Docket No. 50-333

JAN 29 1982

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. 62 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 January 6, 1981.

The amendment modifies the Technical Specifications to reflect scram discharge volume (SDV) long-term modifications. Specifically, SDV vent and drain valves and reactor protection system and rod block limit switches are addressed. Please note that the enclosed Technical Specifications require quarterly testing of vent and drain valves as agreed to by members of your staff.

Copies of the NRC Safety Evaluation, the Franklin Research Center Technical Evaluation Report and the Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Philip J. Polk, Project Manager
Operating Reactors Branch #2
Division of Licensing

cp
1

Enclosures:
1. Amendment No. 62 to DPR-59
2. Safety Evaluation
3. Franklin Research Center
   Technical Evaluation Report
4. Notice

cc: w/enclosures
See next page

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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. 62
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 January 6, 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, 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. 62, 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

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 29, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 62

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages:

Remove	Insert
43	43
44	44
45	45
	45a
46	46
47	47
72	72
73	73
81	81
89	89
89a	89a
96	96

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (cont'd)

C. High Flux IRM

D. Scram Discharge Instrument Volume High Level when any control rod in a control cell containing fuel is not fully inserted

E. APRM 15% Power Trip

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

9. The APRM downscale trip is automatically bypassed when the IRM Instrumentation is operable and not high.

10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.

11. See Section 2.1.A.1.

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP).

where:

FRP - Fraction of rated thermal power (2436 MWt)

MFLPD - Maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 MW/ft for 8x8, 8x8R and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used

W - Loop Recirculation flow in percent of rated (rated is 34.2×10^6 lb/hr)

S_n - Scram setting in percent of initial

13. The Average Power Range Monitor scram function is varied (Figure 1.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.

Table 4.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each refueling outage.
Manual Scram	A	Trip Channel and Alarm	Every 3 months.
RPS Channel Test Switch	A	Trip Channel and Alarm	Every refueling outage or after channel maintenance.
IRM			
High Flux	C	Trip Channel and Alarm(4)	Once per week during refueling or startup and before each startup.
Inoperative	C	Trip Channel and Alarm(4)	Once per week during refueling or startup and before each startup.
APRM			
High Flux	B	Trip output Relays(4)	Once/week.
Inoperative	B	Trip output Relays(4)	Once/week.
Downscale	B	Trip output Relays(4)	Once/week.
Flow Bias	B	Calibrate Flow Bias Signal(4)	Once/month.(1)
High Flux in Startup or Refuel	C	Trip Output Relays(4)	Once per week during refueling or startup and before each startup.
High Reactor Pressure	B	Trip Channel and Alarm(4)	Once/month.(1) (Instrument check once per day)
High Drywell Pressure	A	Trip Channel and Alarm	Once/month(1)
Reactor Low Water Level(5)	A	Trip Channel and Alarm	Once/month(1)
High Water Level in Scram Discharge Instrument Volume	A	Trip Channel and Alarm	Once/month and before each startup(6),(7)
Main Steam Line High Radiation	B	Trip Channel and Alarm(4)	Once/week.

Table 4.1-1 (cont'd)

REACTOR PROTECTION SYSTEM(SCRAM) INSTRUMENT FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/month.(1)
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Once/month.
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every 3 months.(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/month.(1)
Reactor Pressure Permissive	A	Trip Channel and Alarm	Every 3 months.

NOTES FOR TABLE 4.1-1

1. Initially once every month until acceptable 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 operates in an environment similar to that of JAFNPP.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required on the part of the system that is not required to be operable or are tripped.

   If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the instrument channels.

Table 4.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

NOTES FOR TABLE 4.1-1 (cont'd)

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the functional test program.

6. Functional test of the instruments before each startup is required only if a scram has occurred since the last functional test or calibration.

7. The functional test shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.

Table 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency Once/week
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Maximum frequency once/week
APRM High Flux Output Signal Flow Bias Signal	 B B	Heat Balance Internal Power and Flow Test with Standard Pressure Source	Daily Every refueling outage
LPRM Signal	B	TIP System Traverse	Every 1000 effective full power hours
High Reactor Pressure	B	Standard Pressure Source	Once/operating cycle
High Drywell Pressure	A	Standard Pressure Source	Every 3 months
Reactor Low Water Level	A	Pressure Standard	Every 3 months
High Water level in Scram Discharge Instrument Volume	A	Water Column, Note (6)	Once/operating cycle, Note (6)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 months
Turbine Plant Stage Pressure Permissive	A	Standard Pressure Source	Every 6 months
Turbine Control Valve Past Closure Oil Pressure Trip	A	Standard Pressure Source	Once/operating cycle

Table 4.1-2 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency (2)
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Reactor Pressure Permissive	A	Standard Pressure Source	Every 6 months

NOTES FOR TABLE 4.1-2

1. A description of three groups is included in the Bases of this Specification.
2. Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.
3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.
4. Response time is not a part of the routine instrument channel test but will be checked once per operating cycle.
5. Actuation of these switches by normal means will be performed during the refueling outages.
6. Calibration shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action
2	APRM Upscale (Flow Biased)	$S \leq (0.66W+42\%) \times \frac{FRP}{MFLPD}$	6 Inst. Channels	(1)
2	APRM Upscale (Start-up Mode)	$\leq$ 12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Flow Biased)	$S \leq 0.66W+K$ (8)	2 Inst. Channels	(1)
1 (6)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (2)	$\geq$2% of full scale	8 Inst. Channels	(1)
3	IRM Detector not in Start-up Position	(7)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$86.4% of full scale	8 Inst. Channels	(1)
2 (4)	SRM Detector not in Start-up Position	(3)	4 Inst. Channels	(1)
2 (4)(5)	SRM Upscale	$\leq 10^5$ counts/sec	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Water Level	$\leq$ 18 gallons	1 Inst. Channel	(9) (10)

NOTES FOR TABLE 3.2-3

1. For the Start-up and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in run mode, and

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3

the APRM and RBM rod blocks need not be operable in start-up mode. From and after the time it is found that the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter, if this condition lasts longer than seven days, the system shall be tripped. From and after the time it is found that the first column cannot be met for both trip systems, the systems shall be tripped.

2. IRM downscale is bypassed when it is on its lowest range.

3. This function is bypassed when the count rate is $\geq$ 100 cps.

4. One of the four SRM inputs may be bypassed.

5. This SRM Function is bypassed when the IRM range switches are on range 8 or above.

6. The trip is bypassed when the reactor power is $\leq$ 30%.

7. This function is bypassed when the Mode Switch is placed in Run.

8. S = Rod Block Monitor Setting in percent of initial.

   W = Loop recirculation flow in percent of rated, (rated loop recirculation flow is 34.2×10^6 lb/hr).

   K = Intercept values of 39%, 40%, 41% and 42% can be used with appropriate MCPR limits from Section 3.1.B.

9. When the reactor is subcritical and the reactor water temperature is less than 212°F, the control rod block is required to be operable only if any control rod in a control cell containing fuel is not fully inserted.

10. When the control rod block function associated with scram discharge instrument volume high water level is not operable when required to be operable, the trip system shall be tripped.

TABLE 4.2-3

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check (9)
1) APRM - Downscale	(1) (3)	Once/3 months	Once/day
2) APRM - Upscale	(1) (3)	Once/3 months	Once/day
3) IRM - Upscale	(2) (3)	(2)	(2)
4) IRM - Downscale	(2) (3)	(2)	(2)
5) RBM - Upscale	(1) (3)	Once/3 months	Once/day
6) RBM - Downscale	(1) (3)	Once/3 months	Once/day
7) SRM - Upscale	(2) (3)	(2)	(2)
8) SRM - Detector Not in Startup Position	(2) (3)	(2)	
9) IRM - Detector Not in Startup Position	(2) (3)	(2)	
10) Scram Discharge Instrument Volume - High water level	Once/month (2)	Once/operating Cycle (2)	N/A

Logic System Functional Test (4)(6)	Frequency
1) System Logic Check	Once/6 months

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

3.3 (cont'd)

a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to the Cold Shutdown condition within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing, and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.

If investigation demonstrates that the cause of control rod failure is a cracked collet housing, or if this possibility cannot be ruled out, the reactor shall not be started until the affected control rod drive has been replaced or repaired.

4.3 (cont'd)

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30 percent power. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above 30 percent power.

b. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days (these valves may be closed intermittently for testing under administrative control).

c. A second licensed operator shall verify the conformance to Specification 3.3.A.2.d before a rod may be bypassed in the Rod Sequence Control System.

d. Once per week check status of pressure and level alarms for each accumulator.

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in

e. When it is initially determined that a control rod is incapable of normal insertion, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted

shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted. If Specification 3.3.A.1 and 4.3.A.1 are met, reactor startup may proceed.

f. The scram discharge volume drain and vent valves shall each be full travel cycled at least once per quarter.

3.3 (cont'd)

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Control Rod Notch Position Observed	Average Scram Insertion Time (Sec)
46	0.361
38	0.977
24	2.112
04	3.764

4.3 (cont'd)

2. At 8-week intervals, 15 percent of the operable control rod drives shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. All control rods shall be determined operable once each operating cycle by demonstrating the scram discharge volume drain and vent valves operable when the scram test initiated by placing the mode switch in the SHUTDOWN position is performed as required by Table 4.1-1 and by verifying that the drain and vent valves:

   a. Close within 80 seconds after receipt of a signal for control rods to scram , and

   b. Open when the scram signal is reset or the scram discharge instrument volume trip is bypassed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 62 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

As a result of events involving common cause failures of scram discharge volume (SDV) limit switches and SDV drain valve operability, the NRC staff issued IE Bulletin 80-14 on June 12, 1980. In addition, the staff sent a letter dated July 7, 1980 to all operating BWR licensees requesting that they propose Technical Specification changes to provide surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements on SDV limit switches. Model Technical Specifications were enclosed with this letter to provide guidance to licensees for preparation of the requested submittals. By letter dated January 6, 1981 the Power Authority of the State of New York (licensee) requested changes to the Technical Specifications for the James A. FitzPatrick Nuclear Power Plant relating to SDV.

2.0 Evaluation

The enclosed report (TER-C5506-74) was prepared by the Franklin Research Center (FRC) as part of a technical assistance contract program. This FRC report provides the technical evaluation of the compliance of the licensee's submittal with NRC provided criteria.

FRC has concluded that the licensee's response does not meet the explicit requirements of paragraph 3.3-6 and Table 3.3.6-1 of the NRC staff's Model Technical Specifications (TSs). However, the FRC report concludes that technical bases are defined on p. 50 of the staff's "Generic Safety Evaluation Report BWR Scram Discharge System," December 1, 1980 for this departure from the explicit requirements of the Model Technical Specifications. We conclude that these technical bases justify a deviation from the explicit requirements of the Model Technical Specifications.

FRC has concluded that the licensee's proposed Technical Specifications revisions meet our criteria without the need for further revision.

Based upon our review of the contractor's report of his evaluations and discussions with the reviewer, we conclude that the licensee's proposed Technical Specifications satisfy our requirements for surveillance of SDV vent and drain valves and for LCOs and surveillance requirements for SDV limit switches. Consequently, we find the licensee's proposed Technical Specifications acceptable.

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3.0 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 impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 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: January 29, 1982
Author: Phillip J. Polk
Kenneth T. Eccleston

TECHNICAL EVALUATION REPORT

BWR SCRAM DISCHARGE VOLUME LONG-TERM MODIFICATIONS

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

NRC DOCKET NO. 50-333

NRC TAC NO. 42220

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 2

FRC TASK 74

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December 3, 1981

DPL

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SUMMARY

This technical evaluation report reviews and evaluates Phase 1 proposed changes in the FitzPatrick Nuclear Power Plant Technical Specifications for scram discharge volume (SDV) long-term modifications regarding surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for reactor protection system and control rod withdrawal block SDV limit switches. Conclusions were based on the degree of compliance of the Licensee's submittal with criteria from the Nuclear Regulatory Commission (NRC) staff's Model Technical Specifications.

The revised page 89 of the FitzPatrick Technical Specifications, and the Licensee's agreement to add to the proposed specifications changes a requirement to cycle each valve a minimum of one full cycle at least quarterly, comply with the NRC staff's Model Technical Specifications, paragraphs 4.1.3.1.1a and 4.1.3.1.1b. Proposed revisions of pages 43, 44, 45, 45a, 46, 47, 81, 89, 89a, and 96 and unrevised pages 41a and 103 meet the remaining surveillance requirements. Table 5-1 on pages 22 and 23 of this report summarizes the evaluation results.

1. INTRODUCTION

1.1 PURPOSE OF THE TECHNICAL EVALUATION

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of the FitzPatrick Nuclear Power Plant boiling water reactor (BWR) in regard to "BWR Scram Discharge Volume Long Term Modification," specifically:

- o surveillance requirements for scram discharge volume (SDV) vent and drain valves
- o limiting condition for operation (LCO)/surveillance requirements for the reactor protection system SDV limit switches
- o LCO/surveillance requirements for the control rod withdrawal block SDV limit switches.

The evaluation used criteria proposed by the NRC staff in Model Technical Specifications (see Appendix A of this report). This effort is directed toward the NRC's objective of increasing the reliability of installed BWR scram discharge volume systems, the need for which was made apparent by events described below.

1.2 GENERIC ISSUE BACKGROUND

On June 13, 1979, while the reactor at Hatch Unit 1 was in the refuel mode, two SDV high level switches had been modified, tested, and found inoperable. The remaining switches were operable. Inspection of each inoperable level switch revealed a bent float rod binding against the side of the float chamber.

On October 19, 1979, Brunswick Unit 1 reported that water hammer due to slow closure of the SDV drain valve during a reactor scram damaged several pipe supports on the SDV drain line. Drain valve closure time was approximately 5 minutes because of a faulty solenoid controlling the air supply to the valve. After repair, to avoid probable damage from a scram, the unit was started with the SDV vent and drain valves closed except for periodic draining. During this mode of operation, the reactor scrammed due to a high water level in the

SDV system without prior actuation of either the high level alarm or rod block switch. Inspection revealed that the float ball on the rod block switch was bent, making the switches inoperable. The water hammer was reported to be the cause of these level switch failures.

As a result of these events involving common-cause failures of SDV limit switches and SDV drain valve operability, the NRC issued IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capability," on June 12, 1980 [1]. In addition, to strengthen the provisions of this bulletin and to ensure that the scram system would continue to work during reactor operation, the NRC sent a letter dated July 7, 1980 [2] to all operating BWR licensees requesting that they propose Technical Specifications changes to provide surveillance requirements for reactor protection system and control rod block SDV limit switches. The letter also contained the NRC staff's Model Technical Specifications to be used as a guide by licensees in preparing their submittals.

Meanwhile, during a routine shutdown of the Browns Ferry Unit 3 reactor on June 28, 1980, 76 of 185 control rods failed to insert fully. Full insertion required two additional manual scrams and an automatic scram for a total elapsed time of approximately 15 minutes between the first scram initiation and the complete insertion of all the rods. On July 3, 1980, in response to both this event and the previous events at Hatch Unit 1 and Brunswick Unit 1, the NRC issued (in addition to the earlier IE Bulletin 80-14) IE Bulletin 80-17 [3] followed by five supplements [4, 5, 6, 7, 8]. These initiated short-term and long-term programs described in "Generic Safety Evaluation Report BWR Scram Discharge System," NRC staff, December 1, 1980 [9] and "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17 (Continuous Monitoring Systems)" [10].

Analysis and evaluation of the Browns Ferry Unit 3 and other SDV system events convinced the NRC staff that SDV systems in all BWRs should be modified to assure long-term SDV reliability. Improvements were needed in three major areas: SDV-IV hydraulic coupling, level instrumentation, and system isolation. To achieve these objectives, an Office of Nuclear Reactor Regulation (NRR) task force and a subgroup of the BWR Owners Group developed Revised Scram Discharge

System Design and Safety Criteria for use in establishing acceptable SDV systems modifications [9]. Also, an NRC letter dated October 1, 1980 requested all operating BWR licensees to reevaluate installed SDV systems and modify them as necessary to comply with the revised criteria.

In Reference 9, the SDV-IV hydraulic coupling at the Big Rock Point, Brunswick 1 & 2, Duane Arnold, and Hatch 1 & 2 BWRs was judged acceptable. The remaining BWRs will require modification to meet the revised SDV-IV hydraulic coupling criteria, and all operating BWRs may require modification to meet the revised instrumentation and isolation criteria. The changes in Technical Specifications associated with this effort will be carried out in two phases:

Phase 1 - Improvements in surveillance for vent and drain valves and instrument volume level switches.

Phase 2 - Technical Specifications improvements required as a result of long-term modifications made to comply with revised design and performance criteria.

This TER is a review and evaluation of Technical Specifications changes proposed for Phase 1.

1.3 PLANT-SPECIFIC BACKGROUND

The July 7, 1980 NRC letter [2] not only requested all BWR licensees to amend their facilities' Technical Specifications with respect to control rod drive SDV capability, but enclosed the NRC staff's proposed Model Technical Specifications (see Appendix A of this TER) as a guide for the licensees in preparing the requested submittals and as a source of criteria for an FRC technical evaluation of the submittals. In this TER, FRC has reviewed and evaluated Technical Specifications changes for the FitzPatrick Nuclear Power Plant proposed in a January 6, 1981 letter (see Appendix B) by the Licensee, the Power Authority of the State of New York (PASNY), in regard to "BWR Scram Discharge Volume (SDV) Long-Term Modifications" and, specifically, the surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for the reactor protection system and control rod withdrawal block SDV limit switches. FRC assessed the adequacy with which the PASNY information documented compliance of the proposed Technical Specifications changes with the NRC staff's Model Technical Specifications.

2. REVIEW CRITERIA

The criteria established by the NRC staff's Model Technical Specifications involving surveillance requirements of the main SDV components and instrumentation cover three areas of concern:

- o surveillance requirements for SDV vent and drain valves
- o LCO/surveillance requirements for reactor protection system SDV limit switches
- o LCO/surveillance requirements for control rod block SDV limit switches.

2.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

The surveillance criteria of the NRC staff's Model Technical Specifications for SDV drain valves are:

"4.1.3.1.1 - The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days and

b. Cycling each valve at least one complete cycle of full travel at least once per 92 days.

*These valves may be closed intermittently for testing under administrative controls."

The Model Technical Specifications require testing the drain and vent valves, checking at least once every 31 days that each valve is fully open during normal operation, and cycling each valve at least one complete cycle of full travel under administrative controls at least once per 92 days.

Full opening of each valve during normal operation indicates that there is no degradation in the control air system and its components that control the air pressure to the pneumatic actuators of the drain and vent valves. Cycling each valve checks whether the valve opens fully and whether its movement is smooth, jerky, or oscillatory.

During normal operation, the drain and vent valves stay in the open position for very long periods. A silt of particulates such as metal chips and flakes, various fibers, lint, sand, and weld slag from the water or air may accumulate at moving parts of the valves and temporarily "freeze" them. A strong breakout force may be needed to overcome this temporary freeze, producing a violent jerk which may induce a severe water hammer if it occurs during a scram or a scram resetting. Periodic cycling of the drain and vent valves is the best method to clear the effects of particulate silting, thus promoting smooth opening and closing and more reliable valve operation. Also, in case of improper valve operation, cycling can indicate whether excessive pressure transients may be generated during and after a reactor scram which might damage the SDV piping system and cause a loss of system integrity or function.

2.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

The paragraphs of the NRC staff's Model Technical Specifications pertinent to LCO/surveillance requirements for reactor protection system SDV limit switches are:

"3.3.1 - As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

Table 3.3.1-1. Reactor Protection System Instrumentation

Functional Unit	Applicable Operational Conditions	Minimum Operable Channels Per Trip System (a)	Action
8. Scram Discharge Volume Water Level-High	1,2,5 (h)	2	4

Table 3.3.1-2. Reactor Protection System Response Times

Functional Unit	Response Time (Seconds)
8. Scram Discharge Volume Water Level-High	NA"

"4.3.1.1 - Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

Table 4.3.1.1-1. Reactor Protection System Instrumentation Surveillance Requirements

Functional Unit	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
8. Scram Discharge Volume Water Level-High	NA	M	R	1,2,5

Notation (a) A channel may be placed in an inoperable status up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Action 4: In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2."

Paragraph 3.3.1 and Table 3.3.1-1 of the Model Technical Specifications require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, a total of 4 operable channels containing 4 limit switches per two trip systems for the reactor protection system which automatically initiates a scram. The technical objective of these requirements is to provide 1-out-of-2-taken-twice logic for the reactor protection system. The response time of the reactor protection system for the functional unit of SDV water level-high should be measured and kept available (it is not given in Table 3.3.1-2).

Paragraph 4.3.1.1 and Table 4.3.1.1-1 give reactor protection system instrumentation surveillance requirements for the functional unit of SDV water level-high. Each reactor protection system instrumentation channel containing a limit switch should be shown to be operable by the Channel Functional Test monthly and Channel Calibration at each refueling outage.

2.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

The NRC staff's Model Technical Specifications specify the following LCO/ surveillance requirements for control rod withdrawal block SDV limit switches:

"3.3.6 - The control rod withdrawal block instrumentation channel shown in Table 3.3.6-1 shall be OPERABLE with trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

Table 3.3.6-1. Control Rod Withdrawal Block Instrumentation

Trip Function	Minimum Operable Channels Per Trip Function	Applicable Operational Conditions	Action
5. Scram Discharge Volume			
a. Water level-high	2	1, 2, 5**	62
b. Scram trip bypassed	1	(1, 2, 5**)	62

ACTION 62: With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Table 3.3.6-2 Control Rod Withdrawal Block Instrumentation Setpoints

Trip Function	Trip Setpoint	Allowable Value
5. Scram Discharge Volume		
a. Water level-high	To be specified	NA
b. Scram trip bypassed	NA	NA

"4.3.6 - Each of the above control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

Table 4.3.6-1. Control Rod Withdrawal Block Instrumentation Surveillance Requirements

Trip Function	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
5. Scram Discharge Volume				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2."

Paragraph 3.3.6 and Table 3.3.6-1 of the Model Technical Specifications require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high and 1 operable channel containing 1 limit switch for SDV scram trip bypassed. The technical objective of these requirements is to have at least one channel containing one limit switch available to monitor the SDV water level when the other channel with a limit switch is being tested or undergoing maintenance. The trip setpoint for control rod withdrawal block instrumentation monitoring SDV water level-high should be specified as indicated in Table 3.3.6-2. The trip function prevents further withdrawal of any control rod when the control rod block SDV limit switches indicate water level-high.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high and once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

The Surveillance Criteria of the BWR Owners Subgroup given in Appendix A, "Long-Term Evaluation of Scram Discharge System," of "Generic Safety Evaluation Report BWR Scram Discharge System" [9] written by the NRC staff and issued on December 1, 1980, are:

1. Vent and drain valves shall be periodically tested.

2. Verifying and level detection instrumentation shall be periodically tested in place.

3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control rod density.

Analysis of the above criteria indicates that the NRC staff's Model Technical Specifications requirements, the acceptance criteria for the present TER, fully cover the BWR Owners Subgroup Surveillance Criteria 1 and 2 and partially cover Criterion 3.

3. METHOD OF EVALUATION

The PASNY submittal for the FitzPatrick Nuclear Power Plant was evaluated in two stages, initial and final.

During the initial evaluation, only the NRC staff's Model Technical Specifications requirements were used to determine if:

- the Licensee's submittal was responsive to the July 7, 1980 NRC request for proposed Technical Specifications changes involving the surveillance requirements of the SDV vent and drain valves, LCO/surveillance requirements for reactor protection system SDV limit switches, and LCO/surveillance requirements for control rod block SDV limit switches

- the submitted information was sufficient to permit a detailed technical evaluation.

During the final evaluation, in addition to the NRC staff's Model Technical Specifications requirements, background material in References 1 through 10, pertinent sections of FitzPatrick Nuclear Power Plant Final Safety Analysis Report, and FitzPatrick Technical Specifications were studied to determine the technical bases for the design of SDV main components and instrumentation. Subsequently, the Licensee's response was compared directly to the requirements of the NRC staff's Model Technical Specifications. The findings of the final evaluation are presented in Section 4 of this report.

The initial evaluation concluded that the Licensee's submittal was responsive to the NRC request of July 7, 1980, and the submittal contained sufficient information to permit preparation of a TER without a Request for Additional Information.

4. TECHNICAL EVALUATION

4.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 4.1.3.1.1 requires demonstrating that the SDV drain and vent valves are operable by:

a. verifying each valve to be open at least once per 31 days (valves may be closed intermittently for testing under administrative controls)

b. cycling each valve at least one complete cycle of full travel at least once per 92 days.

LICENSEE RESPONSE

The Licensee proposed to revise pages 89 and 96 of the FitzPatrick Technical Specifications as follows (see Appendix B):

"4.3 (Cont'd)

b. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days (these valves may be closed intermittently for testing under administrative control)."

"3. All control rods shall be determined operable once each operating cycle by demonstrating the scram discharge volume drain and vent valves operable when the scram test initiated by placing the mode switch in the SHUTDOWN position is performed as required by Table 4.1-1 and by verifying that the drain and vent valves:

a. Close within 80 seconds after receipt of a signal for control rods to scram, and

b. Open when the scram signal is reset or the scram discharge instrument volume trip is bypassed."

In addition, the Licensee agreed to revise proposed specifications changes to require cycling each valve at least one complete cycle of full travel at least quarterly.

FRC EVALUATION

The added paragraph 4.3b on the revised page 89 of the FitzPatrick Nuclear Power Plant Technical Specifications and the above agreed-upon additional revision comply with the requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b of the NRC staff's Model Technical Specifications.

4.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.1 and Table 3.3.1-1 require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, a total of 4 operable channels containing 4 limit switches per two trip systems for the reactor protection system which automatically initiates scram.

Paragraph 3.3.1 and Table 3.3.1-2 concern the response time of the reactor protection system for the functional unit of SDV water level-high which should be specified for each BWR (it is not specified in the table). Paragraph 4.3.1.1 and Table 4.3.1.1-1 require that each reactor protection system instrumentation channel containing a limit switch be shown to be operable for the functional unit of SDV water level-high by the Channel Functional Test monthly and Channel Calibration at each refueling outage. The applicable operational conditions for these requirements are startup, run, and refuel.

LICENSEE RESPONSE

The NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1 are addressed in the FitzPatrick Technical Specifications original page 41a, Table 3.1-1 (cont'd), Reactor Protection System (Scram) Instrumentation Requirement, which provides the following information for Trip Function High Water Level in Scram Discharge Volume:

"1. Minimum No. of Operable Instrument Channels per Trip System (1): 2

2. Trip Level Setting: $\leq$ 36 gal

3. Modes in Which Function Must be Operable: Refuel (2) (6), Startup, Run

4. Total Number of Instrument Channels Provided by Design for Both Trip Systems: 4 Instrument Channels

5. Action (1): A"

Notes:

"(1) There shall be two operable or tripped trip systems for each function, except as specified in 4.1.D. From and after the time that the minimum number of operable instrument channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

(2) Permissible to bypass in Refuel and Shutdown positions of the Reactor Mode Switch.

(3) When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

A. Mode Switch in Shutdown

B. Manual Scram

C. High Flux IRM

D. Scram Discharge Instrument Volume High Level when any control rod in a control cell containing fuel is not fully inserted.

E. APRM 15% Power Trip."

Note (3)D is taken from the revised page 43 (see Appendix B).

Page 103 of the FitzPatrick Technical Specifications gives the reactor protection system response time as follows:

"In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typical delay of about 210 milliseconds estimated from scram test results."

This fulfills the requirements of paragraph 3.3.1 and Table 3.3.1-2.

The Licensee's response to the requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1 was the submittal of the revised pages 44 through 47. The revised page 44 contains Table 4.1-1, Reactor Protection System (Scram) Instrument Functional Tests, Minimum Functional Test Frequencies for Safety

Instrument and Control Circuits, with the following information in regard to High Water Level in Scram Discharge Instrument Volume:

"1. Group (2): A

2. Functional Test: Trip Channel and Alarm

3. Minimum Frequency (3): Once/month and before each startup (6), (7).

Notes:

(2) A description of the three groups is included in the Bases of this Specification.

(3) Functional tests are not required on the part of the system that is not required to be operable or are tripped.

(6) Functional test of the instruments before each startup is required only if a scram has occurred since the last functional test or calibration.

(7) The functional test shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.

A. on-off sensors that provide a scram trip function."

Notes (2) and (3) are taken from the revised page 45. Notes (6) and (7) were revised and are taken from the revised page 45a.

The revised pages 46 and 47 provide the following information in Table 4.1-2, Reactor Protection System (Scram) Instrument Calibration, Minimum Calibration Frequencies for Reactor Protection Instrument Channels, for Instrument Channel High Water Level in Scram Discharge Instrument Volume:

"1. Group (1): A

2. Calibration (4): Water Column, Note (6)

3. Minimum Frequency (2)*: Once/Operating Cycle, Note (6)

Notes:

(1) A description of the three groups is included in the Bases of this Specification.

*The title of column 4 on the revised page 46, Table 4.1-2, should be "Minimum Frequency (2)," instead of "Minimum Frequency Once/Week"

(2) Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.

(4) Response time is not a part of the routine instrument channel test but will be checked once per operating cycle.

(6) Calibration shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected."

FRC EVALUATION

The original page 41a, Table 3.1-1 of the FitzPatrick Technical Specifications meets the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1. The FitzPatrick reactor protection system SDV water level-high instrumentation consists of 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per two trip systems, making 1-out-of-2-taken-twice logic. The original page 41a, Table 3.1-1 also specifies $\leq$ 36 gal as a trip setting for scram initiation and applicable operating conditions of refuel, startup, and run, which are acceptable.

The reactor protection system response time of 290 milliseconds specified on page 103 of the FitzPatrick Technical Specifications is acceptable and meets the requirements of paragraph 3.3.1 and Table 3.3.1-2.

The revised pages 44 through 47 with Table 4.1-1 and Table 4.1-2 of the FitzPatrick Technical Specifications comply fully with the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1. They prescribe the Channel Functional Test to be performed monthly and the Channel Calibration to be performed once per operating cycle, which is equivalent to refueling outage, as specified.

4.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.6 and Table 3.3.6-1 require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high, and 1 operable channel containing 1 limit switch for SDV trip bypassed. Paragraph 3.3.6 also requires specifying the trip setpoint for control rod withdrawal block instrumentation monitoring SDV water level-high as indicated in Table 3.3.6-2.

Paragraph 4.3.6 and Table 4.3.6-1 require each control rod withdrawal block instrumentation channel containing a limit switch to be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high and once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

LICENSEE RESPONSE

In response to the Model Technical Specifications paragraph 3.3.6, Table 3.3.6-1 and Table 3.3.6-2 requirements, the Licensee proposed revising pages 72 and 73 of the FitzPatrick Technical Specifications, which contain Table 3.2-3, Instrumentation That Initiates Control Rod Blocks, with the following information for Scram Discharge Instrument Volume High Water Level:

"1. Minimum No. of Operable Instrument Channels Per Trip System: 1

2. Trip Level Setting: $\leq$ 18 gallons

3. Total Number of Instrument Channels Provided by Design for Both Channels: 1 Inst. Channel

4. Action: (9), (10)

Notes:

(9) When the reactor is subcritical and the reactor water temperature is less than 212°F, the control rod block is required to be operable only if any control rod in a control cell containing fuel is not fully inserted.

(10) When the control rod block function associated with scram discharge instrument volume high water level is not operable when required to be operable, the trip system shall be tripped."

The Licensee responded to the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1 with proposed revision of page 81, Table 4.2-3, Minimum Test Calibration Frequency for Control Rod Blocks Actuation, which contains the following information for Instrument Channel Scram Discharge Instrument Volume-High Water Level:

"1. Instrument Functional Test: Once/Month (2)

2. Calibration: Once/Operating Cycle (2)

3. Instrument Check: N/A

Note:

(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."

FRC EVALUATION

The existing FitzPatrick Nuclear Plant scram discharge system has six level switches on the scram discharge volume (see FSAR, page 3.5-11) set at three different water levels to guard against operation of the reactor without sufficient free volume present in the scram discharge headers to receive the scram discharge water in the event of a scram. At the first (lowest) level, one level switch initiates an alarm for operator action. At the second level, with the setpoint of $\leq$ 18 gallons (see the revised page 72, Table 3.2-3), one level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, with the setpoint of $\leq$ 36 gallons (see page 41a, Table 3.1-1 of the FitzPatrick Technical Specifications), the four level switches (two for each reactor protection system trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge water. Reference 9, page 50, defines Design Criterion 9 ("Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation"), gives the technical basis for "Long-Term Evaluation of Scram Discharge System," and defines acceptable compliance ("The present alarm and rod block instrumentation meets this criterion given adequate hydraulic coupling with the SDV headers"). Thus, if the FitzPatrick Nuclear Power Plant scram discharge system is modified (long term) so that the hydraulic coupling between scram discharge headers and instrumented volume is adequate and acceptable, then the present alarm and rod block instrumentation consisting of one operable instrument channel with one limit switch for control rod withdrawal block as specified on the revised page 72 is also acceptable.

In the FitzPatrick Nuclear Power Station, "Scram Discharge Volume Scram Trips" cannot be bypassed while the reactor is in operational conditions of startup and run (see FSAR page 7.2-12) and operational condition "refuel with more than one control rod withdrawn" is not applicable, since interlocks are provided which prevent the withdrawal of more than one control rod with the mode switch in the refuel position. Thus, the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 with Table 3.3.6-1 and paragraph 4.3.6 with Table 4.3.6-1 are not applicable to the FitzPatrick Nuclear Power Station for "Trip Function 5. Scram Discharge Volume b. Scram Trip Bypassed" and were not addressed in the proposed revision of pages 72, 73, and 81. This is acceptable.

The 18-gallon trip level setting for control rod withdrawal block instrumentation is acceptable (see revised page 72 of the FitzPatrick Technical Specifications). The Licensee's proposed revision of page 81, Table 4.2-3 to meet the requirements of paragraph 4.3.6 and Table 4.3.6-1 is also acceptable since it prescribes the Channel Functional Test of each control rod withdrawal block instrumentation channel containing a limit switch once per month (required once per 3 months) and Channel Calibration once per operating cycle for SDV water level-high.

5. CONCLUSIONS

Table 5-1 summarizes results of the final review and evaluation of the FitzPatrick Phase 1 proposed Technical Specifications changes for SDV long-term modification in regard to surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements for reactor protection system and control rod block SDV limit switches. The following conclusions were made:

- o The revised page 89 of the FitzPatrick Technical Specifications and the Licensee's agreement to add to the proposed specifications changes a requirement to cycle each valve a minumum of one full cycle at least quarterly comply with the NRC staff's Model Technical Specifications, paragraphs 4.1.3.1.1a and 4.1.3.1.1b.

- o The remaining surveillance requirements are met by revised pages 43, 44, 45, 45a, 46, 47, 72, 73, 81, 89, 89a, and 96 of the FitzPatrick Technical Specifications and by pages 41a and 103 without revision.

Table 5-1. Evaluation of Phase 1 Proposed Technical Specifications Changes for Scram Discharge Volume Long-Term Modifications FitzPatrick Nuclear Power Plant

	Technical Specifications		
Surveillance Requirements	NRC Staff Model (Paragraph)	Proposed by Licensee	Evaluation
SDV DRAIN AND VENT VALVES			
Verify each valve open	Once per 31 days (4.1.3.1.1a)	Once per 31 days (p. 89, revised)	Acceptable
Cycle each valve one complete cycle	Once per 92 days (4.1.3.1.1b)	Quarterly (p. 89, revised)	Acceptable
REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES			
Minimum operable channels per trip system	2 (3.3.1, Table 3.3.1-1)	2 (p. 41a, Table 3.1-1)	Acceptable
SDV water level-high response time	NA (3.3.1, Table 3.3.1-2)	290 millisec. max. 210 millisec. test (p. 103)	Acceptable
SDV water level-high			
Channel functional test	Monthly (4.3.1.1, Table 4.3.1.1-1)	Once per month (p. 44, Table 4.1-1, revised)	Acceptable
Channel calibration	Each refueling (4.3.1.1, Table 4.3.1.1-1)	Once per operating cycle (p. 46, 47, Table 4.1-2, revised)	Acceptable

Table 5-1 (Cont.)

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
CONTROL ROD BLOCK SDV LIMIT SWITCHES			
Minimum operable channels per trip function			
SDV water level-high	2 (3.3.6, Table 3.3.6-1)	1 (p. 72, 73, Table 3.2-3, revised)	Acceptable*
SDV scram trip bypassed	1 (3.3.6, Table 3.3.6-1)	NA (p. 72, 73, Table 3.2-3, revised)	Acceptable*
SDV water level-high			
Trip setpoint	NA (3.3.6, Table 3.3.6-2)	$\leq$ 18 gallons (p. 72, 73 Table 3.2-3, revised)	Acceptable
Channel functional test	Quarterly (4.3.6, Table 4.3.6-1)	Once per month (p. 81, Table 4.2-3, revised)	Acceptable
Channel calibration	Each refueling (4.3.6, Table 4.3.6-1)	Once per operating cycle (p. 81, Table 4.2-3, revised)	Acceptable
SDV scram trip bypassed			
Channel functional test	Monthly (4.3.6, Table 4.3.6-1)	NA	Acceptable*

* See Reference 9, p. 50, and pp. 19 and 20 of this TER.

6. REFERENCES

1. IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capacity"
   NRC, Office of Inspection and Enforcement, June 12, 1980

2. D. G. Eisenhut (NRR), letter "To All Operating Boiling Water Reactors (BWRs)" with enclosure, "Model Technical Specifications," July 7, 1980

3. IE Bulletin 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 3, 1980

4. IE Bulletin 80-17, Supplement 1, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 18, 1980

5. IE Bulletin 80-17, Supplement 2, "Failures Revealed by Testing Subsequent to Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 22, 1980

6. IE Bulletin 80-17, Supplement 3, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, August 22, 1980

7. IE Bulletin 80-17, Supplement 4, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, December 18, 1980

8. IE Bulletin 80-17, Supplement 5, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, February 13, 1981

9. P. S. Check (NRR), memorandum with enclosure, "Generic Safety Evaluation Report BWR Scram Discharge System"
   December 1, 1980

10. P. S. Check (NRR), memorandum with enclosure, "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17"
    June 10, 1981

APPENDIX A

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS*

* Note: Applicable changes are marked by vertical lines in the margins.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days and

b. Cycling each valve through at least one complete cycle of full travel at least once per 92 days.

4.1.3.1.2 When above the preset power level of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

*These valves may be closed intermittently for testing under administrative controls.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position (6), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed (7.0) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding (7.0) seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of (7.0) seconds, or

c. Be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

* If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
8. Scram Discharge Volume Water Level - High	1, 2, 5(h)	2	4
9. Turbine Stop Valve - Closure	1(i)	4(j)	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1(i)	2(j)	7
11. Reactor Mode Switch in Shutdown Position	1, 2, 3, 4, 5	1	8
12. Manual Scram	1, 2, 3, 4, 5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to < (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER, within 2 hours.

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, verify all insertable control rods to be fully inserted within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 9 In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) These functions are automatically bypassed when turbine first stage pressure is ≤ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - Upscale	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, (15)%	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ (0.09)**
c. Fixed Neutron Flux - Upscale, (118)%	≤ (0.09)
d. Inoperative	NA
e. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ (0.55)
4. Reactor Vessel Water Level - Low, Level 3	≤ (1.05)
5. Main Steam Line Isolation Valve - Closure	≤ (0.06)
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	≤ (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ (0.08)#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Trip Oil Pressure - Low	NA	M	Q	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) Within 24 hours prior to startup, if not performed within the previous 7 days.
(c) The IRM and SRM channels shall be determined to overlap for at least (   ) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least (   ) decades during each controlled shutdown, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM readout to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

INSTRUMENTATION

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

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TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[(a)]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Simulated Thermal Power - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale[(c)]	3	2	61
	2	5	61
c. Inoperative[(c)]	3	2	61
	2	5	61
d. Downscale[(d)]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in (e)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative[(e)]	6	2, 5	61
d. Downscale	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypassed	1	(1, 2, 5**)	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. (Comparator) (Downscale)	2	1	62

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TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (2) or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	$\leq$ 0.66 W + (40)%	$\leq$ 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power - Upscale	$\leq$ 0.66 W + (42)%*	$\leq$ 0.66 W + (45)%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ (12)% of RATED THERMAL POWER	$\leq$ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ (2 x 10^5) cps	$\leq$ (5 x 10^5) cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ (3) cps	$\geq$ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ (100/125) of full scale	$\leq$ (110/125) of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ (5/125) of full scale	$\geq$ (3/125) of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	To be specified	NA
b. Scram Trip Bypassed	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ (__/__) of full scale	$\leq$ (__/__) of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	$\leq$ (10)% flow deviation	$\leq$ (__)% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

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TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[(b)],M	Q	1*
b. Inoperative	NA	S/U[(b)],M	NA	1*
c. Downscale	NA	S/U[(b)],M	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U[(b)],M	Q	1
b. Inoperative	NA	S/U[(b)],M	NA	1, 2, 5
c. Downscale	NA	S/U[(b)],M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U[(b)],M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)],W[(c)]	NA	2, 5
b. Upscale	NA	S/U[(b)],W[(c)]	Q	2, 5
c. Inoperative	NA	S/U[(b)],W[(c)]	NA	2, 5
d. Downscale	NA	S/U[(b)],W[(c)]	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)],W[(c)]	NA	2, 5
b. Upscale	NA	S/U[(b)],W[(c)]	Q	2, 5
c. Inoperative	NA	S/U[(b)],W[(c)]	NA	2, 5
d. Downscale	NA	S/U[(b)],W[(c)]	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[(b)],M	Q	1
b. Inoperative	NA	S/U[(b)],M	NA	1
c. (Comparator) (Downscale)	NA	S/U[(b)],M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

APPENDIX B

POWER AUTHORITY OF THE STATE OF NEW YORK LETTER OF JANUARY 6, 1981

AND

SUBMITTAL WITH PROPOSED TECHNICAL SPECIFICATIONS CHANGES

FOR

FITZPATRICK NUCLEAR POWER PLANT

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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SUBJECT: Forwards application for amend to License DPR-59 revising App A Tech Specs & safety evaluation re scram discharge vol. Class II amend fee encl.

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POWER AUTHORITY OF THE STATE OF NEW YORK

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January 6, 1981
JPN-80-2

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Thomas A. Ippolito, Chief
Operating Reactors Branch No. 2
Division of Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Proposed Change to the Technical Specification
Related to Scram Discharge Volume

Dear Sir:

Enclosed for filing are three (3) signed originals and nineteen (19) copies of a document entitled, "Application for Amendment to Operating License", together with forty (40) copies of Attachment I and II thereto, comprising a statement of the proposed changes to the Technical Specifications and the associated Safety Evaluation.

This application seeks to amend Appendix A of the Operating License in accordance with the Commission's July 7, 1980 letter which requested Technical Specification changes to provide surveillance for SDV vent and drain valves and LCO/surveillance requirements for RPS and control rod block SDIV limit switches.

A list of the proposed changes to the Technical Specifications is given below:

1. The proposed change on page 43 (Table 3.1-1) defines specifically the condition when the SDIV High Level trip function needs to be operable during cold shutdown.

2. The first change on page 44 (Table 4.1-1) corrects a typographical error.

3. The second change on page 44 increases the minimum frequency of the Scram Discharge Instrument Volume Water Level trip channel and alarm functional test.

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Commission -2-

4. The change on page 45 would replace AEC with NRC. The change on page 45a is the proposed revised Note (6) and new Note (7). The present form of Note (6) was applicable during the 1977 refueling outage only. The proposed Notes (6) and (7) are based upon Item A.5 of I. E. Bulletin 80-14.

5. The proposed changes to page 46 and 47 (Table 4.1-2) result from I & E Bulletin 80-14 (Item A.5) and the NRC letter dated July 7, 1980.

6. The proposed addition of Table 3.2-2 (page 72), together with the new Notes (9) and (10) on page 73 is written to be consistent with the proposed change to Table 3.1-1 (page 43).

7. Changes to Items 5 and 6 of Table 4.2-3 (page 81) are proposed to achieve consistency with Note 3 of Table 4.2-6. The remaining changes in Table 4.2-3 are proposed as a result of I. E. Bulletin 80-14.

8. The addition proposed to page 89 results from the I. E. Bulletin 80-14. Other changes to page 89 and to page 89a are merely a renumbering of paragraphs following the proposed addition.

9. The proposed addition to page 96 results from I. E. Bulletin 80-14.

The Authority has classified this application for amendment to the operating license as Class III, resulting from the NRC I. E. Bulletin 80-14 on Degradation of the BWR Scram Discharge Volume Capability and from the reference NRC letter. Enclosed is a check in the amount of $4,000 as the filing fee per 10CFR 170.22, which the Authority pays under protest pending a final determination of the legality of the fee schedule.

Very truly yours,

J. P. Bayne
Senior Vice President
Nuclear Generation

BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
POWER AUTHORITY OF THE STATE OF NEW YORK	)	Docket No. 50-333
	)	
James A. FitzPatrick Nuclear Power Plant	)	

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the Nuclear Regulatory Commission, the Power Authority of the State of New York, as holder of Facility Operating License No. DPR-59, hereby applies for an Amendment to the Technical Specifications contained in Appendix A of this license.

The proposed changes to the James A. FitzPatrick Technical Specifications occur in Sections 3.1, 3.2, 4.1, 4.2, and 4.3. These proposed changes result from the NRC letter dated July 7, 1980, and are therefore related to the control rod drive scram discharge volume capability.

The proposed changes to the Technical Specifications are presented in Attachment I to this application. The Safety Evaluation corresponding to this change is included in Attachment II.

POWER AUTHORITY OF THE
STATE OF NEW YORK

BY
J. P. Bayne
Senior Vice President
Nuclear Generation

Subscribed and sworn to before
me this day of 1981.

Notary Public

RUTH G. ZAPF
Notary Public, State of New York
No. 30-4663428
Qualified in Nassau County
Commission Expires March 30, 19

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ATTACHMENT I

PROPOSED OPERATING LICENSE ADDITION

RELATED TO

CONTROL ROD DRIVE

SCRAM DISCHARGE VOLUME CAPABILITY

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (cont'd)

C. High Flux IRM

D. Scram Discharge Instrument Volume High Level when any control rod in a control cell containing fuel is not fully inserted

E. APRM 15% Power Trip

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

9. The APRM downscale trip is automatically bypassed when the IRM Instrumentation is operable and not high.

10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.

11. See Section 2.1.A.1.

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP).

where:

FRP - Fraction of rated thermal power (2436 MWt)

MFLPD - Maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 MW/ft for 8x8, 8x8R and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used

W - Loop Recirculation flow in percent of rated (rated is 34.2×10^6 lb/hr)

S_n - Scram setting in percent of initial

13. The Average Power Range Monitor scram function is varied (Figure 1.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.

Table 4.1-1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each refueling outage.
Manual Scram	A	Trip Channel and Alarm	Every 3 months.
RPS Channel Test Switch	A	Trip Channel and Alarm	Every refueling outage or after channel maintenance.
IRM			
High Flux	C	Trip Channel and Alarm(4)	Once per week during refueling or startup and before each startup.
Inoperative	C	Trip Channel and Alarm(4)	Once per week during refueling or startup and before each startup.
APRM			
High Flux	B	Trip output Relays(4)	Once/week.
Inoperative	B	Trip output Relays(4)	Once/week.
Downscale	B	Trip output Relays(4)	Once/week.
Flow Bias	B	Calibrate Flow Bias Signal(4)	Once/month.(1)
High Flux in Startup or Refuel	C	Trip Output Relays(4)	Once per week during refueling or startup and before each startup.
High Reactor Pressure	B	Trip Channel and Alarm(4)	Once/month.(1) (Instrument check once per day)
High Drywell Pressure	A	Trip Channel and Alarm	Once/month(1)
Reactor Low Water Level(5)	A	Trip Channel and Alarm	Once/month(1)
High Water Level in Scram Discharge Instrument Volume	A	Trip Channel and Alarm	Once/month and before each startup(6),(7)
Main Steam Line High Radiation	B	Trip Channel and Alarm(4)	Once/week.

Table 4.1-1 (cont'd)

REACTOR PROTECTION SYSTEM(SCRAM) INSTRUMENT FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/month.(1)
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Once/month.
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every 3 months.(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/month.(1)
Reactor Pressure Permissive	A	Trip Channel and Alarm	Every 3 months.

NOTES FOR TABLE 4.1-1

1. Initially once every month until acceptable 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 operates in an environment similar to that of JAFNPP.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required on the part of the system that is not required to be operable or are tripped.

   If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the instrument channels.

Table 4.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

NOTES FOR TABLE 4.1-1 (cont'd)

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the functional test program.

6. Functional test of the instruments before each startup is required only if a scram has occurred since the last functional test or calibration.

7. The functional test shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.

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Table 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency Once/week
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Maximum frequency once/week
APRM High Flux			
Output Signal	B	Heat Balance	Daily
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source	Every refueling outage
LPRM Signal	B	TIP System Traverse	Every 1000 effective full power hours
High Reactor Pressure	B	Standard Pressure Source	Once/operating cycle
High Drywell Pressure	A	Standard Pressure Source	Every 3 months
Reactor Low Water Level	A	Pressure Standard	Every 3 months
High Water level in Scram Discharge Instrument Volume	A	Water Column, Note(6)	Once/operating cycle, Note(6)
Main Steam Line Isolation Valve Closure	A	Note(5)	Note(5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 months
Turbine Plant Stage Pressure Permissive	A	Standard Pressure Source	Every 6 months
Turbine Control Valve Fast Closure Oil Pressure Trip	A	Standard Pressure Source	Once/operating cycle

Table 4.1-2 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration(4)	Minimum Frequency (2)
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Reactor Pressure Permissive	A	Standard Pressure Source	Every 6 months

NOTES FOR TABLE 4.1-2

1. A description of three groups is included in the Bases of this Specification.

2. Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

4. Response time is not a part of the routine instrument channel test but will be checked once per operating cycle.

5. Actuation of these switches by normal means will be performed during the refueling outages.

6. Calibration shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action
2	APRM Upscale (Flow Biased)	$S \leq (0.66W+42\%) \times \frac{FRP}{MFLPD}$	6 Inst. Channels	(1)
2	APRM Upscale (Start-up Mode)	$\leq$ 12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Flow Biased)	$S \leq 0.66W+K$ (8)	2 Inst. Channels	(1)
1 (6)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (2)	$\geq$2% of full scale	8 Inst. Channels	(1)
3	IRM Detector not in Start-up Position	(7)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$86.4% of full scale	8 Inst. Channels	(1)
2 (4)	SRM Detector not in Start-up Position	(3)	4 Inst. Channels	(1)
2 (4)(5)	SRM Upscale	$\leq 10^5$ counts/sec	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Water Level	$\leq$ 18 gallons	1 Inst. Channel	(9) (10)

NOTES FOR TABLE 3.2-3

1. For the Start-up and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in run mode, and

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3

the APRM and RBM rod blocks need not be operable in start-up mode. From and after the time it is found that the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter, if this condition lasts longer than seven days, the system shall be tripped. From and after the time it is found that the first column cannot be met for both trip systems, the systems shall be tripped.

2. IRM downscale is bypassed when it is on its lowest range.

3. This function is bypassed when the count rate is $\geq$ 100 cps.

4. One of the four SRM inputs may be bypassed.

5. This SRM Function is bypassed when the IRM range switches are on range 8 or above.

6. The trip is bypassed when the reactor power is $\leq$ 30%.

7. This function is bypassed when the Mode Switch is placed in Run.

8. S = Rod Block Monitor Setting in percent of initial.

   W = Loop recirculation flow in percent of rated,(rated loop recirculation flow is 34.2×10^6 lb/hr).

   K = Intercept values of 39%, 40%, 41% and 42% can be used with appropriate MCPR limits from Section 3.1.B.

9. When the reactor is subcritical and the reactor water temperature is less than 212°F, the control rod block is required to be operable only if any control rod in a control cell containing fuel is not fully inserted.

10. When the control rod block function associated with scram discharge instrument volume high water level is not operable when required to be operable, the trip system shall be tripped.

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TABLE 4.2-3

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

	Instrument Channel	Instrument Functional Test	Calibration	Instrument Check (9)
1)	APRM - Downscale	(1) (3)	Once/3 months	Once/day
2)	APRM - Upscale	(1) (3)	Once/3 months	Once/day
3)	IRM - Upscale	(2) (3)	(2)	(2)
4)	IRM - Downscale	(2) (3)	(2)	(2)
5)	RBM - Upscale	(1) (3)	Once/3 months	Once/day
6)	RBM - Downscale	(1) (3)	Once/3 months	Once/day
7)	SRM - Upscale	(2) (3)	(2)	(2)
8)	SRM - Detector Not in Stattup Position	(2) (3)	(2)	
9)	IRM - Detector Not in Startup Position	(2) (3)	(2)	
10)	Scram Discharge Instrument Volume - High water level	Once/month (2)	Once/operating Cycle (2)	N/A

	Logic System Functional Test (4)(6)	Frequency
1)	System Logic Check	Once/6 months

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

3.3 (cont'd)

a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to the Cold Shutdown condition within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing, and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.

If investigation demonstrates that the cause of control rod failure is a cracked collet housing, or if this possibility cannot be ruled out, the reactor shall not be started until the affected control rod drive has been replaced or repaired.

4.3 (cont'd)

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30 percent power. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above 30 percent power.

b. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days (these valves may be closed intermittently for testing under administrative control).

c. A second licensed operator shall verify the conformance to Specification 3.3.A.2.d before a rod may be bypassed in the Rod Sequence Control System.

d. Once per week check status of pressure and level alarms for each accumulator.

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in

e. When it is initially determined that a control rod is incapable of normal insertion, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted

shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted. If Specification 3.3.A.1 and 4.3.A.1 are met, reactor startup may proceed.

3.3 (cont'd)

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Control Rod Notch Position Observed	Average Scram Insertion Time (Sec)
46	0.361
38	0.977
24	2.112
04	3.764

4.3 (cont'd)

2. At 8-week intervals, 15 percent of the operable control rod drives shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. All control rods shall be determined operable once each operating cycle by demonstrating the scram discharge volume drain and vent valves operable when the scram test initiated by placing the mode switch in the SHUTDOWN position is performed as required by Table 4.1-1 and by verifying that the drain and vent valves:

   a. Close within 80 seconds after receipt of a signal for control rods to scram , and

   b. Open when the scram signal is reset or the scram discharge instrument volume trip is bypassed.

ATTACHMENT II

SAFETY EVALUATION

RELATED TO

SCRAM DISCHARGE VOLUME

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333
JANUARY 6, 1981

Franklin Research Center
A Division of The Franklin Institute

Section I - Description of Modification

The modification provides surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements for the RPS and control rod block SDIV limit switches, in accordance with the NRC letter dated July 7, 1980 to all BWR Licensees.

Section II - Purpose of the Modification

The purpose of the modification is to ensure that the SDV is operable and that the control rod drive system is operable during reactor operation.

Section III - Impact of the Change

These modifications will not alter the conclusion reached in the FSAR and SER accident analysis.

Section IV - Implementation of the Modification

The modification as proposed will not impact the Fire Protection Program at JAF.

Section V - Conclusion

The incorporation of these modifications: a) will not increase the probability nor the consequences of an accident as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; and c) will not reduce the margin of safety as defined in the basis for any Technical Specification, and d) does not constitute an unreviewed safety question.

Section VI - References

(a) JAF FSAR

(b) JAF SER

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. 62 to Operating License No. DPR-59 issued to the Power Authority of the State of New York which revises the Technical Specifications for operation of the James A. FitzPatrick Nuclear Plant (the facility) located in Oswego County, New York. The amendment is effective as of the date of its issuance.

The amendment modifies the Technical Specifications to reflect scram discharge volume (SDV) long-term modifications.

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 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 January 6, 1981, (2) Amendment No. 62 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, N.W., 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 29th day of January 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

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