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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 181
License No. DPR-59

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated May 30, 1990, as supplemented April 18, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 181 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 14, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 181

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

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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. (Deleted)

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.

I. 4kv Emergency Bus Undervoltage Trip

The limiting conditions for operation for the instrumentation that prevents damage to electrical equipment or circuits as a result of either a degraded or loss-of-voltage condition on the emergancy electrical buses are given in Table 3.2-2.

4.2 (cont'd)

E. Drywell Leak Detection

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

F. (Deleted)

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.

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3.2 BASES (cont'd)

the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Flow integrators are used to record the integrated flow of liquid from the drywell sumps. The leak rate is calculated by dividing the integrated volume pumped out of the sumps by the time between sump pump operations. The resultant leak rate value, which is expressed in gallons per minute, is compared to the acceptance criterion specified in Specification 3.6.D.

For each parameter monitored, as listed in Table 3.2-8, by comparing the reading of each channel to the reading on redundant or related instrument channel a near continuous surveillance of instrument performance is available.

3.2 BASES (cont'd)

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 barrier 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 conforms with the acceptance criteria of NUREG-0737, NUREG-0578, and NRC Generic Letter 83-36 and includes Regulatory Guide 1.97, Revision 2 Type A variables.

The Emergency Bus Undervoltage Trip System transfers the 4 kv emergency electrical buses to the Emergency Diesel Generators in the event an undervoltage condition is detected. The system has two levels of protection: (1) degraded voltage protection, and (2) loss-of-voltage protection. Degraded voltage protection prevents a sustained low voltage condition from damaging safety-related equipment. The degraded voltage protection has two time delays. A short time delay coincident with a loss-of-coolant accident (LOCA) and a longer time delay to allow normal plant evolutions without unnecessarily starting the Emergency Diesel Generators. The loss-of-voltage protection prevents a more severe voltage drop from causing a long term interruption of power. Time delays are included in the system to prevent inadvertent transfers due to spurious voltage decreases. Therefore, both the duration and severity of the voltage drop are sensed by the Emergency Bus Undervoltage Trip System.

4.2 BASES

The instrumentation listed in Tables 4.2-1 through 4.2-8 will be functionally tested and calibrated at regularly scheduled intervals. The same design reliability goal as the Reactor Protection System is generally applied. Sensors, trip devices and power supplies are tested, calibrated and checked at the same frequency as comparable devices in the Reactor Protection System.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1 out of n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = \sqrt{\frac{2t}{r}}$$

Where:

i = the optimum interval between tests.

t = the time the trip contacts are disabled from performing their function while the test is in progress.

r = the expected failure rate of the relays.

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 min. to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hr. Using this data and the above operation, the optimum test interval is:

$$i = \sqrt{\frac{2(0.5)}{10^{-6}}} = 1x\,10^{3}\ hr.$$

$$= 40 \text{ days}$$

For additional margin a test interval of once/month will be used initially.

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

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TABLE 3.2-8

ACCIDENT MONITORING INSTRUMENTATION

	Instrument	No. of Channels Provided by Design	Minimum No. of Operable Channels Required	Mode in Which Instrument Must be Operable	Action
1.	Stack High Range Effluent Monitor (17RM-53A) (17RM-53B)	2	1	Note H	Note B
2.	Turbine Building Vent High Range Effluent Monitor (17RM-434A) (17RM-434B)	2	1	Note H	Note B
3.	Radwaste Building Vent High Range Effluent Monitor (17RM-463A) (17RM-463B)	2	1	Note H	Note B
4.	Containment High Range Radiation Monitor* (27RM-104A) (27RM-104B)	2	1	Note H	Note A
5.	Drywell Pressure (narrow range) (27PI-115A1 or 27PR-115A1) (27PI-115B1 or 27PR-115B1)	2	1	Note J	Note A
6.	Drywell Pressure (wide range) (27PI-115A2 or 27PR-115A2) (27PI-115B2 or 27PR-115B2)	2	1	Note J	Note A
7.	Drywell Temperature (16-1TR-107) (16-1TR-108)	2	1	Note J	Note A

* At less than or equal to 450 R/hr, closes vent and purge valves

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TABLE 3.2-8 (cont'd)

ACCIDENT MONITORING INSTRUMENTATION

	Instrument	No. of Channels Provided by Design	Minimum No. of Operable Channels Required	Mode in Which Instrument Must be Operable	Action
8.	Torus Water Level (wide range) (23LI-202A or 23LR-202A/203A) (23LI-202B or 23LR-202B/203B)	2	1	Note J	Note A
9.	Torus Bulk Water Temperature (16-1TI-131A or 16-1TR-131A) (16-1TI-131B or 16-1TR-131B)	2	1	Note J	Note A
10.	Torus Pressure (27PR-101A) (27PR-101B1)	2	1	Note J	Note A
11.	Drywell Hydrogen/Oxygen Concentration (27PCR-101A) (27PCR-101B)	2	1	Note J	Note F
12.	Reactor Vessel Pressure (06PI-61A or 06PR-61A) (06PI-61B or 06PR-61B)	2	1	Note J	Note A
13.	Reactor Water Level (fuel zone) (02-3LI-091) (02-3LR-098)	2	1	Note J	Note A
14.	Reactor Water Level (wide range) (02-3LI-85A) (02-3LR-85B)	2	1	Note J	Note A

TABLE 3.2-8 (cont'd)

ACCIDENT MONITORING INSTRUMENTATION

	Instrument	No. of Channels Provided by Design	Minimum No. of Operable Channels Required	Mode in Which Instrument Must be Operable	Action
15.	Core Spray Flow loop A (14FI-50A) loop B (14FI-50B)	1 per loop	1 per loop	Note J	Note A
16.	Core Spray discharge pressure loop A (14PI-48A) loop B (14PI-48B)	1 per loop	1 per loop	Note J	Note A
17.	LPCI (RHR) Flow loop A (10FI-133A) (10FR-143 - red pen) loop B (10FI-133B) (10FR-143 - blue pen)	2 per loop	1 per loop	Note J	Note A
18.	RHR Service Water Flow loop A (10FI-132A) loop B (10FI-132B)	1 per loop	1 per loop	Note J	Note A
19.	Safety/Relief Valve Position Indicator (See Note C)	2	1	Note J	Notes D, E
20.	Torus Water Level (narrow range) (23LI-201A or 27R-101 - red pen or EPIC A-1258) (EPIC A-1260) (See Note G)	2		Note J	Note B
21.	Drywell-Torus Differential Pressure (16-1DPR-200 or EPIC A-3554) (EPIC A-3551) (See Note G)	2	1	Note J	Note B

TABLE 3.2-8 (Cont´d)

ACCIDENT MONITORING INSTRUMENTATION

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 be in a cold condition within the next 24 hours.

B. With the number of OPERABLE channels less than required by the minimum channels OPERABLE requirements, initiate an alternate method of monitoring the appropriate parameter(s) within 72 hours and: (1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or (2) prepare and submit a Special Report to the Commission within 14 days following the event outlining the cause of the inoperability, the action taken, and the plans and schedule for restoring the system to OPERABLE status.

C. Each Safety/Relief Valve is equipped with two acoustical detectors, one of which is in service. Each SRV also has a backup thermocouple detector. In the event that a thermocouple is inoperable, SRV performance shall be monitored daily with the associated in service acoustical detector.

D. From and after the date that both of the acoustical detectors are inop[illegible]ble, continued operation is permissible until the next outage in which a primary containment entry is made provided that the thermocouple is operable. Both acoustical detectors shall be made operable prior to restart.

E. In the event that both primary (acoustical detectors) and secondary (thermocouple) indicatio[illegible]s parameter for any one valve are disabled and neither indication can be restored in forty-eight (48) hours, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in twelve (12) hours and in a Cold Shutdown within the next twenty-four (24) hours.

F. Refer to Specification 3.7.A.9.

G. This parameter and associated instrumentation are not part of post-accident monitoring.

H. This instrument shall be operable in the Run, Startup/Hot Standby, and Hot Shutdown modes.

J. This instrument shall be operable in the Run and Startup/Hot Standby modes.

TABLE 4.2-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

Instrument Channel (8)	Instrument Functional Test	Calibration Frequency	Instrument Check (4)
1) Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2) Reactor Low-Low-Low Water Level	(1)(5)	(15)	Once/day
3) Main Steam High Temp.	(1)(5)	(15)	Once/day
4) Main Steam High Flow	(1)(5)	(15)	Once/day
5) Main Steam Low Pressure	(1)(5)	(15)	Once/day
6) Reactor Water Cleanup High Temp.	(1)	Once/3 months	None
7) Condenser Low Vacuum	(1)(5)	(15)	Once/day

Logic System Functional Test (7) (9)	Frequency
1) Main Steam Line Isolation valves Main Steam Line Drain Valves Reactor Water Sample Valves	Once/6 months
2) RHR - Isolation Valve Control Shutdown Cooling Valves Head Spray	Once/6 months
3) Reactor Water Cleanup Isolation	Once/6 months
4) Drywell Isolation Valves TIP Withdrawal Atmospheric Control Valves	Once/6 months
5) Standby Gas Treatment System Reactor Building Isolation	Once/6 months

NOTE: See notes following Table 4.2-5

TABLE 4.2-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

	Instrument Channel (8)	Instrument Functional Test	Calibration Frequency	Instrument Check(4)
1)	Reactor Water Level	(1)(5)	(15)	Once/day
2a)	Drywell Pressure (non-ATTS)	(1)	Once/3 months	None
2b)	Drywell Pressure (ATTS)	(1)(5)	(15)	Once/day
3a)	Reactor Pressure (non-ATTS)	(1)	Once/3 months	None
3b)	Reactor Pressure (ATTS)	(1)(5)	(15)	Once/day
4)	Auto Sequencing Timers	None	Once/operating cycle	None
5)	ADS - LPCI or CS Pump Disch.	(1)	Once/3 months	None
6)	Trip System Bus Power Monitors	(1)	None	None
8)	Core Spray Sparger d/p	(1)	Once/3 months	Once/day
9)	Steam Line High Flow (HPCI & RCIC)	(1)(5)	(15)	Once/day
10)	Steam Line/Area High Temp. (HPCI & RCIC)	(1)(5)	(15)	Once/day
12)	HPCI & RCIC Steam Line Low Pressure	(1)(5)	(15)	Once/day
13)	HPCI & RCIC Suction Source Levels	(1)	Once/3 months	None
14)	4kV Emergency Bus Under-Voltage (Loss-of-Voltage, Degraded Voltage LOCA and non-LOCA) 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	None	None

NOTE: See notes following Table 4.2-5.

TABLE 4.2-2 (Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

	Logic System Functional Test		Frequency
1)	Core Spray Subsystem	(7) (9)	Once/6 months
2)	Low Pressure Coolant Injection Subsystem	(7) (9)	Once/6 months
3)	Containment Cooling Subsystem	(9)	Once/6 months
4)	HPCI Subsystem	(7) (9)	Once/6 months
5)	HPCI Subsystem Auto Isolation	(7)	Once/6 months
6)	ADS Subsystem	(7) (9)	Once/6 months
7)	RCIC Subsystem Auto Isolation	(7)	Once/6 months
8)	ADS Relief Valve Bellow Pressure Switch	(7) (9)	Once/operating cycle

NOTE: See notes following Table 4.2-5.

TABLE 4.2-3

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

	Instrument Channel	Instrument Functional Test (5)	Calibration	Instrument Check (4)
1)	APRM - Downscale	(1)	Once/3 months	Once/day
2)	APRM - Upscale	(1)	Once/3 months	Once/day
3)	IRM - Upscale	(2)	(3) (6)	Once/day
4)	IRM - Downscale	(2)	(3) (6)	Once/day
5)	RBM - Upscale	(1)	Once/3 months	Once/day
6)	RBM - Downscale	(1)	Once/3 months	Once/day
7)	SRM - Upscale	(2)	(3) (6)	Once/day
8)	SRM - Detector Not in Startup Position	(2)	(3) (6)	None
9)	IRM - Detector Not in Startup Position	(2)	(3) (6)	None
10)	Scram Discharge Instrument Volume - High Water Level (Group B Instruments)	Once/month (1)	Once/3 months	Once/day

	Logic System Function Test (7) (9)	Frequency
1)	System Logic Check	Once/6 months

NOTE: See notes following Table 4.2-5.

TABLE 4.2-5

MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check (4)
1) Equipment Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
2) Floor Drain Sump Flow Integrator	(1)	Once/3 months	Once/day

NOTE: See notes following Table 4.2-5

NOTES FOR TABLES 4.2-1 THROUGH 4.2-5

1. Initially once every month until acceptance failure rate data are available; thereafter a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instruments operate in a environment similar to that of JAFNPP.

2. Functional tests are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed within seven (7) days prior to each startup.

3. Calibrations are not required when these instruments are not required to be operable or are tripped. Calibration tests shall be performed within seven (7) days prior to each startup or prior to a pre-planned shutdown.

4. Instrument checks are not required when these instruments are not required to be operable or are tripped.

5. This instrumentation is exempt from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

6. These instrument channels will be calibrated using simulated electrical signals once every three months.

7. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

8. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2-1 since they are tested on Table 4.1-2.

9. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

10. At least one (1) Main Stack Dilution Fan is required to be in operation in order to isokinetically sample the Main Stack.

11. Uses same instrumentation as Main Steam Line High Radiation. See Table 4.1-2.

12. (Deleted)

13. Calibration and instrument check surveillance for SRM and IRM Instruments are as specified in Tables 4.1-1, 4.1-2, 4.2-3.

14. Functional test is performed once each operating cycle.

15. Sensor calibration once per operating cycle. Master/slave trip unit calibration once per 6 months.

TABLE 4.2-7

MINIMUM TEST AND CALIBRATION FREQUENCY FOR ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	TRIP UNIT CALIBRATION	CHANNEL CALIBRATION	SIMULATED AUTO ACTUATION & LOGIC FUNCTIONAL TEST
1 - Reactor Pressure-High	Once/day	Once/31 days	Once/6 months	Once/Operating cycle	Once/Operating cycle
2 - Reactor Water Level-Low Low	Once/day	Once/31 days	Once/6 months	Once/Operating cycle	Once/Operating cycle

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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.	Drywell Pressure (narrow range)	N/A	Once/Operating Cycle	Once/day
6.	Drywell Pressure (wide range)	N/A	Once/Operating Cycle	Once/day
7.	Drywell Temperature	N/A	Once/Operating Cycle	Once/day
8.	Torus Water Level (wide range)	N/A	Once/Operating Cycle	Once/day
9.	Torus Bulk Water Temperature	N/A	Once/Operating Cycle	Once/day
10.	Torus Pressure	N/A	Once/Operating Cycle	Once/day
11.	Drywell Hydrogen/Oxygen Concentration Analyzer	N/A	Once/3 months	Once/day
12.	Reactor Vessel Pressure	N/A	Once/Operating Cycle	Once/day
13.	Reactor Water Level (fuel zone)	N/A	Once/Operating Cycle	Once/day
14.	Reactor Water Level (wide range)	N/A	Once/Operating Cycle	Once/day

TABLE 4.2-8 (cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR ACCIDENT MONITORING INSTRUMENTATION

	Instrument	Instrument Functional Test	Calibration Frequency	Instrument Check
15.	Core Spray Flow	N/A	Once/Operating Cycle	Once/day
16.	Core Spray Discharge Pressure	N/A	Once/Operating Cycle	Once/day
17.	LPCI (RHR) Flow	N/A	Once/Operating Cycle	Once/day
18.	RHR Service Water Flow	N/A	Once/Operating Cycle	Once/day
19.	Safety/Relief Valve Position Indicator (Primary and Secondary)	Once/Operating Cycle	N/A	Once/month
20.	Torus Water Level (narrow range)	N/A	Once/Operating Cycle	Once/day
21.	Drywell-Torus Differential Pressure	N/A	Once/Operating Cycle	Once/day

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3.3.A.2 (cont'd)

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

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

d. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

e. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5 X 5 array may be inoperable (at least 4 operable control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a cold condition within 24 hours.

4.3.A.2 (cont'd)

e. The scram discharge volume drain and vent valves shall be full-travel cycled at least once per quarter to verify that the valves close in less than 30 seconds and to assure proper valve stroke and operation.

f. An instrument check of control rod position indication shall be performed once/day.

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The volume and temperature of the water in the torus shall be maintained within the following limits whenever the reactor is critical or whenever the reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel:

   a. Maximum vent submergence level of 53 inches.

   b. Minimum vent submergence level of 51.5 inches.

   The torus water level may be outside the above limits for a maximum of four (4) hours during required operability testing of HPCI, RCIC, RHR, CS, and the Drywell-Torus Vacuum System.

   c. Maximum water temperature

      (1) During normal power operation maximum water temperature shall be 95°F.

4.7 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary, and secondary containment systems.

Specification:

A. Primary Containment

1. The torus water level and temperature shall be monitored as specified in Table 4.2-8. The accessible interior surfaces of the drywell and above the water line of the torus shall be inspected at each refueling outage for evidence of deterioration. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the torus shall be conducted before resuming power operation.

3.7 (cont'd)

6. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than four percent oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.6.b.

b. Within the 24 hr. period subsequent to placing the reactor in the run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4 percent by weight and maintained in this condition. De-inerting may commence 24 hr. prior to a shutdown.

7. Drywell-Torus Differential Pressure

a. Differential pressure between the drywell and torus shall be maintained at equal to or greater than 1.7 psid except as specified in (1) and (2) below:

4.7 (cont'd)

6. Oxygen Concentration

a. The primary containment oxygen concentration shall be monitored as specified in Table 4.2-8.

7. Drywell-Torus Differential Pressure

a. The pressure differential between the drywell and torus shall be monitored as specified in Table 4.2-8.

3.7 (cont'd)

9. Primary containment atmosphere shall be continuously monitored for hydrogen and oxygen when containment integrity is required. The exception to this is when the Post-Accident Sampling System is to be operated. In this instance, the containment atmosphere monitoring systems may be isolated for a period not to exceed 3 hours in a 24-hour period. The monitoring system shall be considered operable if at least one monitor is operable.

   a) From and after the time the primary containment atmosphere monitoring instruments are found or made to be inoperable for any reason, continued reactor operation is permissible for the succeeding thirty (30) days unless one instrument monitoring each parameter is sooner made operable, provided an appropriate grab sample is obtained and analyzed at least once each twenty-four (24) hour period.

   b) If specification 3.7.A.9.a cannot be met, the reactor shall be placed in the cold condition within twenty-four (24) hours.

B. Standby Gas Treatment System

1. Except as specified 7.B.2 below both circuits of the Standby Gas Treatment System shall be operable at all times when secondary containment integrity is required.

4.7 (cont'd)

9. Primary Containment Atmosphere Monitoring Instruments

   a. Instrumentation shall be functionally tested and calibrated as specified in Table 4.2-8.

B. Standby Gas Treatment System

1. Standby Gas Treatment System surveillance shall be performed as indicated below:

   a. At least once per operating cycle, it shall be demonstrated that:

      (1) Pressure drop across the combined high-efficiency and charcoal filters is less than 5.7 in. of water at 6,000 scfm, and

      (2) Each 39kW heater shall dissipate greater than 29kW of electric power as calculated by the following expression:

      $P = \sqrt{3}EI$

      where:

      P = Dissipated Electrical Power;
      E = Measured line-to-line voltage in volts (RMS);
      I = Average measured phase current in amperes (RMS).

3.7 BASES (cont'd)

Using the minimum or maximum downcomer submergence levels given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the design of 56 psig. The minimum downcomer submergence of 51.5 in. results in a minimum suppression chamber water volume of 105,600 ft.^{3}. The majority of the Bodega tests (9) were run with a submerged length of 4 ft. and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. Additional JAFNPP specific analyses done in connection with the Mark I Containment-Suppression Chamber Integrity Program indicate the adequacy of the specified range of submergence to ensure that dynamic forces associated with pool swell do not result in overstress of the suppression chamber or associated structures. Level instrumentation is provided for operator use to maintain downcomer submergence within the specified range.

The maximum temperature at the end of blowdown tested during the Humboldt Bay (10) and Bodega Bay tests was 170°F, and this is conservatively taken to be the limit for complete condensation of the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Using a 40°F rise (Section 5.2 FSAR) in the suppression chamber water temperature and a maximum initial temperature of 95°F, a temperature of 145°F is achieved, which is well below the 170°F temperature which is used for complete condensation.

For an initial maximum suppression chamber water temperature of 95°F and assuming the normal complement of containment cooling pumps (two LPCI pumps and two RHR service water pumps) containment pressure is not required to maintain adequate net positive suction head (HPSH) for the core spray LPCI and HPCI pumps.

Limiting suppression pool temperature to 130°F during RCIC, HPCI, or relief valve operation, when decay heat and stored energy are removed form the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for a potential blowdown any time during RCIC, HPCI, or relief valve operation.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

3.7 BASES (cont'd)

are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24 hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration.

Drywell to Torus Vacuum Breakers

The capacity of the five drywell to torus vacuum relief valves are sized to limit the pressure differential between the torus and drywell during post-accident drywell cooling operations to well under the design limit of 2 psi. They are sized on the basis of the Bodega Bay pressure suppression system test. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows a 2 psi differential. With one vacuum relief valve secured in the closed position and four operable valves, containment integrity is not impaired.

The drywell to torus bypass leakage is limited to 71 scfm to provide assurance that steam released to the drywell will flow to and be condensed in the torus. The maximum allowable bypass leakage is determined to be that which would limit the maximum containment pressure rise to the design value of 56 psig and is a function of the break size, reactor pressure decay, and time till containment sprays are actuated. The allowable test leakage, 71 scfm, is approximately 10 times less than the maximum allowable bypass capacity. A test leakage of 71 scfm corresponds to a pressure transient of 0.25 in. water/min. over a 10 min period with the drywell/torus equal to 1 psid.

A drywell-torus minimum differential pressure of 1.7 psid has been established as being adequate to ensure that appropriate torus and torus support system safety margins are maintained following postulated design basis accidents. This differential lowers the water level in the torus to drywell vent downcomers thereby reducing dynamic forces as a result of a LOCA. Instrumentation is provided for operator use to maintain drywell-torus differential pressure.

B. Standby Gas Treatment System and

C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides

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