Core Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, which will remain at Level 2. However, with the MSIVs now remaining open until Level 1 is reached, steam for the reactor feed pump turbines would remain available for an additional period of time. For certain operating transients, this would give the operator the option of adding water to the reactor vessel via continued operation of the condensate and feedwater systems rather than by use of the HPCI and/or RCIC systems. Since the HPCI and RCIC turbines exhaust to the suppression pool, the heat load on the suppression pool would be reduced.

The instruments which set the level for MSIV closure also initiate closure of the MSLDVs and RWSVs. Hence the level setpoint for the MSIVs, MSLDVs and RWSVs would all be lowered from Level 2 to Level 1 as a result of the proposed change.

2.0 EVALUATION

The licensee provided a General Electric (GE) report entitled "Safety Review of Safety Relief Valve Simmer Margin Analysis and Water Level Setpoint Change for James A. Fitzpatrick Nuclear Power Plant," Rev. 1. This report evaluated the change in reactor water level isolation setpoint from Level 2 to Level 1 for the MSIVs, MSLDVs and RWSVs. The staff evaluation of the change in setpoint level for each group of valves and the proposed TS changes follows.

2.1 Main Steam Isolation Valves (MSIVs)

The effect of change in level setpoint for the MSIVs was considered for abnormal operational transients, loss of coolant accidents (LOCAs), including a main steam line (MSL) break, and anticipated transients without scram (ATWS). The specific events analyzed, with the exception of ATWS, were the same as those previously considered in Chapter 14 of the FitzPatrick Final Safety Analysis Report (FSAR).

Of the abnormal operational transients discussed in the FSAR, only two were affected by the level setpoint change; i.e., the loss of feedwater flow transient and a feedwater controller failure at maximum demand. For both these events, the reactor feed pump would be unavailable and HPCI and RCIC would be initiated for core cooling. The licensee found that the setpoint change would not cause: a) a reduction in the minimum critical power ratio, b) an increase in peak reactor vessel pressure, c) an increase in radiation release, d) equipment damage, e) a reduction in plant shutdown capability, or f) a decrease in core cooling capability. The licensee, therefore, concluded that no new safety concerns are introduced by the setpoint change with regard to abnormal operating transients.

Large and intermediate LOCA events were qualitatively analyzed. For these events, there would be a rapid depressurization and inventory loss within the reactor vessel, resulting in a fast actuation of the MSIVs. The reactor feed pump would be unavailable in this case and the core would be cooled by the low-pressure Emergency Core Cooling Systems. The licensee reported that the MSIVs would close before any fuel failure could occur, that the lower MSIV setpoint would not increase the reactor core inventory loss or radiation loss to the environment, and that the maximum average planar linear heat generation rate (MAPLHGR) would not be changed. Thus, the licensee concluded that the setpoint change would not affect the design basis accident (DBA) for large and intermediate LOCA events.

In the event of a small break LOCA, with the condensate and feedwater system unavailable, there is a potential for initiation of MSIV closure at the proposed lower level setpoint. Upon MSIV closure, safety relief valve actuation would follow. This event was analyzed using the approved GE Appendix K evaluation model for the lowered MSIV setpoint. The licensee stated that the highest peak cladding temperature would be substantially less than the 2200°F limit. Therefore, the setpoint change would have no effect on the limiting MAPLHGR.

The licensee also reported that a MSL break would be unaffected by the setpoint change since other MSIV isolation signals (e.g. high flow) would occur well before a low reactor water level isolation signal.

For the ATWS event, the licensee concluded that the lowered setpoint level for the MSIVs would not introduce any new safety concerns. This is based on the fact that the recirculation pump trip at Level 2 results in a significant decrease in reactor power, and the HPCI and RCIC systems initiation at Level 2 would provide makeup water to the reactor vessel.

We have reviewed the licensee's analysis of the effects of a lowered MSIV water level setpoint on abnormal operational transients, LOCAs and the ATWS event and find the above determinations appropriate and conclusions acceptable.

2.2 Main Steam Line Drain Valves (MSLDVs)

The MSLDVs are considered part of the MSL isolation system. The drain lines which they isolate are normally closed and represent a flow area of about 1.8% of the MSL. The licensee concludes that the amount of radiation release and inventory loss through the MSLDVs will be insignificant when compared to that through the MSIVs when the setpoint is lowered from Level 2 to Level 1. We find the above determinations appropriate and conclusions acceptable.

2.3 Reactor Water Sample Valves (RWSVs)

In the event of a decreasing water level, the RWSVs presently isolate the sample line at Level 2 to reduce inventory loss. The RWSVs are 3/4 inch in size. The licensee states that the additional inventory loss through this line, in going from Level 2 to Level 1, would be insignificant. They also conclude that, since the lower level setpoint for the MSIVs will not increase the amount of radiation release for a DBA (as discussed above), isolation of the RWSVs at Level 1 will not affect the calculated radiation doses. We concur with these findings.

2.4 Technical Specification Changes

The licensee proposed to change TS Tables 3.2-1, 3.7-1 and Bases 3.2 to reflect a change in setpoint from Level 2 to Level 1 for the MSIVs, MSLDVs and RWSVs. We find the above determinations appropriate and conclusions acceptable.

3.0 SUMMARY

The licensee has proposed TS changes which would lower the reactor water level setpoint from Level 2 to Level 1 for initiation of closure of the MSIVs, MSLDVs and RWSVs. In the event of a scram, the lower setpoint would allow steam flow from the reactor vessel through the bypass valves to the condenser for a longer period of time. Also, steam would remain available longer for continued operation of the reactor feed pump. These are advantages which result in fewer challenges to the SRVs, a reduced heatup of the suppression pool, and in certain circumstances, a preferred source of water addition to the reactor vessel via the reactor feed pump. We therefore, find the licensee proposed TS changes to be acceptable.

4.0 ENVIRONMENTAL CONSIDERATIONS

This 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. 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding. 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

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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Katze

Dated: December 19, 1986