ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO NUREG-0737 ITEMS

NEW YORK POWER AUTHORITY

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

JAFNPP

TECHNICAL SPECIFICATIONS

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Table 4.1-1

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

INSTRUMENT CHANNEL		GROUP	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	Once/month (7)
High Water Level in Scram Discharge Instrument Volume		B	Trip Channel and Alarm (4)	Once/month
Main Steam Line High Radiation		B	Trip Channel and Alarm (4)	Once/week

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 INSTRUMENTATION

Applicability:

Applies to the plant instrumentation which either (1) initiates and controls a protective function, or (2) provides information to aid the operator in monitoring and assessing plant status during normal and accident conditions.

Objective:

To assure the operability of the aforementioned instrumentation.

Specifications:

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table B.3.2-1.

B. Core and Containment Cooling Systems - Initiation & Control

The limiting conditions for operation for the instrumentation that initiates or controls the Core and Containment Cooling Systems are given in Table 3.2-2. This instrumentation must be operable when the system(s) it initiates or

4.2 SURVEILLANCE REQUIREMENTS

4.2 INSTRUMENTATION

Applicability:

Applies to the surveillance requirement of the instrumentation which either (1) initiates and controls a protective function, or (2) provides information to aid the operator in monitoring and assessing plant status during normal and accident conditions.

Objective:

To specify the type and frequency of surveillance to be applied to the aforementioned instrumentation.

Specifications:

A. Primary Containment Isolation Functions

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-1.

System logic shall be functionally tested as indicated in Table 4.2-1.

B. Core and Containment Cooling Systems - Initiation and Control

Instrumentation shall be functionally tested, calibrated, and checked as indicated in Table 4.2-2.

3.2 (cont'd)

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-5.

F. Surveillance Information Readouts

The limiting conditions for the instrumentation that provide(s) surveillance information readouts are given in Table 3.2-6.

G. Recirculation Pump Trip

The limiting conditions for operation for the instrumentation that trip(s) the recirculation pumps as a means of limiting the consequences of a failure to scram during an anticipated transient are given in Table 3.2-7.

H. Accident Monitoring Instrumentation

The limiting conditions for operation of the instrumentation that provides accident monitoring are given in Table 3.2-8.

4.2 (cont'd)

E. Drywell Leak Detection

Instrumentation shall be calibrated and checked as indicated in Table 4.2-5.

F. Surveillance Information Readouts

Instrumentation shall be calibrated and checked as indicated in Table 4.2-6.

G. Recirculation Pump Trip

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-7.

System logic shall be functionally tested as indicated in Table 4.2-7.

H. Accident Monitoring Instrumentation

Instrumentation shall be demonstrated operable by performance of a channel check and channel calibration as indicated in Table 4.2-8.

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report, NEDO-10349, dated March, 1971.

Accident monitoring instrumentation provides additional information which is helpful to the operator in assessing plant conditions following an accident by (1) providing information needed to permit the operators to take preplanned manual actions to accomplish safe plant shutdown; (2) determining whether systems are performing their intended functions; (3) providing information to the operators that will enable them to determine the potential for a breach of the barriers to radioactivity release and if a barrier has been breached; (4) furnishing data for deciding on the need to take unplanned action if an automatic or manually initiated safety system is not functioning properly or the plant is not responding properly to the safety systems in operation; and (5) allowing for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any problem. This instrumentation has been upgraded to conform with the acceptance criteria of NUREG-0737 and NRC Generic Letter 83-36.

TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Item No.	Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Remarks
22	2	Condensate Storage Tank Low Level	≥ 59.5 inches above tank bottom (= 15,600 gal. avail)	2 Inst. Channels	Transfers RCIC pump suction to suppression chamber
23					
24					
25	1	Core Spray Sparger to Reactor Pressure vessel d/p	≤ 0.5 psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break
26	2	Condensate Storage Tank Low Level	≥ 59.5 in. above tank bottom (= 15,600 gal avail)	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.
27	2	Suppression Chamber High Level	≤ 6 in. above normal level	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.
28	1	RCIC Turbine Steam Line High Flow	≤ 282 in. H_2O psid	2 Inst. Channels	Close Isolation Valves in RCIC Subsystem.

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TABLE 3.2-8

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	NO. OF CHANNELS PROVIDED BY DESIGN	MINIMUM NO. OF OPERABLE CHANNELS REQUIRED	ACTION	MEASUREMENT RANGE
1. Stack High Range Effluent Monitor	2	1	A	10^{-1} to 10^{7} mR/hr.
2. Turbine Building Vent High Range Effluent Monitor	2	1	A	10^{-1} to 10^{7} mR/hr.
3. Radwaste Building Vent High Range Effluent Monitor	2	1	A	10^{-1} to 10^{7} mR/hr.
4. Containment High Range Radiation Monitor*	2	1	A	1 to 10^{8} rads/hr.
5. Containment Pressure Transmitter	2 wide range 2 narrow range	1 1	A A	0 to 250 psig -5 + 5 psig
6. Drywell Level Transmitter	2	1	A	22 to 100 ft. (H_2O)
7. Suppression Pool Level Transmitter	2	1	A	1.7 to 27.5 ft. (H_2O)
8. Reactor Vessel Pressure Transmitter	2	1	A	0 to 1500 psig
9. Drywell Hydrogen Concentration Monitor	2	1	A	0 to 30% H_2

* At $\leq$450 R/hr, closes vent and purge valves

TABLE 3.2-8(cont'd)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	NO. OF CHANNELS PROVIDED BY DESIGN	MINIMUM NO. OF OPERABLE CHANNELS REQUIRED	ACTION	MEASUREMENT RANGE
10. Post-Accident Containment and Reactor Coolant Radioactivity Sampling Components				
a) Gaseous Radioactivity Sampling Tray Components	1	1	B	10^{-5} uCi/cc to 10^{6} uCi/cc
b) Liquid Radioactivity Sampling Tray Components	1	1	B	10 uCi/ml to 10 Ci/ml

NOTES FOR TABLE 3.2-8

A. With the number of operable channels less than the required minimum, either restore the inoperable channels to operable status within 30 days, or: (1) initiate an alternate method of monitoring the appropriate parameter(s), or (2) be in a cold condition within the next 24 hours.

B. 1. When the ability to provide a liquid and/or gaseous sample to the Post-Accident Sample Station has been lost, restore the capability within 7 days or be in at least a cold condition within the next 24 hours.

2. When the ability to sample and analyze a liquid and/or gaseous sample has been lost, within 7 days confirm that alternate arrangements for sampling and analysis of the required samples can be made available within 24 hours of the need to perform sampling and analysis. If alternate arrangements for sampling and analysis can not be confirmed within 7 days, be in the cold condition within the next 24 hours.

3. When operating in accordance with 2 above, restore complete sampling and analysis capability within 90 days. If complete sampling and analysis capability cannot be restored within 90 days, be in the cold condition within the next 24 hours.

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TABLE 4.2-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Water Level	(1)	Once/3 months	Once/day
2) Drywell Pressure	(1)	Once/3 months	None
3) Reactor Pressure	(1)	Once/3 months	None
4) Auto Sequencing Timers	N/A	Once/operating cycle	None
5) ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	N/A	None
8) Core Spray Sparger d/p	(1)	Once/3 months	Once/day
9) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
10) Steam Line/Area High Temp. (HPCI & RCIC)	(1)	Once/operating Cycle	Once/day
12) HPCI & RCIC Steam Line Low Pressure	(1)	Once/3 months	None
13) HPCI & RCIC Suction Source Levels	(1)	Once/3 months	None
14) 4kV Emergency Power Under-Voltage Relays and timers	Once/operating cycle	Once/operating cycle	None
15) HPCI & RCIC Exhaust Diaphragm Pressure High	(1)	Once/3 months	None
17) LPCI/Cross Connect Valve Position	Once/operating cycle	NA	NA

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

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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 - Downscale	(1) (3)	Once/3 months	Once/Day
6) RBM - Upscale	(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- Dectector Not in Startup Position	(2) (3)	(2)	
10) Scram Discharge Instrument Volume - High Water Level	Once/month (2)(3)	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.

TABLE 4.2-8
MINIMUM TEST AND CALIBRATION FREQUENCY FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	INSTRUMENT FUNCTIONAL TEST	CALIBRATION FREQUENCY	INSTRUMENT CHECK
1. Stack High Range Effluent Monitor	Once/Operating Cycle	Once/Operating Cycle	Once/day
2. Turbine Building Vent High Range Effluent Monitor	Once/Operating Cycle	Once/Operating Cycle	Once/day
3. Radwaste Building Vent High Range Effluent Monitor	Once/Operating Cycle	Once/Operating Cycle	Once/day
4. Containment High Range Radiation Monitor	Once/Operating Cycle	Once/Operating Cycle	Once/day
5. Containment Pressure Transmitter	N/A	Once/Operating Cycle	Once/day
6. Drywell Level Transmitter	N/A	Once/Operating Cycle	Once/day
7. Suppression Pool Level Transmitter	N/A	Once/Operating Cycle	Once/day
8. Reactor Vessel Pressure Channel	N/A	Once/Operating Cycle	Once/day
9. Drywell Hydrogen Concentration Analyzer	N/A	Once/Operating Cycle	Once/day
10. Post-Accident Containment and Reactor Coolant Radioactivity Sampling Components			
a) Gaseous Radioactivity Sampling Tray Components	Once/Operating Cycle	Once/Operating Cycle	Once/quarter
b) Liquid Radioactivity Sampling Tray Components	Once/Operating Cycle	Once/Operating Cycle	Once/quarter

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E. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 150 psig and prior to a reactor startup from a cold condition, except from the time that the RCIC System is made or found to be inoperable for any reason, continued reactor power operation is permissible during the succeeding 7 days unless the system is made operable earlier provided that during these 7 days the HPCI System is operable.

2. If the requirements of 3.5.E cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hours.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.5.E.2 above, provided that reactor coolant temperature is $\leq$212°F.

E. Reactor Core Isolation Cooling (RCIC) System

1. RCIC System testing shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within ten days of continuous operation from the time steam becomes available.

	Item	Frequency
a.	Simulated Automatic Actuation (and Restart*) Test	Once/operating cycle
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month
d.	Flow Rate	Once/3 months
e.	Testable Check Valves	Tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

* Automatic restart on a low water level signal which is subsequent to a high water level trip.

3.6 (cont'd)

E. Safety and Safety/Relief Valves

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

4.6 (cont'd)

E. Safety and Safety/Relief Valves

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

E. Safety and Safety/Relief Valves

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

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

E. Safety and Safety/Relief Valves

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

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

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

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

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

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

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

4. An annual report of safety/relief valve failures and challenges will be sent to the NRC in accordance with Section 6.9.A.2.b

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

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

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

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

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3.6 (cont'd)

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

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

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

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

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

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

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When the primary containment is inerted, it shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. The monitoring system may be taken out of service for maintenance, but shall be returned to service as soon as possible.

4. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.4.b below, two Pressure Suppression Chamber-Reactor Building Vacuum Breakers shall be operable at all times when the primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber reactor building vacuum breakers shall be $\leq$ 0.5 psi external pressure.

b. From and after the date that one of the pressure suppression chamber - reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days, unless

4. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentations including setpoint shall be checked for proper operation every three months.

TABLE 4.7-2
EXCEPTION TO TYPE C TESTS

Certain Type C tests will be perfo~med or omitted as follows:

Penetration	System	Valve	Local Leak Rate Test Performed
X-7A, B, C and D	Main Steam	29-AOV-80A, B C, and D 29-AOV-86A, B, C, and D	These valves are air-operated globe valves - pressurized in reverse direction and measurement of leakage will be equivalent to results from pressure appli~d in the same direction as when the valves would be required to perform its safety function. Therefore, pressure will be applied between the isolation valves and leakage measured. A water seal of 25 psig will be used on the inboard valve to determine the outboard valve's leak rate. (limit 11.5 SCFH at 25 psig)
X-10	RCIC	13-MOV-15	See X-25 (27-AOV-131A, B)
X-11	HPCI	23-MOV-15	See X-25 (27-AOV-131A, B)
X-25	Dry Well Inerting CAD and Purge	27-AOV-112	This valve is a butterfly valve - pressurization in reverse direction and measurement of leakage will be equivalent to results from pressure applied in the same direction as that when the valve would be required to perform its safety function.

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TABLE 4.7-2
EXCEPTION TO TYPE C TESTS

(CONTINUED)

Penetration	System	Valve	Local Leak Rate Test Performed
X-25	Dry Well Inerting CAD and Purge	27-AOV-131A 27 AOV-131B	These valves will be tested in the reverse direction, since the system was not designed for test pressure to be applied in the same direction as that when the valve would be required to perform its safety function. Basis - The pressurization direction was not a requirement at the time of plant design.
X-26 A/B	Dry Well Inerting CAD and Purge	27-AOV-113 27-MOV-122	See X-25 (27-AOV-112) This globe valve will be tested in the reverse direction. See X-25 (27-AOV-131A, B)

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TABLE 4.7-2 (CONT'D)

Penetration	System	Valve	Local Leak Rate Test Performed
		27-SOV-120B 27-SOV-121B 27-SOV-122B	See X-25 (27-AOV-131A, B)
X-31 Bd	Dry Well Inerting CAD and Purge	27-SOV-125B	See X-25 (27-AOV-131A)
X-39A	Cont. Spray	10-MOV-31A	This valve will be pressurized in the reverse direction and leakage measured. See X-25 (27-SOV-131A, B)
X-39B	Cont. Spray	10-MOV-31A	See X-39A
X-45	ILRT	VSM-100T	See X-25 (27-AOV-131A, B)
X-59	Dry Well Inerting Cad and Purge	27-SOV-123A	See X-25 (27-AOV-131A, B)
X-202	Torus Vacuum Breakers	AOV-101A/B	See X-25 (27-AOV-112)
X-203A	Dry Well Inerting CAD and Purge	27-SOV-119B	See X-25 (27-AOV-131A, B)
X-203B	Dry Well Inerting CAD and Purge	27-SOV-124A	See X-25 (27-AOV-131A)
X-205	Dry Well Inerting CAD and Purge	27-AOV-117 27-MOV-117	See X-25 (27-AOV-112) See X-25 (27-MOV-113)
X-210 A/B	RCIC, RHR		Will not be tested as lines are water sealed by suppression chamber water See X-25 (27-AOV-131A, B)
X-211A	RHR	10-MOV-38A	This valve will be tested in the reverse direction. See X-25 (27-AOV-131A, B)
X-211B	RHR	10-MOV-38B	This valve will be tested in the reverse direction.
X-212	RCIC	13-MOV-130	See X-25 (27-AOV-131A/B)
X-218	ILRT	VSM-100T	See X-25 (27-AOV-131A/B)
X-220	Dry Well Inerting CAD and Purge	27-AOV-116 27-SOV-132A 27-SOV-132B	See X-25 (27-AOV-112) See X-25 (27-AOV-131A/B)
X-222	HPCI		See X-210 A/B
X-224	RHR		See X-210 A/B
X-225	RHR		See X-210 A/B

TABLE 4.7-2 (CONT'D)

Penetration	System	Valve	Local Leak Rate Test Performed
X-226	HPCI		See X-210 A/B
X-227	Core Spray		See X-210 A/B
X-228	Condensate		See X-210 A/B

6.0 ADMINISTRATIVE CONTROLS

Administrative Controls are the means by which plant operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, plant organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of these measures is necessary to ensure safe and efficient facility operation.

6.1 RESPONSIBILITY

The Resident Manager is responsible for safe operation of the plant. During periods when the Resident Manager is unavailable, the Superintendent of Power will assume his responsibilities. In the event both are unavailable, the Resident Manager may delegate this responsibility to other qualified supervisory personnel. The Resident Manager reports directly to the Executive Vice President Nuclear Generation, as shown in Fig. 6.1-1.

6.2 PLANT STAFF ORGANIZATION

The plant staff organization is shown graphically in Fig. 6.2-1 and functions as follows:

1. A licensed senior reactor operator shall be on site at all times when there is fuel in the reactor.

2. In addition to item 1 above, a licensed reactor operator shall be in the control room at all times when there is fuel in the reactor.

3. In addition to items 1 & 2 above, a licensed reactor operator and a licensed senior reactor operator shall be readily available on site whenever the reactor is in other than cold condition.

4. Two licensed reactor operators shall be in the control room during start-ups and scheduled shutdowns.

5. A licensed senior reactor operator shall be responsible for all movement of new and irradiated fuel within the site boundary. A licensed reactor operator will be required to manipulate or directly supervise the manipulation of the controls of all fuel moving equipment, except the reactor building crane. All fuel movements by the reactor building crane, except new fuel movements from receipt through dry storage, shall be under the direct supervision of a licensed reactor operator. All fuel movements within the core shall be directly monitored by a member of the reactor analyst group.[a]

SEE NEXT PAGE FOR FOOTNOTE

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

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

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

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

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

11. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and maintenance personnel who are working on safety-related systems.

    Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

    a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

    b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

    c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

    d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

    Any deviation from the above guidelines shall be authorized by the Resident Manager or the Superintendent of Power, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting

the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Resident Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

6.3 PLANT STAFF QUALIFICATIONS

The minimum qualifications with regard to educational background and experience for plant staff positions shown in Fig. 6.2.1 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions; except for the Radiation and Environmental Services Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Coordinator to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI N18.1-1971.

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

6.5 REVIEW AND AUDIT

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

6.5.1 PLANT OPERATING REVIEW COMMITTEE (PORC)

(A) Membership

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

(A) ROUTINE REPORTS AND REPORTABLE OCCURRENCES (Continued)

1. STARTUP REPORT (Continued)

b. Startup Reports shall be submitted within (1) 90 days following completion of the startup test program, or (2) 90 days following resumption or commencement of commercial power operation, or whichever is earliest. If the Startup Report does not cover both events, i.e., completion of startup test program and resumption or commencement of commercial power operation, supplementary reports shall be submitted at least every three months until both events are completed.

2. ANNUAL REPORTS

a. Annual Occupational Exposure Tabulation

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources, shall be assigned to specific major work functions.

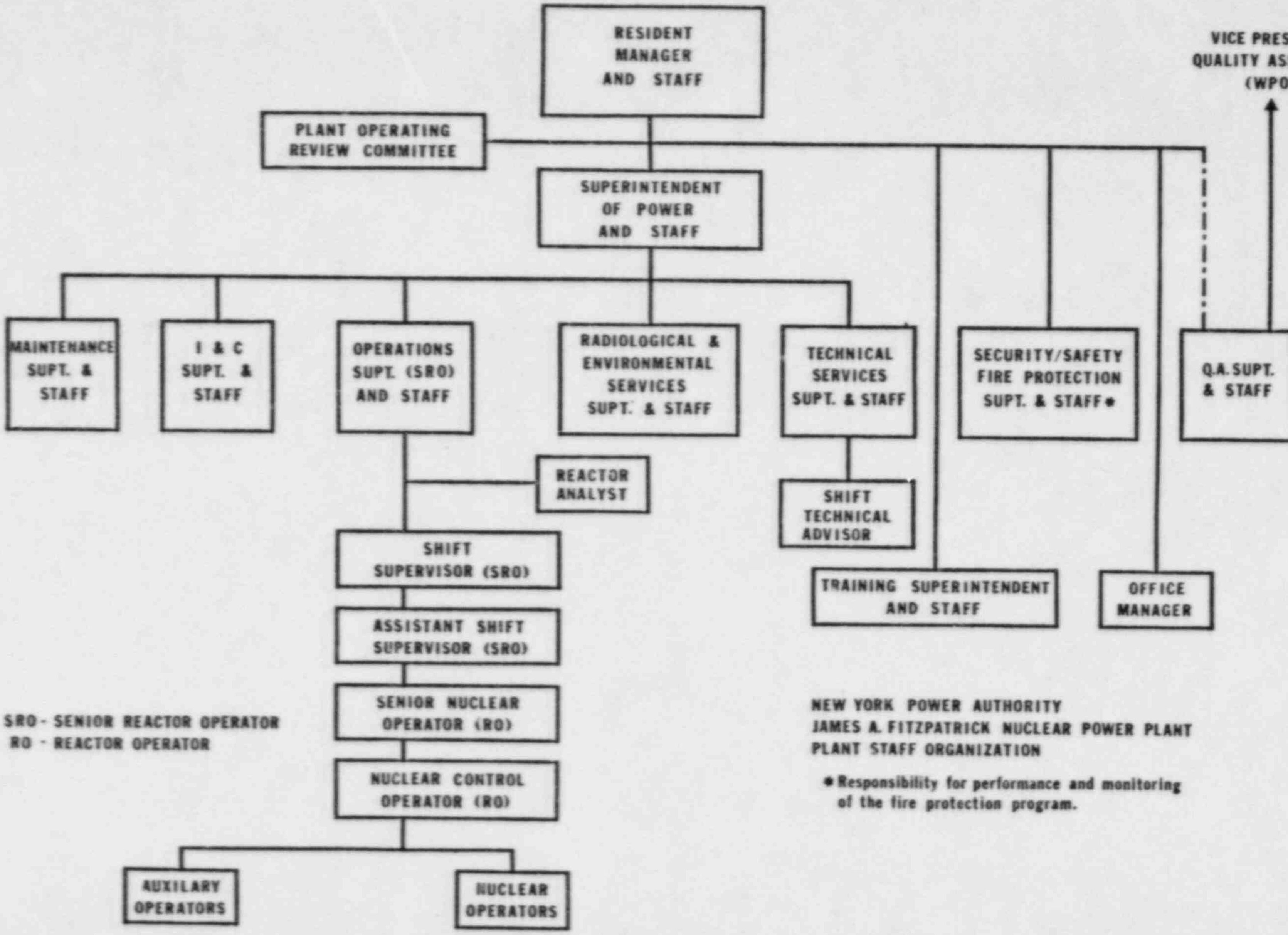
b. Annual Report of S/RV Failures and Challenges

An annual report of safety/relief valve failures and challenges will be submitted prior to March 1 of each year. An S/RV failure is defined to be any one of the following: (1) Failure of a valve to actuate when automatically or manually signaled to do so, for any reason, including testing, (2) Failure of a valve to close when the actuation signal is removed, (3) Spontaneous operation of a valve in the absence of an actuation signal, and (4) Determination that a valve is, or was inoperable in any operating mode except cold shutdown or refueling. An S/RV challenge is defined as an automatic or manual signaling of the valve to actuate for the purpose of controlling physical parameters of the primary coolant system. Test actuations of S/RV's are not considered challenges and will not be reported unless the test results in a failure.

3. MONTHLY OPERATING REPORT

A report providing a narrative summary of facility operating experience, major safety-related maintenance, and other pertinent information, should be submitted not later than the 15th of each month following the calendar month covered to the USNRC Director, Office of Management Information and Program Control.

1/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20



SRO - SENIOR REACTOR OPERATOR
RO - REACTOR OPERATOR

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

* Responsibility for performance and monitoring of the fire protection program.

FIGURE 6.2-1

ATTACHMENT II

Safety Evaluation

for

Technical Specifications Related to NUREG-0737 Items

(JPTS-84-011)
(JPTS-84-015)

New York Power Authority

James A. Fitzpatrick Nuclear Power Plant

Docket No. 50-333

I. Description of the Proposed Changes

The proposed changes to the FitzPatrick Technical Specifications relate to items identified in NUREG-0737, "Clarification of TMI Action Plan Requirements."

Specifically, the following changes are being proposed:

The Table of Contents on pages i to iv has been retyped to restore margins. In addition, the following changes are incorporated. Page numbers refer to the retyped revision of the Table. On page i, the following line is added:

3.0	General	4.0	3.0

On page i, the page number for Section 3.1 is changed from "30" to "30f."

On page i, "Protective Instrumentation" is changed to "Instrumentation."

On page i, the following line is added:

H. Accident Monitoring Instrumentation H. 54

On page ii, page number "136" is added for Section 3.6.

On page iii, add the following line:

G. Reactor Protection System Electrical Protection Assemblies G. 222c

On page iii, add the following:

3.12	Fire Protection Systems	4.12	244a
	A. High Pressure Water	A.	244a
	B. Water Spray/and Sprinkler Systems	B.	244e
	C. Carbon Dioxide Systems	C.	244e
	D. Manual Fire Hose Stations	D.	244f
	E. Fire Protection Systems Smoke & Heat Detectors	E.	244g
	F. Fire Barrier Penetration Seals	F.	244g

On page iv, the page number for Section 6.5 is changed from "249" to "248."

On page iv, the page number for Section 6.9 is changed from "254A" to "254a."

On page iv, the page number for Section 6.10 is changed from "254H" to "254g."

On page iv, the following line is added:

6.15 Environmental Qualification 258a

On page v, the List of Tables is revised to include Table 3.2-8.

On page vi, the List of Tables is revised to include Tables 4.2-8, 3.12-1, 3.12-3, 4.12-1, 4.12-2 and 4.12-3, and to delete Table 6.11-1.

On page vii, a typographic error in the fuel type listed for Figure 3.5-9 is corrected by changing "P8DRB284L" to P8DRB284H."

On page 44, add Note number (4) to "Trip Channel and Alarm" for High Water Level in Scram Discharge Instrument Volume," Group "B."

On page 49, in Sections 3.2 and 4.2, change titles from "Protective Instrumentation" to "Instrumentation." Revise "Applicability" paragraphs to add instrumentation which provides information to aid in monitoring and assessing plant status. In "Objective" paragraphs change "protective" to "aforementioned."

On page 54, the following is added to Section 3.2:

H. Accident Monitoring Instrumentation

The limiting conditions for operation of the instrumentation that provides accident monitoring are given in Table 3.2-8.

On page 54, the following is added to Section 4.2:

H. Accident Monitoring Instrumentation

Instrumentation shall be demonstrated operable by performance of a channel check and channel calibration as indicated in Table 4.2-8.

On page 60, the following is added to Section 3.2:

Accident monitoring instrumentation provides additional information which is helpful to the operator in assessing plant conditions following an accident by (1) providing information needed to permit the operators to take preplanned manual actions to accomplish safe plant shutdown; (2) determining whether systems are performing their intended functions; (3) providing information to the operators that will enable them to determine the

potential for a breach of the barriers to radioactivity release and if a barrier has been breached; (4) furnishing data for deciding on the need to take unplanned action if an automatic or manually initiated safety system is not functioning properly or the plant is not responding properly to the safety systems in operation; and (5) allowing for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any problem. This instrumentation has been upgraded to conform with the acceptance criteria of NUREG-0737 and NRC Generic Letter 83-36.

On page 70a Table 3.2-2, "Instrumentation that Initiates or Controls the Core and Containment Cooling System" is revised by adding a new Item 22 condensate storage tank low level. For Item No. 22, the following information is added: minimum number of operable instrument channels per trip system, trip function, trip level setting, and total number of instrument channels provided by design for both trip systems. Remarks for Item 26 are revised to read, "Transfers HPCI pump suction to suppression chamber."

New pages 77a and 77b contain a newly created Table 3.2-8, "Accident Monitoring Instrumentation".

On page 79 Table 4.2-2, "Minimum Test and Calibration Frequency for Core and Containment Cooling System," is revised to include RCIC Suction Source Levels. Item 13 is, therefore, changed to read "HPCI & RCIC Suction Source Levels".

On page 81 "3" is added to item 10.

New page 86a contains newly created Table 4.2-8, "Accident Monitoring Instrumentation".

On page 121, Section 4.5.E.1.a is revised to read "Simulated Automatic Actuation (and Restart) Test *".

On page 121, the following is added: "*Automatic restart on a low water level signal which is subsequent to a high water level trip."

On page 142a the note in the lower right corner concerning effective date is deleted. Page 142b is deleted.

On page 143, in Section 4.6.E, item 3 and the note in the lower right corner are deleted, item 4 is renumbered, and the following new item is added:

4. An annual report of safety/relief valve failures and challenges will be sent to the NRC as specified in Section 6.9.A.2.b.

Pages 143a and 143b are deleted.

On page 177 change "vacuum breakers shall be 0.5 psid" to "vacuum breakers shall be $\leq$ 0.5 psi external pressure."

On page 211 the last phrase in the remarks for penetrations X7A, B, C & D is revised to read, "Limit 11.5 SCFH at 25 psig)."

On page 211, the note at the bottom of the page concerning cycle 3 and the superscript referring to it are deleted.

On page 211, "Table 3.7-2" is changed to "Table 4.7-2". A portion of the description of penetration X-25 and all of the description of penetration X-26 are moved to a new page 211a.

On page 211a, the word "test" is added to remark for X-25 and two lines which are no longer applicable are deleted.

On page 211a, for penetration X-26 A/B, CAD and Purge system valve 27-MOV-113 is changed to 27-MOV-122.

On page 247, Section 6.0, "are" is changed to "is" in the sixth line.

The following item is added to Section 6.2 (page 248):

11. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

    Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

    a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

    b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Resident Manager or the Superintendent of Power, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Resident Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

In Section 6.3 on page 248a correct "ANSI NI8.1-1971" to "ANSI N18.1-1971".

In Section 6.4 on page 248a correct "ANSI NI8.1-1971" to "ANSI N18.1-1971".

In Section 6.5 on page 248a correct "seperate" to "separate".

On page 254-b, item 2 is reformatted to read as follows:

2. ANNUAL REPORTS

a. Annual Occupational Exposure Tabulation

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, [1]e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources, shall be assigned to specific major work functions.

b. Annual Report of S/RV Failures and Challenges

An annual report of safety/relief valve failures and challenges will be submitted prior to March 1 of each

year. An S/RV failure is defined to be any one of the following: (1) Failure of a valve to actuate when automatically or manually signaled to do so, for any reason, including testing, (2) Failure of a valve to close when the actuation signal is removed, (3) Spontaneous operation of a valve in the absence of an actuation signal, and (4) Determination that a valve is, or was inoperable in any operating mode except cold shutdown or refueling. An S/RV challenge is defined as an automatic or manual signaling of the valve to actuate for the purpose of controlling physical parameters of the primary coolant system. Test actuations of S/RV's are not considered challenges and will not be reported unless the test results in a failure.

The wording in paragraph 2a is not changed. The wording in paragraph 2b is new.

On page 260, "Asst. Shift Supervisor" is added to Fig. 6.2-1, and "Vice President Quality Assurance (NYO)" is changed to "Vice President Quality Assurance (WPO)."

The proposed changes on pages i through 44 are editorial in nature.

II. Purpose of the Proposed Changes

The proposed changes on page 49, 54, 60, 77a, 77b and 86a are necessary as a result of the following NUREG-0737 items.

II.B.3	Postaccident Sampling Capability
II.F.1.1	Noble Gas Effluent Monitor
II.F.1.3	Containment High-Range Radiation Monitor
II.F.1.4	Containment Pressure Monitor
II.F.1.5	Containment Water Level Monitor
II.F.1.6	Containment Hydrogen Monitor

The proposed changes on pages 70a and 79 are necessary to be consistent with NUREG-0737 Item II.K.3.22.

The proposed changes on pages 121 and 121a are intended to incorporate in Technical Specifications the requirements of NUREG-0737 Item II.K.3.13 and Reference 1 for automatic restart of the RCIC System on low-low reactor water level following trip of the system on high reactor water level.

The proposed deletion of item 3 of Section 4.6.E is necessary since the safety/relief valves have been modified and do not have a bellows arrangement.

The renumbering of item 4 is editorial in nature.

The proposed addition to page 143 is intended to incorporate in Technical Specifications the items discussed in NUREG-0737, Item II.K.3.3, and Reference 1 concerning safety relief valve challenges and/or failures.

The proposed change to the title of Table 4.7-2 and the deletion of note(a) and its superscript are editorial in nature.

The proposed change to the content of Table 4.7-2 incorporates in Technical Specifications the items discussed in NUREG-0737, Item II.E.4.1, and Reference 1 concerning the addition of containment isolation valves.

The proposed change to Section 6.0 is editorial in nature.

The proposed changes to Section 6.2 are intended to incorporate in Technical Specifications the items discussed in NUREG-0737, Item I.A.1.3.2, and Reference 1 concerning shift manning.

The proposed changes to Section 6.2 on page 248 relate to overtime policy. This policy has been in effect at the JAF NPP as per Plant Standing Order #26. It is now being incorporated into the Technical Specifications as required by Reference 1.

The proposed changes to Sections 6.3, 6.4 and 6.5 are editorial in nature.

III. Impact of the Proposed Changes

The additional instrumentation described in these proposed Technical Specification changes should improve safety at the FitzPatrick plant by providing additional information which is helpful to the operator in assessing plant conditions following an accident. The proposed changes related to overtime restrictions should improve safety by limiting overtime for safety related functions. The proposed changes related to additional reporting should improve safety by increasing reporting of safety and relief valve challenges. The editorial changes proposed will have no impact since their only purpose is to correct errors in the Technical Specifications as currently written.

The Commission has provided guidance concerning the application of the standards for making a "no significant hazard considerations" determination by providing certain examples in the Federal Register (F.R.) Vol. 48, No. 67 dated April 6, 1984, page 14870. The proposed changes all match at least one of these examples.

The addition proposed to page 143 and the change proposed on page 254b were identified by the NRC in Reference 1 as needed to be consistent with NUREG-0737 Item II.K.3.3, "Reporting Safety and Relief Valve Failures and Challenges." The proposed changes are, therefore, considered to match Commission example (i), "A purely administrative change to Technical Specifications; for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature."

The deletion of item 3 of Section 4.6.E on page 143 and the renumbering of the items in that section was necessary as a result of previous changes to achieve consistency throughout the Technical Specifications. The proposed changes are, therefore, considered to match Commission example (i).

The proposed changes on pages 142a, 142b, 143a, 143b, and the minor changes on page 143 are also editorial and considered to match commission example (i).

The proposed change on page 177 would make the wording consistent with Standard Technical Specifications, and is, therefore, considered to match example (i).

The proposed changes on pages i through 44, page 81, and on pages 211, 212, and 213 the proposed change to the title of Table 4.7-2 and the deletion of note(a) and its superscript can be classified as not likely to involve significant hazard considerations, since these changes are editorial in nature, and, therefore, fit Commission example (i).

On page 211, the change to penetrations X7A, B, C and D is typographical and matches example (i).

The changes to page 211a are typographical and match example (i).

The proposed change to the content of Table 4.7-2 was identified by the NRC in Reference 1 as a change needed to be consistent with NUREG-0737, Item II.E.4.1, "Dedicated Hydrogen Penetrations." This change is not expected to result in any major change to facility operations. Since it is similar, therefore, to Commission example (vii), "A change to make a license conform to regulations where the license change results in very minor changes to facility operations," it is determined to involve no significant hazard considerations.

The proposed change to Section 6.0 on page 247 can be classified as not likely to involve significant hazard considerations since it is editorial in nature. It, therefore, fits Commission example (i).

The proposed new Item 11 in Section 6.2 on page 248 was identified by the NRC in Reference 1 as a change needed to be consistent with NUREG-0737 Item I.A.1.3.1, Limit Overtime. The proposed change is considered to match Commission examples (vii) and (ii). Therefore, the proposed change is determined to involve no significant hazard considerations.

The proposed changes to Sections 6.3, 6.4 and 6.5 can be classified as not likely to involve significant hazard considerations, since they are editorial in nature, and, therefore, fit Commission example (i).

The changes proposed on pages 121 and 121a were identified by the NRC in Reference 1 as changes needed to be consistent with NUREG-0737 Item II.K.3.13, RCIC Restart. The proposed changes are considered to match the above example (vii) and, therefore, to involve no significant hazard considerations.

The proposed changes to page 70a and 79 identified by the NRC in Reference 1 as changes needed to be consistent with NUREG-0737 Item II.K.3.22, RCIC Suction. The proposed changes are considered to match the above example (vii) and, therefore, to involve no significant hazard considerations.

The proposed changes to pages 54, 60, 77a, 77b and 86a can be classified as not likely to involve significant hazard considerations, since they are changes to make a license conform to regulations where the license change results in very minor changes to facility operations, and, therefore, comply with example (vii).

The proposed change to Figure 6.2-1 on page 260 is needed to be consistent with NUREG-0737 Item I.A.1.3.2., "Shift Manning". The proposed change is considered to match Commission example (vii).

IV. Implementation of the Changes

Implementation of the changes, as proposed, will not impact the fire protection program at FitzPatrick, nor will the changes impact the environment.

V. Conclusion

The incorporation of these changes:

a) will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report;

b) will not increase the possibility of an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report;

c) will not reduce the margin of safety as defined in the basis for any Technical Specifications

d) does not constitute an unreviewed safety question, and

e) involves no Significant Hazard Considerations, as defined in 10.CFR.50.92.

VI. References

1. NRC Generic Letter No. 83-02, dated January 10, 1983.
2. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER).
3. NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.
4. James A. FitzPatrick Nuclear Power Plant Final Safety Analysis Report (FSAR), Rev. 1, July 1983.