Attachment I to JPN-93-068

PROPOSED OPERATING TECHNICAL SPECIFICATION CHANGES
MISCELLANEOUS ADMINISTRATIVE CHANGES

(JPTS-90-018)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

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3.2 BASES

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

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

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

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

The low water level instrumentation set to trip at 177 in. above the top of the active fuel closes all isolation valves except those in Group 1. Details of the isolation valve grouping are given in Section 7.3 of the updated FSAR. For valves which isolate at this level, this trip setting is adequate to prevent uncovering the core in the case of a break in the largest line.

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

3.2 BASES (cont'd)

initiates the HPCI and RCIC systems and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18 in. above the top of active fuel. This trip activates the remainder of the ECCS subsystems, closes the main steam isolation valves, main steam line drain valves and reactor water sample line isolation valves, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate ECCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 24 in. recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 of the updated FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating ECCS, it causes isolation of Groups B and C isolation valves. For the breaks discussed above, this instrumentation will generally initiate ECCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. Details of the isolation valve closure group are given in Section 7.3 of the updated FSAR. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperature peak at approximately 1,000°F and release of radioactivity to the environs is below 10 CFR 100 guidelines. Reference Section 14.6.5 of the updated FSAR.

Table 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action(2)
2 (6)	Reactor Low Water Level	≥ 177 in. above TAF	4	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2	D
2	Reactor Low-Low-Low Water Level	≥ 18 in. above the TAF	4	A
2 (6)	High Drywell Pressure	≤ 2.7 psig	4	A
2	High Radiation Main Steam Line Tunnel	≤ 3 x Normal Rated Full Power Background (9)	4	B
2	Low Pressure Main Steam Line	≥ 825 psig (7)	4	B
2	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4	B
2	Main Steam Line Leak Detection High Temperature	≤ 40°F above max ambient	4	B
4	Reactor Cleanup System Equipment Area High Temperature	≤ 40°F above max ambient	8	C
2	Low Condenser Vacuum Closes MSIV's	≥ 8" Hg. Vac (7)(8)	4	B

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3.3.C (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 (Seconds)
46	0.361
38	0.977
24	2.112
04	3.764

3. The maximum scram insertion time for 90 percent insertion of any operable control rod shall not exceed 7.00 sec.

4.3.C (cont'd)

2. At 16-week intervals, 10 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 in less that 30 seconds after receipt of a signal for control rods to scram, and

   b. Open when the scram signal is reset.

3.5 (cont'd)

2. Should one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining components of the containment cooling mode subsystems are operable.

3. Should one of the containment cooling subsystems become inoperable or should one RHRSW pump in each subsystem become inoperable, continued reactor operation is permissible for a period not to exceed 7 days.

4. If the requirements of 3.5.B.2 or 3.5.B.3 cannot be met, the reactor shall be placed in a cold condition within 24 hr.

5. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature $<212°F$ with an inoperable component(s) as specified in 3.5.B above.

4.5 (cont'd)

2. When it is determined that one RHRSW pump of the components required in 3.5.B.1 above is inoperable, the remaining components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.

3. When one containment cooling subsystem becomes inoperable, the redundant containment cooling subsystem shall be verified to be operable immediately and daily thereafter. When one RHRSW pump in each subsystem becomes inoperable, the remaining components of the containment cooling subsystems shall be verified to be operable immediately and daily thereafter.

3.6 (cont'd)

5. With the Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring or Floor Drain Sump Monitoring) inoperable, restore the system to operable status within 24 hours or be in at least hot shutdown within the next 12 hours and in the cold condition within the following 24 hours.

6. With the Primary Containment Atmosphere Radioactivity Monitoring System (gaseous) or the Primary Containment Atmosphere Radioactivity Monitoring System (particulate) inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours. Otherwise be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

4.6 (cont'd)

3. Drywell Continuous Atmosphere Radioactivity Monitoring System instrumentation shall be functionally tested and calibrated as specified in Table 4.6-2.

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

G. Jet Pumps

Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

F. Structural Integrity

1. Nondestructive inspections shall be performed on the ASME Boiler and Pressure Vessel Code Class 1, 2 and 3 components and supports in accordance with the requirements of the weld and support inservice inspection program. This inservice inspection program is based on an NRC approved edition of, and addenda to, Section XI of the ASME Boiler and Pressure Vessel Code which is in effect 12 months or less prior to the beginning of the inspection interval.

2. An augmented inservice inspection program is required for those high stressed circumferential piping joints in the main steam and feedwater lines larger than 4 inches in diameter, where no restraint against pipe whip is provided. The augmented in-service inspection program shall consist of 100 percent inspection of these welds per inspection interval.

3. An Inservice Inspection Program for piping identified in the NRC Generic Letter 88-01 shall be implemented in accordance with NRC staff positions on schedules, methods, personnel, and sample expansion included in this Generic Letter, or in accordance with alternate measures approved by the NRC staff.

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

3.6 and 4.6 BASES (cont'd)

than 100,000 lb/hr, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Fig. 4.6-1 are not exceeded. At steaming rates of at least 100,000 lb/hr, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal ranges. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This is not necessarily the case. Conductivity could be high due to the presence of a neutral salt; e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are, in fact, high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the condition include operation of the Reactor Cleanup System, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the Reactor Water Cleanup System to reestablish the purity of the reactor coolant.

During startup periods, which are in the category of less than 100,000 lb/hr, conductivity may exceed 2 μmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μmho/cm (other than short-term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses of the reactor coolant required by Specification 4.6.C.1 may be performed by a gamma scan.

D. Coolant Leakage

Allowable leakage rates of coolant from the Reactor Coolant System have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to make up Reactor Coolant System leakage in the event of loss of off-site a-c power. The normally expected background leakage due to equipment design and the detection capability for determining system

Table 4.6-2

Minimum Test and Calibration Frequency for Drywell Continuous Atmosphere Radioactivity Monitoring System

Inst. Channel	Inst. Functional Test	Calibration	Sensor Check
1. Air Particulate Analyzer	None	Once / 3 mos.	once / day
2. Gaseous Activity Analyzer	None	Once / 3 mos.	once / day
3. Iodine Analyzer	None	Once / 3 mos.	once / day

3.7 (cont'd)

2. With one or more of the containment isolation valves inoperable, maintain at least one isolation valve operable in each affected penetration that is open and:

   a. Restore the inoperable valve(s) to operable status within 4 hours; or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the closed position. Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control; or

   c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or a blind flange.

3. If Specifications 3.7.D.1 or 3.7.D.2 cannot be met the reactor shall be in the cold condition within 24 hrs.

4.7 (cont'd)

(2.) With the reactor at a reduced power level, fast close each main steam isolation valve, one at a time, and verify closure time.

d. At least twice per week, the main steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

e. The RBCLCWS isolation valves shall be fully closed and reopened any time the reactor is in the cold condition exceeding 48 hours, if the valves have not been fully closed and reopened during the preceding 92 days.

2. Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

3. Not Used

3.7 BASES (cont'd)

of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the Pressure Suppression System. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

The containment isolation valves on the containment vent and purge lines may be open for safety related reasons. Safety related reasons include, but are not limited to, the following: inerting or de-inerting primary containment; maintaining containment oxygen concentration; maintaining drywell and suppression pool atmospheric pressures; and maintaining the differential pressure between the drywell and suppression pool. These valves have been modified to limit the maximum angle of opening as shown in 3.7.D.1.

Nine remote manual isolation valves have been added to the Reactor Building Closed Loop Cooling Water System (RBCLCWS) in order to comply with 10 CFR 50 Appendix A GDC 57; These valves are air operated (with solenoid pilot valves), normally open, and are designed to fail "open" on loss of electrical power or "as is" upon loss of instrument air. Each AOV is provided with a Seismic Class I accumulator tank to allow operation of the valves upon loss of instrument air up to 2 full valve cycles. The fail-open design permits continued operation of the system to supply water to the recirculation pump-motor coolers and drywell coolers during normal operation and as necessary under accident conditions. If there is a postulated accident, and indications of leakage from RBCLCWS appear, the operator will selectively close the AOV's affected to provide containment isolation.

A list of containment isolation valves, including a brief description of each valve is included in Section 7.3 of the updated FSAR.

4.7 BASES (cont'd)

operability results in a more reliable system.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 in. restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

The RBCLCWS valves are excluded from the quarterly surveillance requirements because closure of these valves will eliminate the coolant flow to the drywell air and recirculation pump-motor coolers. Without cooling water, the drywell air and equipment temperature will increase and may cause damage to the equipment during normal plant operations. Therefore, testing of these valves would only be conducted in the cold condition.

A list of containment isolation valves, including a brief description of each valve is included in Section 7.3 of the updated FSAR.

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 are 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, one of the three General Managers will assume this responsibility. In the event all four 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.

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities that affect the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR.

2. The Resident Manager shall be responsible for overall plant operation, and shall have control over those onsite activities that are necessary for safe operation and maintenance of the plant.

3. The Executive Vice President - Nuclear Generation shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Plant Staff

The plant staff organization shall be as follows:

1. Each shift crew shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

Attachment II to JPN-93-068

PROPOSED RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATION CHANGES
MISCELLANEOUS ADMINISTRATIVE CHANGES

(JPTS-90-018)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
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NOTES FOR TABLE 3.2-1 (continued)

(d) Main stack gaseous sampling and analysis shall also be performed following shutdown, startup, or a thermal power change exceeding 20% of rated thermal power in one hour.

1. This requirement applies only if:

   - Analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3; and

   - The noble gas monitor shows that effluent activity has increased more than a factor of 3; and

   - Correction for increases due to changes in thermal power level have been made in both cases.

(e) Main stack iodine and particulate sampling shall also be performed daily following each shutdown, startup or thermal power change exceeding 20% of rated thermal power in one hour.

1. Daily sampling is not required for thermal power changes if the off gas charcoal filters are in service.

2. In addition, this requirement applies only if:

   - Analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3; and

   - The noble gas monitor shows that effluent activity has increased more than a factor of 3; and

   - Corrections for increases due to changes in thermal power level have been made in both cases.

3. Daily sampling shall be performed until two consecutive samples show no increase in concentration but not to exceed 7 consecutive days.

4. LLDs may be increased by a factor of 10 for analysis of daily samples.

5. Analysis of daily and weekly samples shall be completed within 48 hours of changing.

(f) Incinerated oil may be discharged via points other than the main stack and building vents (i.e., auxiliary boiler). Release shall be accounted for based on pre-release grab sample data.

(g) Samples of incinerated oil releases shall be collected from and representative of filtered oil in liquid form. Whenever oil samples cannot be filtered such as No. 6 bunker fuel oil, raw oil samples shall be collected and analyzed.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

treatment system under the following conditions:

1. The offgas dilution steam flow instrumentation shall alarm and automatically isolate the offgas recombiner system at a low flow setpoint greater than or equal to 6300 pounds per hour and at a high flow setpoint less than or equal to 7900 pounds per hour.

2. The offgas recombiner inlet temperature sensor shall alarm and automatically isolate the offgas recombiner system at a temperature setpoint of greater than or equal to 125°C.

3. The offgas recombiner outlet temperature sensor shall alarm and automatically isolate the offgas treatment system at a temperature setpoint of greater than or equal to 150°C.

c. In lieu of continuous hydrogen or oxygen monitoring, the condenser offgas treatment system recombiner effluent shall be analyzed to verify that it contains less than or equal to 4% hydrogen by volume.

d. With the requirements of the above specifications not satisfied, restore the recombiner system to within operating specifications or suspend use of the charcoal treatment system within 48 hours.

1. An instrument check shall be performed daily when the offgas treatment system is in operation.

2. An instrument channel functional test shall be performed once per operating cycle.

3. An instrument channel calibration shall be performed once per operating cycle.

c. With condenser offgas treatment system recombiner in service, in lieu of continuous hydrogen or oxygen monitoring, the hydrogen content shall be verified weekly to be less than or equal to 4 % by volume.

In the event that the hydrogen content cannot be verified, operation of this system may continue for up to 14 days.

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels	Trip Function	Trip Level Settings	Total Number of Instrument Channels Provided by Design	Actions
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)
1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)
1(a)	SJAE Radiation Monitors	$\leq$ 500,000 μCi/sec	2	(e)
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)
1(a)	Main Control Room Ventilation	$\leq 4 \times 10^3$ cpm[i]	1	(g)
(h)	Mechanical Vacuum Pump Isolation	≤ 3 x Normal Full Power Background	4	(h)

NOTES FOR TABLE 3.10-1

(a) Whenever the systems are required to be operable, there shall be one operable or tripped instrument channel per system. From and after the time it is found that this cannot be met, the indicated action shall be taken.

(b) Trip level setting is in accordance with the methods and procedures of the ODCM.

(c) Cease operation of the refueling equipment.

(d) Isolate secondary containment and start the SBGTS.

(e) Bring the SJAE release rate within the limit within 72 hours or be in hot standby within the next 12 hours.

(f) Refer to Appendix B LCO 3.1.d.

(g) Control room isolation is manually initiated.

(h) Uses same sensors as primary containment isolation on high main steam line radiation. Refer to Appendix A Table 3.2-1 for minimum number of operable instrument channels and action required.

(i) Conversion factor is 8.15×10^7 cpm - 1μCi/cc.

Amendment No. ~~93~~, ~~127~~,

7.0 ADMINISTRATIVE CONTROLS

7.1 RESPONSIBILITY

a. The Resident Manager shall have direct responsibility for assuring the operation of the James A. FitzPatrick Plant is conducted in such a manner as to provide continuing protection to the environment. During periods when the Resident Manager is unavailable, one of the three General Managers will assume this responsibility. In the event all four are unavailable, the Resident Manager may delegate this responsibility to other qualified supervisory personnel.

b. Implementation of the Radiological Effluent Technical Specifications is the responsibility of the General Manager - Operations, with the assistance of the plant staff organization.

7.2 PROCEDURES

Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 "Facility Administrative Policies and Procedures" of ANSI 18.7-1972 and Regulatory Guide 1.33, November 1972, Appendix A. In addition, procedures shall be established, implemented and maintained for the PCP, ODCM, and Quality Control Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1.

7.3 REPORTING REQUIREMENTS

a. Planned Liquid and Gaseous Releases

The limits for radioactive materials contained in liquid and gaseous effluents are contained in Specifications 2.3, 3.3 and 3.4.

b. Environmental Samples Exceeding Limits of Table 6.1-2

When the limits of Table 6.1-2 are exceeded, refer to Specification 6.1.b for reporting requirements.

c. Semiannual Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

1. The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit using as guidance Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof.

SAFETY EVALUATION FOR
PROPOSED TECHNICAL SPECIFICATION CHANGES
MISCELLANEOUS ADMINISTRATIVE CHANGES (JPTS-90-018)

I. DESCRIPTION OF THE PROPOSED CHANGES

This application for an amendment to the James A. FitzPatrick Technical Specifications proposes to make miscellaneous administrative changes to the Appendix A Operating Technical Specifications and Appendix B Radiological Effluent Technical Specifications (RETS). The proposed changes are addressed below.

Minor changes in format, such as type font, margins or hyphenation, are not described in this submittal. These changes are typographical in nature and do not affect the content of the Technical Specifications.

Appendix A Operating Technical Specifications

1. Page 55, Bases 3.2

   a. In the first paragraph, replace the phrase:

      "In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which"

      with the following:

      "Besides reactor protection instrumentation which initiates a reactor scram, additional protective instrumentation is also provided. This protective instrumentation."

   b. In the fifth paragraph, replace the sentence:

      "Details of valve grouping are given in the JAF FSAR section 7.3."

      with the sentence:

      "Details of the isolation valve grouping are given in Section 7.3 of the updated FSAR."

2. Page 56, Bases 3.2

   a. At the top of the left column, add the column identifier "3.2 BASES (cont'd)."

   b. In the first paragraph, insert "systems" after RCIC.

   c. In the first paragraph, replace the phrase "guidelines of 10CFR100 will" with the phrase "guidelines of 10 CFR 100 will."

d. In the first paragraph, replace the phrase "paragraph 6.5.3.1 FSAR" with the phrase "paragraph 6.5.3.1 of the updated FSAR."

e. In the second paragraph, replace the phrase "the JAF FSAR section 7.3" with the phrase "Section 7.3 of the updated FSAR."

f. In the third paragraph, replace the phrase "below 10CFR100 guidelines" with the phrase "below 10 CFR 100 guidelines."

g. In the third paragraph, replace the phrase "Section 14.6.5 FSAR" with the phrase "Section 14.6.5 of the updated FSAR."

3. <u>Page 64, Table 3.2-1</u>

In the fourth column, delete the phrase "Inst. Channels" for all occurrences.

4. <u>Page 96, Specification 4.3.C.3</u>

a. In Specification 4.3.C.3, replace "scam" with "scram."

b. In Specification 4.3.C.3.b, delete the phrase "or the scram discharge instrument volume trip is bypassed."

5. <u>Page 116, Specification 3.5.B.3</u>

Replace the phrase "two of the RHRSW pumps" with "one RHRSW pump in each subsystem."

6. <u>Page 116, Specification 4.5.B.3</u>

a. Delete the word "loop" at both locations.

b. Replace the phrase "two of the RHRSW pumps become" with "one RHRSW pump in each subsystem becomes."

c. Replace the phrase "subsystem(s) shall be demonstrated" with "subsystems shall be verified."

7. <u>Page 142, Specification 4.6.D.3</u>

Replace table identifier "4.6.2" with "4.6-2."

8. Page 144, Specification 4.6.F.3

Replace the word "acordance" with "accordance."

9. Page 150, Bases 3.6 and 4.6

In the third paragraph, delete the duplicate words "and the initial evolution of gases."

10. Page 162a, Table 4.6-2

In item 1, replace the table entry "Air Particle Analyzer" with "Air Particulate Analyzer."

11. Page 186, Specification 4.7.D.1.c.(2.)

Replace the phrase "reduced power level, trip main steam isolation valves" with "a reduced power level, fast close each main steam isolation valve, one at a time,."

12. Page 192, Bases 3.7

In the last paragraph, replace the phrase "the updated JAF FSAR section 7.3" with "Section 7.3 of the updated FSAR."

13. Page 197, Bases 4.7

In the last paragraph, replace the phrase "the updated JAF FSAR section 7.3" with "Section 7.3 of the updated FSAR."

14. Page 247, Specification 6.1

Replace the words "his responsibilities" with the words "this responsibility."

15. Page 247, Specification 6.2.1

a. In Specification 6.2.1.2, replace the word "resident" with "Resident."

b. In Specification 6.2.1.3, replace the phrase "of the stall in operating" with "of the staff in operating."

Appendix B Radiological Effluent Technical Specifications

1. Page 23, Notes for Table 3.2-1

   a. In item (f), replace the last sentence:

      "Whenever oil samples cannot be filtered such as No. 6 bunker fuel oil, raw oil samples shall be collected and analyzed"

      with:

      "Release shall be accounted for based on pre-release grab sample data."

   b. In item (g), replace the word "oils" with "oil."

2. Page 33, Limiting Condition for Operation 3.7.b

   a. For Limiting Condition for Operation (LCO) 3.7.b.1, replace the phrase:

      "low flow less than or equal to 6300 pounds per hour or high flow greater"

      with

      "a low flow setpoint greater than or equal to 6300 pounds per hour and at a high flow setpoint less."

   b. For LCO 3.7.b.2, replace the phrase "of not less than" with "setpoint of greater than or equal to."

   c. For LCO 3.7.b.3, insert the word "sensor" between the words "temperature shall."

   d. For LCO 3.7.b.3, replace the phrase "of not less than" with "setpoint of greater than or equal to."

3. Page 37, Table 3.10-1

   In item 6, replace the table entry "$\leq 4 \times 10^9$ cpm(i)" with the table entry "$\leq 4 \times 10^3$ cpm(i)."

4. Page 66, Specification 7.1

   a. In Specification 7.1.a, replace the phrase:

"he may delegate his responsibilities to the Superintendent of Power, or in his absence,"

with:

"one of the three General Managers will assume this responsibility. In the event all four are unavailable, the Resident Manager may delegate this responsibility."

b. In Specification 7.1.b, replace the phrase "the Superintendent of Power" with the phrase "the General Manager - Operations."

II. PURPOSE OF THE PROPOSED CHANGES

This application makes miscellaneous administrative changes including typographical and editorial corrections to the Appendix A Operating Technical Specifications and Appendix B RETS. The proposed changes will clarify and improve the quality of the Technical Specifications.

Appendix A Operating Technical Specifications

1. The proposed change (item 1.a) editorializes a sentence. This editorial change now makes the sentence grammatically correct. No change to the intent of the sentence has been made.

2. The proposed changes (items 1.b, 2.c, 2.d, 2.e, 2.f, 2.g, 12 and 13) to pages 55, 56, 192 and 197 revise the wording used in referencing either the updated Final Safety Analysis Report (FSAR) or the Code of Federal Regulations. These changes clarify these sentences and provide internal consistency within the Technical Specifications.

3. The proposed change (item 2.a) to page 56 adds the column identifier "3.2 BASES (cont'd)" to improve overall consistency in the Technical Specifications.

4. The proposed change (item 2.b) to page 56 adds the word "system" to improve overall clarity in the sentence.

5. The proposed change (item 3) to Table 3.2-1 deletes the phrase "Inst. Channels" from each entry to improve clarity within the table.

6. The proposed change (item 4.a) to page 96 corrects the spelling of "scram."

7. The proposed change (item 4.b) to page 96 deletes an erroneous surveillance requirement from Specification 4.3.C.3.b which required verification that the scram discharge instrument volume (SDIV) drain and vent valves open when the SDIV trip is bypassed.

   The surveillance test calls for the mode switch to be placed in the shutdown position which also de-energizes the manual scram circuit. In addition to

scramming the control rods it also de-energizes the scram discharge volume isolation valve solenoids to vent air pressure and close the SDIV drain and vent valves.

At the end of the surveillance test, the operator must clear all signals to reset the Reactor Protection System (RPS), including the SDIV high level signal if necessary. Once all signals are cleared, the operator can reset the scram which will reopen the drain and vent valves. It is the scram reset and not the SDIV high level trip bypass that causes the SDIV vent and drain valves to open.

This proposed change is consistent with Standard Technical Specification (STS) Surveillance Requirement 3.1.8.3 (Reference 1) for the BWR/4. The words "or the scram discharge instrument volume trip is bypassed" are found in the BWR/5 STS (Reference 2). The proposed change will, therefore, correct the Technical Specifications to reflect the approved plant design.

8. The proposed changes (items 5, 6a, and 6b) to page 116 are clarifications of the LCO and the associate Surveillance Requirement concerning RHRSW system/subsystem availability, contained in Specifications 3.5.B.3 and 4.5.B.3, respectively. By definition, if one subsystem is inoperable than both pumps of that subsystem are considered inoperable. The other situation in which two pumps may be inoperable would be if one pump from each subsystem were to be inoperable. These changes clarifies the second situation in which two pumps may be inoperable. These changes also deletes the word "loop" for consistency in usage concerning the containment cooling subsystems. The removal of the parentheses and making the word "subsystems" plural is consistent with the revised text concerning verification of both subsystems.

9. The proposed change (item 6.c) to page 116 replaces the term "demonstrate" with "verify" in Specification 4.5.B.3. This change corrects an oversight in which two Technical Specification revision submittals (References 3 and 4) were submitted simultaneously affecting this page. One submittal established the definitions and usage in the Technical Specifications for the words "demonstrate" and "verify" and was issued as Amendment 148 (Reference 5). Amendment 151 (Reference 6) was developed and submitted to the NRC prior to the issuance of Amendment 148 utilizing the then current definition/usage of the word "demonstrate." With the issuance of Amendment 148 establishing the acceptable usage of the words "demonstrate" and "verify," pending subsequent amendments should have been updated prior to NRC issuance. During the pre-issuance review of Amendment 151, the need for updating "demonstrate" to "verify" in the new Surveillance Requirement was overlooked.

It is clearly not the intent of the Surveillance Requirement to require a specific test as evidenced by the use of the word "verify" throughout Surveillance Requirements 4.5.A and 4.5.B as well as the observation in Reference 5 that component failure in the Core Spray System (Specification 4.5.A.2), LPCI subsystem (Specification 4.5.A.3.a), RHR pump or RHRSW pump (Specification 4.5.B.2) and Containment Cooling mode of the RHR system will require verification of the operability of redundant components and not a demonstration

of operability. This change corrects Specification 4.5.B.3 as intended in Amendment 148.

10. The proposed change (item 7) to page 142 makes an editorial correction to a Table 4.6-2 reference by replacing a "." with a "-" for consistency with the table numbering.

11. The proposed change (item 8) corrects the spelling of the word "accordance."

12. The proposed change (item 9) corrects a sentence by removing a typographical error introduced by Amendment 179 (Reference 7). The error inadvertently duplicated part of a preceding phrase. This change removes the duplication.

13. The proposed change (item 10) to Table 4.6-2 replaces the word "Particle" with the word "Particulate" to reflect proper reference to the Drywell Continuous Atmosphere Radioactivity Monitoring System.

14. The proposed change (item 11) to page 186 revises Specification 4.7.D.1.c.(2.) to reflect the original intent of the surveillance requirement and physical operations performed in surveillance testing (Reference 8). A literal interpretation of the current surveillance requirement would result in a reactor scram with each surveillance test.

15. The proposed change (item 14) to page 247 is grammatical in nature. No change to the intent of the sentence has been made.

16. The proposed changes (items 15a and 15b) to page 247 corrects the spelling of the words "Resident" and "staff."

Appendix B Radiological Effluent Technical Specifications

1. The proposed change (item 1.a) to page 23 restores the last sentence in Note (g) of Table 3.2-1 which was inadvertently replaced with the last sentence of Note (h) by Amendment 127 (Reference 9). This change restores the correct sentence from Amendment 93 (Reference 10).

2. The proposed change (item 1.b) to page 23 corrects a typographical error introduced by Amendment 127. This change to Note (g) of Table 3.2-1 restores the correct word from Amendment 93.

3. The proposed changes (items 2.a, b, c, and d) to page 33 are editorial in nature for the purpose of consistency and clarification. The changes make this specification consistent with other similar portions of the Technical Specifications. These changes revise the LCO to read as originally intended and as practiced by applying the Technical Specification limits on the instrumentation rather than the monitored parameter. The proposed changes do not alter the current instrument settings or parameter limits.

4. The proposed change (item 3) to page 37 corrects a typographical error in Table 3.10-1 introduced by Amendment 127. This change restores the correct number from Amendment 93.

5. The proposed changes (items 4.a and 4.b) to page 66 update the RETS to reflect changes made by Amendment 178 (Reference 11) to the Appendix A Operating Technical Specifications. Amendment 178 revised the Technical Specifications to reflect a management reorganization. The change to item 3.a replaces text associated with delegation of Resident Manager responsibility during absences. The change makes RETS Specification 7.1.a consistent with Technical Specification 6.1. The proposed change to item 3.b replaces the position title of Superintendent of Power with the position title of General Manager - Operations which were changed by Amendment 178.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGES

The proposed changes to the James A. FitzPatrick Technical Specifications will not affect plant safety or operations. The proposed changes make miscellaneous administrative changes that correct typographical and editorial errors. The proposed changes involve no limiting conditions for operation, surveillance requirements (the change to Surveillance Requirement 4.5.B.3 is only the correction of an oversight and does not change the intent of the requirement), setpoint or safety limit changes, nor do they affect the environmental monitoring program. The proposed changes do not change any system or subsystem and will not alter the conclusions of either the FSAR or the SER.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated.

   The intent of the proposed changes is to clarify and correct the Technical Specifications. The changes are administrative in nature and include: clarifying a specification to reflect system design; changing specifications for consistency with previous Amendments; revising a specification to accurately reflect surveillance testing, and; correction of typographical and editorial errors. There are no setpoint changes, safety limit changes, surveillance requirement changes, or limiting conditions for operation. These changes have no impact on plant safety or operations. The changes will have no impact on previously evaluated accidents.

2. create the possibility of a new or different kind of accident from those previously evaluated.

   The proposed changes are purely administrative in nature and involve only correcting typographical and editorial errors. These proposed changes are intended to clarify and improve the quality of the Technical Specifications. This cannot create the possibility of a new or different kind of accident.

3. involve a significant reduction in the margin of safety.

   The proposed changes correct errors which currently exist in the Technical Specifications. The changes are all administrative in nature and will clarify the Technical Specifications by eliminating errors such as typographical and editorial errors. These changes do not change any setpoint or safety limit changes regarding isolation or alarms. The proposed changes do not affect the environmental monitoring program. These changes do not affect the plants safety systems and do not reduce any safety margins.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes do not adversely affect the ALARA or Fire Protection Programs at the FitzPatrick plant, nor the environment. The proposed changes are administrative and by their nature can have no affect.

VI. CONCLUSION

The changes, as proposed, do not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, they:

1. 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;

2. will not create the possibility of an accident or malfunction of a type different from any previously evaluated in the Safety Analysis Report; and

3. will not reduce the margin of safety as defined in the basis for any technical specification.

The changes involve no significant hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. NUREG-1433, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/4)" Revision 0, dated September 1992.

2. NUREG-0123, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5)", Revision 3, dated Fall 1980.

3. NYPA letter, J.C. Brons to NRC, dated May 31, 1989, (JPN-89-034), "Proposed Changes to the Technical Specifications Regarding Low Pressure Coolant Injection Pump Flow Surveillance and Demonstrate/Verify Terminology (JPTS-80-011)."

4. NYPA letter, J.C. Brons to NRC, dated May 31, 1989, (JPN-89-037), "Proposed Change to the Technical Specifications Regarding Clarification of RHR Containment Cooling Mode Requirements (JPTS-84-006)."

5. NRC letter, D.E. LaBarge to J.C. Brons dated December 26, 1989, (JAF-90-002), Transmittal of Amendment 148 to the Technical Specifications.

6. NRC letter, D.E. LaBarge to J.C. Brons dated February 15, 1990, (JAF-90-054). Transmittal of Amendment 151 to the Technical Specifications.

7. NRC letter, B.C. McCabe to R.E. Beedle dated March 9, 1992, (JAF-92-069). Transmittal of Amendment 179 to the Technical Specifications.

8. James A. FitzPatrick Nuclear Power Plant, Operations Surveillance Test Procedure ST-1B, "Main Steam Isolation Valve (MSIV) Fast Closure," dated August 22, 1991, Revision 14.

9. NRC letter, D.E. LaBarge to J.C. Brons dated May 9, 1989, (JAF-89-213). Transmittal of Amendment 127 to the Technical Specifications.

10. NRC letter, H.I. Abelson to J.C. Brons dated May 29, 1985, (JAF-85-166). Transmittal of Amendment 93 to the Technical Specifications.

11. NRC letter, B.C. McCabe to R.E. Beedle dated March 9, 1992, (JAF-92-068). Transmittal of Amendment 178 to the Technical Specifications.

12. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report, Sections 4.10.3.4 "Leakage Detection System" and 7.2 "Reactor Protection System," and Figures 7.2-1 and 7.7-1, through Revision 5, dated January 1992.

13. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER), dated November 20, 1972, and Supplements.

Attachment IV to JPN-93-068

PROPOSED OPERATING TECHNICAL SPECIFICATION CHANGES
MISCELLANEOUS ADMINISTRATIVE CHANGES
MARKUP OF OPERATING TECHNICAL SPECIFICATION PAGES

(JPTS-90-018)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

3.2 BASES

Replace with Insert "A"

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

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

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

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

The low water level instrumentation set to trip at 177 in. above the top of the active fuel closes all isolation valves except those in Group 1. Details of valve grouping are given in the JAF FSAR section 7.3. For valves which isolate at this level, this trip setting is adequate to prevent uncovering the core in the case of a break in the largest line.

Details of the isolation valve grouping are given in Section 7.3 of the updated FSAR.

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

Amendment No. ~~103~~, ~~179~~, 1/3,

3.2 BASES (cont'd)

systems

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

of the updated

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating ECCS, it causes isolation of Groups B and C isolation valves. For the breaks discussed above, this instrumentation will generally initiate ECCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. Details of the isolation valve closure group are given in the JAF FSAR section 7.3. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents.

Section 7.3 of the updated FSAR

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

Amendment No. ~~38~~, ~~48~~, ~~58~~, ~~103~~, 1/9

TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (2)
2 (6)	Reactor Low Water Level	≥ 177 in. above TAF	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low-Low Water Level	≥ 18 in. above TAF	4 Inst. Channels	A
2 (6)	High Drywell Pressure	≤ 2.7 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	≤ 3 x Normal Rated Full Power Background (9)	4 Inst. Channels	B
2	Low Pressure Main Steam Line	≥ 825 psig (7)	4 Inst. Channels	B
2	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Leak Detection High Temperature	≤ 40°F above max ambient	4 Inst. Channels	B
4	Reactor Cleanup System Equipment Area High Temperature	≤ 40°F above max ambient	8 Inst. Channels	C
2	Low Condenser Vacuum Closes MSIV's	≥ 8" Hg. Vac (7)(8)	4 Inst. Channels	B

Amendment No. ~~15~~, ~~31~~, ~~46~~, ~~61~~, ~~90~~, ~~106~~, ~~119~~, ~~122~~, ~~185~~,

3.3.C (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 (Seconds)
46	0.361
38	0.977
24	2.112
04	3.764

3. The maximum scram insertion time for 90 percent insertion of any operable control rod shall not exceed 7.00 sec.

4.3.C (cont'd)

2. At 16-week intervals, 10 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.

scram

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 in less that 30 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.

3.5 (cont'd)

2. Should one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining components of the containment cooling mode subsystems are operable.

3. Should one of the containment cooling subsystems become inoperable or should two of the RHRSW pumps become inoperable, continued reactor operation is permissible for a period not to exceed 7 days.

4. If the requirements of 3.5.B.2 or 3.5.B.3 cannot be met, the reactor shall be placed in a cold condition within 24 hr.

5. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature < 212°F with an inoperable component(s) as specified in 3.5.B above.

4.5 (cont'd)

2. When it is determined that one RHRSW pump of the components required in 3.5.B.1 above is inoperable, the remaining components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.

3. When one containment cooling subsystem loop becomes inoperable, the redundant containment cooling subsystem loop shall be verified to be operable immediately and daily thereafter. When two of the RHRSW pumps becomes inoperable, the remaining components of the containment cooling subsystem(s) shall be demonstrated to be operable immediately and daily thereafter.

one RHRSW pump in each subsystem

verified

3.6 (cont'd)

5. With the Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring or Floor Drain Sump Monitoring) inoperable, restore the system to operable status within 24 hours or be in at least hot shutdown within the next 12 hours and in the cold condition within the following 24 hours.

6. With the Primary Containment Atmosphere Radioactivity Monitoring System (gaseous) or the Primary Containment Atmosphere Radioactivity Monitoring System (particulate) inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours. Otherwise be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

4.6 (cont'd)

3. Drywell Continuous Atmosphere Radioactivity Monitoring System instrumentation shall be functionally tested and calibrated as specified in Table 4.6-2.

Amendment No. ~~26~~, ~~56~~, ~~102~~, ~~192~~,

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

G. Jet Pumps

Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

F. Structural Integrity

1. Nondestructive inspections shall be performed on the ASME Boiler and Pressure Vessel Code Class 1, 2 and 3 components and supports in accordance with the requirements of the weld and support inservice inspection program. This inservice inspection program is based on an NRC approved edition of, and addenda to, Section XI of the ASME Boiler and Pressure Vessel Code which is in effect 12 months or less prior to the beginning of the inspection interval.

2. An augmented inservice inspection program is required for those high stressed circumferential piping joints in the main steam and feedwater lines larger than 4 inches in diameter, where no restraint against pipe whip is provided. The augmented in-service inspection program shall consist of 100 percent inspection of these welds per inspection interval.

3. An Inservice Inspection Program for piping identified in the NRC Generic Letter 88-01 shall be implemented in accordance with NRC staff positions on schedules, methods, personnel, and sample expansion included in this Generic Letter, or in acordance with alternate measures approved by the NRC staff.

accordance

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

3.6 and 4.6 BASES (cont'd)

than 100,000 lb/hr, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Fig. 4.6-1 are not exceeded. At steaming rates of at least 100,000 lb/hr, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal ranges. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This is not necessarily the case. Conductivity could be high due to the presence of a neutral salt; e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are, in fact, high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the condition include operation of the Reactor Cleanup System, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the Reactor Water Cleanup System to reestablish the purity of the reactor coolant.

During startup periods, which are in the category of less than 100,000 lb/hr, conductivity may exceed 2 μmho/cm because of the initial evolution of gases and the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μmho/cm (other than short-term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses of the reactor coolant required by Specification 4.6.C.1 may be performed by a gamma scan.

D. Coolant Leakage

Allowable leakage rates of coolant from the Reactor Coolant System have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to make up Reactor Coolant System leakage in the event of loss of off-site a-c power. The normally expected background leakage due to equipment design and the detection capability for determining system

Table 4.6-2

Minimum Test and Calibration Frequency for Drywell Continuous Atmosphere Radioactivity Monitoring System

Inst. Channel	Inst. Functional Test	Calibration	Sensor Check
1. Air Particle Analyzer Particulate	None	Once/3 mos.	once/day
2. Gaseous Activity Analyzer	None	Once/3 mos.	once/day
3. Iodine Analyzer	None	Once/3 mos.	once/day

3.7 (cont'd)

2. With one or more of the containment isolation valves inoperable, maintain at least one isolation valve operable in each affected penetration that is open and:

   a. Restore the inoperable valve(s) to operable status within 4 hours; or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the closed position. Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control; or

   c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or a blind flange.

3. If Specifications 3.7.D.1 or 3.7.D.2 cannot be met the reactor shall be in the cold condition within 24 hrs.

4.7 (cont'd)

Replace with Insert "B"

(2.) With the reactor at reduced power level, trip main steam isolation valves and verify closure time.

d. At least twice per week, the main steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

e. The RBCLCWS isolation valves shall be fully closed and reopened any time the reactor is in the cold condition exceeding 48 hours, if the valves have not been fully closed and reopened during the preceding 92 days.

2. Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

3. Not Used

Amendment No. 134, 154, 173, 192,

3.7 BASES (cont'd)

of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the Pressure Suppression System. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

The containment isolation valves on the containment vent and purge lines may be open for safety related reasons. Safety related reasons include, but are not limited to, the following: inerting or de-inerting primary containment; maintaining containment oxygen concentration; maintaining drywell and suppression pool atmospheric pressures; and maintaining the differential pressure between the drywell and suppression pool. These valves have been modified to limit the maximum angle of opening as shown in 3.7.D.1.

Nine remote manual isolation valves have been added to the Reactor Building Closed Loop Cooling Water System (RBCLCWS) in order to comply with 10 CFR 50 Appendix A GDC 57. These valves are air operated (with solenoid pilot valves), normally open, and are designed to fail "open"on loss of electrical power or "as is" upon loss of instrument air. Each AOV is provided with a Seismic Class I accumulator tank to allow operation of the valves upon loss of instrument air up to 2 full valve cycles. The fail open design permits continued operation of the system to supply water to the recirculation pump motor coolers and drywell coolers during normal operation and as necessary under accident conditions. If there is a postulated accident, and indications of leakage from RBCLCWS appear, the operator will selectively close the AOV's affected to provide containment isolation.

A list of containment isolation valves, including a brief description of each valve is included in the updated (FA) FSAR Section 7.3

Section 7.3 of

4.7 BASES (cont'd)

operability results in a more reliable system.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 in. restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

The RBCLCWS valves are excluded from the quarterly surveillance requirements because closure of these valves will eliminate the coolant flow to the drywell air and recirculation pump motor coolers. Without cooling water, the drywell air and equipment temperature will increase and may cause damage to the equipment during normal plant operations. Therefore, testing of these valves would only be conducted in the cold condition.

A list of containment isolation valves, including a brief description of each valve is included in the updated JAF FSAR Section 7.3.

Section 7.3 of

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 are 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, one of the three General Managers will assume his responsibilities. In the event all four 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.

this responsibility

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities that affect the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR.
2. The Resident Manager shall be responsible for overall plant operation, and shall have control over those onsite activities that are necessary for safe operation and maintenance of the plant.

   staff
3. The Executive Vice President - Nuclear Generation shall take any measures needed to ensure acceptable performance of the stall in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.
4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Plant Staff

The plant staff organization shall be as follows:

1. Each shift crew shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

INSERT "A"

Besidus reactor protection instrumentation which initiates a reactor scram, additional protective instrumentation is also provided. This protective instrumentation

INSERT "B"

a reduced power level, fast close each main ste ., isolation valve, one at a time,

(d) Main stack gaseous sampling and analysis shall also be performed following shutdown, startup, or a thermal power change exceeding 20% of rated thermal power in one hour.

1. This requirement applies only if:

   - Analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3; and
   - The noble gas monitor shows that effluent activity has increased more than a factor of 3; and
   - Corrections for increases due to changes in thermal power level have been made in both cases.

(e) Main stack iodine and particulate sampling shall also be performed daily following each shutdown, startup or thermal power change exceeding 20% of rated thermal power in one hour.

1. Daily sampling is not required for thermal power changes if the off gas charcoal filters are in service.

2. In addition, this requirement applies only if:

   - Analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3; and
   - The noble gas monitor shows that effluent activity has increased more than a factor of 3; and
   - Corrections for increases due to changes in thermal power level have been made in both cases.

Replace with Insert "C"

3. Daily sampling shall be performed until two consecutive samples show no increase in concentration but not to exceed 7 consecutive days.

4. LLDs may be increased by a factor of 10 for analysis of daily samples.

5. Analysis of daily and weekly samples shall be completed within 48 hours of changing.

(f) Incinerated oil may be discharged via points other than the main stack and building vents (i.e., auxiliary boiler). Whenever oil samples cannot be filtered such as No. 6 bunker fuel oil, raw oil samples shall be collected and analyzed.

(g) Samples of incinerated oil releases shall be collected from and representative of filtered oil in liquid form. Whenever oil samples cannot be filtered such as No. 6 bunker fuel oil, raw oils samples shall be collected and analyzed.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

treatment system under the following conditions:

1. The offgas dilution steam flow instrumentation shall alarm and automatically isolate the offgas recombiner system at low flow less than or equal to 6300 pounds per hour or high flow greater than or equal to 7900 pounds per hour.
2. The offgas recombiner inlet temperature sensor shall alarm and automatically isolate the offgas recombiner system at a temperature of not less than 125°C.
3. The offgas recombiner outlet temperature sensor shall alarm and automatically isolate the offgas treatment system at a temperature of not less than 150°C.

setpoint of greater than or equal to

c. In lieu of continuous hydrogen or oxygen monitoring, the condenser offgas treatment system recombiner effluent shall be analyzed to verify that it contains less than or equal to 4% hydrogen by volume.

d. With the requirements of the above specifications not satisfied, restore the recombiner system to within operating specifications or suspend use of the charcoal treatment system within 48 hours.

1. An instrument check shall be performed daily when the offgas treatment system is in operation.
2. An instrument channel functional test shall be performed once per operating cycle.
3. An instrument channel calibration shall be performed once per operating cycle.

a low flow setpoint greater than or equal to 6300 pounds per hour and at a high flow setpoint less

c. With condenser offgas treatment system recombiner in service, in lieu of continuous hydrogen or oxygen monitoring, the hydrogen content shall be verified weekly to be less than or equal to 4 % by volume.

In the event that the hydrogen content cannot be verified, operation of this system may continue for up to 14 days.

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TABL .10 1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design	Action
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)
1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)
1(a)	SJAE Radiation Monitors	$\leq$500,000 µCi/sec	2	(e)
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)
1(a)	Main Control Room Ventilation	$\leq 4 \times 10^{9}$ cpm(i) 3	1	(g)
(h)	Mechanical Vacuum Pump Isolation	$\leq$3 x Normal Full Power Background	4	(h)

NOTES FOR TABLE 3.10-1

(a) Whenever the systems are required to be operable, there shall be one operable or tripped instrument channel per system. From and after the time it is found that this cannot be met, the indicated action shall be taken.

(b) Trip level setting is in accordance with the methods and procedures of the ODCM.

(c) Cease operation of the refueling equipment.

(d) Isolate secondary containment and start the SBGTS.

(e) Bring the SJAE release rate within the limit within 72 hours or be in hot standby within the next 12 hours.

(f) Refer to Appendix B LCO 3.1.d.

(g) Control room isolation is manually initiated.

(h) Uses same sensors as primary containment isolation on high main steam line radiation. Refer to Appendix A Table 3.2-1 for minimum number of operable instrument channels and action required.

(i) Conversion factor is 8.15×10^{7} cpm - 1 µCi/cc.

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7.0 ADMINISTRATIVE CONTROLS

7.1 RESPONSIBILITY

a. The Resident Manager shall have direct responsibility for assuring the operation of the James A. FitzPatrick Plant is conducted in such a manner as to provide continuing protection to the environment. During periods when the Resident Manager is unavailable, he may delegate his responsibilities to the Superintendent of Power, or in his absence, to other qualified supervisory personnel.

Replace with Insert "D"

b. Implementation of the Radiological Effluent Technical Specifications is the responsibility of the Superintendent of Power, with the assistance of the plant staff organization.

General Manager - Operations

7.2 PROCEDURES

Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 "Facility Administrative Policies and Procedures" of ANSI 18.7-1972 and Regulatory Guide 1.33, November 1972, Appendix A. In addition, procedures shall be established, implemented and maintained for the PCP, ODCM, and Quality Control Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1.

7.3 REPORTING REQUIREMENTS

a. Planned Liquid and Gaseous Releases

The limits for radioactive materials contained in liquid and gaseous effluents are contained in Specifications 2.3, 3.3 and 3.4.

b. Environmental Samples Exceeding Limits of Table 6.1-2

When the limits of Table 6.1-2 are exceeded, refer to Specification 6.1.b for reporting requirements.

c. Semiannual Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

1. The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit using as guidance Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof.

INSERT "C"

Release shall be accounted for based on pre-release grab sample data.

INSERT "D"

one of the three General Managers will assume this responsibility. In the event all four are unavailable, the Resident Manager may delegate this responsibility