October 23, 1987

Docket Nos. 50-335
and 50-389

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Mr. C. O. Woody
Group Vice President
Nuclear Energy
Florida Power & Light Company
P. O. Box 14000
Juno Beach, Florida 33408

Dear Mr. Woody:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 63057 AND 63058)

The Commission has issued the enclosed Amendment Nos. 86 and 25 to Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your applications dated March 17, 1987 (for Unit No. 1) and March 31, 1987 (for Unit No. 2).

These amendments (1) change the unit of reactivity from "delta k/k" to "pcm," (2) delete requirements that are currently outdated, (3) correct typographical errors, (4) provide the currently correct titles and composition of the Company Nuclear Review Board, and (5) delete specific titles of NRC addressees.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/s/

E. G. Tourigny, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 86 to DPR-67
2. Amendment No. 25 to NPF-16
3. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

*LA:PD22	*PM:PD22	*OGC	D:PD22
DMiller	ETourigny	MYoung	HBerkow
10/13/87	10/13/87	10/16/87	10/23/87

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Mr. C. O. Woody
Florida Power & Light Company

St. Lucie Plant

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated March 17, 1987, 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, Facility Operating License No. DPR-67 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 2.C.(2) to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 86 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 23, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 86

TO FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 1-1	3/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
3/4 1-5	3/4 1-5
3/4 1-10	3/4 1-10
3/4 1-18	3/4 1-18
3/4 3-25	3/4 3-25
3/4 3-54	3/4 3-54
3/4 4-58	3/4 4-58
3/4 7-8	3/4 7-8
3/4 7-10	3/4 7-10
3/4 7-11	3/4 7-11
3/4 7-12	3/4 7-12
B3/4 1-1	B3/4 1-1
B3/4 1-2	B3/4 1-2
B3/4 7-3	B3/4 7-3
6-9	6-9
6-15	6-15
6-16	6-16
6-19	6-19
6-23	6-23

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 3600 pcm.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN < 3600 pcm, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3600 pcm:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, at least once during CEA withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

#With $K_{eff} \geq 1.0$.

##With $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,*
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1000 pcm at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

*For Modes 3 and 4, during calculation of shutdown margin with all CEA's verified fully inserted, the single CEA with the highest reactivity worth need not be assumed to be stuck in the fully withdrawn position.

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

$\geq$ 2000 pcm, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

APPLICABILITY: MODE 5.

ACTION:

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

---

* Breaker racked-out.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentation is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant to the reactor pressure vessel < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be determined to be ≥ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm to the reactor pressure vessel.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than +7 pcm/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than +2 pcm/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -28 pcm/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each refueling.

b. At any THERMAL POWER, within 7 EFPD after initially reaching a RATED THERMAL POWER equilibrium boron concentration.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Two flow paths from the boric acid makeup tanks via either a boric acid pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to at least 2000 pcm at 200°F; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following three borated water sources shall be OPERABLE:

a. Two boric acid makeup tanks and one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water tank with:

1. A minimum contained volume of 401,800 gallons of water,

2. A minimum boron concentration of 1720 ppm,

3. A maximum solution temperature of 100°F,

4. A minimum solution temperature of 55°F when in MODES 1 and 2, and

5. A minimum solution temperature of 40°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to at least 2000 pcm at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

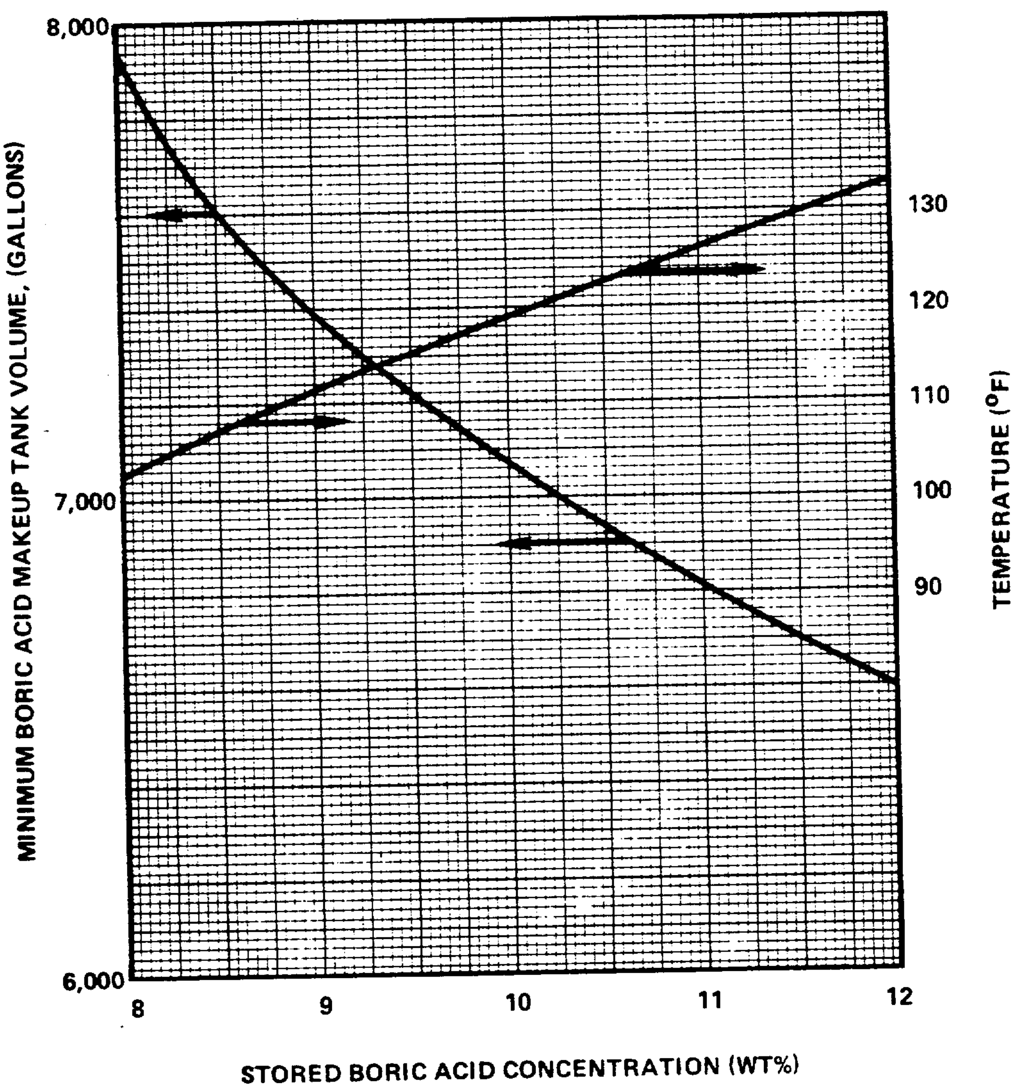

Figure 3.1-1 Minimum Boric Acid Makeup Tank Volume and Temperature as a Function of Stored Boric Acid Concentration

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of three OPERABLE rhodium detectors.

APPLICABILITY: When the incore detection system is used for:

a. Recalibration of the excore axial flux offset detection system,

b. Monitoring the AZIMUTHAL POWER TILT,

c. Calibration of the power level neutron flux channels, or

d. Monitoring the linear heat rate.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Recalibration of the excore axial flux offset detection system,

2. Monitoring the linear heat rate pursuant to Specification 4.2.1.3,

SURVEILLANCE REQUIREMENTS (Continued)

3. Monitoring the AZIMUTHAL POWER TILT, or

4. Calibration of the Power Level Neutron Flux Channels.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

TABLE NOTATION

* At all times.

** During waste gas system operation.

***At all times when air ejector exhaust is not directed to plant vent.

ACTION 1 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup (one performs, one verifies).

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 2 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 3 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for Isotopic activity within 24 hours.

ACTION 4 - Maximum system flows shall be utilized in the determination of the instantaneous release monitor alarm setpoint.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 30 days provided samples of O_2 are analyzed by the lab gas partitioner at least once per 24 hours.

ACTION 6 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS DECAY TANKS					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	*
2. WASTE GAS DECAY TANKS EXPLOSIVE GAS MONITORING SYSTEM					
a. Oxygen Monitor	D	N.A.	Q(4)	M	**
b. Oxygen Monitor (alternate)	D	N.A.	Q(4)	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	***
4. PLANT VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.12 Each Power Operated Relief Valve (PORV) Block Valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.12 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed after the inoperable valve is restored to OPERABLE status or the isolation valve is maintained closed; otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve that is open shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full closure within 6 seconds on any closure actuation signal while in HOT STANDBY with $T_{avg} \geq 515°F$ during each reactor shutdown except that verification of full closure within 6 seconds need not be determined more often than once per 92 days.

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3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 3600 pcm is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transient resulting from a boron dilution event with a partially drained Reactor Coolant System requires a 2000 pcm SHUTDOWN MARGIN and restrictions on charging pump operation to provide adequate protection. A 2000 pcm SHUTDOWN MARGIN is 1000 pcm conservative for Mode 5 operation with total RCS volume present, however LCO 3.1.1.2 is written conservatively for simplicity.

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 11,400 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limiting values assumed for the MTC used in the accident and transient analyses were +7 pcm/°F for THERMAL POWER levels $\leq$ 70% of RATED THERMAL POWER, +2 pcm/°F for THERMAL POWER levels > 70% of RATED THERMAL and -28 pcm/°F at RATED THERMAL POWER. Therefore, these limiting values are included in this specification. Determination of MTC at the specified conditions ensures that the maximum positive and/or negative values of the MTC will not exceed the limiting values.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 2000 pcm after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 7,925 gallons of 8.0% boric acid solution from the boric acid tanks or 13,700 gallons of 1720 ppm borated water from the refueling water tank.

The requirements for a minimum contained volume of 401,800 gallons of borated water in the refueling water tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

THIS SECTION LEFT BLANK INTENTIONALLY

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200-psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 INTAKE COOLING WATER SYSTEM

The OPERABILITY of the intake cooling water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitation on minimum water level is based on providing an adequate cooling water supply to safety related equipment until cooling water can be supplied from Big Mud Creek.

Cooling capacity calculations are based on an ultimate heat sink temperature of 95°F. It has been demonstrated by a temperature survey conducted from March 1976 to May 1981 that the Atlantic Ocean has never risen higher than 86°F. Based on this conservatism, no ultimate heat sink temperature limitation is specified.

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility will be adequately protected from flooding.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

FUNCTION

6.5.2.1 The Company Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechnical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The CNRB shall be composed of the following members:

Member: Group Vice President
Member: Group Vice President - Nuclear Energy
Member: Vice President - Engineering, Projects & Construction
Member: Vice President - Nuclear Operations
Member: Director - Nuclear Licensing
Member: Director - Quality Assurance
Member: Chief Engineer - Power Plant Engineering
Member: Manager - Nuclear Energy Services
Member: Manager - Nuclear Fuel
Member: Senior Project Manager - Power Plant Engineering

The Chairman shall be a member of the CNRB and shall be designated in writing.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter and as convened by the CNRB Chairman or his designated alternate.

QUORUM

6.5.2.6 The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The CNRB shall review:

a. The safety evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments completed under the provisions of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions, and

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring.

e. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel,

(ii) Procedures for sampling and analysis, and

(iii) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS 1/

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,2/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC, no later than the 15th of each month following the calendar month covered by the report.

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

report period, including a comparison, as appropriate, with preoperational studies, with operational controls and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of the Interlaboratory Comparison Program required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of reactor tests and experiments.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those unit components identified in Table 5.9-1.

6.13 PROCESS CONTROL PROGRAM (PCP)

Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information.

   b. A determination that the change did not reduce the overall conformance of the dewatered bead resin to existing criteria for radioactive wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Facility Review Group. The discussion of each shall contain:

    a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

    b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

    c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems.

    d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

    e. An evaluation of the change which shows the expected maximum exposures to individuals in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

    f. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period prior to when the changes are to be made;

    g. An estimate of the exposure to plant operating personnel as a result of the change; and

    h. Documentation of the fact that the change was reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

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*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

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

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated March 31, 1987, 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, Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 2.C.2 to read as follows:

    2. Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 25 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: October 23, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 25

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 1-1	3/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
3/4 1-5	3/4 1-5
3/4 1-8	3/4 1-8
3/4 1-10	3/4 1-10
3/4 1-12	3/4 1-12
3/4 1-14	3/4 1-14
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-34	3/4 3-34
3/4 3-39	3/4 3-39
3/4 3-53	3/4 3-53
3/4 3-57	3/4 3-57
3/4 4-27	3/4 4-27
3/4 4-38	3/4 4-38
3/4 5-6	3/4 5-6
3/4 6-5	3/4 6-5
3/4 6-15	3/4 6-15
3/4 6-21	3/4 6-21
3/4 6-23	3/4 6-23
3/4 7-4	3/4 7-4
3/4 7-39	3/4 7-39
3/4 8-1	3/4 8-1
3/4 8-2	3/4 8-2
3/4 9-5	3/4 9-5
3/4 11-6	3/4 11-6
3/4 11-10	3/4 11-10
B3/4 0-3	B3/4 0-3
B3/4 1-1	B3/4 1-1
B3/4 1-2	B3/4 1-2
B3/4 1-4	B3/4 1-4
B3/4 3-4	B3/4 3-4
6-10	6-10
6-15	6-15
6-16	6-16
6-17	6-17
6-20	6-20
6-23	6-23

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 5000 pcm.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN less than 5000 pcm, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5000 pcm:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1000 pcm at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPDs after each fuel loading.

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 3000 pcm.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 3000 pcm, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 3000 pcm:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,and
6. Samarium concentration.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2b. and by verifying at least two charging pumps are rendered inoperable by racking out their motor circuit breakers.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be $\geq$ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant to the reactor pressure vessel < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be determined to be $\geq$ 3000 gpm within 1 hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying $\geq$ 3000 gpm to the reactor pressure vessel.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than +5 pcm/°F at $\leq$ 70% RATED THERMAL POWER,

b. Less positive than +3 pcm/°F at > 70% RATED THERMAL POWER, and

c. Less negative than -27 pcm/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 800 ppm.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exceptions 3.10.2 and 3.10.5.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 515°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 515°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 515°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 525°F.

#With K_{eff} greater than or equal to 1.0.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid makeup tank via either a boric acid makeup pump or a gravity feed connection and charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a. is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid makeup tanks is used.

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Two flow paths from the boric acid makeup tanks via either a boric acid makeup pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3000 pcm at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid makeup tanks is above the temperature limit line shown on Figure 3.1-1.

b. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on an SIAS test signal.

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a delivers at least 40 gpm to the Reactor Coolant System.

CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least the above required pump shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2854 ft when tested pursuant to Specification 4.0.5.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3000 pcm at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying that each pump develops a flow rate of greater than or equal to 40 gpm when tested pursuant to Specification 4.0.5.

4.1.2.4.2 At least once per 18 months verify that each charging pump starts automatically on an SIAS test signal.

BORIC ACID MAKEUP PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid makeup pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1a. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid makeup pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a., suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required boric acid makeup pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 90 psig when tested pursuant to Specification 4.0.5.

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 3000 pcm at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid makeup pump(s) shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump(s) develop a discharge pressure of greater than or equal to 90 psig when tested pursuant to Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank and at least one associated heat tracing circuit with a minimum contained volume of 4150 gallons of 8 weight percent boron.

b. The refueling water tank with:

1. A minimum contained borated water volume of 125,000 gallons,

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and 120°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid makeup tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is outside the range of 40°F and 120°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one boric acid makeup tank and at least one associated heat tracing circuit per tank with the contents of the tank in accordance with Figure 3.1-1, and

b. The refueling water tank with:

1. A minimum contained borated water volume of 417,100 gallons,
2. A boron concentration of between 1720 and 2100 ppm of boron, and
3. A solution temperature between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 3000 pcm at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,
2. Verifying the contained borated water volume of the water source, and
3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is outside the range of 55°F and 100°F.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area					
i. Criticality and Ventilation System Isolation Monitor	4	*	≤ 20 mR/hr	10^{-1} - 10^{4} mR/hr	22
b. Containment Isolation	3	6	≤ 90 mR/hr	1 - 10^{7} mR/hr	25
c. Control Room Isolation	1 per intake	ALL MODES	≤ 2x background	10^{-7} - 10^{-2} μCi/cc	26
d. Containment Area - Hi Range	1	1, 2, 3 & 4	Not Applicable	1 - 10^{7} R/hr	27
2. PROCESS MONITORS					
a. Fuel Storage Pool Area Ventilation System					
i. Gaseous Activity	1	**	***	10^{-7} - 10^{-2} μCi/cc	24
ii. Particulate Activity	1	**	***	1 - 10^{6} cpm	24
b. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{-7} - 10^{-2} μCi/cc	23
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	1 - 10^{6} cpm	23

*With fuel in the storage pool or building.

**With irradiated fuel in the storage pool or whenever there is fuel movement within the pool or crane operation with loads over the storage pool.

***The Alarm/Trip Setpoints are determined and set in accordance with the requirements of Specification 3.3.3.10.

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
PROCESS MONITORS (Continued)					
c. Noble Gas Effluent Monitors					
i. Reactor Auxiliary Building Exhaust System (Plant Vent Low Range Monitor)	1	1, 2, 3, & 4	***	$10^{-7} - 10^{-2}$ µCi/cc	27
ii. Reactor Auxiliary Building Exhaust System (Plant Vent High Range Monitor)	1	1, 2, 3, & 4	***	$10^{-2} - 10^{5}$ µCi/cc	27
iii. Steam Generator Blowdown Treatment Facility Building Exhaust System	1	1, 2, 3, & 4	***	$10^{-7} - 10^{-2}$ µCi/cc	27
iv. Steam Safety Valve Discharge#	1/steam header	1, 2, 3, & 4	***	$10^{-1} - 10^{3}$ µCi/cc	27
v. Atmospheric Steam Dump Valve Discharge#	1/steam header	1, 2, 3, & 4	***	$10^{-1} - 10^{3}$ µCi/cc	27
vi. ECCS Exhaust	1	1, 2, 3, & 4	***	$10^{-7} - 10^{5}$ µCi/cc	27

***The Alarm/Trip Setpoints are determined and set in accordance with the requirements of Specification 3.3.3.10.

\# The steam safety valve discharge monitor and the atmospheric steam dump valve discharge monitor are the same monitor.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION
(Instrumentation located in St. Lucie Unit 1)

INSTRUMENT CHANNEL	SENSOR LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. STRONG MOTION TRIAXIAL ACCELEROGRAPHS			
a. SMR-42-1	R.B. Elev. 23.0'	0-1 g	1*
b. SMR-42-2	R.B. Elev. 62.0'	0-1 g	1
c. SMR-42-3	R.A.B. Elev. -0.5'	0-1 g	1
d. SMR-42-4	R.A.B. Elev. 43.0'	0-1 g	1
2. PEAK RECORDING ACCELEROGRAPHS			
a. SMR-42-6	R.B. Piping from S.I.T.1A2-c Elev. 46' 10 9/16"	0-2 g	1
b. SMR-42-7	R.B. Equipment on S.I.T.1A2	0-2 g	1
c. SMR-42-8	R.A.B.-Sh. Dn. Ht. XCHR Supports	0-2 g	1
3. PEAK SHOCK RECORDERS			
a. SMR-42-9	R.B. Elev. 23.0'	-	1
b. SMR-42-10	R.B. M.S. Pipe Restraints - S.G.1B1	-	1
4. EARTHQUAKE FORCE MONITOR			
a. SMI-42-11	Control Room	0-0.2 g	1
5. SEISMIC SWITCH			
a. SMS-42-12	R.B. Elev. 23.0'	-	1*

*With St. Lucie Unit 2 reactor control room alarm

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS
(Instrumentation located in St. Lucie Unit 1)

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. STRONG MOTION TRIAXIAL ACCELEROGRAPHS			
a. SMR-42-1	M*	R	SA
b. SMR-42-2	M*	R	SA
c. SMR-42-3	M*	R	SA
d. SMR-42-4	M*	R	SA
e. SMR-42-5	M*	R	SA
2. PEAK RECORDING ACCELEROGRAPHS			
a. SMR-42-6	NA	R	NA
b. SMR-42-7	NA	R	NA
c. SMR-42-8	NA	R	NA
3. PEAK SHOCK RECORDERS			
a. SMR-42-9	NA	R	NA
b. SMR-42-10	NA	R	NA
4. EARTHQUAKE FORCE MONITOR			
a. SMI-42-11	M	R	SA
5. SEISMIC SWITCH			
a. SMS-42-12	NA	R	SA

*Except seismic trigger

TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	CHANNELS RANGE	REQUIRED OF NUMBER CHANNELS	MINIMUM CHANNELS OPERABLE
1. Power Range Neutron Flux	Hot Shutdown Panel	$2 \times 10^{-8}\%$ - 200%	2	1
2. Reactor Trip Breaker Indication	Reactor Trip Switch Gear (RB)	OPEN-CLOSE	1/trip breaker	1/trip breaker
3. Reactor Coolant Temperature - T_{Cold}	Hot Shutdown Panel	0° - 600°F	2	1
4. Pressurizer Pressure	Hot Shutdown Panel	0 - 3000 psia	2	1
5. Pressurizer Level	Hot Shutdown Panel	0 - 100% level	2	1
6. Steam Generator Pressure	Hot Shutdown Panel	0 - 1200 psia	1/steam generator	1/steam generator
7. Steam Generator Level	Hot Shutdown Panel	0 - 100% level	2/steam generator	1/steam generator
8. Shutdown Cooling Flow Rate	Hot Shutdown Panel	0 - 5000 gpm	2	1
9. Shutdown Cooling Temperature	Hot Shutdown Panel	0° - 350°F	2	1
10. Diesel Generator Voltage	Hot Shutdown Panel	0 - 5250 V	1/diesel generator	1/diesel generator
11. Diesel Generator Power	Hot Shutdown Panel	0 - 5000 kW	1/diesel generator	1/diesel generator
12. Atmospheric Dump Valve Pressure	Hot Shutdown Panel	0 - 1200 psig	1/steam generator	1/steam generator
13. Charging Flow/Pressure	Hot Shutdown Panel	0 - 150 gpm/ 0 - 3000 psia	2	1
CONTROLS/ISOLATE SWITCHES				
1. Atmospheric Stm Dump Controllers	Hot Shutdown Panel/RAB431	N.A.	2/steam generator	1/steam generator
2. Aux. Spray Valves	Hot Shutdown Panel/RAB431	N.A.	2	1
3. Charging Pump Controls	Hot Shutdown Panel/RAB431	N.A.	3	2
4. Letdown Isol Valve	Hot Shutdown Panel/RAB431	N.A.	3	2
5. AFW Pump/Valve Controls	Hot Shutdown Panel/RAB431	N.A.	3	2
6. AFW Pump Steam Inlet Valve	Hot Shutdown Panel/RAB431	N.A.	2	1
7. Pzr Heater Controls	Hot Shutdown Panel/RAB431	N.A.	6	3

TABLE 4.3-6

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Power Range Neutron Flux	M	Q
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Coolant Temperature-T_{Cold}	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Level	M	R
7. Steam Generator Pressure	M	R
8. Shutdown Cooling Flow Rate	M	R
9. Shutdown Cooling Temperature	M	R
10. Diesel Generator Voltage	M	R
11. Diesel Generator Power	M	R
12. Atmospheric Dump Valve Pressure	M	R
13. Charging Flow/Pressure	M	R

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. If the inoperable instruments are not returned to operable status within 30 days, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS DECAY TANKS			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	35
2. WASTE GAS DECAY TANKS EXPLOSIVE GAS MONITORING SYSTEM			
a. Oxygen Monitors	1	**	39
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor	1	***	37
4. PLANT VENT SYSTEM			
a. Noble Gas Activity Monitor (Low Range)	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate	NA	*	38
e. Sampler Flow Rate Monitor	1	*	36

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS DECAY TANKS					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	*
2. WASTE GAS DECAY TANKS EXPLOSIVE GAS MONITORING SYSTEM					
a. Oxygen Monitor	D	N.A.	Q(4)	M	**
b. Oxygen Monitor (alternate)	D	N.A.	Q(4)	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	***
4. PLANT VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. FUEL STORAGE AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*
6. LAUNDRY AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*
7. STEAM GENERATOR BLOWDOWN BUILDING VENT					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 micro-Ci/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ micro-Ci/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

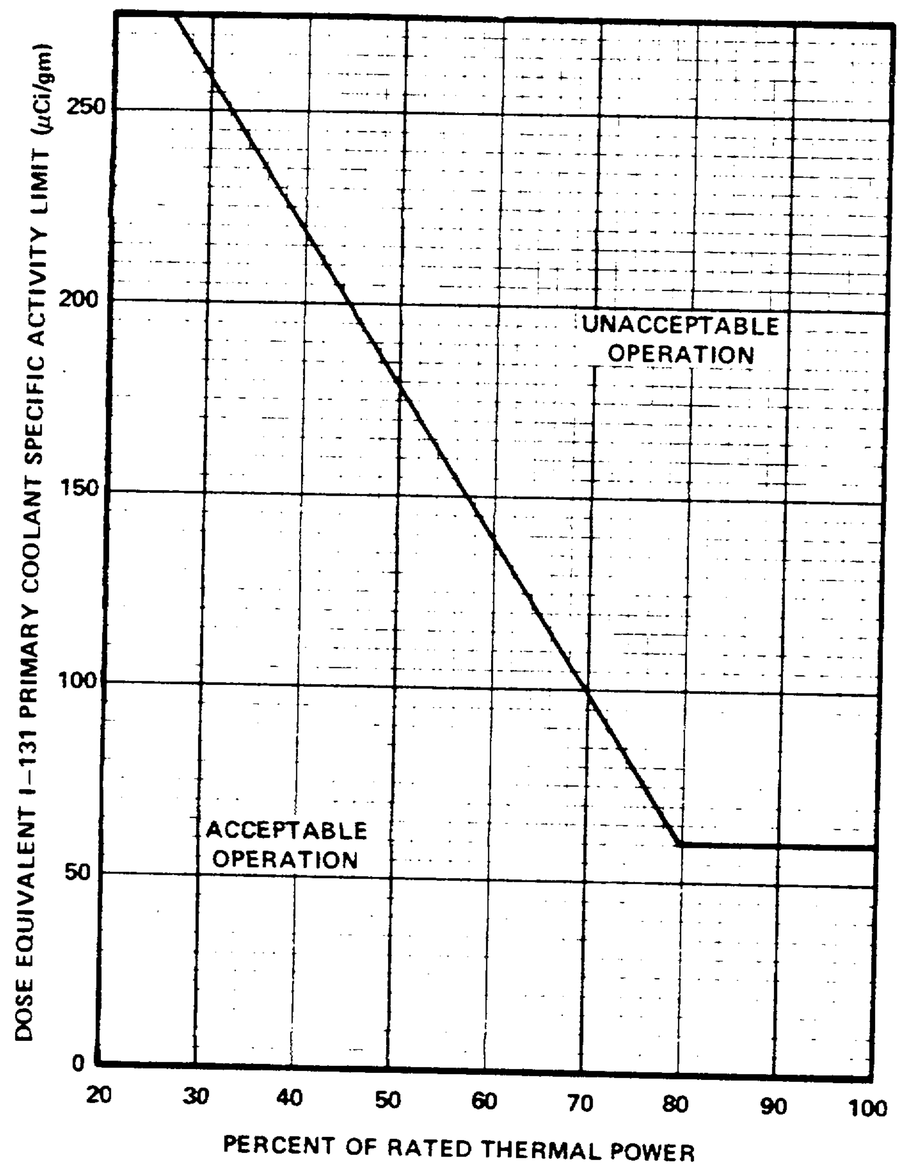

FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

c. Performance of a CHANNEL CALIBRATION on the PORV actuation channel, at least once per 18 months.

d. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.10 At least one Reactor Coolant System vent path consisting of two vent valves and one block valve powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Pressurizer steam space, and

b. Reactor vessel head.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable, maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.1 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each vent valve through at least one complete cycle of full travel from the control room.

3. Verifying flow through the Reactor Coolant System vent paths during venting.

SURVEILLANCE REQUIREMENTS (Continued)

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 173 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 70.5 ± 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 ± 0.1 gallons of 120 ± 10°F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and/or RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

3. Verifying that on a Sump Recirculation Actuation Test Signal, the containment sump isolation valves open and the recirculation valve to the refueling water tank closed.

g. By verifying that each of the following pumps develops the specified total developed head on recirculation flow when tested pursuant to Specification 4.0.5:

1. High-Pressure Safety Injection pumps: greater than or equal to 2854 ft.

2. Low-Pressure Safety Injection pump: greater than or equal to 374 ft.

h. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1. During valve stroking operation or following maintenance on the valve and prior to declaring the valve OPERABLE when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

HPSI System Valve Number	LPSI System Valve Number
a. HCV 3616/3617	a. HCV 3615
b. HCV 3626/3627	b. HCV 3625
c. HCV 3636/3637	c. HCV 3635
d. HCV 3646/3647	d. HCV 3645
e. V3523/V3540	

i. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics. The test shall measure the individual leg flow rates and pump total developed head to verify the following conditions:

1. HPSI Pump 2A:
   The sum of the three lowest cold leg flow rates shall be greater than or equal to 476 gpm with total developed head greater than or equal to 1150 ft but less than or equal to 1290 ft.

2. HPSI Pump 2B:
   The sum of the three lowest cold leg flow rates shall be greater than or equal to 484 gpm with total developed head greater than or equal to 910 ft but less than or equal to 1040 ft.

3. With the system operating in hot/cold leg injection mode, the hot leg flow shall be greater than or equal to 317 gpm and within 10% of the cold leg header flow and:

   HPSI Pump 2A:
   The pump shall be producing total developed head greater than or equal to 1297 ft but less than or equal to 1500 ft.

   HPSI Pump 2B:
   The pump shall be producing total developed head greater than or equal to 1042 ft but less than 1250 ft.

4. LPSI System - Each Pump:

   The flow through each injection leg shall be greater than or equal to 1763 gpm at a total developed head greater than or equal to 298 ft but less than or equal to 337 ft.

TABLE 3.6-1

CONTAINMENT LEAKAGE PATHS

Penetration	System	Valve Tag Number/Type	Location to Containment	Service	Test Type*
7	Makeup Water	I-HCV-15-1 Globe I-V-15-328 Check	Outside Inside	Primary Makeup Water	BYPASS/ TYPE C
8	Station Air	I-V-18-794 Globe I-V-18-1270 Check I-V-18-797 Globe I-HCV-18-2 Globe	Outside Inside Annulus Outside	Station Air Supply	BYPASS/ TYPE C
9	Instrument Air	I-HCV-18-1 Globe I-V-18-195 Check	Outside Inside	Instrument Air Supply	BYPASS/ TYPE C
10	Containment Purge	I-FCV-25-5 B'FLY I-FCV-25-4 B'FLY	Annulus Inside	Containment Purge Exhaust	TYPE C
11	Containment Purge	I-FCV-25-2 B'FLY I-FCV-25-3 B'FLY	Annulus Inside	Containment Purge Supply	TYPE C
14	Waste Management	V-6741 Globe V-6792 Check	Outside Inside	N_2 Supply to Safety Inj. Tanks	BYPASS/ TYPE C
23	Component Cooling	I-HCV-14-7 B'FLY I-HCV-14-1 B'FLY	Outside Inside	RC Pump Cooling Water Supply	BYPASS/ TYPE C
24	Component Cooling	I-HCV-14-6 B'FLY I-HCV-14-2 B'FLY	Outside Inside	RC Pump Cooling Water Return	BYPASS/ TYPE C
25	Fuel Transfer Tube	Double Gasket Flange	Inside	Fuel Transfer	BYPASS/ TYPE C
26	CVCS	I-V-2516 Globe I-V-2522 Globe	Inside Outside	Letdown Line	BYPASS/ TYPE C

TABLE 3.6-1 (Continued)

CONTAINMENT LEAKAGE PATHS

Penetration	System	Valve Tag Number/Type	Location to Containment	Service	Test Type*
28A	Sampling	ISE-05-1A,1B,1C,1D Globe ISE-05-1E Globe	Inside Outside	Safety/Injection Tank Sample	BYPASS TYPE C
28B	Sampling	I-V-5200 Globe I-V-5203 Globe	Inside Outside	RCS Hot Leg Sample	BYPASS TYPE C
29A	Sampling	I-V-5204 Globe I-V-5201 Globe	Outside Inside	Pressure Surge Sample	BYPASS TYPE C
29B	Sampling	I-V-5205 Globe I-V-5202 Globe	Outside Inside	Pressure Steam Sample	BYPASS TYPE C
31	Waste Management	I-V-6718 Diaph I-V-6750 Diaph	Inside Outside	Containment Vent Header	BYPASS TYPE C
41	Safety Injection	I-SE-03-2A,2B Globe I-V-3463 Gate	Inside Outside	Safety Injection Tank Fill/Drain and Sampling	BYPASS TYPE C
42	Waste Management	I-LCV-07-11A Globe I-LCV-07-11B Globe	Inside Outside	Reactor Cavity Sump Pump Discharge	BYPASS TYPE C
43	Waste Management	I-V-6341 Diaph I-V-6342 Diaph	Inside Outside	Reactor Drain Tank Pump Suction	BYPASS TYPE C
44	CVCS	I-V-2524 Globe I-V-2505 Globe	Inside Outside	Reactor Coolant Pump Controlled Bleedoff	BYPASS TYPE C
46	Fuel Pool	I-V-07-206 Gate I-V-07-189 Gate	Outside Inside	Fuel Pool Cleanup (inlet)	BYPASS TYPE C
47	Fuel Pool	I-V-07-170 Gate I-V-07-188 Gate	Outside Inside	Fuel Pool Cleanup (outlet)	BYPASS TYPE C

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a Containment Spray Actuation Signal and automatically transferring suction to the containment sump on a Sump Recirculation Actuation Signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is positioned to take suction from the RWT on a Containment Pressure--High-High test signal.

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 200 psig when tested pursuant to Specification 4.0.5.

c. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a CSAS test signal.

   2. Verifying that upon a Recirculation Actuation Test Signal (RAS), the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

*Applicable only when pressurizer pressure is ≥ 1750 psia.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump starts automatically on a CSAS test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Maximum Isolation Time (Sec)
A) Containment Isolation				
I-HCV-15-1	7	Primary Makeup Water (CIS)	Yes	5
I-HCV-18-2	8	Station Air Supply	Yes	5
I-HCV-18-1	9	Instrument Air Supply (CIS)	No	5
I-FCV-25-5,4	10	Containment Purge Exhaust (CIS)	No	3
I-FCV-25-2,3	11	Containment Purge Makeup (CIS)	No	3
V-6741	14	Nitrogen Supply to Safety Injection Tanks (CIS)	Yes	5
I-HCV-14-7 I-HCV-14-1	23	Reactor Coolant Pump Cooling Water Supply (SIAS)	No	5
I-HCV-14-6 I-HCV-14-2	24	Reactor Coolant Pump Cooling Water Return (SIAS)	No	5
I-HCV-2516 I-HCV-2522	26	Letdown Line (CIS)	No	5
I-SE-05-1A,1B, 1C,1D,1E	28A	Safety Injection Tank Sample	Yes	5
I-V-5200 I-V-5203	28B	Reactor Coolant System Hot Leg Sample (CIS)	Yes	5
I-V-5204 I-V-5201	29A	Pressurizer Surge Sample (CIS)	Yes	5
I-V-5205 I-V-5202	29B	Pressurizer Steam Sample (CIS)	Yes	5

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Maximum Isolation Time (Sec)
I-V-6718 I-V-6750	31	Containment Vent Header (CIS)	Yes	5
I-SE-03-2A,2B	41	Safety Injection Tank Test Line (CIS/SIAS)	Yes	5
I-LCV-07-11A I-LCV-07-11B	42	Reactor Cavity Sump Pump Discharge (CIS/SIAS)	Yes	5
I-V-6341 I-V-6342	43	RCDT Pump Suction (CIS)	Yes	5
I-V-2524 I-V-2505	44	RCP Controlled Bleed-off (CIS)	No	5
I-FCV-26-1 I-FCV-26-2	52A	Containment Radiation Monitoring (CIS)	Yes	10
I-FCV-26-3 I-FCV-26-4	52B	Containment Radiation Monitoring (CIS)	Yes	10
I-FCV-26-5 I-FCV-26-6	52C	Containment Radiation Monitoring (CIS)	Yes	10
I-FCV-25-26 I-FCV-25-36	56	Cont. Containment/H_2 Purge Makeup Inlet (CIS)	Yes	3
I-FCV-25-20 I-FCV-25-21	57	Cont. Containment/H_2 Purge Exhaust (CIS)	Yes	3

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Maximum Isolation Time (Sec)
B) Manual CR Remote Manual				
I-V-18-797	8	Station Air Supply (Manual)	Yes	NA
I-V-18-1270		Station Air Supply (Check)	No	NA
I-V-3463	41	Safety Injection Tank Test Line (Manual)	Yes	NA
I-V-07-206 I-V-07-189	46	Fuel Pool Cleanup (Inlet) (Manual)	Yes	NA
I-V-07-170 I-V-07-188	47	Fuel Pool Cleanup (Outlet) (Manual)	Yes	NA
I-FSE-27-8,9,10, 11,15,16	48	H_2 Sampling (Remote Manual)	Yes	NA
I-FSE-27-12,13,14, 17,18	51	H_2 Sampling (Remote Manual)	Yes	NA
I-V-00-140 I-V-00-143	52D	ILRT (Manual)	Yes	NA
I-V-00-139 I-V-00-144	52E	ILRT (Manual)	Yes	NA
I-V-00-101	54	ILRT (Manual)	Yes	NA

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent hydrogen, balance nitrogen and oxygen.

b. Four volume percent hydrogen, balance nitrogen and oxygen.

TABLE 4.7-0

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	Header A	Header B	LIFT SETTING (± 1%)	ORIFICE SIZE
a.	8201	8205	1000 psia	16 $in.^2$
b.	8202	8206	1000 psia	16 $in.^2$
c.	8203	8207	1000 psia	16 $in.^2$
d.	8204	8208	1000 psia	16 $in.^2$
e.	8209	8213	1040 psia	16 $in.^2$
f.	8210	8214	1040 psia	16 $in.^2$
g.	8211	8215	1040 psia	16 $in.^2$
h.	8212	8216	1040 psia	16 $in.^2$

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1270 psig on recirculation flow.

2. Verifying that the turbine-driven pump develops a discharge pressure of greater than or equal to 1260 psig on recirculation flow when the secondary steam supply pressure is greater than 50 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3/4.7.12 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.12 All fire rated assemblies (walls, floor/ceilings, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire dampers, cable, piping, and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire damper and associated hardware.

c. Performing a visual inspection of at least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.12.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

b. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours.

c. That each locked closed fire door is closed at least once per 7 days.

4.7.12.3 At least once per 18 months, perform a functional test of each automatic hold-open and release mechanism.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. Two separate engine-mounted fuel tanks containing a minimum volume of 200 gallons of fuel each,

   2. A separate fuel storage system containing a minimum volume of 40,000 gallons of fuel, and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With either an offsite circuit other than the conditions delineated in Action 3.8.1.1f. or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, or 3, the steam-driven auxiliary feed pump is OPERABLE.

ACTION (Continued)

If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With one Unit 2 startup transformer (2A or 2B) inoperable and with a Unit 1 startup transformer (1A or 1B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 1 require the use of the startup transformer administratively available to both units, Unit 2 shall demonstrate the operability of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore the inoperable startup transformer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane shall be used for movement of fuel assemblies, with or without CEAs, and shall be OPERABLE with:

a. A minimum capacity of 2000 pounds, and

b. An overload cut off limit of less than or equal to 3000 pounds.

APPLICABILITY: During movement of fuel assemblies, with or without CEAs, within the reactor pressure vessel.

ACTION:

With the requirements for crane OPERABILITY not satisfied, suspend use of any inoperable manipulator crane from operations involving the movement of CEAs and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The manipulator crane used for movement of fuel assemblies, with or without CEAs, within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 2000 pounds and demonstrating an automatic load cut off when the crane load exceeds 3000 pounds.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases and radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose or dose commitment to an individual from these releases is within 3 mrems to the total body and 10 mrems to any organ.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site to UNRESTRICTED AREAS (see Figure 5.1-1) when averaged over 31 days, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases to UNRESTRICTED AREAS shall be projected at least once per 31 days, in accordance with the ODCM unless the liquid radwaste treatment system is being used.

4.11.1.3.2 The liquid radwaste treatment system shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 30 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid effluents during the previous 92 days.

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

c. Samples shall be changed at least 4 times a month and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

d. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 5000 pcm is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. At earlier times in core life, the minimum SHUTDOWN MARGIN required for the most restrictive conditions is less than 5000 pcm. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from any postulated accident are minimal and a 3000 pcm SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 gpm provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 gpm will circulate an equivalent Reactor Coolant System volume of 10,931 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 3000 pcm after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 72,000 gallons of 1720 ppm - 2100 ppm borated water from the refueling water tank.

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 3000 pcm SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4,150 gallons of 1720 ppm - 2100 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

The contained water volume limits includes allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.2.9 BORON DILUTION

The simultaneous use of the boronometer and RCS sampling at intervals dependent upon the MODE and the number of OPERABLE charging pumps to monitor the RCS boron concentration provides diverse and redundant indications of an inadvertent boron dilution. This will allow detection with sufficient time for termination of the boron dilution event before a complete loss of SHUTDOWN MARGIN occurs.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs and to a large misalignment (greater than or equal to 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 15 inches) of the CEAs, there is (1) a small effect on the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

MOVABLE CONTROL ASSEMBLIES (Continued)

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

The requirement to reduce power in certain time limits depending upon the previous F_r^T is to eliminate a potential nonconservatism for situations when a CEA has been declared inoperable. A worst-case analysis has shown that a DNBR SAFDL violation may occur during the second hour after the CEA misalignment if this requirement is not met. This potential DNBR SAFDL violation is eliminated by limiting the time operation is permitted at full power before power reductions are required. These reductions will be necessary once the deviated CEA has been declared inoperable. This time allowed for continued operation at a reduced power level can be permitted for the following reasons:

1. The margin calculations which support the Technical Specifications are based on a steady-state radial peak of F_r^T = 1.70.
2. When the actual F_r^T < 1.70, significant additional margin exists.
3. This additional margin can be credited to offset the increase in F_r^T with time that can occur following a CEA misalignment.
4. This increase in F_r^T is caused by xenon redistribution.
5. The present analysis can support allowing a misalignment to exist for up to 63 minutes without correction, if the initial $F_r^T \leq$ 1.54.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

RESPONSIBILITIES (Continued)

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President Nuclear Operations and to the Company Nuclear Review Board.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and RADWASTE TREATMENT SYSTEMS.

m. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last FRG meeting.

AUTHORITY

6.5.1.7 The Facility Review Group shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specifications 6.5.1.6a. through d. and m. above.

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6a. through e. above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President Nuclear Operations, the Group Vice President Nuclear Energy, and the Company Nuclear Review Board of disagreement between the FRG and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The Facility Review Group shall maintain written minutes of each FRG meeting that, at a minimum, document the results of all FRG activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Vice President Nuclear Operations, the Group Vice President Nuclear Energy, and the Chairman of the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

FUNCTION

6.5.2.1 The Company Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

FUNCTION (Continued)

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The CNRB shall be composed of the following members:

Member: Group Vice President
Member: Group Vice President - Nuclear Energy
Member: Vice President - Engineering, Projects & Construction
Member: Vice President - Nuclear Operations
Member: Director - Nuclear Licensing
Member: Director - Quality Assurance
Member: Chief Engineer - Power Plant Engineering
Member: Manager - Nuclear Energy Services
Member: Manager - Nuclear Fuel
Member: Senior Project Manager - Power Plant Engineering

The Chairman shall be a member of the CNRB and shall be designated in writing.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter and as convened by the CNRB Chairman or his designated alternate.

QUORUM

6.5.2.6 The quorom of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman and his designated alternate and at least four CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

PROCEDURES AND PROGRAMS (Continued)

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions, and

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring.

e. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel,

(ii) Procedures for sampling and analysis, and

(iii) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures

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[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

ANNUAL REPORTS (Continued)

greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC, no later than the 15th of each month following the calendar month covered by the report.

[2/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

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

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

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

Every 2 years using the previous 6 months release history for isotopes and historical meteorolgical data determine the controlling age group for both liquid and gaseous pathways. If changed from current submit change to ODCM to reflect new tables for these groups and use the new groups in subsequent dose calculations.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases for the previous

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

** In lieu of submission with the Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

calendar year. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, March 1976.

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms)

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.8 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and information based on trend analysis of the results of the radiological environmental surveillance activities for the report period, including a comparison, as appropriate, with preoperational studies, with operational controls and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the

*A single submittal may be made for a multiple unit station.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

* One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

6.13 PROCESS CONTROL PROGRAM (PCP)

Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. Sufficiently detailed information to totally suport the rationale for the change without benefit of additional or supplemental information;

    b. A determination that the change did not reduce the overall conformance of the dewatered bead resin to existing criteria for radioactive wastes; and

    c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

    a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

    b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

    c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Facility Review Group. The discussion of each shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

   g. An estimate of the exposure to plant operating personnel as a result of the change; and

   h. Documentation of the fact that the change was reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

*Licensees may chose to submit the information called for in this Specification as part of the annual FSAR update.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 86 AND 25

TO FACILITY OPERATING LICENSE NOS. DPR-67 AND NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By applications dated March 17, 1987 and March 31 1987, the Florida Power and Light Company (FP&L), the licensee, requested various changes to the technical specifications (TS) for the St. Lucie Plant, Unit Nos. 1 and 2. The March 17, 1987 application addressed TS changes for Unit No. 1, and the March 31, 1987 application addressed TS changes for Unit No. 2. The staff's evaluation of the proposed changes follows.

2.0 EVALUATION - OVERVIEW

The proposed TS changes can be grouped into five broad categories. Category 1 changes deal with changing the unit of reactivity from "delta-k/k" to "pcm." Category 2 changes delete requirements that are currently outdated. Category 3 changes correct typographical errors. Category 4 changes provide the currently correct titles and composition of the Company Nuclear Review Board. Category 5 changes delete specific titles of NRC addressees. The staff's evaluation of the changes by category is contained below.

2.1 Evaluation - Unit of Reactivity

The unit of reactivity currently used in both TS is "delta-k/k." The licensee proposed to change the unit to "pcm." One percent delta k/k equals 1000 pcm's, and one delta k/k equals 100,000 pcm's. The affected specifications are TS 3/4.1.1.1 (Boration Control - Shutdown Margin - Tave greater than 200°F), TS 3/4.1.1.2 (Boration Control - Shutdown Margin - Tave less than or equal to 200°F), TS 3/4.1.1.4 (Moderator Temperature Coefficient), TS 3/4.1.2.2 (Flow Paths - Operating), TS 3/4.1.2.4 (Charging Pumps - Operating) (Unit 2 only), TS 3/4 1.2.6 (Boric Acid Makeup Pumps - Operating) (Unit 2 only), and TS 3/4.1.2.8 (Borated Water Sources - Operating). Similar changes are proposed in the associated bases sections. Although the unit of reactivity is proposed to be changed, the actual reactivity required by the TS and the safety analyses will not change. Thus, the proposed change is acceptable.

2.2 Evaluation - Requirements That are Currently Outdated

The licensee proposed to delete two requirements from the Unit No. 1 TS which were in effect until October 1, 1981. The first requirement was a special operability requirement for the incore detection system (TS 3/4.3.3.2). This special requirement, which expired on October 1, 1981, was specified as a footnote to the TS operability statement. The original requirement contained in the body of the TS is currently utilized. Therefore, deletion of the footnote is acceptable.

The second requirement was a special action statement requirement for the power-operated relief valve block valves (TS 3/4.4.12). This special requirement, which expired on October 1, 1981, was specified as a footnote to the TS action statement. The original requirement contained in the body of the TS is currently utilized. Therefore, deletion of the footnote is acceptable.

TS 6.13 for each unit specifies requirements for the process control program (PCP). TS 6.14 for each unit specifies requirements for the offsite dose calculation manual (ODCM). These specifications require the PCP and ODCM to be approved by the Commission prior to implementation. The licensee proposed to delete these requirements on the basis that the PCP and ODCM were approved by the Commission before FP&L implementation. The PCP was approved for both units on May 10, 1983 by letter from G. Knighton to R. Uhrig. The ODCM was approved for both units on July 28, 1982 by letter from G. Knighton to R. Uhrig. On this basis, it is acceptable to delete the PCP and ODCM implementation requirements.

The licensee proposes to delete a secondary water chemistry TS for Unit No. 1 which was never fully specified (TS 3/4.7.1.6). The associated bases statement would also be deleted. The partial specification was added to the TS when the unit was licensed (1976), and was never fully specified or deleted. The requirements for secondary water chemistry are currently contained in the administrative control section of the TS (TS 6.8.4.C). The detailed program is defined in TS 6.8.4.C and its purpose is to monitor secondary water chemistry, and to inhibit steam generator tube degradation. Since an acceptable secondary water chemistry program is in effect, there is no need for the additional TS on this subject; therefore, TS 3/4.7.1.6 may be deleted.

A number of specifications were put in place at the time of licensing for Unit No. 2 that required plant modifications to be completed before initial criticality in 1983, before exceeding 5% rated thermal power in 1983, and before cycle 2 startup subsequent to 1983. In regard to radiation monitoring instrumentation (Table 3.3-6 of TS 3/4.3.3), the containment area high range monitors, reactor auxiliary building exhaust system (plant vent high range monitor), and atmospheric steam dump valve discharge monitors were to be completely installed and made operable prior to exceeding 5% rated thermal power. The licensee proposed to delete these footnotes on the basis that the requirements were met prior to exceeding 5% rated thermal power. The licensee advised the staff by letter dated July 6, 1983 that the containment high range radiation monitors were installed and operable prior to exceeding 5% rated thermal power. NRC Inspection Reports 50-389/83-25 dated April 6, 1983, and 50-389/83-59 dated September 27, 1983 confirmed that the other two monitors were installed and operable prior to exceeding 5% rated thermal power. On the basis of the above-described documentation, the footnotes may be deleted.

In addition, the reactor coolant system vents were to be installed and made operable prior to exceeding 5% rated thermal power. This requirement was placed as a footnote to the RCS vent TS (TS 3/4.4.10). The licensee proposed that this footnote be deleted because the requirement was met. The licensee advised the staff by letter dated June 6, 1983 that the reactor coolant system vents were installed and operable prior to exceeding 5% rated thermal power. On this basis, the deletion of the footnote is acceptable. Finally, the sound-powered telephone system was to be installed and made operable prior to exceeding 5% rated thermal power. This requirement was placed as a footnote to the refueling operations-communications TS (TS 3/4.9.5). The licensee proposed to delete this footnote because the requirement was met. This requirement was satisfied as documented in NRC Inspection Report 50-389/83-47, dated July 28, 1983. Therefore, the deletion of the footnote is acceptable.

The licensee proposed to delete a footnote relating to the installation and operability of the auxiliary feedwater system automatic initiation system. This requirement called for the system to be operable prior to initial criticality and is contained in TS 3/4.7.1.2, auxiliary feedwater system. This requirement was satisfied as documented in NRC Inspection Report 50-389/83-47, dated July 28, 1983. Therefore, the deletion of this footnote is acceptable.

The licensee also proposed to delete a footnote dealing with fire-rated assemblies (TS 3/4.7.12). This footnote states that the assemblies would be installed and made operable prior to exceeding 5% rated thermal power, with the exception of the permanent flame impingement shields in containment, which would be installed and made operable prior to startup following the first refueling outage. These requirements were discussed in the staff's Supplemental Safety Evaluation Report No. 3 issued in April 1983 (see Sections 9.5.1.11(a) and 9.5.1.11(b)). In addition, a license condition was placed in the original license to assure that these modifications would be completed on the schedule specified (license condition 2.C.13). By letter dated November 20, 1984, the staff was provided a status of the completion of the license condition. In regard to license condition 2.C.13, the licensee stated, "The fire protection program was implemented as specified in Sections 9.5.1.11(a) and (b) of Supplement No. 3 to the Safety Evaluation Report. This license condition has been satisfied." The removal of the license condition itself is the subject of a separate licensing action. The proposed removal of the two footnotes are acceptable on the basis of the licensee's statement that the fire-rated assemblies and flame impingement shields were installed and operable as required.

The licensee proposed to delete footnotes associated with Valve Tables 3.6-1 and 3.6-2 from the Unit No. 2 TS. Valve Table 3.6-1 is entitled, "Containment Leakage Paths." Valve Table 3.6-2 is entitled, "Containment Isolation Valves." These footnotes were placed in the Unit No. 2 TS by Amendment No. 10, issued on March 15, 1985, and were associated with valve changes to be made at a future date. The valves were part of the station air line associated with Containment Penetration No. 8. By letter dated March 13, 1987, the licensee advised the staff that the modification was complete. Thus, the footnotes are being removed and the currently correct valve tag numbers associated with Containment Penetration No. 8 are being incorporated into the body of the tables. Thus, the proposed changes are acceptable.

2.3 Evaluation - Correction of Typographical Errors

A number of typographical errors exist in the TS of both units. The licensee proposed that these be corrected. The Unit No. 1 TS typographical errors are contained on pages 3/4 5-4 and 3/4 7-8. The Unit No. 2 TS typographical errors are contained on pages 3/4 3-34, 3/4 3-39, 3/4 3-53, 3/4 3-57, 3/4 4-27, 3/4 5-6, 3/4 6-15, 3/4 8-1, 3/4 8-2, 3/4 11-6, 3/4 11-10, B3/4 0-3, B3/4 1-4, B3/4 2-1, and B3/4 3-4. The staff has reviewed each individual error as well as the licensee's rationale for correcting these errors and concludes that these corrections are acceptable.

2.4 Evaluation - Currently Correct Titles and Composition of the Company Nuclear Review Board

TS 6.5.2.2 for each unit contains the titles and composition of the Company Nuclear Review Board. The number of members changed from 8 to 10. The following members remain on the board:

Group Vice President - Nuclear Energy
Vice President - Nuclear Operations
Director - Quality Assurance
Chief Engineer - Power Plant Engineering
Manager - Nuclear Fuel
Senior Project Manager - Power Plant Engineering

The following members were added:

Group Vice President
Director - Nuclear Licensing
Manager - Nuclear Energy Services
Vice President - Engineering, Projects and Construction

The following two members were deleted:

Vice President, Advanced Systems and Technology
Power Plant Engineering Principal Engineer

It should be noted that the Vice President - Engineering, Projects and Construction is a higher official than the Power Plant Engineering Principal Engineer in the same FP&L organization. Therefore, the discipline area of the Power Plant Engineering Principal Engineer remains. As a result of the changes, the staff believes that a higher level of collective talents are now on the board and a higher quality of independent review and audit of designated activities should result. On this basis, the changes are acceptable.

2.5 Evaluation - Deletion of Specific NRC Addressees

The licensee proposes to delete specific NRC addressees from the TS for both units. The specific NRC addressees are the Regional Administrator and the Director of the Office of Resource Management. The addressees are contained on pages 6-15, 6-16 and 6-19 for the Unit No. 1 TS, and on pages 6-16, 6-17, and 6-20 for the Unit No. 2 TS. The licensee proposed to keep the words, ". . . submitted to the NRC." The licensees used a recent change to

10 CFR 50.4, effective January 5, 1987, as the basis for these changes. The licensee stated that the revised regulation directs that Part 50 reports be addressed to the Document Control Desk, Washington, D.C. 20555. Thus, when the TS state ". . . submit to the NRC," the licensee will send the report to the Document Control Desk. The proposed change by the licensee is acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments involve changes to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 amendments also involve changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 23, 1987

Principal Contributor:

E. Tourigny