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

June 29, 1993

Docket No. 50-333

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M86248)

The Commission has issued the enclosed Amendment No.190 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 9, 1993.

The amendment involves editorial changes, corrects typographical errors, and adjusts line spacing and text formats. In addition, the amendment deletes pertinent portions of the TSs that related to exceptions that are no longer applicable. The amendment does not make any substantive changes to the TSs..

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

John E. Menning

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.190 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

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DATED: June 29, 1993

AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
J. Menning
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
Wanda Jones, P-370
C. Grimes, 11/F/23
ACRS (10)
OPA
OC/LFMB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 190
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 March 9, 1993, 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. 190, 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: June 29, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 190

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
i	i
v	v
8	8
20	20
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TECHNICAL SPECIFICATIONS

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1.1 (cont'd)

B. Core Thermal Power Limit (Reactor Pressure ≤785 psig)

When the reactor pressure is ≤785 psig or core flow is less than or equal to 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 (cont'd)

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

APRM - The APRM flux scram setting shall be ≤15 percent of rated neutron flux with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

c. APRM Flux Scram Trip Settings (Run Mode)

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be less than or equal to the limit specified in Table 3.1-1. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.

For no combination of recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

2.1 BASES (Cont'd)

B. Not Used

C. <u>References</u>

1. (Deleted)

2. "General Electric Standard Application for Reactor Fuel", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

3. (Deleted)

4. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO-24281, August, 1980.

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575°F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575° for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% x 1,250 = 1,375 psig) and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% x 1,150 = 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The current reload analysis shows that the main steam isolation valve closure transient, with flux scram, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. (See current reload analysis for the curve produced by this analysis.) Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

The numerical distribution of safety/relief valve setpoints shown in 2.2.1.B (2 @ 1090 psi, 2 @ 1105 psi, 7 @ 1140 psi) is justified by analyses described in the General Electric report NEDO-24129-1, Supplement 1, and assures that the structural acceptance criteria set forth in the Mark I Containment Short Term Program are satisfied.

3.1 BASES

A. The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the Reactor Coolant System.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The Reactor Protection System is of the dual channel type (Reference subsection 7.2 FSAR). The System is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE-279 (1971) for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the average power range monitor (APRM) channel the intermediate range monitor (IRM) channels, the scram discharge volume, the main steam isolation valve closure and the turbine stop valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

Three APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other

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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 emergency 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.

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	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.

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TABLE 4.2-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

	Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check (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.

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3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes of satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types. These values are specified in the Core Operating Limits Report. If at anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been maintained in a filled condition; the discharge piping of the affected subsystem shall be vented from the high point of the system and water flow observed.

3. Whenever the HPCI or RCIC System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI or RCIC shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested each month.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the Reactor Coolant System.

Objective:

To assure the integrity and safe operation of the Reactor Coolant System.

Specification:

A. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

   The reactor vessel head bolting studs shall not be under tension unless the temperatures of the reactor vessel flange and the reactor head flange are at least 90°F.

2. In-Service Hydrostatic and Leak Tests

   During in-service hydrostatic or leak testing the Reactor Coolant System pressure and temperature shall be on or to the right of curve A shown in Figure 3.6-1 Part 1, 2, or 3 and the maximum temperature change during any one hour period shall be:

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the Reactor Coolant System.

Objective:

To determine the condition of the Reactor Coolant System and the operation of the safety devices related to it.

Specification:

A. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

   When in the cold condition, the reactor vessel head flange and the reactor vessel flange temperatures shall be recorded:

   a. Every 12 hours when the reactor vessel head flange is $\leq$120°F and the studs are tensioned.

   b. Every 30 minutes when the reactor vessel head flange is $\leq$100°F and the studs are tensioned.

   c. Within 30 minutes prior to and every 30 minutes during tensioning of reactor vessel head bolting studs.

2. In-Service Hydrostatic and Leak Tests

   During hydrostatic and leak testing the Reactor Coolant System pressure and temperature shall be recorded every 30 minutes until two consecutive temperature readings are within 5°F of each other.

3.6 (cont'd)

B. Deleted

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed the equilibrium value of 3.1 μCi/gm of dose equivalent I-131. This limit may be exceeded, following a power transient, for a maximum of 48 hours. During this iodine activity transient the iodine concentrations shall not exceed the equilibrium limits by more than a factor of 10 whenever the main steamline isolation valves are open. The reactor shall not be operated more than 5 percent of its annual power operation under this exception to the equilibrium limits. If the iodine concentration exceeds the equilibrium limit by more than a factor of 10, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

7. Reactor Vessel Flux Monitoring

The reactor vessel Flux Monitoring Surveillance Program complies with the intent of the May, 1983 revision to 10 CFR 50, Appendices G and H. The next flux monitoring surveillance capsule shall be removed after 15 effective full power years (EFPYs) and the test procedures and reporting requirements shall meet the requirements of ASTM E 185-82.

B. Deleted

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for gross gamma activity.

   b. Isotopic analysis of a sample of reactor coolant shall be made at least once/month.

   c. A sample of reactor coolant shall be taken prior to startup and at 4 hour intervals during startup and analyzed for gross gamma activity.

   d. During plant steady state operation and following an offgas activity increase (at the Steam Jet Air Ejectors) of 10,000 μCi/sec within a 48 hour period or a power level change of ≥20 percent of full rated power/hr reactor coolant samples shall be taken and analyzed for gross gamma activity. At least three samples will be taken at 4 hour intervals. These sampling requirements may be omitted whenever the equilibrium I-131 concentration in the reactor coolant is less than 0.007 μCi/ml.

4.6 (cont'd)

e. If the gross activity counts made in accordance with a, c, and d above indicate a total iodine concentration in excess of 0.007 μCi/ml, a quantative determination shall be made for I-131 and I-133.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 lb/hr except as specified in 3.6.C.3:

   Conductivity 2 μmho/cm
   Chloride ion 0.1 ppm

2. During startups and at steaming rates below 100,000 lb/hr, and when the conductivity of the reactor coolant exceeds 2 μmhos/cm, a sample of reactor coolant shall be taken every 4 hr and analyzed for conductivity and chloride content.

3. For reactor startups the maximum value for conductivity shall not exceed 10 μmho/cm and the maximum value for chloride ion concentration shall not exceed 0.1ppm, for the first 24 hours after placing the reactor in the power operating condition. During reactor shutdowns, specification 3.6.C.4 will apply.

3. a. With steaming rates greater than or equal to 100,000 lb/hr, a reactor coolant sample shall be taken at least every 96 hours and whenever the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

   b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

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.

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)

B. Deleted

C. Coolant Chemistry

A radioactivity concentration limit of 20 μCi/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Radiological Effluent Technical Specification Section 3.2.a if there is a failure or a prolonged shutdown of the cleanup demineralizer.

In the event of a steam line rupture outside the drywell, with this coolant activity level, the resultant radiological dose at the site boundary would be 33 rem to the thyroid, under adverse meteorological conditions assuming no more than 3.1 μCi/gm of dose equivalent I-131. The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hours. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant. Also during reactor startups and large power changes which could affect iodine levels, samples of reactor coolant shall be analyzed to insure iodine concentrations are below allowable levels. Analysis is required whenever the I-131 concentration is within a factor of 100 of its allowable equilibrium value. The necessity for continued sampling following power and offgas transients will be reviewed within 2 years of initial plant startup.

The surveillance requirements 4.6.C.1 may be satisfied by a continuous monitoring system capable of determining the total iodine concentration in the coolant on a real time basis, and annunciating at appropriate concentration levels such that sampling for isotopic analysis can be initiated. The design details of such a system must be submitted for evaluation and accepted by the Commission prior to its implementation and incorporation in these Technical Specifications.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

Materials in the Reactor Coolant System are primarily 304 stainless steel and Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. The attached graph, Fig. 4.6-1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup, and hot standby. During these periods with steaming rates less

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

3.6 and 4.6 BASES (cont'd)

leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC-sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.D, on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.6.D, the experimental and analytical data suggest a reasonable margin of safety such that leakage of this magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the Plant should be shut down to allow further investigation and corrective action.

The capacity of the drywell sump pumps is 100 gpm, and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

The performance of the Reactor Coolant Leakage Detection System will be evaluated during the first 5 years of plant operation, and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detectors are capable of detecting a leak on the order of 3,500 lb/hr. The system performance will be evaluated during the first 5 years of plant operation, and the conclusions of the evaluation will be reported to the NRC.

The reactor coolant leakage detection systems consist of the drywell sump monitoring system and the drywell continuous atmosphere monitoring system. The drywell continuous atmosphere monitoring system utilizes a three-channel monitor to provide information on particulate, iodine and noble gas activities in the drywell atmosphere. Two independent and redundant systems are provided to perform this function. This system supplements the drywell sump monitoring system in detecting abnormal leakage that could occur from the reactor coolant system. In the event that the drywell continuous atmosphere monitoring system is inoperable, grab sample will be taken on a periodic basis to monitor drywell activity.

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.

4.7 (cont'd)

Type A test shall be performed at each plant shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria.

b. Type B tests (Local leak rate testing of containment penetrations)

(1.) All preoperational and periodic Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa, and the gas flow to maintain Pa shall be measured.

(2.) Acceptance criteria

The combined leakage rate of all penetrations and valves subject to Type B and C tests shall be less than 0.60 La, with the exception of the valves sealed with fluid from a seal system.

4.7 (cont'd)

(5) Type C test.

Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.

(6) Other leak rate tests specified in Section 4.7d shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.

f. Containment modification

Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in the test report. The acceptance criteria as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

3.7 BASES

A. Primary Containment

The integrity of the primary containment and operation of the Emergency Core Cooling Systems in combination limit the offsite doses to values less than those specified in 10 CFR 100 in the event of a break in the Reactor Coolant System piping. Thus, containment integrity is required whenever the potential for violation of the Reactor Coolant System integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception to the requirement to maintain primary containment integrity is allowed during core loading and during low power physics testing when ready access to the reactor vessel is required. There will be no pressure on the system at this time, which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period, however, restrictive operating procedures and operation of the RWM in accordance with Specification 3.3.B.3 minimize the probability of an accident occurring. Procedures in conjunction with the Rod Worth Minimizer Technical Specifications limit individual control worth such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm. In the unlikely event that an excursion did occur, the reactor building and Standby Gas Treatment System, which shall be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the Reactor Coolant System energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1,020 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 56 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (updated FSAR Section 5.2).

3.7 BASES (cont'd)

complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The Standby Gas Treatment System is designed to filter and exhaust the reactor building atmosphere to the main stack during secondary containment isolation conditions with a minimum release of radioactive materials from the reactor building to the environs. Both standby gas treatment fans are designed to automatically start upon containment isolation; however, only one fan is required to maintain the reactor building pressure at approximately a negative 1/4 in. water gage pressure; all leakage should be in-leakage. Each of the two fans has 100 percent capacity. If one Standby Gas Treatment System circuit is inoperable, the other circuit must be verified operable daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable, the Plant is brought to a condition where the system is not required.

While only a small amount of particulates is released from the Pressure Suppression Chamber System as a result of the loss-of-coolant accident, high-efficiency particulate filters are specified to minimize potential particulate release to the environment and to prevent clogging of the iodine filter. The high-efficiency filters have an efficiency greater than 99 percent for particulate matter larger than 0.3 micron. The minimum iodine removal efficiency is 99 percent. Filter banks will be replaced whenever significant changes in filter efficiency occur. Tests (11) of impregnated charcoal identical to that used in the filters indicated that shelf life up to 5 yr leads to only minor decreases in methyl iodine removal efficiency.

The 99 percent efficiency of the charcoal and particulate filters is sufficient to prevent exceeding 10CFR100 guidelines for the accidents analyzed. The analysis of the loss-of-coolant accident assumed a charcoal filter efficiency of 90 percent, and TID 14844 fission product source term. Hence, requiring 99 percent efficiency for both the charcoal and particulate filters provides adequate margin. A heater maintains relative humidity below 70 percent in order to assure the efficient removal of methyl iodine on the impregnated charcoal filters.

The operability of the Standby Gas Treatment System (SGTS) must be assured if a design basis loss of coolant accident (LOCA) occurs while the containment is being purged or vented through the SGTS. Flow from containment to the SGTS is via 6 inch Valve Number 27MOV-121. Since the maximum flow through the 6 inch line following a design basis LOCA is within the design capabilities of the SGTS, use of the 6 inch line assures the operability of the SGTS.

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space

3.9 (cont'd)

3. From and after the time that one of the Emergency Diesel Generator Systems is made or found to be inoperable, continued reactor operation is permissible for a period not to exceed 7 days provided that the two incoming power sources are available and that the remaining Diesel Generator System is operable. At the end of the 7 day period, the reactor shall be placed in a cold condition within 24 hours, unless the affected diesel generator system is made operable sooner.

4. When both Emergency Diesel Generator Systems are made or found to be inoperable, a reactor shutdown shall be initiated within two hours and the reactor placed in a cold condition within 24 hours after initiation of shutdown.

4.9 (cont'd)

3. The emergency diesel generator system instrumentation shall be checked during the monthly generator test.

4. Once each operating cycle, the conditions under which the Emergency Diesel Generator System is required will be simulated to demonstrate that the pair of diesel generators will start, accelerate, force parallel, and accept the emergency loads in the prescribed sequence.

5. Once within one hour and at least once per twenty-four hours thereafter while the reactor is being operated in accordance with Specifications 3.9.B.1, 3.9.B.2, or 3.9.B.3 the availability of the operable Emergency Diesel Generators shall be demonstrated by manual starting and force paralleling where applicable.

3.9 (cont'd)

F. LPCI MOV Independent Power Supplies

1. Reactor shall not be made critical unless both independent power supplies, including the batteries, inverters and chargers and their associated buses (MCC-155 and MCC-165) are in service, except as specified below.

2. During power operation, if one independent power supply becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days unless the unavailable train is made operable sooner. From and after the date one of the independent power supplies is made or found to be inoperable for any reason, the following would apply:

   a. The other independent power supply including its charger, inverter, battery and associated bus is operable.

   b. Pilot cell voltage, specific gravity and temperature and overall battery voltage are measured immediately and weekly thereafter for the operable independent power supply battery.

   c. The inoperable independent power supply shall be isolated from its associated LPCI MOV bus, and this bus will be manually switched to its alternate power source.

3.9 BASES (cont'd)

C. Diesel Fuel

Minimum on-site fuel oil requirements are based on operation of the emergency diesel generator systems at rated load for 7 days.

Additional diesel fuel can be delivered to the site within 48 hours.

If one of the Emergency Diesel Generator Systems is not operable, the plant shall be permitted to run for 7 days provided both sources of reserve power are operational. This is based on the following:

1. The operable Emergency Diesel Generator System is capable of carrying sufficient engineered safeguards and emergency core cooling system equipment to cover all loss-of-coolant accidents.

2. The reserve (offsite) power is highly reliable.

D. Not Used

E. Battery System

125 v DC power is supplied from two plant batteries each sized to supply the required equipment at design power following a loss-of-coolant accident with a concurrent loss of normal and reserve power. Each battery is provided with a charger sized to maintain the battery in a fully charged state while supplying normal operating loads.

F. LPCI MOV Independent Power Supplies

There are two LPCI MOV Independent Power Supplies each consisting of a charger, rectifier, inverter and battery. Each independent power supply charger-rectifier is normally fed from the emergency A-C power supply system to maintain the battery in a fully charged state. In the event of a LOCA each independent power supply is automatically isolated from the Emergency A-C power system. The battery and inverter have sufficient capacity to power the MOV's essential to the operation of the LPCI System. An alternate power source is provided for each LPCI MOV bus whereby in the event its independent power supply is out of service, the LPCI MOV bus may be energized directly from the Emergency A-C Power System.

3.9 BASES (cont'd)

G. Reactor Protection System Power Supplies

Each of two RPS divisions may be supplied power from it's respective RPS MG set or from an alternate source which derives power from the same electrical division. The MG sets and alternate sources for both divisions are provided with redundant, seismic qualified, class 1E electrical protection assemblies between the power source and the RPS bus. Any abnormal output type failure in either of the MG sets or alternate sources (if in service) would result in a trip of one or both of the electrical protection assemblies producing a half scram on that RPS division and retaining full scram capability in the other RPS division.

Limiting operating conditions in Section 3.9.G provide a high degree of assurance that RPS buses are protected as described above.

4.9 BASES (con't)

D. Not Used

E. Battery System

Measurements and electrical tests are conducted at specified intervals to provide indication of cell condition and to determine the discharge capability of the batteries. Performance and service tests are conducted in accordance with the recommendations of IEEE 450-1987.

F. LPCI MOV Independent Power Supply

Measurement and electrical tests are conducted at specified intervals to provide indication of cell condition, to determine the discharge capability of the battery. Performance and service tests are conducted in accordance with the recommendations of IEEE 450-1987.

G. Reactor Protection Power Supplies

Functional tests of the electrical protection assemblies are conducted at specified intervals utilizing a built-in test device and once per operating cycle by performing an instrument calibration which verifies operation within the limits of Section 4.9.G.

A. High Pressure Water Fire Protection System (Cont'd)

3. If 1. above cannot be fulfilled, place the reactor in Hot Standby within six (6) hours and in Cold Shutdown within the following thirty (30) hours.

A. High Pressure Water Fire Protection System (Cont'd)

	Item	Frequency
h.	Fire pump diesel engine by verifying the fuel storage tank contains at least 172 gallons of fuel.	Once/Month
i.	Diesel fuel from each tank obtained in accordance with ASTM-D270-65 is within the acceptable limits for quality as per the following:	Once/Quarter
	Flash Point - °F	125°F min.
	Pour Point - °F	10°F max.
	Water & Sediment	0.05% max.
	Ash	0.01% max.
	Distillation 90% Point	540 min.
	Viscosity (SSU) @ 100°F	40 max.
	Sulfur	1% max.
	Copper Strip Corrosion	No. 3 max.
	Cetane #	35 min.
j.	Fire pump diesel engine by inspection during shut down in accordance with procedures prepared in conjunction with manufacturers recommendations and verifying the diesel, starts from ambient conditions on the auto start signal and operates for ≥20 minutes while loaded with the fire pump.	Once/18 months

2. If the fire protection systems smoke and/or heat detectors in Tables 3.12.1 and 3.12.2 cannot be restored to an operable status within 14 days, a written report to the Commission outlining the action taken, the cause of inoperability and plans and schedule for restoring the detectors to an operable status shall be prepared and submitted within 30 days.

F. Fire Barrier Penetration Seals

1. All fire barrier penetrations, including cable penetration barriers, fire doors and fire dampers, in fire zone boundaries protecting safety related areas shall be functional.

2. With one or more of the required fire barrier penetrations non-functional, within one hour establish a continuous fire watch on at least one side of the affected penetration or verify the operability of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.A, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

F. Fire Barrier Penetration Seals

1. All fire barrier penetration seals for each protected area shall be visually inspected once/1.5 years to verify functional integrity. For those fire barrier-penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

2. Any repair of fire barrier penetration seals shall be followed by a visual inspection.

3.12 and 4.12 BASES

The Fire Protection System specifications provide pre-established minimum levels of operability to assure adequate fire protection during any operating condition including a design basis accident or safe shutdown earthquake.

A. The high pressure water fire protection system is supplied by redundant vertical turbine pumps, one diesel driven and one electric motor driven, each design rated 2500 gpm at 125 psig discharge pressure. Both pumps take suction from the plant intake cooling water structures from Lake Ontario. The high pressure water fire protection header is normally maintained at greater than 115 psig by a pressure maintenance subsystem. If pressure decreases, the fire pumps are automatically started by their initiation logic to maintain the fire protection system header pressure. Each pump, together with its manual and automatic initiation logic combined makes up a redundant high pressure water fire pump.

   A third fire pump, diesel-driven, has been installed and is set to automatically actuate upon decreasing pressure after the actuation of the first two fire pumps. No credit is taken for this pump in any analyses and the requirements of Technical Specifications 3.12 and 4.12 do not apply.

   Pressure Maintenance subsystem checks, valve position checks, system flushes and comprehensive pump and system flow and/or performance tests including logic and starting subsystem tests provide for the early detection and correction of component failures thus ensuring high levels of operability.

B. Safety related equipment areas protected by water spray or sprinklers are listed in Table 3.12.1. Whenever any of the protected areas, spray or sprinklers are inoperable continuous fire detection and backup fire protection equipment is available in the area where the water spray and/or sprinkler protection was lost.

   Performance of the tests and inspections listed in Table 4.12.1 will prevent and detect nozzle blockage or breakage and verify header integrity to ensure operability.

C. The carbon dioxide systems provide total flood protection for eight different safety related areas of the plant from either a 3 ton or 10 ton storage unit as indicated in Table 3.12.2. Both CO_2 storage units are equipped with mechanical refrigeration units to maintain the storage tank content at 0°F with a resultant pressure of 300 psig. Automatic smoke and heat detectors are provided in the CO_2 protected areas and initiation is automatic and/or manual as indicated in Table 3.12.2. For any area in which the CO_2 protection is made or found to be inoperable, continuous fire detection is available and one or more large wheeled CO_2 fire extinguisher is also available for each area in which protection was lost.

   Weekly checks of storage tank pressure and level verify proper operation of the tank refrigeration units and availability of sufficient volume of CO_2 to extinguish a fire in any of the protected areas.

5.5.B Bases

The spent fuel pool and high density fuel storage racks are Class I structures designed to store up to 2,797 fuel bundles. The storage racks are designed to maintain a subcritical configuration having a multiplication factor (k_{eff}) less than 0.95 for all possible operational and abnormal conditions. The nuclear criticality analyses for the Spent Fuel Racks (References 1 and 3) conclude that fresh fuel bundles with 3.3 w/o U-235 meet the 0.95 k_{eff} limit. This design basis bundle was reanalyzed to determine its infinite lattice multiplication factor, k_{∞}, when in a reactor core geometry (Reference 2). This k_{∞} was obtained under conservative calculational assumptions and reduced by 2.33 times the standard deviation in the calculation resulting in the Technical Specification limit of 1.36.

References:

1) Increased Spent Fuel Storage Modification, Stone & Webster Engineering Corporation, Boston, Mass. March 15, 1978.

2) General Electric letter, P. Van Dieman to G. Rorke, FitzPatrick Fuel Storage K-infinity Conversion, Revision 1, dated July 10, 1986.

3) Increased Storage Capacity for FitzPatrick Spent Fuel Pool, Holtec International, Mount Laurel, New Jersey, February, 1989.

2. An SRO or an SRO with a license limited to fuel handling shall directly supervise all Core Alterations. This person shall have no other duties during this time;

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

4. In the event of illness or unexpected absence, up to two (2) hours is allowed to restore the shift crew or fire brigade to the minimum complement.

5. The Operations Manager, Assistant Operations Manager, Shift Supervisor and Assistant Shift Supervisor shall hold a SRO license and the Senior Nuclear Operator and the Nuclear Control Operator shall hold a RO license or an SRO license.

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

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating.

   However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major modifications, on a temporary basis, the following guidelines shall be followed:

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

   b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

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

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

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

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in FSAR Figure 13.2-7 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions; except for the Radiological and Environmental Services Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 The Shift Technical Advisor (STA) shall meet or exceed the minimum requirements of either Option 1 (Combined SRO/STA Position) or Option 2 (Continued use of STA Position), as defined in the Commission Policy Statement on Engineering Expertise on Shift, published in the October 28, 1985 Federal Register (50 FR 43621). When invoking Option 1, the STA role may be filled by the Shift Supervisor or Assistant Shift Supervisor. (1)

6.3.3 Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

NOTE:

(1) The 13 individuals who hold SRO licenses, and have completed the FitzPatrick Advanced Technical Training Program prior to the issuance of License Amendment 111, shall be considered qualified as dual-role SRO/STAs.

6.4 RETRAINING AND REPLACEMENT TRAINING

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

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55.59. In addition, fire brigade training shall meet or exceed the requirements of NFPA 27-1975, except for Fire Brigade training sessions which shall be held at least quarterly. The effective date for implementation of fire brigade training is March 17, 1978.

6.5 REVIEW AND AUDIT

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

7.0 REFERENCES

(1) E. Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M. Backer, "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) FSAR Section 11.2.2.

(4) FSAR Section 4.4.3.

(5) I.M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Vol. 9, No. 4, July-August 1968, pp 310-312.

(6) Benjamin Epstein, Albert Shiff, UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, July 16, 1968, p. 10, Equation (24), Lawrence Radiation Laboratory.

(7) I.M. Jacobs and P.W. Mariott, APED Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards - April 1969.

(8) Bodega Bay Preliminary Hazards Report, Appendix 1, Docket 50-205, December 28, 1962.

(9) C.H. Robbins, "Tests of a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(10) "Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, Progress Report for Period Ending December 31, 1966, ORNL-4071."

(11) Section 5.2 of the FSAR.

(12) TID 20583, "Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations."

(13) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements," USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

(14) Section 14.6 of the FSAR.

(15) ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III. Maximum allowable internal pressure is 62 psig.

(16) 10 CFR 50.54, Appendix J, "Reactor Containment Testing Requirements."

(17) 10 CFR 50, Appendix J, February 13, 1973.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated March 9, 1993, the Power Authority of the State of New York (the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant, Technical Specifications (TSs). The proposed revisions to the TSs involve editorial changes, correct typographical errors, and adjust line spacings and text formats. In addition, pertinent portions of the TSs that related to exceptions that are no longer applicable would also be deleted.

2.0 EVALUATION

The licensee has proposed revisions to the TSs that would incorporate editorial changes, correct typographical errors, and adjust line spacings and text formats. The staff has reviewed these proposed changes and has determined that they are acceptable since they do not involve any substantive changes to requirements.

In addition to the above noted editorial changes, typographical corrections, and spacing and format changes, the following changes were also proposed to delete exceptions that are no longer applicable:

Technical Specification	Description of Proposed Change
4.7.A.2.a.(10.)	Delete the footnote added by License Amendment No. 125 that eliminated the requirement to conduct a Type A primary containment integrity leak rate test during the 1988 refueling outage.
4.7.A.2	Delete the footnote added by License Amendments 125 and 140 that eliminated the requirement to conduct Type A, B, and C leak rate tests for two plant modifications during outages in 1988 and 1989.

4.12.F.1 Delete the footnote added by License Amendment No. 177 that provided a one-time extension to the fire barrier penetration seal visual inspection interval. The extension ended on May 15, 1992.

The staff has reviewed the above listing of proposed changes and has concluded that they are acceptable since they only delete exceptions that are no longer applicable and do not involve any substantive changes to requirements.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 30198). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
John E. Menning

Date: June 29, 1993

June 29, 1993

Docket No. 50-333

DISTRIBUTION:
See attached sheet

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M86248)

The Commission has issued the enclosed Amendment No. 190 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 9, 1993.

The amendment involves editorial changes, corrects typographical errors, and adjusts line spacing and text formats. In addition, the amendment deletes pertinent portions of the TSs that related to exceptions that are no longer applicable. The amendment does not make any substantive changes to the TSs..

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original Signed By:
John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.190 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDI-1:LA	PDI-1:PM	OGC C. Marco	PDI-1:D	
NAME	CVogan CV	JMenning JEM avl		RACapra RAC	
DATE	6/24/93	6/24/93	6/25/93	6/29/93	

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