

DCS MS014

AUG 18 1983

Docket No. 50-335

Dr. Robert E. Uhrig
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Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 59 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated March 15, 1979 as revised March 29, 1983.

The amendment revises the TS to require that releases of radioactive materials to unrestricted areas during normal operations, be kept as low as is reasonably achievable. The revised TS update those portions of the TS addressing radioactive waste management and make them consistent with current NRC positions.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular monthly Federal Register Notice.

Sincerely,

Original signed by

Donald E. Sells, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:

- 1. Amendment No. 59 to DPR-67
- 2. Safety Evaluation

cc w/enclosures:
See next page

Immediately Before it is issued check to see if there were any petitions or comments. If any come back to ERD

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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. 59
License No. DPR-67

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Florida Power & Light Company, (the licensee) dated March 15, 1979, as revised March 29, 1983, 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.

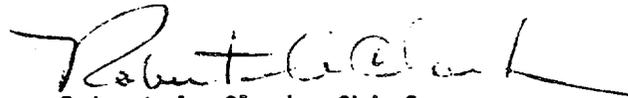
2. Accordingly, Facility Operating License No. DPR-67 is amended by changes to the Technical Specifications as indicated in the Attachment to this license amendment, and by amending paragraph 2.C(2) to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 59, 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



Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 18, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 59

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.

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DEFINITIONS

STAGGERED TEST BASIS

- 1.21 A STAGGERED TEST BASIS shall consist of:
- a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and
 - b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U}$$

$$Y_I = AY_E + B$$

UNRODDED PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.24 The UNRODDED PLANAR RADIAL PEAKING FACTOR is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding tilt.

SHIELD BUILDING INTEGRITY

- 1.25 SHIELD BUILDING INTEGRITY shall exist when:
- 1.25.1 Each door is closed except when the access opening is being used for normal transit entry and exit, and
 - 1.25.2 The shield building ventilation system is OPERABLE.

DEFINITIONS

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.27 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.28 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.29 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

LOAD FOLLOW OPERATION

1.30 LOAD FOLLOW OPERATION shall be daily power level changes of more than 10% of RATED THERMAL POWER or daily insertion of CEAs below the Long Term Insertion Limit.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

DEFINITIONS

MEMBER(S) OF THE PUBLIC

1.32 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.33 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and shall include the Radiological Environmental Sample point locations.

PROCESS CONTROL PROGRAM (PCP)

1.34 The PROCESS CONTROL PROGRAM shall contain the provisions, based on full scale testing, to assure that dewatering of spent bead resins results in a waste form with the properties that meet the requirements of 10 CFR Part 61 (as implemented by 10 CFR Part 20) and of the low level radioactive waste disposal site at the time of disposal.

PURGE - PURGING

1.35 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

SITE BOUNDARY

1.36 The SITE BOUNDARY shall be that line beyond which the land is neither owned, leased, nor otherwise controlled by the licensee.

SOURCE CHECK

1.37 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

TABLE 1.1
OPERATIONAL MODES

<u>MODE</u>	<u>REACTIVITY CONDITION, K_{eff}</u>	<u>%RATED THERMAL POWER*</u>	<u>AVERAGE COOLANT TEMPERATURE</u>
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 325^{\circ}\text{F}$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 325^{\circ}\text{F}$
3. HOT STANDBY	< 0.99	0	$\geq 325^{\circ}\text{F}$
4. HOT SHUTDOWN	< 0.99	0	$325^{\circ}\text{F} > T_{avg}$ $> 200^{\circ}\text{F}$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200^{\circ}\text{F}$
6. REFUELING**	≤ 0.95	0	$\leq 140^{\circ}\text{F}$

* Excluding decay heat.

**Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2
FREQUENCY NOTATION

<u>NOTATION</u>	<u>FREQUENCY</u>
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
4/M*	At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year
M	At least once per 31 days
Q	At least once per 92 days
SA	At least once per 184 days
R	At least once per 18 months
S/U	Prior to each reactor startup
P**	Completed prior to each release
N.A.	Not applicable

* For Radioactive Effluent Sampling
 ** For Radioactive Batch Releases Only

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ALARM SETPOINT</u>	<u>MEASUREMENT RANGE</u>	<u>ACTION</u>
1. AREA MONITORS					
a. Fuel Storage Pool Area	1	*	≤ 15 mR/hr	10 ⁻¹ - 10 ⁴ mR/hr	13
b. Containment (CIS)	3	6	≤ 90 mR/hr	1 - 10 ⁵ mR/hr	16
c. Containment Area - Hi Range	1	1, 2, 3 & 4	≤ 10 R/hr	1 - 10 ⁷ R/hr	15
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 ⁻⁷ - 10 ⁻² μCi/cc	14
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	1 - 10 ⁶ cpm/hr	14
b. Fuel Storage Pool Area Ventilation System					
i. Gaseous Activity	1	**	***	10 ⁻⁷ - 10 ⁵ μCi/cc	12
ii. Particulate Activity	1	**	***	1 - 10 ⁶ cpm	12

*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 Setpoints are determined and set in accordance with requirements of Specification 3.3.3.10.

TABLE 3.3-6 (Continued)
RADIATION MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ALARM SETPOINT</u>	<u>MEASUREMENT RANGE</u>	<u>ACTION</u>
2. PROCESS MONITORS (Continued)					
c. Noble Gas Effluent Monitors					
i. Radwaste Building Exhaust System (Plant Vent Exhaust Monitor)	1	1, 2, 3 & 4	***	$10^{-7} - 10^5$ $\mu\text{Ci/cc}$	15
ii. Steam Generator Blowdown Treatment Facility Building Exhaust System	1	1, 2, 3 & 4	***	$10^{-7} - 10^{-2}$ $\mu\text{Ci/cc}$	15
iii. Steam Safety Valve Discharge	1/Header	1, 2, 3 & 4	***	$10^{-1} - 10^3$ $\mu\text{Ci/cc}$	15
iv. ECCS Exhaust	1/Train	1, 2, 3 & 4	***	$10^{-7} - 10^5$ $\mu\text{Ci/cc}$	15

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

TABLE 3.3-6 (Continued)

TABLE NOTATION

- ACTION 12 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.
- ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.
- ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.
- ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:
- 1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and
 - 2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.
- ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>MODES IN WHICH SURVEILLANCE REQUIRED</u>
1. AREA MONITORS				
a. Fuel Storage Pool Area	S	R	M	*
b. Containment (CIS)	S	R	M	6
c. Containment Area - High Range	S	R	M	1, 2, 3 & 4
2. PROCESS MONITORS				
a. Fuel Storage Pool Area - Ventilation System				
i. Gaseous Activity	S	R	M	**
ii. Particulate Activity	S	R	M	**
b. Containment				
i. Gaseous Activity RCS Leakage Detection	S	R	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection	S	R	M	1, 2, 3 & 4

*With fuel in the storage pool or building.

**With irradiated fuel in the storage pool.

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>MODES IN WHICH SURVEILLANCE IS REQUIRED</u>
2. PROCESS MONITORS (Continued)				
c. Noble Gas Effluent Monitors				
i. Radwaste Building Exhaust System Plant Vent Monitor	S	R	M	1, 2, 3 & 4
ii. Steam Generator Blowdown Treatment Building Exhaust System	S	R	M	1, 2, 3 & 4
iii. Steam Safety Valve Discharge	S	R	M	1, 2, 3 & 4
iv. ECCS Exhaust	S	R	M	1, 2, 3 & 4

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION*

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.**

APPLICABILITY: At all times.

ACTION:

- a. With the number of OPERABLE meteorological monitoring channels less than required by Table 3.3-8, suspend all release of gaseous radioactive material from the radwaste gas decay tanks until the inoperable channel(s) is restored to OPERABLE status.
- b. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.
- c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

* The Meteorological Instrumentation system is shared between St. Lucie - Unit 1 and St. Lucie - Unit 2.

** The emergency power source may be inoperable in Modes 5 or 6.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

<u>INSTRUMENT & ELEVATION</u>	<u>INSTRUMENT MINIMUM ACCURACY</u>	<u>MINIMUM CHANNELS OPERABLE</u>
1. WINDSPEED		
a) Nominal Elev (10 meters)	± 0.5 *mph	1 ^A
b) Nominal Elev (57.9 meters)	± 0.5 *mph	N.A.
2. WIND DIRECTION		
a) Nominal Elev (10 meters)	± 5°	1 ^B
b) Nominal Elev (57.9 meters)	± 5°	N.A.
3. AIR TEMPERATURE (Delta T)		
a) Nominal Elev (10 meters)	± 0.1°C**	1 ^C
b) Nominal Elev (57.9 meters)	± 0.1°C**	1 ^C
c) Nominal Elev (33.5 meters)	± 0.1°C**	N.A.

* Starting speed of anemometer shall be < 1 mph.

** ΔT measurement channels only.

^A The 57.9-meter channel may be substituted for the 10-meter wind speed for up to 30 days in the event the 10-meter channel is inoperable. Wind speed data from the 57.9-meter elevation should be adjusted using the wind speed power law:

$$S_{10 \text{ meters}} = S_{57.9 \text{ meters}} (0.1727)^n$$

where:

S = wind speed in mph

n = 0.25 for Pasquill Vertical Stability Classes A,B,C,D.

n = 0.50 for Pasquill Vertical Stability Classes E,F,G.

1.727×10^{-1} = constant = 10 meters/57.9 meters

^B The 57.9-meter channel may be substituted for the 10-meter wind direction channel for up to 30 days in the event the 10-meter channel is inoperable.

^C The 33.5-meter channel may be substituted for one of the 10-meter or 57.9-meter temperature channels for up to 30 days if one of the channels is inoperable. The data should always be normalized to °C/100 meters to determine the vertical stability class.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL</u> <u>CHECK</u>	<u>CHANNEL</u> <u>CALIBRATION</u>
1. WIND SPEED		
a) Nominal Elev (10 meters)	D	SA
b) Nominal Elev (57.9 meters)	D*	SA*
2. WIND DIRECTION		
a) Nominal Elev (10 meters)	D	SA
b) Nominal Elev (57.9 meters)	D*	SA*
3. AIR TEMPERATURE (DELTA T)		
a) Nominal Elev (10 meters)	D	SA
b) Nominal Elev (57.9 meters)	D.	SA
c) Nominal Elev (33.5 meters)	D*	SA*

*Required only if these channels are being substituted for one of the Minimum Channels Operable as per Table 3.3-8.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

- a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.
- b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. 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, 3.0.4 and 6.9.1.9 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid 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 4.3-8.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>ACTION</u>
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	1	1
b. Steam Generator Blowdown Effluent Line	1/SG	2
2. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	NA	3
b. Discharge Canal	NA	3
c. Steam Generator Blowdown Effluent Lines	NA	3

TABLE 3.3-12 (Continued)

TABLE NOTATION

- ACTION 1 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:
- a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and
 - b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving (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 grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 2×10^{-7} microcuries/gram:
- a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram DOSE EQUIVALENT I-131.
 - b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131.
- ACTION 3 - Minimum system design flow of required running pumps shall be utilized for MPC calculations for discharge canal flow and maximum system design flow shall be utilized for MPC calculations for effluent line flow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>SOURCE CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	P	R(2)	Q(1)
b. Steam Generator Blowdown Effluent Line	D	M	R(2)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line	D	N.A.	R	Q

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

- (1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:
 1. Instrument indicates measured levels above the alarm/trip setpoint.
 2. Circuit failure.
 3. Instrument indicates a downscale failure.
 4. Instrument controls not set in operate mode.

- (2) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards traceable to the National Bureau of Standards or using standards that have been calibrated against standards certified by the NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities that are typical of normal plant operation. For subsequent CHANNEL CALIBRATION, button sources that have been related to the initial calibration may be used.

INSTRUMENTATION

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, 3.0.4, and 6.9.1.9 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

	<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABILITY</u>	<u>ACTION</u>
1.	WASTE GAS DECAY TANKS			
a.	Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	1
2.	WASTE GAS DECAY TANKS EXPLOSIVE GAS MONITORING SYSTEM			
a.	Oxygen Monitors	1	**	5
3.	CONDENSER EVACUATION SYSTEM			
a.	Noble Gas Activity Monitor	1	***	3
4.	PLANT VENT SYSTEM			
a.	Noble Gas Activity Monitor (Low Range)	1	*	3
b.	Iodine Sampler	1	*	6
c.	Particulate Sampler	1	*	6
d.	Flow Rate	NA	*	4
e.	Sampler Flow Rate Monitor	1	*	2

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABILITY</u>	<u>ACTION</u>
5. FUEL STORAGE AREA VENTILATION SYSTEM			
a. Noble Gas Activity Monitor (Low Range)	1	*	3
b. Iodine Sampler	1	*	6
c. Particulate Sampler	1	*	6
d. Flow Rate Monitor	NA	*	4
e. Sampler Flow Rate Monitor	1	*	2
6. STEAM GENERATOR BLOWDOWN BUILDING VENT SYSTEM			
a. Noble Gas Activity Monitor	1	*	3
b. Iodine Sampler	1	*	6
c. Particulate Sampler	1	*	6
d. Flow Rate Monitor	NA	*	4
e. Sampler Flow Rate Monitor	1	*	2

TABLE 3.3-13 (Continued)

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

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>SOURCE CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>MODES IN WHICH SURVEILLANCE REQUIRED</u>
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(5)	M	^^
b. Oxygen Monitor (alternate)	D	N.A.	Q(5)	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

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>SOURCE CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>MODES IN WHICH SURVEILLANCE REQUIRED</u>
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. 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 4.3-9 (Continued)

TABLE NOTATION

* At all times other than when the line is valved out and locked.

** During waste gas holdup system operation.

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

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards traceable to the National Bureau of Standards or using standards that have been calibrated against standards certified by the NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities that are typical of normal plant operation. For subsequent CHANNEL CALIBRATION, button sources that have been related to the initial calibration may be used, at intervals of at least once per 18 months.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1. Also, results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1.

4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance with Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ($\mu\text{Ci/ml}$) ^a
A. Batch Waste Release Tanks ^c	P Each Batch	P Each Batch	Principal Gamma Emitters ^e	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
			P Each Batch	M Composite ^b
	Gross Alpha	1×10^{-7}		
	P Each Batch	Q Composite ^b	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-5}
	B. Continuous Releases ^{dg}	Daily	4/M Composite	Principal Gamma Emitters ^e
I-131				1×10^{-5}
D Grab Sample Daily		4/M Composite	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
			Daily	M Composite
Gross Alpha		1×10^{-7}		
Daily		Q Composite	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
C. Settling Basin ⁱ		W Grab Sample	W	Principal Gamma Emitters ^e
	I-131			1×10^{-5}

TABLE 4.11-1 (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 4.11-1 (Continued)

TABLE NOTATION

- b. A composite sample is one in which the quantity of liquid samples is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.
- c. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.
- d. A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.
- e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.
- f. Grab samples to be taken when there is confirmed primary to secondary system leakage indicated by the air ejector monitor indicating $\geq 2x$ background.
- g. If Component Cooling Water activity is $>1 \times 10^{-5}$ $\mu\text{Ci/ml}$, perform a weekly gross activity on the Intake Cooling Water System outlet to ensure the activity level is \leq a 2×10^{-7} $\mu\text{Ci/ml}$ LLD limit. If ICW is $>2 \times 10^{-7}$ $\mu\text{Ci/ml}$, perform analysis in accordance with a Plant Continuous Release on this Table.

RADIOACTIVE EFFLUENTS

DOSE

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, in lieu of any other report required by Specification 6.9.1, 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.

RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT

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, in lieu of any other report required by Specification 6.9.1, 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.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in UNRESTRICTED AREAS due to radioactive materials released in gaseous effluents from the site (see Figure 5.1-1) shall be limited to the following:

- a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and
- b. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ^a ($\mu\text{Ci/ml}$)
A. Waste Gas Storage Tank	^P Each Tank Grab Sample	^P Each Tank	Principal Gamma Emitters ^e	1×10^{-4}
B. Containment Purge	^P Each Purge ^b Grab Sample	^P Each Purge ^b	Principal Gamma Emitters ^e	1×10^{-4}
			H-3	1×10^{-6}
C. (1) Plant Vent (2) Fuel Building Vent (3) Steam Generator Blowdown Building Vent	4/M Grab Sample	4/M	Principal Gamma Emitters ^e	1×10^{-4}
			H-3	1×10^{-6}
D. All Release Types as listed in A,B, C above.	Continuous ^d	4/M ^c Charcoal Sample	I-131	1×10^{-12}
	Continuous ^d	4/M ^c Particulate Sample	Principal Gamma Emitters ^e (I-131, Others)	1×10^{-11}
	Continuous ^d	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous ^d	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous ^d	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

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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.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

- a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation and,
- b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION

- a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, 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 the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit to areas at and beyond the SITE BOUNDARY, (see Figure 5.1-1) shall be limited to the following:

- a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,
- b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

- a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the GASEOUS RADWASTE TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site to UNRESTRICTED AREAS (see Figure 5.1-1), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-1) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

- a. With the GASEOUS RADWASTE TREATMENT SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, 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 the 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

4.11.2.4.1 Doses due to gaseous releases from the site to UNRESTRICTED AREAS shall be projected at least once per 31 days, in accordance with the ODCM unless the GASEOUS RADWASTE TREATMENT SYSTEM is being used.

4.11.2.4.2 The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 30 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

- a. With the concentration of oxygen in the waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.
- b. With the concentration of oxygen in the waste gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.
- c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the waste gas decay tank shall be determined to be within the above limits by continuously monitoring the waste gases in the on service waste gas decay tank with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 285,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

- a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, and provide prompt notification to the Commission.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank when reactor coolant system activity exceeds $\frac{100}{E}$.

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive bead resins shall be dewatered, as appropriate, in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

- a. With dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately dewatered bead resin and correct the PROCESS CONTROL PROGRAM, the procedures and/or the dewatering system as necessary to prevent recurrence.
- b. With dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) if the dewatered bead resin has not already been shipped for disposal, verify each container to ensure that it meets burial ground and shipping requirements and (2) take appropriate administrative action to prevent recurrence.
- c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 Prior to disposal, each container of radioactive bead resins shall be tested for free standing liquids in accordance with the PROCESS CONTROL PROGRAM to assure that it meets shipping, transportation, and disposal site requirements.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

APPLICABILITY: At all times.

ACTION:

- a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a, 3.11.1.2b, 3.11.2.2a, 3.11.2.2b, 3.11.2.3a or 3.11.2.3b, calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels of concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Surveillance Requirements 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This Surveillance Requirement shall be required only in the event the above Action a. requires the applicable calculations.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

- a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.11, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.
- b. With the confirmed* level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \dots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report shall include the methodology for calculating the cumulative potential dose contributions for the calendar year from radionuclides detected in environmental samples and can be determined in accordance with the methodology and parameters in the ODCM. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

- c. With milk or broadleaf vegetation samples unavailable from one or more of the sample locations listed by Table 3.12-1, identify locations

*A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis but in any case within 30 days.

RADIOLOGICAL ENVIRONMENTAL MONITORING

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.10, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

- d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM^{a)}

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations^{b) c)}</u>	<u>Sampling and Collection Frequency^{d)}</u>	<u>Type and Frequency^{d)} of Analysis</u>
1. DIRECT RADIATION ^{e)}	27 Monitoring Locations	Continuous monitoring with sample collection quarterly	Gamma exposure rate - quarterly.
2. AIRBORNE Radioiodine and Particulates	5 Locations	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading	<u>Radioiodine Filter:</u> I-131 analysis weekly. <u>Particulate Filter:</u> Gross beta radioactivity analysis \geq 24 hours following a filter change ^{g)} Gamma isotopic ^{h)} analysis of composite ^{j)} (by location) quarterly
3. WATERBORNE			
a. Surface ^{h)}	1 Location ⁱ⁾	Weekly	Gamma isotopic ^{h)} & tritium analyses weekly
	1 Location ^{j)}	Monthly	Gamma isotopic ^{h)} & tritium analyses monthly
b. Sediment from shoreline	2 Locations	Semiannually	Gamma isotopic ^{h)} analysis semiannually

TABLE 3.12-1 (Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM^{a)}

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations^{b) c)}</u>	<u>Sampling and Collection Frequency^{d)}</u>	<u>Type and Frequency^{d)} of Analysis</u>
4. INGESTION			
a. Fish and Invertebrates			
1. Crustacea	2 Locations	Semiannually	Gamma isotopic ^{h)} analyses semiannually
2. Fish	2 Locations	Semiannually	Gamma isotopic ^{h)} analyses semiannually
b. Food Products			
1. Broad leaf vegetation	3 Locations ¹⁾	Monthly when available	Gamma isotopic ^{h)} and I-131 analyses monthly

TABLE 3.12-1 (continued)

TABLE NOTATION

- a. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment or other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction corrective action shall be taken prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.11.1.
- b. Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each sample location in Table 3.12-1 in a Table and figure(s) in the ODCM.
- c. At times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program.
- d. The following definition of frequencies shall apply to Table 3.12-1 only:
- Weekly - Not less than once per calendar week. A maximum interval of 11 days is allowed between the collection of any two consecutive samples.
- Semi-Monthly - Not less than 2 times per calendar month with an interval of not less than 7 days between sample collections. A maximum interval of 24 days is allowed between collection of any two consecutive samples.
- Monthly - Not less than once per calendar month with an interval of not less than 10 days between sample collections.
- Quarterly - Not less than once per calendar quarter.
- Semiannually - One sample each between calendar dates (January 1 - June 30) and (July 1 - December 31). An interval of not less than 30 days will be provided between sample collections.
- The frequency of analyses is to be consistent with the sample collection frequency.
- e. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters.

TABLE 3.12-1 (Continued)

TABLE NOTATION

- f. Refers to normal collection frequency. More frequent sample collection is permitted when conditions warrant.
- g. Airborne particulate sample filters are analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. In addition to the requirement for a gamma isotopic on a composite sample a gamma isotopic is also required for each sample having a gross beta radioactivity which is $> 1.0 \text{ pCi/m}^3$ and which is also > 10 times that of the most recent control sample.
- h. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.
- i. Atlantic Ocean, in the vicinity of the public beaches along the eastern shore of Hutchinson Island near the St. Lucie Plant (grab sample).
- j. Atlantic Ocean, at a location beyond influence from plant effluents (grab sample).
- k. Discharges from the St. Lucie Plant do not influence drinking water or ground water pathways.
- l. Samples of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q, and one sample of similar broad leaf vegetation at an available location 15-30 km distant in the least prevalent wind direction based upon historical data in the ODCM.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m ³)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	30,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95***	400				
I-131	2**	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140***	200			300	

* Since no drinking water pathway exists, a value of 30,000 pCi/l is used. For drinking water samples, a value of 20,000 pCi/l is used. This is 40 CFR, Part 141 value.

** Applies to drinking water.

*** An equilibrium mixture of the parent and daughter isotopes which corresponds to the reporting value of the parent isotope.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS^aLOWER LIMIT OF DETECTION (LLD)^{b,c}

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m ³)	Fish (pCi/kg,wet)	Milk (pCi/l)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	0.01				
H-3	3000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15 ^e					
I-131	1 ^d	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15 ^e			15 ^e		

* Since no drinking water pathway exists, a value of 3000 pCi/l may be used. For drinking water samples, a value of 2000 pCi/l is used.

TABLE 4.12-1 (Continued)

TABLE NOTATION

^aThis list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

^bRequired detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

^cThe 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 4.12-1 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

^d LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

^e An equilibrium mixture of the parent and daughter isotopes which corresponds to 15 pCi/l of the parent isotope.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m²(500 ft²) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

- a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.10.
- b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.10, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).
- c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

*Broad leaf vegetation sampling may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4b shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.*

APPLICABILITY: At all times.

ACTION:

- a. With analyses not being performed as required above, report the corrective actions to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

*This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA).

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The Safety Injection Actuation Signal (SIAS) provides direct actuation of the Containment Isolation Signal (CIS) to ensure containment isolation in the event of a small break LOCA.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served

INSTRUMENTATION

BASES

RADIATION MONITORING INSTRUMENTATION (Continued)

by the individual channels; and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) 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 Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

INSTRUMENTATION

BASES

3/4.3.3.6 DELETED

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 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 during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

INSTRUMENTATION

BASES

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT 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 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.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A. design objectives of Appendix I, 10 CFR Part 50, to an individual, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," March 1976 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

RADIOACTIVE EFFLUENTS

BASES

3/4.11.1.3 LIQUID WASTE TREATMENT

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrem/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

RADIOACTIVE EFFLUENTS

BASES

This specification applies to the release of gaseous effluents from all reactors at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", March 1976, and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors", Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions or real atmospheric conditions.

This specification applies to the release of gaseous effluents from each reactor at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I

RADIOACTIVE EFFLUENTS

BASES

be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", March 1976, and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric or real meteorological conditions. The release rate specifications for Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

This specification applies to the release of gaseous effluents from each reactor at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

RADIOACTIVE EFFLUENTS

BASES

3/4.11.2.6 GAS STORAGE TANKS

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure."

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance relates only to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry", Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m² provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity 26 kg/year of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m².

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area is shown on Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone is shown on Figure 5.1-1.

FLOOD CONTROL

5.1.3 The flood control provisions (dunes and slope protection) shall be designed and maintained in accordance with the original design provisions contained in Section 2.4.2.2 of the FSAR.

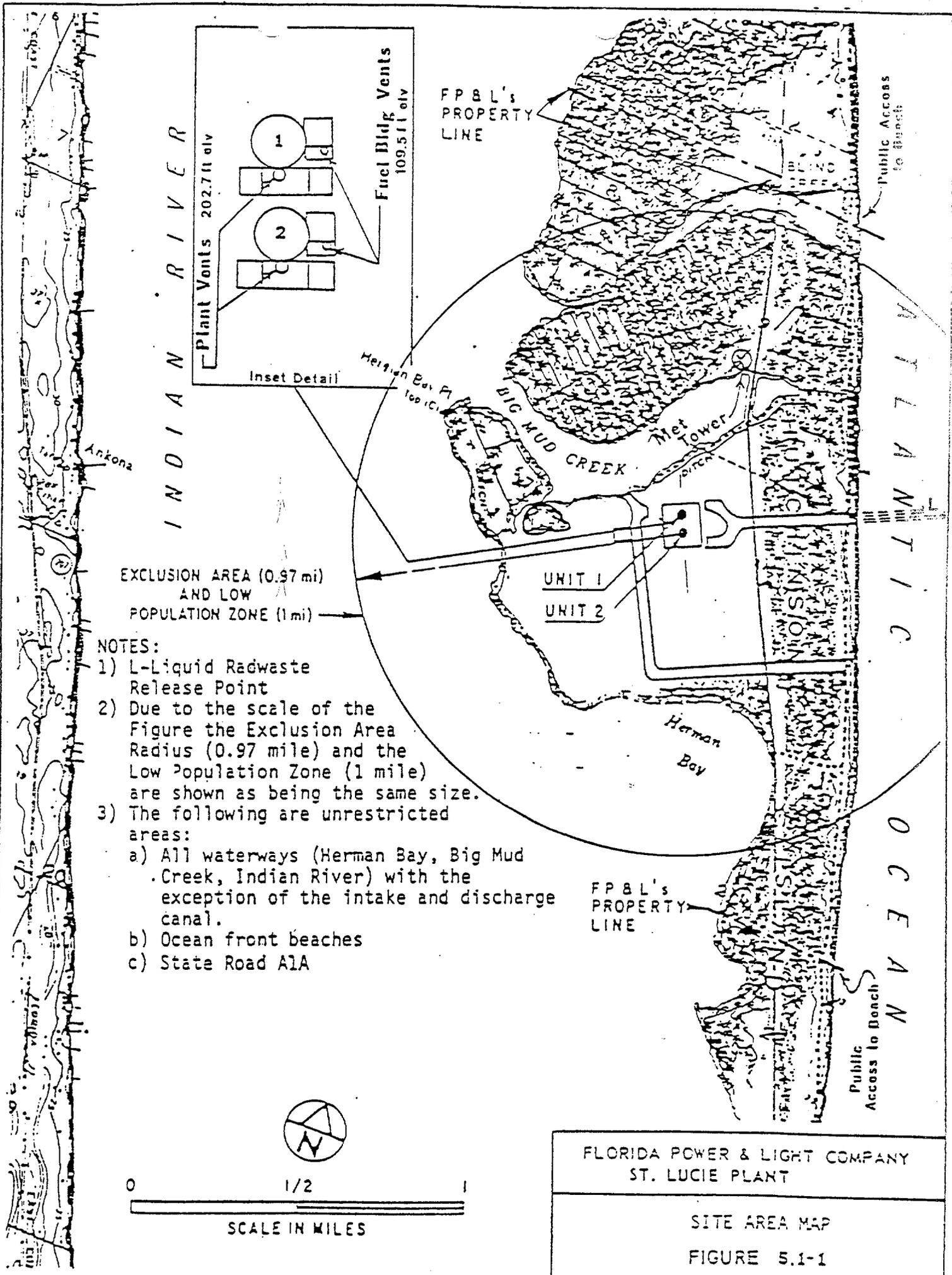
5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel, having the shape of a right circular cylinder with a hemispherical dome and ellipsoidal bottom, surrounded by a reinforced concrete shield building. The radius of the shield building is at least 4 feet greater than the radius of circular cylinder portion of the containment vessel at any point.

5.2.1.1 CONTAINMENT VESSEL

- a. Nominal inside diameter = 140 feet.
- b. Nominal inside height = 232 feet.
- c. Net free volume = 2.5×10^6 cubic feet.
- d. Nominal thickness of vessel walls = 2 inches.
- e. Nominal thickness of vessel dome = 1 inch.
- f. Nominal thickness of vessel bottom = 2 inches.



EXCLUSION AREA (0.97 mi)
AND LOW
POPULATION ZONE (1 mi)

NOTES:

- 1) L-Liquid Radwaste Release Point
- 2) Due to the scale of the Figure the Exclusion Area Radius (0.97 mile) and the Low Population Zone (1 mile) are shown as being the same size.
- 3) The following are unrestricted areas:
 - a) All waterways (Herman Bay, Big Mud Creek, Indian River) with the exception of the intake and discharge canal.
 - b) Ocean front beaches
 - c) State Road A1A

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

SITE AREA MAP
FIGURE 5.1-1

FIGURE 5.1-2

(Deleted)

DESIGN FEATURES

5.2.1.2 SHIELD BUILDING

- a. Minimum annular space = 4 feet.
- b. Annulus nominal volume = 543,000 cubic feet.
- c. Nominal outside height (measured from top of foundation base to the top of the dome) = 230.5 feet.
- d. Nominal inside diameter = 148 feet.
- e. Cylinder wall minimum thickness = 3 feet.
- f. Dome minimum thickness = 2.5 feet.
- g. Dome inside radius = 112 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment vessel is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

PENETRATIONS

5.2.3 Penetrations through the containment structure are designed and shall be maintained in accordance with the original design provisions contained in Sections 3.8.2.1.10 and 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 2250 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

5.3.2 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

ADMINISTRATIVE CONTROLS

- e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager of Power Resources Nuclear, the Vice President of Power Resources and to the Chairman of the Company Nuclear Review Board.
- f. Review of those REPORTABLE OCCURRENCES requires 24 hour notification to the Commission.
- g. Review of facility operations to detect potential safety hazards.
- h. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Company Nuclear Review Board.
- i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Company Nuclear Review Board.
- j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Company Nuclear Review Board.
- 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 Energy 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.

AUTHORITY

6.5.1.7 The Facility Review Group shall:

- a. Recommend to the Plant Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.
- b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.
- c. Provide written notification within 24 hours to the Vice President of Power Resources 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 6.1.1 above.

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RECORDS

6.5.1.8 The Facility Review Group shall maintain written minutes of each meeting and copies shall be provided to the Vice President of Power Resources and 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
- e. instrumentation and control
- f. radiological safety
- g. mechanical and electrical engineering
- h. quality assurance practices

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AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

- a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.
- b. The performance, training and qualifications of the entire facility staff at least once per 12 months.
- c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.
- d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.
- e. Any other area of unit operation considered appropriate by the CNRB or the Executive Vice President.
- f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.
- g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection specialist or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.
- h. The radiological environmental monitoring program and the results thereof at least once per 12 months.
- i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.
- j. The PROCESS CONTROL PROGRAM and implementing procedures for dewatering of radioactive bead resin at least once per 24 months.

AUTHORITY

6.5.2.9 The CNRB shall report to and advise the Executive Vice President on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

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RECORDS

6.5.2.10 Records of CNRB activities shall be prepared, approved and distributed as indicated below:

- a. Minutes of each CNRB meeting shall be prepared, approved and forwarded to the Executive Vice President within 14 days following each meeting.
- b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Executive Vice President within 14 days following completion of the review.
- c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Executive Vice President and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

- a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.
- b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the FRG and submitted to the CNRB and the Vice President of Power Resources.

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6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

- a. The facility shall be placed in at least HOT STANDBY within one hour.
- b. The Safety Limit violation shall be reported to the Commission, the Vice President of Power Resources and to the CNRB within 24 hours.
- c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRG. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.
- d. The Safety Limit Violation Report shall be submitted to the Commission, the CNRB and the Director of Power Resources within 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972. .
- b. Refueling operations.
- c. Surveillance and test activities of safety related equipment.
- d. Security Plan implementation.
- e. Emergency Plan implementation.
- f. Fire Protection Program implementation.
- g. PROCESS CONTROL PROGRAM implementation.
- h. OFFSITE DOSE CALCULATION MANUAL implementation.
- i. Quality Control Program for effluent monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974.
- j. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each procedure of Specification 6.8.1a through i. above, and changes thereto, shall be reviewed by the FRG and shall be approved by the Plant Manager prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

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6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

- a. The intent of the original procedure is not altered.
- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.
- c. The change is documented, reviewed by the FRG and approved by the Plant Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the application reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

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

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- d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any planned criticality.
- e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.
- f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.
- g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by technical specifications.
- h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.
- i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.
- j. Offsite releases of radioactive materials in liquid and gaseous effluents that exceed the limits of Specification 3.11.1.1 or 3.11.2.1.
- k. Exceeding the limits in Specification 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written followup report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

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THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

- a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.
- b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.
- c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.
- d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.
- e. An unplanned offsite release of (1) more than 1 curie of radioactive material in liquid effluents, (2) more than 150 curies of noble gas in gaseous effluents, or (3) more than 0.05 curie of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:
 1. A description of the event and equipment involved.
 2. Cause(s) for the unplanned release.
 3. Actions taken to prevent recurrence.
 4. Consequences of the unplanned release.

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SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.10 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 from 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 meteorological 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.

* 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.

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SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

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 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 off-site 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*

6.9.1.11 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 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

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

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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.

6.9.1.12 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 Regional Administrator of the Regional Office of the NRC.

6.9.1.13 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 Regional Administrator of the Regional Office of the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

- a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

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

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SPECIAL REPORTS (Continued)

- b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.
- c. Inservice Inspection Program Reviews, Specifications 4.4.10.1 and 4.4.10.2.
- d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.
- e. Sealed Source leakage in excess of limits, Specification 4.7.9.1.3.
- f. Seismic event analysis, Specification 4.3.3.3.2.
- g. Beach survey results, Specification 4.7.6.1.1.
- h. Core Barrel Movement, Specifications 3.4.11 and 4.4.11.
- i. Fire Detection Instrumentation, Specification 3.3.3.7.
- j. Fire Suppression Systems, Specification 3.7.11.1.
- k. Radiological Effluent, Specifications 3.11.1.2, 3.11.1.3, 3.11.2.2, 3.11.2.3, 3.11.2.4, 3.11.4, and 3.12.1.

6.10 RECORD RETENTION

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

- a. Records and logs of facility 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 OCCURRENCES submitted to the Commission.
- d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
- e. Records of reactor tests and experiments.
- f. Records of changes made to Operating Procedures.
- g. Records of radioactive shipments.
- h. Records of sealed source leak tests and results.
- i. 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 Facility Operating License:

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

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- b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
- c. Records of facility radiation and contamination surveys.
- 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 facility components identified in Table 5.9-1.
- g. Records of training and qualification for current members of the plant staff.
- h. Records of in-service inspections performed pursuant to these Technical Specifications.
- i. Records of Quality Assurance activities required by the QA Manual.
- j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
- k. Records of meetings of the FRG and the CNRB.
- l. Records of Environmental Qualification which are covered under the provisions of paragraph 6.13.
- m. Records of the service lives of all hydraulic and mechanical snubbers listed on Tables 3.7-2a and 3.7-2b including the date at which the service life commences and associated installation and maintenance records.
- n. Annual Radiological Environmental Operating Reports; and records of analyses transmitted to the licensee which are used to prepare the Annual Radiological Environmental Monitoring Report.
- o. Meteorological data, summarized and reported in a format consistent with the recommendation of Regulatory Guides 1.21 and 1.23.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

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6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance there-to shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

- a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.
- c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

**Measurement made at 18 inches from source of radioactivity.

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6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979.

Copies of these documents are attached to Order for Modification of License DPR-67 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 SYSTEMS INTEGRITY

6.14.1 The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and
2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.15 IODINE MONITORING

6.15.1 The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

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6.16 BACKUP METHOD FOR DETERMINING SUBCOOLING MARGIN

6.16.1 The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and
2. Procedures for monitoring.

6.17 PROCESS CONTROL PROGRAM (PCP)

6.17.1 The PCP shall be approved by the FRG prior to implementation.

6.17.2 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.

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6.18 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.18.1 The ODCM shall be approved by the Commission prior to implementation.

6.18.2 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.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. DPR-67

ST. LUCIE PLANT, UNIT NO. 1

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, the Florida Power & Light Company has filed with the Commission plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable. The Florida Power & Light Company filed this information with the Commission by letter dated March 29, 1983, which requested changes to the Technical Specifications appended to Facility Operating License No. DPR-67 for St. Lucie Plant, Unit No. 1 (St. Lucie 1). The proposed technical specifications update those portions of the technical specifications addressing radioactive waste management and make them consistent with the current staff positions as expressed in NUREG-0472. These revised technical specifications would reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR 50.36a, as supplemented by Appendix I to 10 CFR Part 50, with 10 CFR 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64; and with 10 CFR Part 50, Appendix B.

2.0 BACKGROUND AND DISCUSSION

2.1 Regulations

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Section 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors", provides that each license authorizing operation of a nuclear power reactor will include technical specifications that (1) require compliance with applicable provisions of 10 CFR 20.106, "Radioactivity in Effluents to Unrestricted Areas"; (2) require that operating procedures developed for the control of effluents be established and followed; (3) require that equipment installed in the radioactive waste system be maintained and used; and (4) require the periodic submission of reports to the NRC specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents, any quantities of radioactive materials released that are significantly above design objectives, and such other information as may be required by the Commission to estimate maximum potential radiation dose to the public resulting from the effluent releases.

10 CFR Part 20, "Standards for Protection Against Radiation," Sections 20.105(c), 20.106(g), and 20.405(c), require that nuclear power plant and other licensees comply with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations" and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded.

10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants - contains Criterion 60, Control of releases of radioactive materials to the environment; Criterion 63, Monitoring fuel and waste storage; and Criterion 64, Monitoring radioactivity releases. Criterion 60 requires that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

Criterion 63 requires that appropriate systems be provided in radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. Criterion 64 requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

10 CFR Part 50, Appendix B, establishes quality assurance requirements for nuclear power plants.

10 CFR Part 50, Appendix I, Section IV, provides guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.

2.2 Standard Radiological Effluent Technical Specifications

NUREG-0472 provides radiological effluent technical specifications for pressurized water reactors which the staff finds to be an acceptable standard for licensing actions. Further clarification of these acceptable methods is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." NUREG-0133 describes methods found acceptable to the staff of the NRC for the calculation of certain key values required in the preparation of proposed radiological effluent technical specifications for light-water-cooled nuclear power plants. NUREG-0133 also provides guidance to licensees in preparing requests for changes to existing radiological effluent technical specifications for operating reactors. It also describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment systems.

The above NUREG documents address all of the radiological effluent technical specifications needed to assure compliance with the guidance and requirements provided by the regulations previously cited. However, alternative approaches to the preparation of radiological effluent technical specifications and alternative radiological effluent technical specifications may be acceptable if the staff determines that the alternatives are in compliance with the regulations and with the intent of the regulatory guidance.

The standard radiological effluent technical specifications can be grouped under the following categories:

- (1) Instrumentation
- (2) Radioactive effluents
- (3) Radiological environmental monitoring
- (4) Design features
- (5) Administrative controls.

Each of the specifications under the first three categories is comprised of two parts: the limiting condition for operation and the surveillance requirements. The limiting condition for operation provides a statement of the limiting condition, the times when it is applicable, and the actions to be taken in the event that the limiting condition is not met.

In general, the specifications established to assure compliance with 10 CFR Part 20 standards provide, in the event the limiting conditions of operation are exceeded, that without delay conditions are restored to within the limiting conditions. Otherwise, the facility is required to effect approved shutdown procedures. In general, the specifications established to assure compliance with 10 CFR Part 50 provide, in the event the limiting conditions of operation are exceeded, that within specified times corrective actions are to be taken, alternative means of operation are to be employed, and certain reports are to be submitted to the NRC describing these conditions and actions.

The specifications concerning design features and administrative controls contain no limiting conditions of operation or surveillance requirements.

Table 1 indicates the standard radiological effluent technical specifications that are needed to assure compliance with the particular provisions of the regulations described in Section 1.0.

3.0 EVALUATION

The attached Technical Evaluation Report EG&G-PHYS-6241 (TER) was prepared for the staff by EG&G Idaho, Inc. (EG&G) as part of the NRC technical assistance contract program. This report provides the technical evaluation of the compliance of the licensee's submittal with NRC provided criteria. The staff has reviewed this TER and agrees with the evaluation.

3.1 SAFETY CONCLUSIONS

The proposed radiological effluent technical specifications for St. Lucie 1 have been reviewed, evaluated, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0472 and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications.

The proposed changes will not remove or relax any existing requirement related to the probability or consequences of accidents previously

considered or needed to provide reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

4.0 ENVIRONMENTAL CONSIDERATION

The staff has determined that the issuance of the proposed amendment to the Technical Specifications appended to Facility Operating License No. DPR-67 for St. Lucie 1 would not authorize a significant change in the types, or a significant increase in the amounts, of effluents or in the authorized power level, and that the amendment will not result in any significant environmental impact. Having made these determinations, the staff has further concluded that the amendment involves an action that is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 GENERAL 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: **AUG 18 1983**

Attachment: TER

Principal Contributors:

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