FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT
CHEMISTRY PROCEDURE NO. C-200
REVISION 14

1.0 TITLE:

OFFSITE DOSE CALCULATION MANUAL (ODCM)

2.0 REVIEW AND APPROVAL:

Reviewed by Facility Review Group ____ April 22, 1982

Approved by C. M. Wethy Plant General Manager April 27, 1982

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INTRODUCTION

The ODCM consists of the Controls Section followed by the Methodology Section.

The Controls Section provides the Control Statements, Limits, ACTION Statements, Surveillance Requirements, and BASES for ensuring that Radioactive Liquid and Gaseous Effluents released to UNRESTRICTED AREAS and/or the SITE BOUNDARY will be maintained within the requirements of 10 CFR Part 20, 40 CFR Part 190, 10 CFR 50.36.a, and 10 CFR Part 50 Appendix-I radioactive release criteria. All Control Statements and most Administrative Control Statements in the ODCM are directly tied to, and reference the Plant Technical Specification (TS) Administrative Section. The Administrative Control for Major Changes to Radioactive Liquid, Gaseous and Solid Treatment Systems is as per the guidance of NUREG-1301, April 1991, Supplement No. 1 to NRC Generic Letter 89-01. The numbering sequences of Control Statements also follow the guidance of NUREG-1301 as applicable, to minimize differences.

The Methodology Section uses the models suggested by NUREG-0133, November, 1978, and Regulatory Guide 1.109 to provide calculation methods and parameters for determining results in compliance with the Controls Section of the ODCM. Simplifying assumptions have been applied where applicable to provide a more workable document for implementing the Control requirements. Alternate calculation methods may be used from those presented as long as the overall methodology does not change or as long as most up-to-date revisions of the Regulatory Guide 1.109 dose conversion factors and environmental transfer factors are substituted for those currently included and used in this document.

RECORDS AND NOTIFICATIONS

All records of reviews performed for changes to the ODCM shall be maintained in accordance with QI 17-PR/PSL-1. All FRG approved changes to the ODCM, with required documentation of the changes per TS 6.14, shall be submitted to the NRC in the Annual Effluent Release Report. Procedures that directly implement, administer or supplement the requirements of the ODCM Controls and Surveillances are:

C-01 Schedule for Periodic Test
C-02 Schedule for Test Calibrations
C-70 Processing Aerated Liquid Waste
C-72 Processing Gaseous Wastes
The Radiological Environmental Monitoring Program is performed by the State of Florida as per FPL Juno Nuclear Operations Corporate Environmental Procedures.

CONTROLS

AND

SURVEILLANCE REQUIREMENTS

1.0 DEFINITIONS for CONTROLS SECTION OF ODCM

The defined terms of this section appear in capitalized type and are applicable throughout these Controls.

ACTION

1.1 ACTION shall be that part of a Control that prescribes remedial measures required under designated conditions.

CHANNEL FUNCTIONAL TEST

1.3 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The CHANNEL FUNCTIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the setpoints are within the required range and accuracy.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

1.0 DEFINITIONS for CONTROLS SECTION OF ODCM

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in [Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977].

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

MEMBER (S) OF THE PUBLIC

1.16 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

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by TS section 6.8.4g and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by TS 6.9.1.7 and 6.9.1.8.

1.0 DEFINITIONS for CONTROLS SECTION OF ODCM

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2 of the St. Lucie Plant TS.

PURGE - PURGING

1.23 PURGE or PURGING shall be any 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.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

1.0 DEFINITIONS for CONTROLS SECTION OF ODCM

SITE BOUNDARY

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

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

THERMAL POWER

1.35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

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.

VENTILATION EXHAUST TREATMENT SYSTEM

1.39 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

1.0 DEFINITIONS for CONTROLS SECTION OF ODCM

VENTING

1.40 VENTING shall be 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 not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

WASTE GAS HOLDUP SYSTEM

1.41 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

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TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
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 minimum of 48 per year.
M	At least once per 31 days.
Q	At lease 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.
N.A.	Not Applicable.
P**	Completed prior to each release

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

3/4 CONTROLS AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

CONTROLS

3.0.1 Compliance with the Controls contained in the succeeding controls is required during the conditions specified therein; except that upon failure to meet the Control, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Control shall exist when the requirements of the Control and associated ACTION requirements are not met within the specified time intervals. If the Control is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the conditions specified for individual Controls unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

CONTROLS

3.3.3.9 In accordance with St. Lucie Plant TS 6.8.4.f.1), 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 Control 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted 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 control, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why this inoperability was not corrected in a timely manner.

c. Report all deviations in the Annual Radioactive Effluent Release Report.

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 at the frequencies shown in Table 4.3-8.

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TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a) Liquid Radwaste Effluent Line	1	35
b) Steam Generator Blowdown Effluent Line	1/SG	36
2. Flow Rate Measurement Devices		
a) Liquid Radwaste Effluent Line	N.A.	38
b) Discharge Canal	N.A.	38
c) Steam Generator Blowdown Effluent Lines	N.A.	38

SG - Denotes Steam Generator

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 35 - 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 the Surveillance Requirement for concentration limit of Control 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.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - 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.E-07 micro-Curie/ml:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 micro-Curies/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 micro-Curies/gram DOSE EQUIVALENT I-131.

ACTION 38 - Minimum system design flow of required running pumps shall be utilized for MPC calculations for discharge canal flow and maximum system design flow be utilized for MPC calculations for effluent line flow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. 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 (3)	N.A.	R	Q
b) Discharge Canal	D (3)	N.A.	R	Q
c) Steam Generator Blowdown Effluent Line	D (3)	N.A.	R	Q

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

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

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

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards traceable to the National Institute of Standards & Technology (NIST) or using standards that have been calibrated against standards certified by the NIST. 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.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

CONTROLS

3.3.3.10 In accordance with St. Lucie Plant TS 6.8.4.f.1), 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 Control 3.11.2.1.are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted 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 control, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why this inoperability was not corrected in a timely manner.

c. Report all deviations in the Annual Radioactive Effluent Release Report.

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 at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Waste Gas Holdup System			
a) Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1/Rx	*	45
2. Condenser Evacuation System			
a) Noble Gas Activity Monitor	1/Rx	**	47
3. Plant Vent System			
a) Noble Gas Activity Monitor (Low Range)	1/Rx	*	47
b) Iodine Sampler	1/Rx	*	51
c) Particulate Sampler	1/Rx	*	51
d) Flow Rate Monitor	N.A.	*	53
e) Sampler Flow Rate Monitor	1/Rx	*	46
4. Fuel Storage Area Ventilation System			
a) Noble Gas Activity Monitor (Low Range)	1/Rx	*	47
b) Iodine Sampler	1/Rx	*	51
c) Particulate Sampler	1/Rx	*	51
d) Flow Rate Monitor	N.A.	*	53
e) Sampler Flow Rate Monitor	1/Rx	*	46

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. Laundry Area Ventilation System			
a) Noble Gas Activity Monitor (Low Range)	1/Rx	*	47
b) Iodine Sampler	1/Rx	*	51
c) Particulate Sampler	1/Rx	*	51
d) Flow Rate Monitor	N.A.	*	53
e) Sampler Flow Rate Monitor	1/Rx	*	46
6. Steam Generator Blowdown Building Vent			
a) Noble Gas Activity Monitor (Low Range)	1	*	47
b) Iodine Sampler	1	*	51
c) Particulate Sampler	1	*	51
d) Flow Rate Monitor	N.A.	*	53
e) Sampler Flow Rate Monitor	1	*	46

TABLE 3.3-13 (Continued)
TABLE NOTATIONS

- * - At all times while making releases via this pathway
- ** - At all times when air ejector exhaust is not directed to plant vent.
- Rx - Denotes reactor

ACTION STATEMENTS

ACTION 45 - 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 a 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.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - 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 47 - 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 51 - 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.

ACTION 53 - Maximum system flows shall be utilized in the determination of the instantaneous release monitor alarm setpoint.

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TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	Modes in which surveillance required
1. Waste Gas Holdup System					
a) Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R (3)	Q (1)	*
2. Condenser Evacuation System					
a) Noble Gas Activity Monitor	D	M	R (3)	Q (2)	**
3. 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) Flow Rate Monitor	D	N.A.	R	Q	*
e) Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*
4. 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) Flow Rate Monitor	D	N.A.	R	Q	*
e) Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

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TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	Modes in which surveillance required
5. Laundry Area Ventilation System					
a) Noble Gas Activity Monitor	D	M	R (3)	Q (2)	*
b) Iodine Sampler	W	N.A.	N.A.	N.A.	*
c) Particulate Sampler	W	N.A.	N.A.	N.A.	*
d) Flow Rate Monitor	D	N.A.	R	Q	*
e) 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) Flow Rate Monitor	D	N.A.	R	Q	*
e) Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* - At all times when making releases via this pathway.

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

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

1. Instrument indicates measured levels above the alarm/trip setpoint, or
2. Circuit failure, or
3. Instrument indicates a downscale failure, or
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 exist:

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

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards traceable to the National Institute of Standards & Technology (NIST) or using standards that have been calibrated against standards certified by the NIST. 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.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

CONTROLS

3.11.1.1 In accordance with the St. Lucie Plant TS 6.8.4.f.2) and 3), the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see TS 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.E-04 micro-Curie/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Control 3.11.1.1.

4.11.1.1.3 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1, and 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 Control 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 (1) (µCi/ml)
A. Batch Waste Release Tanks (2)	P Each Batch	Each Batch	P.G.E. (3)	5.E-07
			I-131	1.E-06
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1.E-05
	P Each Batch	M Composite (4)	H-3	1.E-05
			Gross Alpha	1.E-07
	P Each Batch	Q Composite (4)	Sr-89, Sr-90	5.E-08
			Fe-55	1.E-06
B. Continuous Releases (5, 6)	Daily	4/M Composite	P.G.E.(3)	5.E-07
			I-131	1.E-06
	Daily Grab Sample	4/M Composite	Dissolved and Entrained Gases (Gamma Emitters)	1.E-05
	Daily	M Composite	H-3	1.E-05
			Gross Alpha	1.E-07
	Daily	Q Composite	Sr-89, Sr-90	5.E-08
			Fe-55	1.E-06
C. Settling Basin (7)	W Grab Sample	W	P.G.E. (3)	5.E-07
			I-131	1.E-06

P.G.E. - Denotes Principal Gamma Emitter

TABLE 4.11-1 (Continued)

TABLE NOTATION

(1) The LLD is defined for purposes of these controls, 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:

$$LLD = \frac{4.66\ S_b}{E \cdot V \cdot 2.22E{+}06 \cdot Y \cdot \exp(-\lambda \cdot \Delta T)}$$

Where:

LLD	=	the "a priori" lower limit of detection (micro-Curie per unit mass or volume),
S_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),
E	=	the counting efficiency (counts per disintegration),
V	=	the sample size (units of mass or volume),
2.22E+06	=	the number of disintegrations per minute per micro-Curie.,
Y	=	the fractional radiochemical yield, when applicable,
λ	=	the radioactive decay constant for the particular radionuclide (sec^{-1}), and
ΔT	=	the elapsed time between the midpoint of sample collection and the time of counting (sec).

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

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.

TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(2) 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 by a method described in the ODCM to assure representative sampling.

(3) The principal gamma emitters for which the LLD control applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report pursuant to Control 3.11.2.6 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(5) 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.

(6) If Component Cooling Water activity is > 1.E-5 μCi/ml, perform a weekly gross activity on the Intake Cooling Water System outlet to ensure the activity level is less than or equal to 2.E-07 μCi/ml LLD limit. If ICW is >2.E-07 μCi/ml, perform analysis in accordance with a Plant Continuous Release on this Table.

(7) Grab samples to be taken when there is confirmed primary to secondary system leakage indicated by the air ejector monitor indicating greater than or equal to 2x background.

RADIOACTIVE EFFLUENTS

DOSE

CONTROLS

3.11.1.2 In accordance with St. Lucie Plant TS 6.8.4.f.4) and 6.8.4.f.5), the dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see TS Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole 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 whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Plant TS 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) 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.

SURVEILLANCE REQUIREMENTS

4.11.1.2 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 RADWASTE TREATMENT SYSTEM

CONTROLS

3.11.1.3 In accordance with St. Lucie Plant TS 6.8.4.f.6), the Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see TS Figure 5.1-1) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Plant TS 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any 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.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed 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

CONTROLS

3.11.2.1 In accordance with St. Lucie Plant TS 6.8.4.f.3), and 7), the dose rate in UNRESTRICTED AREAS due to radioactive materials released in gaseous effluents from the site (see TS Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, immediately restore 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 & ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (1) (μCi/cc)
1. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Noble Gas P.G.E. (2)	1.E-04
2. Containment Purge	P Each Purge (6) Grab Sample	P Each Purge (6)	Noble Gas P.G.E. (2)	1.E-04
			H-3	1.E-06
3. Vents: a. Plant b. Fuel Bldg (5) c. Laundry d. S/G Blowdown Bldg.	4/M Grab Sample	4/M	Noble Gas P.G.E. (2)	1.E-04
			H-3	1.E-06
4. All Release Types as listed in 3. above	Continuous (3)	4/M Charcoal Sample (4)	I-131	1.E-12
		4/M Particulate Sample (4)	P.G.E.	1.E-11
		M Composite Particulate Sample	Gross Alpha	1.E-11
		Q Composite Particulate Sample	Sr-89, Sr-90	1.E-11
		Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1.E-06

P.G.E. - Denotes Principal Gamma Emitters

TABLE 4.11-2 (Continued)

TABLE NOTATIONS

(1) The LLD is defined for purposes of these controls, 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:

$$LLD = \frac{4.66\ S_b}{E \cdot V \cdot 2.22E{+}06 \cdot Y \cdot \exp(-\lambda \cdot \Delta T)}$$

Where:

LLD	=	the "a priori" lower limit of detection (micro-Curie per unit mass or volume),
S_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),
E	=	the counting efficiency (counts per disintegration),
V	=	the sample size (units of mass or volume),
2.22E+06	=	the number of disintegrations per minute per micro-Curie.,
Y	=	the fractional radiochemical yield, when applicable,
λ	=	the radioactive decay constant for the particular radionuclide (sec^{-1}), and
ΔT	=	the elapsed time between the midpoint of sample collection and the time of counting (sec).

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

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.

TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(2) The principal gamma emitters for which the LLD control applies include the following radionuclides: Kr-87,Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141, and Ce-144 in Iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report pursuant to Control 3.11.2.6 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3) 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 Controls 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(4) Samples shall be changed at least four times per 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 within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased by more than a factor of 3.

(5) Tritium grab samples shall be taken at least 4/M from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

(6) 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.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

CONTROLS

3.11.2.2 In accordance with St. Lucie Plant TS 6.8.4.f.5), and 8), the air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see TS Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads 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, prepare and submit to the Commission within 30 days, pursuant to Plant TS 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases 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 RADIOACTIVE MATERIAL IN PARTICULATE FORM

CONTROLS

3.11.2.3 In accordance with St. Lucie Plant TS 6.8.4.f.5), and 9), 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 unit, to areas at and beyond the SITE BOUNDARY (see TS 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, prepare and submit to the Commission within 30 days, pursuant to Plant TS 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to assure that subsequent releases will be in compliance with the above limits.

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 SYSTEM

CONTROLS

3.11.2.4 In accordance with St. Lucie Plant TS 6.8.4.f.6), the VENTILATION EXHAUST Treatment System and the WASTE GAS HOLDUP SYSTEM shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see TS Figure 5.1-1) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Plant TS 6.9.2, a Special Report that includes the following information:

1. Identification of any 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.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

4.11.2.4.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and WASTE GAS HOLDUP SYSTEM shall be demonstrated OPERABLE by operating the WASTE GAS HOLDUP 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

3/4.11.4 TOTAL DOSE

CONTROLS

3.11.4 In accordance with St. Lucie Plant TS 6.8.4.f.10), 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 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

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 Control 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations shall be made including direct radiation contributions from the units (including outside storage tanks etc.) to determine whether the above limits of Control 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Plant TS 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 Subpart M of 10 CFR Part 20, 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 or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 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. /R14

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE (Continued)

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Controls 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 units (including outside storage tanks etc.) shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a. of Control 3.11.4.

RADIOACTIVE EFFLUENTS

3/4.11.5 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

ADMINISTRATIVE CONTROLS

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

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

   a) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.
   b) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
   c) A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;
   d) An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;
   e) An evaluation of the change which shows the expected maximum exposure to individuals in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;
   f) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period when the changes are to be made;
   g) An estimate of the exposure to plant operating personnel as a result of the change; and
   h) Documentation of the fact that the change was reviewed and found acceptable by the FRG.

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

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* Licensees may choose to submit the information called for in this Administrative Control as part of the annual FUSAR update.

RADIOACTIVE EFFLUENTS

3/4.11.6 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT TO THE COMMISSION*

ADMINISTRATIVE CONTROLS

3.11.2.6 As per Technical Specification 6.9.1.7, a Annual Radioactive Effluent Release Report covering the operation of each unit during the previous 12 months of operation shall be submitted within 60 days after January 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from each unit. The material provided shall be (1) consistent with the objectives outlined in by items a) through f) below, using the example report format in the ODCM, and (2) be in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

a. 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.

b. 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

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* - 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.

RADIOACTIVE EFFLUENTS

3/4.11.6 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT TO THE COMMISSION (Continued)

ADMINISTRATIVE CONTROLS

3.11.2.6 (Continued)

b. (Continued)

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 (see TS 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 ODCM.

c. 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.

d. 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.

RADIOACTIVE EFFLUENTS

3/4.11.6 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT TO THE COMMISSION (Continued)

ADMINISTRATIVE CONTROLS

3.11.2.6 (Continued)

e. The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:
   1. Volume
   2. Total Curie quantity (specify whether determined by measurement or estimate)
   3. Principal radionuclides (specify whether determined by measurement or estimate)
   4. Type of waste (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms)
   5. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and
   6. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

f. 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.

g. 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 of ODCM Control 3.12.2.

h. The format for an Annual Radioactive Effluent Release Report is provided in ODCM Section 4.0. The information contained in an annual report shall not apply to any ODCM Control Dose Limit(s) since the methodology for the annual report is based on actual meteorological data, instead of historical conditions that the ODCM Controls and Control required calculations are based on.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

CONTROLS

3.12.1 In accordance with St. Lucie Plant TS 6.8.4.g.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, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Control 3.12.4, 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, prepare and submit to the Commission within 30 days, pursuant to Plant TS 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 limit of Controls 3.11.1.2, 3.11.2.2, or 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:

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

When radionuclides other than those in Table 3.12-2 are

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* 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.

** The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

Controls (Continued)

Action b. (Continued)

detected and are the result of plant effluents, this report shall be submitted if the potential annual dose, to a MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Control 3.11.1.2, 3.11.2.2, or 3.11.2.3. 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 required by Control 3.12.4.

c. With milk or broad leaf vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Control 3.11.2.6, submit in the next Annual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

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.

ST. LUCIE PLANT
CHEMISTRY OPERATING PROCEDURE NO. C-200, REVISION 14
OFFSITE DOSE CALCULATION MANUAL (ODCM)

TABLE 3.12-1
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM[a)]

EXPOSURE PATHWAY and/or SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[b) c)]	SAMPLING AND COLLECTION FREQUENCY[d)]	TYPE AND FREQUENCY [d)] OF ANALYSIS
1. Direct Radiation[e)]	27 Monitoring Locations	Continuous monitoring with sample collection quarterly [f)]	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	Radioiodine filter: I-131 analysis weekly Particulate Filter: Gross beta radioactivity analysis ≥24 hours following a filter change[g)] Gamma isotopic[h)] analysis of composite [g)] (by location) quarterly
3. Waterborne			
a) Surface[k)]	1 Location[m)]	Weekly	Gamma isotopic[h)] & tritium analyses weekly
	1 Location[n)]	Monthly	Gamma isotopic[h)] & tritium analyses monthly
b) Sediment from shoreline	2 Locations	Semiannually	Gamma isotopic[h)] analyses semiannually
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[p)]	Monthly when available	Gamma isotopic[h)] and I-131 analyses monthly

TABLE 3.12-1 (Continued)

TABLE NOTATIONS

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 Control 3.12.4.

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 required by Table 3.12-1, in Appendix-E and applicable figures.

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.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

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.

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 pCi per cubic meters 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.

k. Discharges from the St. Lucie Plant do not influence drinking water or ground water pathways.

m. Atlantic Ocean, in the vicinity of the public beaches along the eastern shore of Hutchinson Island near the St. Lucie Plant (grab sample)

n. Atlantic Ocean, at a location beyond influence from plant effluents (grab sample).

p. 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 kilometers distant in the least prevalent wind direction based upon historical data in the ODCM.

[i, j, l (lower case), and o are not used on notation for clarity reasons]

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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	

l - as in pCi/l denotes liter

* - 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 pathway exists, 2 pCi/l is the limit for drinking water.

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

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[(1) (2)]

LOWER LIMIT OF DETECTION (LLD)[(3)]

ANALYSIS	WATER pCi/l	AIRBORNE PARTICULATE OR GASES 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, Co-60	15		130			
Zn-65	30		260			
Zr-95, Nb-95[(4)]	15					
I-131	1**	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140, La-140[(4)]	15			15		

* No drinking water pathway exists, a value of 2000 pCi/l is for drinking water.

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

TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1) This 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 Control 3.12.4.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

(3) The LLD is defined for purposes of these controls, 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:

$$LLD = \frac{4.66\ S_b}{E \bullet V \bullet 2.22 \bullet Y \bullet \exp(-\lambda \bullet \Delta T)}$$

Where:

LLD = the "a priori" lower limit of detection (pico-Curie per unit mass or volume),

S_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per pico-Curie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

ΔT = the elapsed time between the midpoint of sample collection and the time of counting (sec).

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

TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

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 Control 3.12.4.

(4) An equilibrium mixture of the parent and daughter isotopes which corresponds to 15 pCi/Liter of the parent isotope.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

CONTROLS

3.12.2 In accordance with St. Lucie Plant TS 6.8.4.g.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 square meters (500 square feet) 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 Control 4.11.2.3, pursuant to Control 3.11.2.6, identify the new location(s) in the next Annual Radioactive Effluent Release Report.

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 Control 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. 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. Pursuant to TS 6.14, submit in the next Annual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

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* 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. Controls for broad leaf vegetation sampling in Table 3.12-1, Part 4.b., shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS (Continued)

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 Control 3.12.4.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

CONTROLS

3.12.3 In accordance with St. Lucie Plant TS 6.8.4.g.3), analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission*, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective action taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Control 3.12.4.

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 Control 3.12.4. If the Interlaboratory Comparison Program is other than the program conducted by the EPA, then the Interlaboratory Comparison Program shall be described in the ODCM.

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

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RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.4 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (AREOR)*

ADMINISTRATIVE CONTROLS

3.12.4 In accordance with St. Lucie Plant TS 6.9.1.8, an Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and information based on trend analysis of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided in the AREOR shall be consistent with the objectives outlined below, and with Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

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

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

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

BASES

for the

CONTROLS

AND

SURVEILLANCE REQUIREMENTS

NOTE
The BASES contained in succeeding pages summarize the reasons for the Controls in Section 3.0 and 4.0, but are not part of these Controls.

INSTRUMENTATION

BASES

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluent during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters 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.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluent during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters 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.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This control is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table 2, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20. 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. /R14

This control applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid 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 Currie, L.A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300.

3/4.11.1.2 DOSE

This control is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Control 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 to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation

3/4.11 RADIOACTIVE EFFLUENTS (Continued)

BASES

3/4.11.1 LIQUID EFFLUENTS (Continued)

3/4.11.1.2 DOSE (Continued)

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, "Revision 1, October 1977 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 control applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared Radwaste Systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

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 control implements the requirements of 10 CFR 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.

This control applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

RADIOACTIVE EFFLUENTS

BASES

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This control 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 concentration of 10 CFR Part 20, Appendix B, Table 2, Column I. 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 an annual average concentration exceeding the limits specified in Appendix B, Table 2 of 10 CFR Part 20 (Subpart D of 10 CFR Part 20). 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 mrems/year to the whole 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 mrems/year. /R14

This control applies to the release of radioactive materials in gaseous effluents from all units at the site.

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 Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300.

RADIOACTIVE EFFLUENTS

BASES

3/4.11.2.1 DOSE - NOBLE GASES

This control is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The control implements the guides set forth in Section I.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, "Revision 1, October 1977 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.

This control applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the gaseous effluents from the shared system are to be proportioned among the units sharing that system.

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

This control is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Controls 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 material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the

RADIOACTIVE EFFLUENTS

BASES

3/4.11.2.1 DOSE - NOBLE GASES (Continued)

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM (Continued)

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 ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject material 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, "Revision 1, October 1977 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 conditions. The release rate controls 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 the 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 control applies to the release of radioactive materials in gaseous effluents from each unit 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.

RADIOACTIVE EFFLUENTS

BASES

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensure 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 control implements the requirements of 10 CFR 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 systems were specified as a suitable fraction of the dose design objectives set forth in Section II.B and II.C of Appendix I, 10 CFR Part 50 for gaseous effluents.

This control applies to the release of radioactive materials in gaseous effluents from each unit 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.

3/4.11.2.5 NOT USED

3/4.11.2.6 NOT USED

3/4.11.3 NOT USED

3/4.11.4 TOTAL DOSE

This control is provided to meet the dose limitations of 10 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The control requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrems to the whole 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 units (including outside storage tanks, etc.) are kept small. The Special Report will describe a course of action that should result in the limitation

RADIOACTIVE EFFLUENTS

BASES

3/4.11.4 TOTAL DOSE (Continued)

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 kilometers 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 190.11 and Subpart M of 10 CFR Part 20, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates 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 Controls 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. /R14

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this control provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposure of MEMBERS OF THE PUBLIC resulting from the plant 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. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first three 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 Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300.

3/4.12.2 LAND USE CENSUS

This control 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 given in the ODCM 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.

RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.2 LAND USE CENSUS (Continued)

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 square meters 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 kilograms/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 kilograms per square meter.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

This 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 materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

METHODOLOGY

for the

CONTROLS

AND

SURVEILLANCE REQUIREMENTS

GLOSSARY OF COMMONLY USED TERMS IN METHODOLOGY SECTION

D_B	-	Dose from Beta Radiation
CC or cc	-	Cubic centimeter
Ci	-	Curies - a unit of radioactivity see µCi
C_i	-	Activity or concentration of a nuclide in the release source. Units of µCi, µCi/cc, or µCi/ml
CFR	-	Code of Federal Regulations
Control(s)	-	Regulations for operating, controlling, monitoring, and reporting radioactive effluent related activity as indicated by the Controls Section of the ODCM.
Dose	-	The exposure, in mrem or mrad, the organ or the individual receives from radioactive effluents
Dose Factor	-	Normally, a factor that converts the effect of ingesting radioactive material into the body, to dose to a specific organ. Body elimination, radioactive decay, and organ uptake are some of the factors that determine a dose factor for a given nuclide
Dose Pathway	-	A specific path that radioactive material physically travels through prior to exposing an individual to radiation. The Grass-Cow-Milk-Infant is a dose pathway
Dose Rate	-	The dose received per unit time
(D/Q)	-	A long term D over Q - a factor with units of $1/m^2$ which describes the deposition of particulate matter from a plume at a point downrange from the source. It can be thought of as what part of the cloud is going to fallout and deposit over one square meter of ground. (See Appendix F).
FUSAR	-	Final Updated Safety Analysis Report.

GLOSSARY OF COMMONLY USED TERMS IN METHODOLOGY SECTION
(continued)

Y	-	A gamma photon - The dose from Gammas in air, etc.
Ground Plane	-	Radioactive material deposited uniformly over the ground emits radiation that produces an exposure pathway when an individual is standing, sitting, etc., in the area. It is assumed that an adult receives the same exposure as an infant, regardless of the physical height differences. Only the whole body is considered for the ODCM.
H-3	-	Hydrogen-3, or Tritium, a weak Beta emitter
I&8DP	-	Radioiodines and particulates with half-lives greater than 8 days
m^3	-	Cubic Meters
m^2	-	Square Meters
MPC	-	Maximum Permissible Concentration
nuclide	-	For the purposes of this manual, a radioactive isotope. Nuclide (i) signifies a specific nuclide, the 1st, 2nd, 3rd one under consideration. If nuclide (i) is I-131, then the Mi (dose factor) under consideration should be M_{I-131} for example.
Organ	-	For the ODCM either the bone, liver, thyroid, kidney, lung, GI-LLI, or the Whole Body. Whole Body is considered an organ for ease of writing the methodology in the ODCM.
pCi	-	1 pico-Curie = 1.E-12 Curies.
$(Q\ Dot)_i$	-	$(Q\ Dot)_i$ - Denotes a release rate in µCi/sec for nuclide (i).
Q_i	-	Denotes µCi of nuclide (i) released over a specified time interval.
Radioiodines	-	Iodine-131 and Iodine I-133 for gaseous release pathways.

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GLOSSARY OF COMMONLY USED TERMS IN METHODOLOGY SECTION
(continued)

Receptor - The individual receiving the exposure in a given location or who ingests food products from an animal for example. A receptor can receive dose from one or more pathways.

Release Source(s) - A subsystem, tank, or vent where radioactive material can be released independently of other radioactive release points.

TS - The St. Lucie Plant Standard Technical Specifications

Total Body - Same as Whole Body in Control Statements

µCi - micro Curies. 1 µCi = 10^{-6} Curies. The µCi is the standard unit of radioactivity for all dose calculations in the ODCM.

(X/Q) - A long term Chi over Q. It describes the physical dispersion characteristics of a semi-infinite cloud of noble gases as the cloud traverses downrange from the release point. Since Noble Gases are inert, they do not tend to settle out on the ground. (See Appendix F).

$(X/Q)_D$ - A long term Depleted Chi over Q. It describes the physical dispersion characteristics of a semi-infinite cloud of radioactive iodines and particulates as the cloud travels downrange. Since Iodines and particulates tend to settle out (fallout of the cloud) on the ground, the $(X/Q)_D$ represents what physically remains of the cloud and its dispersion qualities at a given location downrange from the release point. (See Appendix F).

dt, Δt, or delta t - A specific delta time interval that corresponds with the release interval data etc.

1.0 LIQUID RELEASES METHODOLOGY

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1.1 Radioactive Liquid Effluent Model Assumptions

The FUSAR contains the official description of the site characteristics. The description that follows is a brief summary for dose calculation purposes:

The St. Lucie Plant is located on an island surrounded on two sides by the Atlantic Ocean and the Indian River, an estuary of the Atlantic Ocean. Normally, all radioactive liquid releases enter the Atlantic Ocean where the Circulating Water Discharge Pipe terminates on the ocean floor at a point approximately 1200 feet offshore. No credit is taken for subsequent mixing of the discharge flume with the ocean. The diffusion of radioactive material into the ocean is dependent on the conditions of tide, wind, and some eddy currents caused by the Gulf Stream. The conditions are sufficiently random enough to distribute the discharges over a wide area and no concentrating effects are assumed.

There are no direct discharge paths for liquid effluents to either of the north or south private property boundary lines. The Big Mud Creek (part of the Indian River) does connect to a normally locked shut dam, that is intended to provide an emergency supply of circulating water to the Intake Cooling Water Canal in the event a Hurricane causes blockage of the Intake Canal. No radioactive water could be discharged directly into the Intake Cooling Water Canal because all plant piping is routed to the discharge canal and no back flow can occur. Consult the FUSAR for a detailed description of characteristics of the water bodies surrounding the plant site.

Only those nuclides that appear in the Liquid Dose Factor Tables will be considered for dose calculation.

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1.2 Determining the Fraction F of 10 CFR Part 20 MPC Limits for A Liquid Release Source

Discussion - Control 3.11.1.1 requires that the sampling and analysis results of liquid waste (prior to discharge) be used with calculation methods in the in-plant procedures to assure that the concentration of liquid radioactive material in the unrestricted areas will not exceed the concentrations specified in 10 CFR Part 20, Appendix B, Table II. Chemistry Procedure C-70 "Processing Aerated Liquid Waste" provides instruction for ensuring batch release tanks will be sampled after adequate mixing. This section presents the calculation method to be used for this determination. This method only addresses the calculation for a specific release source. The in-plant procedures will provide instructions for determining that the summation of each release source's F values do not exceed the site's 10 CFR Part 20 MPC limit. The values for release rate, dilution rate, etc., will also have to be obtained from in-plant procedures. The basic equation is:

$$F_L = \frac{R}{D} \sum_{i=1}^{n} \frac{C_i}{(MPC)\ i}$$

Where:

F_L = the fraction of 10 CFR Part 20 MPC that would result if the release source was discharged under the conditions specified.

R = The undiluted release rate in gpm of the release source.
Liquid Rad Waste = 170 gpm
Steam Generator = 125 gpm/Steam Generator

D = The dilution flow in gpm of Intake Cooling Water or Circulating Water Pumps
Intake Cooling flow is 14,500 gpm/pump
Circulating Water flow is 121,000 gpm/pump

C_i = The undiluted concentration of nuclide (i) in µCi/ml from sample assay

$(MPC)_i$ = The maximum permissible concentration of nuclide (i) in µCi/ml from Table L-1. For dissolved or entrained noble gases the MPC value is 2×10^{-4} µCi/ml for the sum of all gases.

1.2 (continued)

The fraction of the 10 CFR Part 20 MPC limit may be determined by a nuclide-by-nuclide evaluation or for purposes of simplifying the calculation by a cumulative activity evaluation. If the simplified method is used, the value of 3×10^{-8} µCi/ml (unidentified MPC value) should be substituted for $(MPC)_i$ and the cumulative concentration (sum of all identified radionuclide concentrations) or the gross concentration should be substituted for C_i. As long as the diluted concentration (C_{total} R/D) is less than 3×10^{-8} µCi/ml, the nuclide-by-nuclide calculation is not required to demonstrate compliance with the 10 CFR Part 20 MPC limit. The following section provides a step-by-step procedure for determining the MPC fraction.

1. Calculation Process for Solids

   A. Obtain from the in-plant procedures, the release rate value (R) in gpm for the release source.

   B. Obtain from the in-plant procedures, the dilution rate (D) in gpm. No credit is taken for any dilution beyond the discharge canal flow.

   C. Obtain (C_i), the undiluted assay value of nuclide (i), in µCi/ml. If the simplified method is used, the cumulative concentration (C_{total}) is used.

   D. From Table L-1, obtain the corresponding (MPC) for nuclide (i) in µCi/ml. The value of 3×10^{-8} µCi/ml should be used for the simplified method.

   E. Divide C_i by $(MPC)_i$ and write down the quotient

   F. If the simplified method is used, proceed to the next step. If determining the MPC fraction by the nuclide-by-nuclide evaluation, repeat steps 1.2.1.C through 1.2.1.E for each nuclide reported in the assay, for H_3 from previous month composite, and for SR89/90 and Fe55 from previous quarter composite.

1.2 (continued)

1. (continued)

G. Add each C_i/(MPC) quotient from step 1.2.1.E and solve for F_L as follows:

$$F_L = \frac{R}{D} \sum_{i=1}^{n} \frac{C_i}{(MPC)\ i}$$

F_L = a unit-less value where:

the value of F_L could be $\leq$ or >1. The purpose of the calculation is to determine what the initial value of F_L is for a given set of release conditions. If F_L is >1, administrative steps are taken to ensure that the actual release conditions for dilution will ensure that F_L is $\leq$1 during the actual release. F_L is called the fraction of 10 CFR Part 20 MPC because it should never be allowed to be >1.

2. Calculation Process for Gases in Liquid

A. Sum the µCi/ml of each noble gas activity reported in the release.

B. The values of R and D from 1.2.1 above shall be used in the calculations below:

$$F_g = \frac{(sum\ of\ 1.2.2.A)\ \mu Ci/ml}{1} \times \frac{R}{D}$$

C. F_g shall be less than 2 X 10^{-4} µCi/ml for the site for all releases in progress. Each release point will be administratively controlled. Consult in-plant procedures for instructions.

1.3 Determining Setpoints for Radioactive Liquid Effluent Monitors

Discussion - Control 3.3.3.10 requires that the liquid effluent monitoring instrumentation alarm/trip setpoints be set to initiate an alarm or trip so that the radioactivity concentration in water in the unrestricted area does not exceed the concentration of 10 CFR Part 20, Appendix B, Table II as a result of radioactivity in liquid effluents (Control 3.11.1.1). This section presents the method to be used for determining the instrumentation setpoints.

Gross cpm vs. total liquid activity curves are available for Liquid Effluent Monitors based on a composite of real release data. A direct correlation between gross cpm and the concentrations that would achieve 10 CFR Part 20 MPC levels in the discharge canal can be estimated. The 1978 liquid release data from annual reports was used to determine the average undiluted release concentration. These concentrations were then projected to a diluted concentration in the discharge canal assuming a 1 gpm release rate and a constant dilution flow of 121,000 gpm from 1 circ. water pump. This diluted activity was divided by the nuclide's respective 10 CFR Part 20 MPC value (Table L-1) to obtain the Mi column on the table that follows:

1.3 (continued)

TABLE 1.3

NUCLIDE SYMBOL	1978 UNDILUTED μCi/ml[1]	M_i[2] (no units)
I-131	4.43 E-5	3.66 E-4
I-132	2.23 E-7	1.84 E-8
I-133	3.17 E-6	3.74 E-6
I-135	1.31 E-6	3.61 E-7
Na-24	1.72 E-7	2.84 E-8
Cr-51	2.51 E-5	4.15 E-7
Mn-54	5.64 E-6	1.55 E-6
Mn-56	1.11 E-9	1.31 E-10
Co-57	3.69 E-7	5.08 E-8
Co-58	1.51 E-4	6.24 E-5
Fe-59	2.92 E-6	2.41 E-6
Co-60	3.66 E-5	1.01 E-4
Zn-65	4.55 E-7	7.52 E-7
Ni-65	8.23 E-7	6.8 E-8
Ag-110	1.96 E-6	2.70 E-6
Sn-113	5.75 E-7	1.58 E-7
Sb-122	2.15 E-6	1.78 E-6
Sb-124	8.40 E-6	9.92 E-6
W-187	3.51 E-6	9.67 E-7
Np-239	1.57 E-7	6.49 E-8
Br-82	3.64 E-7	7.52 E-8
Zr-95	2.82 E-5	1.17 E-5
Zr-97	4.05 E-6	3.72 E-6
Mo-99	3.24 E-6	1.34 E-6
Ru-103	3.84 E-8	1.06 E-8
Sb-125	2.26 E-6	6.23 E-7
Cs-134	2.14 E-5	1.97 E-4
Cs-136	7.82 E-7	1.08 E-6
Cs-137	4.85 E-5	4.01 E-4
Ba-140	6.44 E-7	6.65 E-7
Ce-141	3.04 E-8	8.38 E-9
Ce-144	2.37 E-6	6.53 E-6
A_{tot}=	4.01 E-4	
M_{Total}=		1.18 E-3

(1) 1978 Undiluted Release Volume = 7 E 9 ml.

(2) $$M_i = \frac{1978\ Undil.\ Act\ Nuclide\ (i)}{MPC_i\ (from\ Table\ L\text{-}1)} \times \frac{1\ gpm\ (release\ rate)}{121000\ gpm\ (dil\ rate)}$$

1.3 (continued)

A_{Tot} is the total average µCi/ml concentration of the reference mixture and M_{Tot} is the fraction of the MPC of all nuclides for the release conditions specified. Dividing A_{Tot} by M_{Tot} yields A_{Max}, which is the maximum total activity concentration equivalent to the MPC limit for the nuclide distribution typical of radwaste discharges.

$$A_{Max} = \frac{A_{Tot}}{M_{Tot}} = \frac{4.01\ E\text{-}4}{1.18\ E\text{-}3} = 0.34\ \mu Ci/ml$$

The assumption that the mixture does not change is only used for calculational purposes.

1. If the effluent monitor requires counts per minute units, a (C_{max}) value in cpm should be obtained for the A_{max} (0.34 µCi/ml) from the release sources radioactive liquid effluent monitor curve of cpm vs. µCi/ml.

NOTE
This setpoint is for a specified release of 1 gpm into 121000 gpm dilution flow.

2. For establishing the setpoint prior to liquid radwaste discharges, the A_{max} (or C_{max}) will be adjusted as needed to account for actual release conditions (i.e., actual design maximum discharge flow rate, dilution flow rate, and the contribution of dissolved and entrained Nobles Gas Activity to the Monitor Activity Level).

1.4 Determining the Dose for Radioactive Liquid Releases

Discussion - Control 3.11.1.2 requires calculations be performed at least once per 31 days to verify that cumulative radioactive liquid effluents do not cause a dose in excess of 1.5 mrem to the whole body and 5 mrem to any organ during any calendar quarter and not in excess of 3 mrem to the whole body and 10 mrem to any organ during any calendar year. This section presents calculational method to be used for this verification.

1.4 (continued)

This method is based on the methodology suggested by sections 4.3 and 4.3.1 of NUREG-0133 Revision 1, November, 1978. The dose factors are a composite of both the fish and shellfish pathways so that the fish-shellfish pathway is the only pathway for which dose will be calculated. For St. Lucie Plant, the adult is the most limiting age group, but the dose for child, and teenager can also be calculated by this method provided that their appropriate dose factors are available for the organ of interest. Only those nuclides that appear in the Tables of this manual will be considered.

1. This method provides for a dose calculation to the whole body or any organ for a given age group based on real release conditions during a specified time interval for radioactive liquid release sources. The equation is:

$$D_{1_T} = \frac{A_{iT}\ dt_1\ Q_{i1}}{(DF)_1}$$

Where:

D_{1_T} = dose commitment in mrem received by organ T of age group (to be specified) during the release time interval dt_1.

A_{iT} = the composite dose factor for the fish-shellfish pathway for nuclide (i) for organ T of age group (to be specified). The A_{iT} values listed in the Tables in this manual are independent of any site specific information and have the units $\frac{\text{mrem-ml}}{\mu\text{Ci-hr}}$

dt_1 = the number of hours that the release occurs.

Q_{i1} = The total quantity of nuclide (i) release during dt_1 (μCi)

$(DF)_1$ = The total volume of dilution that occurred during the release time period dt_1 (i.e., the circulating water flow times time)

1.4 (continued)

1. (continued)

The doses associated with each release may then be summed to provide the cumulative dose over a desired time period (e.g., sum all doses for release during a 31 day period, calendar quarter or a year).

$$D_{total_T} = \Sigma D_{1_T}$$

Where:

D_T = the total dose commitment to organ $_T$ due to all releases during the desired time interval (mrem)

Based on the radionuclide distribution typical in radioactive effluents, the calculated doses to individuals are dominated by the radionuclides Fe-59, Co-58, Co-60, Zn-65, Nb-95, Cs-134 and Cs-137. These nuclides typically contribute over 95% of the whole body dose and over 90% of the GI-LLI dose, which is the critical organ. Therefore, the dose commitment due to radioactivity in liquid effluents may be reasonably evaluated by limiting the dose calculation process to these radionuclides for the adult whole body and adult GI-LLI. To allow for any unexpected variability in the radionuclide distribution, a conservatism factor of 0.6 is introduced into the equation. After calculating the dose based on these 7 nuclides, the cumulative dose should be divided by 0.6, the conservatism factor. (i.e., $D_T = D_T/0.6$). Refer to Appendix B for a detailed evaluation and explanation of this limited analysis approach.

The methodology that follows is a step-by-step breakdown to calculate doses based on the above equation. Refer to the in-plant procedures to determine the applicable organs, age groups, and pathway factors. If the limited analysis approach is used, the calculation should be limited to the Adult whole body dose and Adult GI-LLI dose from the fish and shellfish pathways. Only the 7 previously specified radionuclides should be evaluated. For the dose calculations to be included in annual reports, the doses to the adult groups and all organs should be evaluated for all radionuclides identified in the liquid effluents.

1.4 (continued)

1. (continued)

NOTE
Table 1.4 provides a convenient form for compiling the dose accounting.

A. Determine the time interval dt_l that the release took place. The in-plant procedures shall describe the procedure for calculating dt_l for official release purposes.

B. Obtain $(DF)_l$ for the time period dt_l from Liquid Waste Management Records for the release source(s) of interest.

C. Obtain Q_{il} for nuclide (i) for the time period dt_l from the Liquid Waste Management Records

D. Obtain A_{iT} from the appropriate Liquid Dose Factor Table

1.4 (continued)

1. (continued)

TABLE 1.4
FISH AND SHELLFISH PATHWAY

TIME/DATE START:____:____ ___/___/___ TIME/DATE STOP:____:____ ___/___/___HOURS

TOTAL DILUTION VOLUME:____________mls

AGE GROUP:__________ ORGAN:_______________ DOSE FACTOR TABLE #:________

NUCLIDE (i)	C_i (µCi)	A_{iT}	DOSE (i) mrem
Fe-59			
Co-58			
Co-60			
Zn-65			
Nb-95			
Cs-134			
Cs-137			
OTHERS			
		TOTAL DOSE $_T$ =	mrem
		If based on limited analysis, divide by 0.6	mrem

E. Solve for Dose (i)

$$Dose\ (i) = \frac{Q_{i1}\ dt_1\ A_{iT}}{(DF)_1}$$

1.4 (continued)

1. (continued)

F. Repeat steps 1.4.1.C through 1.4.1.E for each nuclide reported and each organ required. If the limited analysis method is used, limit the radionuclides to Fe-59, Co-58, Co-60, Zn-65, Nb-95, Cs-134, and Cs-137 and determine the adult whole body dose and the adult GI-LLI dose.

G. Sum the Dose (i) values to obtain the total dose to organ T from the fish-shellfish pathway. If the limited analysis method is being used, divide the cumulative dose by a conservatism factor of 0.6 to account for any unexpected variability in radionuclide distribution

1.5 Projecting Dose for Radioactive Liquid Effluents

Discussion - Control 3.11.1.3 requires that appropriate subsystems of the liquid radwaste treatment system be used to reduce radioactive material in liquid effluents when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see TS Figure 5.1-1) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31 day period. The following calculation method is provided for performing this dose projection. The method is based on dose as calculated in section 1.4 with the adult as the bases for projecting.

1. Obtain the latest result of the monthly calculation of the adult whole body dose and the adult's highest organ dose. These doses can be obtained from the in-plant records.

2. Divide each dose by the number of days the reactor plant was operational during the month.

3. Multiply the quotient of each dose by the number of days the reactor plant is projected to be operational during the next month. The products are the projected dose for the next month. These values should be adjusted as needed to account for any changes in failed fuel or other identifiable operating conditions that could significantly alter the actual releases.

4. If the projected dose is greater than 0.06 mrem to the whole body or greater than 0.2 mrem to the adults highest exposed organ, the liquid radwaste system shall be used.

2.0 GASEOUS RELEASES METHODOLOGY

2.1 Gaseous Effluent Model Assumptions

Description of Site - (The FUSAR contains the official description of the site characteristics. The description that follows is a brief summary for dose calculation purposes only). The St. Lucie Plant is located on an island surrounded on two sides by the Atlantic Ocean and the Indian River, an estuary of the Atlantic Ocean. Private property adjoins the plant site in the north and south directions. A meteorological tower is located north of the plant near the site property line. There are 16 sectors, for dose calculation purposes, divided into 22.5° each. The MET tower is calibrated such that a zero degree bearing coincides with TRUE NORTH. A bearing of zero degrees dissects the north sector such that bearings of 348.75° and 11.25° define the boundaries of the north sector. The nearest distance to private property occurs in the north sector at approximately 0.97 miles. For ease of calculation, this 0.97 mile radius is assumed in all directions, although the real Unrestricted Area Boundary is defined in Figure 5.1-1 of the TS. Doses calculated over water areas do not apply to Controls or the annual report and may be listed as O.W. (over water) in lieu of performing calculations. The 0.97 mile range in the NW sector is O.W., but it was chosen as the worst sector for conservative dose calculations using the historical MET data.

Historical MET Data - MET data, between September 1, 1976 and August 31, 1978, from the St. Lucie MET Tower was analyzed by Dames & Moore of Washington, D.C. The methodology used by Dames & Moore was consistent with methods suggested by Regulatory Guide 1.111, Revision 1. Recirculation correction factors were also calculated for the St. Lucie Site and are incorporated into the historical MET tables (Tables M5, M6, and M7) in Appendix A of this manual. It was determined that these two years are representative data for this locale.

Dose Calculations - Dose calculations for Control dose limits are normally calculated using historical MET data and receptor location(s) which yield calculated doses no lower than the real location(s) experiencing the most exposure. Actual MET data factors are calculated and used in dose calculations for the annual reports.

Live MET data and hour-by-hour dose calculations are beyond the scope of this manual. Historical information and conservative receptor locations, etc., are only used for ease of Control dose limit calculations. Dose calculations for Control dose limits may be performed using actual MET data and real receptor locations. Any dose calculations performed with actual data should note the source of the data in the annual report. Actual MET data reduction should be performed in accordance with Regulatory Guide 1.111, Revision 1 and should incorporate Recirculation Correction Factors from Table M-4 of this manual.

2.1 (continued)

Dose Calculations - (continued)

The St. Lucie site uses the long term ground release model for all gaseous effluents. Only those radionuclides that appear in the gaseous effluent dose factor tables will be considered in any dose calculations. Radioiodines are defined as Iodine-131 and I-133 for application to Controls. Other nuclides of Iodine may be included in dose calculations for ease of performing calculations, but their dose contribution does not have to be included in the Control requirements. Land Census information will apply to the calendar year following the year that the census was taken in to avoid splitting quarters, etc.

2.2 Determining the Whole Body and Skin Dose Rates for Noble Gas Releases And Establishing Setpoints for Effluent Monitors

Discussion - Control 3.11.2.1 limits the dose rate from noble gases in airborne releases to <500 mrem/yr - whole body and <3000 mrem/yr - skin. Control 3.3.3.11 requires that the gaseous radioactive effluent monitoring instrumentation be operable with alarm/trip setpoints set to ensure that these dose rate limits are not exceeded. The results of the sampling and analysis program of Control Table 4.11-2 are used to demonstrate compliance with these limits.

The following calculation method is provided for determining the dose rates to the whole body and skin from noble gases in airborne releases. The alarm/trip setpoints are based on the dose rate calculations. The Controls apply to all airborne releases on the site but all releases may be treated as if discharged from a single release point. Only those noble gases appearing in Table G-2 will be considered. The calculation methods are based on Sections 5.1 and 5.2 of NUREG-0133, November 1978. The equations are:

For WHOLE BODY Dose Rate:

$$DR_{WB} = \sum_{i}^{n} K_i \, (X/Q) \, (Q\ DOT)_i$$

For TOTAL SKIN Dose Rate:

$$DR_{skin} = \sum_{i}^{n} [L_i + 1.1\ M_i] \, (X/Q) \, (Q\ DOT)_i$$

2.2 (continued)

Where:

DR_{WB} = whole body dose rate from noble gases in airborne releases (mrem/yr)

DR_{skin} = skin dose rate from noble gases in airborne releases (mrem/yr)

$\sum_{i}^{n}$ = a mathematical symbol to signify the operations to the right of the symbol are to be performed for each noble gas nuclide (i) through (n), and the individual nuclide doses are summed to arrive at the total dose rate for the release source.

K_i = the whole body dose factor due to gamma emissions for each noble gas nuclide reported in the release source. (mrem-m^3/µCi-yr)

L_i = the skin dose factor due to beta emissions for each noble gas nuclide (i) reported in the assay of the release source. (mrem-m^3/µCi-yr)

M_i = the air dose factor due to gamma emissions for each noble gas nuclide (i) reported in the assay of the release source. The constant 1.1 converts mrad to mrem since the units of M_i are in (mrad-m^3/µCi-yr)

(X/Q) = for ground level, the highest calculated annual long term historic relative concentration for any of the 16 sectors, at or beyond the exclusion area boundary (sec/m^3)

$(Q\ DOT)_i$ = The release rate of noble gas nuclide (i) in µCi/sec from the release source of interest

1. Simplified Whole Body Dose Rate Calculation

From an evaluation of past releases, an effective whole body dose factor (K_{eff}) can be derived. This dose factor is in effect a weighted average whole body dose factor, i.e., weighted by the radionuclide distribution typical of past operation. (Refer to Appendix C for a detailed explanation and evaluation of K_{eff}). The value of K_{eff} has been derived from the radioactive noble gas effluents for the years 1978, 1979, and 1980. The value is:

$$K_{eff} = 6.8\ X\ 10^2\ \frac{mrem\text{-}m^3}{\mu Ci\text{-}yr}$$

2.2 (continued)

1. (continued)

This value may be used in conjunction with the total noble gas release rate $(Q\ DOT)_i$ to verify that the dose rate is within the allowable limits. To allow for any unexpected variability in the radionuclide distribution, a conservatism factor of 0.8 is introduced into the calculation. The simplified equation is:

$$DR_{WB} = \frac{K_{eff}\ (X/Q)}{0.8} \sum_{i} (Q\ DOT)_i$$

To further simplify the determination, the historical annual average meteorological X/Q of 1.6×10^{-6} sec/m^3 (From Table M-1) may be substituted into the equation. Also, the dose limit of 500 mrem/yr may be substituted for DR_{WB}. Making these substitutions yields a single cumulative (or gross) noble gas release rate limit. This value is:

Noble gas release rate limit = 3.5×10^5 µCi/sec

As long as the noble gas release rates do not exceed this value (3.5×10^5 µCi/sec), no additional dose rate calculations are needed to verify compliance with Control 3.11.2.1.

2.2 (continued)

2. Setpoint Determination

To comply with Control 3.3.3.11, the alarm/trip setpoints are established to ensure that the noble gas releases do not exceed the value of 3.5 X 10^5 µCi/sec, which corresponds to a whole body dose rate of 500 mrem/yr. The method that follows is a step-by-step procedure for establishing the setpoints. To allow for multiple sources of releases from different or common release points, the allowable operating setpoints will be controlled administratively by allocating a percentage of the total allowable release to each of the release sources.

A. Determine (V) the maximum volume release rate potential from the in-plant procedures for the release source under consideration. The units of (V) are ft^3/min.

B. Solve for A, the activity concentration in µCi/cc that would produce the Y control dose rate Limit.

$$A = \frac{3.5 \ X\ 10^5 \ \ \mu Ci}{sec} \ X \ \frac{min}{(V)\ ft^3} \ X \ \frac{ft_3}{2.8317 \ X\ 10^4 cc} \ X \ \frac{60 \ sec}{min}$$

A = µCi/cc

C. Refer to the µCi/cc vs. cpm curve for the Release Source's Gaseous Effluent Monitor cpm value (C), corresponding to the value of A above.

D. C is the 100% setpoint, assuming that there are no other release sources on the site.

E. Obtain the current % allocated to this release source from the gaseous waste management logs.

2.2 (continued)

2. (continued)

F. The Operating setpoint SP

$$SP = (C)\ cpm\ X\ \frac{\%\ allotted\ by\ in\text{-}plant\ procedures}{100\%}$$

The whole body dose is more limiting than the calculated skin dose. (Refer to Appendix C for a detailed evaluation.) Therefore, the skin dose rate calculations are not required if the simplified dose rate calculation is used (i.e., use of K_{eff} to determine release rate limits).

The calculation process of the following Section (2.2.3) is to be used if actual releases of noble gases exceed the above limit of 3.5 X 10^5 µCi/sec.

Under these conditions, a nuclide-by-nuclide evaluation is required to evaluate compliance with the dose rate limits of Control 3.11.2.1.

3. Whole Body and Skin Nuclide Specific Dose Rate Calculations

The following outline provides a step-by-step explanation of how the whole body dose rate is calculated on a nuclide-by-nuclide basis to evaluate compliance with Control 3.11.2.1. This method is only used if the actual releases exceed the value of 3.5 X 10^5 µCi/sec.

A. The (X/Q) value =__________sec/m^3 and__________is the most limiting sector at the exclusion area. (See Table M-1 for value and sector.)

B. Enter the release rate in ft^3/min of the release source and convert it to:

$$= \frac{(\qquad)ft^3}{min}\ X\ \frac{2.8317\ X\ 10^4 cc}{ft^3}\ X\ \frac{min}{60\ sec}$$

= cc/sec volume release rate

2.2 (continued)

3. (continued)

C. Solve for $(Q\ DOT)_i$ for nuclide (i) by obtaining the µCi/cc assay value of the release source and multiplying it by the product of 2.2.3.B above.

$(Q\ DOT)_i$ = (nuclide [i])

$$\frac{(assay)\quad \mu Ci}{cc} \; X \; \frac{(2.2.3.B\ value)\ cc}{\text{sec}}$$

$(Q\ DOT)_i$ = µCi/sec for nuclide (i)

D. To evaluate the whole body dose rate obtain the K_i value for nuclide (i) from Table G-2.

E. Solve for DR_{WBi}

$$DR_{WBi} = K_i(X/Q)\ (Q\ DOT)_i = \frac{mrem\text{-}m^3}{\mu Ci\text{-}yr} \; X \; \frac{\text{sec}}{m^3} \; X \; \frac{\mu Ci}{\text{sec}}$$

$DR_{WBi} = \frac{\text{mrem}}{\text{yr}}$ whole body dose from nuclide (i) for the specified release source

F. To evaluate the skin dose rate, obtain the L_i and M_i values from Table G-2 for nuclide (i).

G. Solve for $DR_{skin\ i}$

$DR_{skin\ i} = [L_i + 1.1\ M_i]\ (X/Q)(Q\ DOT)_i$

$DR_{skin\ i} = \frac{\text{mrem}}{\text{yr}}$ skin dose from nuclide (i) for the specified release source

2.2 (continued)

3. (continued)

H. Repeat steps 2.2.3.D through 2.2.3.G for each noble gas nuclide (i) reported in the assay of the release source.

I. The Dose Rate to the Whole Body from radioactive noble gas gamma radiation from the specified release source is:

$$DR_{WB} = \sum_{i}^{n} DR_{WBi}$$

J. The Dose Rate to the skin from noble gas radiation from the specified release source is:

$$DR_{skin} = \sum_{i}^{n} DR_{skin\ i}$$

The dose rate contribution of this release source shall be added to all other gaseous release sources that are in progress at the time of interest. Refer to in-plant procedures and logs to determine the Total Dose Rate to the Whole Body and Skin from noble gas effluents.

2.3 Determining the Radioiodine & Particulate Dose Rate to Any Organ From Gaseous Releases

Discussion - Control 3.11.2.1 limits the dose rate from I-131, I-133, tritium and all radionuclides in particulate form with half lives >eight days to $\leq$1500 mrem/yr to any organ. The following calculation method is provided for determining the dose rate from radioiodines (see 2.1) and particulates and is based on Section 5.2.1 and 5.2.1.1 through 5.2.1.3 in NUREG-0133, November 1978. The Infant is the controlling age group in the inhalation, ground plane, and cow/goat milk pathways, which are the only pathways considered for releases. The long term $(X/Q)_D$ (depleted) and (D/Q) values are based on historical MET data prior to implementing Appendix I. Only those nuclides that appear on their respective table will be considered. The equations are:

For Inhalation Pathway (excluding H-3):

$$DR_{I\&DP_T} = \sum_{i}^{n} R^{*}_{i_T} \, (X/Q)_D \, (Q\ DOT)_i$$

For Ground Plane:

$$DR_{I\&DP_T} = \sum_{i}^{n} P_{i_T} \, (D/Q)(Q\ DOT)_i$$

For Grass-Cow/Goat-Milk:

$$DR_{I\&DP_T} = \sum_{i}^{n} R^{*}_{i_T} \, (D/Q)(Q\ DOT)_i$$

For Tritium Releases (Inhalation & Grass-Cow/Goat-Milk):

$$DR_{H3_T} = R^{*}_{H\text{-}3_T} \, (X/Q)_D \, (Q\ DOT)_{H\text{-}3}$$

* Normally should be Pi_T, but Ri_T values are the same, thus use Ri_T tables in Appendix A.

2.3 (continued)

For Total Dose Rate from I & 8DP and H-3 To An Infant Organ T:

$$DR_T = \sum_Z [DR_{I\&8DP_T} + DR_{H\text{-}3_T}]$$

Where:

T	=	The organ of interest for the infant age group
z	=	The applicable pathways
$DR_{I\&8DP_T}$	=	Dose Rate in mrem/yr to the organ T from iodines and 8 day particulates
$DR_{H\text{-}3T}$	=	Dose Rate in mrem/yr to organ T from Tritium
DR_T	=	Total Dose Rate in mrem/yr to organ T from all pathways under consideration
$\sum_i^n$	=	A mathematical symbol to signify the operations to the right of the symbol are to be performed for each nuclide (i) through (n), and the individual nuclide dose rates are summed to arrive at the total dose rate from the pathway.
$\sum_Z$	=	A mathematical symbol to indicate that the total dose rate D_T to organ T is the sum of each of the pathways dose rates
R_i	=	The dose factor for nuclide (i) for organ T for the pathway specified (units vary by pathway)
P_i	=	The dose factor for instantaneous ground plane pathway in units of $\frac{\text{mrem-m}^2\text{ sec}}{\mu\text{Ci-yr}}$

2.3 (continued)

From an evaluation of the radioactive releases and environmental pathways, the grass-cow/goat-milk pathway has been identified as the most limiting pathway with the infant's thyroid being the critical organ. This pathway typically contributes >90% of the total dose received by the infant's thyroid and the radioiodine contribute essentially all of this dose. Therefore, it is possible to demonstrate compliance with the release rate limit of Control 3.11.2.1 for radioiodines and particulates by only evaluating the infant's thyroid dose for the release of radioiodines via the grass-cow/goat-milk pathway. The calculation method of Section 2.3.3 is used for this determination. If this limited analysis approach is used, the dose calculations for other radioactive particulate matter and other pathways need not be performed. Only the calculations of Section 2.3.3 for the radioiodines need be performed to demonstrate compliance with the Control dose rate limit.

The calculations of Sections 2.3.1, 2.3.2, 2.3.4, and 2.3.5 may be omitted. The dose rate calculations as specified in these sections are included for completeness and are to be used only for evaluating unusual circumstances where releases of particulate materials other than radioiodines in airborne releases are abnormally high. The calculations of Sections 2.3.1, 2.3.2, 2.3.4, and 2.3.5 will typically be used to demonstrate compliance with the dose rate limit of Control 3.11.2.1 for radioiodines and particulates when the measured releases of particulate material (other than radioiodines and with half lives >8 days) are >10 times the measured releases of radioiodines.

1. The Inhalation Dose Rate Method:

NOTE
The H-3 dose is calculated as per 2.3.4.

A. The controlling location is assumed to be an Infant located in the _______________ sector at the _______________ mile range. The $(X/Q)_D$ for this location is _______________ sec/m^3. This value is common to all nuclides. (See Table M-2 for value, sector and range.)

B. Enter the release rate in ft^3/min of the release source and convert to cc/sec.

$$= \frac{\_\_\_\_\_\_\_\_\_\_\_\_\_\_ ft^3}{min} \times \frac{2.8317 \times 10^4 cc}{ft^3} \times \frac{min}{60\ sec.} = cc/sec$$

2.3 (continued)

1. (continued)

C. Solve for $(Q\ DOT)_i$ for nuclide (i) by obtaining the µCi/cc assay value of the release source activity and multiplying it by the product of 2.3.1.B above.

$$(Q\ DOT)_i = \frac{(nuclide\ [i]\ assay)\ \mu Ci}{cc} \times \frac{(Value\ 2.3.1.B)\ cc}{\sec}$$

$(Q\ DOT)_i =$ µCi/sec for nuclide (i)

D. Obtain the R_i value from Table G-5 for the organ T.

E. Solve for DR_i

$$DR_{iT} = R_{iT}\ (X/Q)_D\ (Q\ DOT)_i = \frac{mrem\text{-}m^3}{\mu Ci\text{-}yr} \times \frac{\sec}{m^3} \times \frac{\mu Ci}{\sec}$$

$DR_{iT} = \frac{mrem}{yr}$ The Dose Rate to organ T from nuclide (i)

F. Repeat steps 2.3.1.C through 2.3.1.E for each nuclide (i) reported in the assay of the release source.

G. The Dose Rate to the Infants organ T from the Inhalation Pathway is:

$$DR_{Inhalation_T} = DR_1 + DR_2 + \_\_\_\_\_\_ + DR_n$$

for all nuclides except H-3. This dose rate shall be added to the other pathways as per 2.3.5 - Total Organ Dose.

NOTE
Steps 2.3.1.C through 2.3.1.G need to be completed for each organ T of the infant.

2.3 (continued)

2. The Ground Plane Dose Rate Method:

NOTE
Tritium dose via the ground plane is zero.

A. The controlling location is assumed to be an Infant located in the __________ sector at the __________ mile range. The (D/Q) for this location is __________ $1/m^2$. This value is common to all nuclides. (See Table M-2 for sector, range and value.)

B. Enter the release rate in ft^3/min of the release source and convert to cc/sec.

$$= \frac{\_\_\_\_\_\_ ft^3}{min} \; X \; \frac{2.8317 \; X \; 10^4 cc}{ft^3} \; X \; \frac{min}{60 \; sec.} = cc/sec$$

C. Solve for $(Q\ DOT)_i$ for nuclide (i) by obtaining the µCi/cc assay value from the release source activity and multiplying it by the product of 2.3.2.B above.

$$(Q\ DOT)_i = \frac{(nuclide\ [i]\ assay)\ \mu Ci}{cc} \; X \; \frac{(Value\ 2.3.2.B)\ cc}{sec}$$

$(Q\ DOT)_i$ = µCi/sec for nuclide (i)

D. Obtain the P_i value from Table G-3

E. Solve for DR_i

$$DR_i = P_{iT}\ (D/Q)\ (Q\ DOT)_i = \frac{mrem\text{-}m^2 - sec}{\mu Ci\text{-}yr} \; X \; \frac{1}{m^2} \; X \; \frac{\mu Ci}{sec}$$

$DR_i = \frac{mrem}{yr}$ The Dose Rate to organ T from nuclide (i)

2.3 (continued)

2. (continued)

F. Repeat steps 2.3.2.C through 2.3.2.E for each nuclide (i) reported in the assay of the release source.

G. The Dose Rate to the Infant's Whole Body from the Ground Plane Pathway is:

$$DR_{Gr\,Pl} = DR_1 + DR_2 + \_\_\_\_\_\_\_\_ + DR_n$$

for all nuclides. This dose rate shall be added to the other pathways as per 2.3.5.

3. The Grass-Cow/Goat-Milk Dose Rate Method:

NOTE
H-3 dose is calculated as per 2.3.4.

A. The controlling animal was established as a __________ located in the __________ sector at __________ miles. The (D/Q) for this location is __________ $1/m^2$. This value is common to all nuclides. (See Table M-3 for sector, range, and value.)

B. Enter the anticipated release rate in ft^3/min of the release source and convert to cc/sec.

$$= \frac{\_\_\_\_\_\_ ft^3}{min} \; X \; \frac{2.8317 \; X \; 10^4 cc}{ft_3} \; X \; \frac{min}{60 \; sec.} = cc/sec$$

C. Solve for $(Q\ DOT)_i$ for nuclide (i) by obtaining the µCi/cc assay value of the release source activity and multiplying it by the product of 2.3.3.B above.

$$(Q\ DOT)_i = \frac{(nuclide\ [i]\ assay)\ \ \mu Ci}{cc} \; X \; \frac{(value\ 2.3.3.B)\ cc}{sec}$$

$(Q\ DOT)_i$ = µCi/sec for nuclide (i)

2.3 (continued)

3. (continued)

D. Obtain the R_i value from Table G-6(7) (whichever is the controlling animal, cow/goat, for infant).

If the limited analysis approach is being used, limit the calculation to the infant thyroid.

E. Solve for DR_{iT}

$$DR_{iT} = R_{iT}\,(D/Q)\,(Q\ DOT)_i = \frac{mrem\text{-}m^2 - sec}{\mu Ci\text{-}yr} \times \frac{1}{m^2} \times \frac{\mu Ci}{sec}$$

$DR_{iT} = \frac{mrem}{yr}$ the Dose Rate to organ T from nuclide (i)

F. Repeat steps 2.3.3.C through 2.3.3.E for each nuclide (i) reported in the assay of the release source.

Only the radioiodines need to be included if the limited analysis approach is being used.

G. The Dose Rate to the Infant's organ T from Grass-__________-Milk pathway is:

DR_{grass}-________-$Milk_T$ = $DR_1 + DR_2 +$ ______ $+ DR_n$

for all nuclides. This dose rate shall be added to the other pathways as per 2.3.5 - Total Organ Dose.

> NOTE
> Steps 2.3.3.C through 2.3.3.G need to be completed for each organ of the Infant. Limit the calculation to the infant thyroid if the limited analysis approach is being used.

2.3 (continued)

4. The H-3 Dose Rate Method:

A. The controlling locations and their $(X/Q)_D$ values for each pathway are:

Inhalation - Infant at __________ range in the __________ sector.

$(X/Q)_D$ = sec/m^3 (See Table M-2 for range, sector and value)

Ground Plane - Does not apply to H-3

Grass-Cow/Goat-Milk-__________ located in the __________ sector at _____miles with an Infant at the exclusion area in the __________ sector drinking the milk. The $(X/Q)_D$ for the __________ location is $(X/Q)_D$ = __________ sec/m^3. (From Table M-6 at the range and sector corresponding to the location of the Milk Animal above.)

B. Enter the anticipated release rate in ft^3/min of the release source and convert it to cc/sec.

$$= \frac{\_\_\_\_ ft^3}{min} X \frac{2.8317 X 10^4 cc}{ft^3} X \frac{min}{60 \text{ sec.}}$$

= cc/sec volume release rate

C. Solve for $(Q\ DOT)_{H-3}$ for Tritium, by obtaining the µCi/cc assay value of the release source, and multiplying it by the product of 2.3.4.B above.

$$(Q\ DOT)_{H-3} = \frac{(H-3)\ \mu Ci}{cc} X \frac{(2.3.4.B\ value)\ cc}{sec}$$

$(Q\ DOT)_{H-3}$ = µCi/sec activity release rate

D. Obtain the Tritium dose factor (R_i) for Infant organ T from:

PATH	TABLE #
Inhalation	G-5
Grass-Cow/Goat-Milk	G-6(7)

2.3 (continued)

4. (continued)

E. Solve for D_{H-3} (Inhalation) using the $(X/Q)_D$ for inhalation from 2.3.4.A and R_{H-3} (Inhalation) from 2.3.4.D.

$$DR_{H-3_{inh_T}} = R_{H-3}\,(X/Q)_D\,(Q\ DOT)_{H-3}$$

$DR_{H-3_{inh_T}}$ = *mrem/yr from H-3 Infant Inhalation for organ T*

F. Solve for D_{H-3} (Grass-__________-Milk) using the $(X/Q)_D$ for Grass-__________-Milk from 2.3.4.A and R_{H-3} (Grass-__________-Milk) from 2.3.4.D

$$DR_{H-3_{G-\ -M_T}} = R_{H-3_{G-\ -M_T}}\,(X/Q)_D\,(Q\ DOT)_{H-3}$$

$DR_{H-3_{G-\ -M_T}}$ = *mrem/yr from H-3 Infant*

G. Repeat steps 2.3.4.D through 2.3.4.F for each Infant organ T of interest.

H. The individual organ dose rates from H-3 shall be added to the other organ pathway dose rates as per 2.3.5.

2.3 (continued)

5. Determining the Total Organ Dose Rate from Iodines, 8D-Particulates, and H-3 from Release Source(s)

   A. The following table describes all the pathways that must be summed to arrive at the total dose rate to an organ T:

PATHWAY	DOSE RATE	STEP # REF.
Inhalation (I&8DP)		2.3.1.G
Ground Plane (I&8DP)	(Whole Body only)	2.3.2.G
Gr-________Milk (I&8DP)		2.3.3.G
Inhalation (H-3)		2.3.4.E
Gr-________-Milk (H-3)		2.3.4.F
DR_T=	(sum of above)	

   B. Repeat the above summation for each Infant organ T.

   C. The DR_T above shall be added to all other release sources on the site that will be in progress at any instant. Refer to in-plant procedures and logs to determine the Total DR_T to each organ.

2.4 Determining the Gamma Air Dose for Radioactive Noble Gas Release Source(s)

Discussion - Control 3.11.2.2 limits the air dose due to noble gases in gaseous effluents for gamma radiation to <5 mrads for the quarter and to <10 mrads in any calendar year. The following calculation method is provided for determining the noble gas gamma air dose and is based on section 5.3.1 of NUREG-0133, November 1978. The dose calculation is independent of any age group. The equation may be used for Control dose calculation, the dose calculation for the annual report or for projecting dose, provided that the appropriate value of (X/Q) is used as outlined in the detailed explanation that follows. The equation for gamma air dose is:

$$D_\gamma\text{-}air = \sum_{i}^{n} 3.17 \times 10^{-8} M_i (X/Q) Q_i$$

2.4 (continued)

Where:

D_γ-air = gamma air dose in mrad from radioactive noble gases.

Σ = A mathematical symbol to signify the operations to the right side of the symbol are to be performed for each nuclide (i) through (n), and summed to arrive at the total dose, from all nuclides reported during the interval. No units apply.

3.17×10^{-8} = the inverse of the number of seconds per year with units of year/sec.

M_i = the gamma air dose factor for radioactive noble gas nuclide (i) in units of $\frac{\text{mrad-m}^3}{\mu\text{Ci-yr}}$

(X/Q) = the long term atmospheric dispersion factor for ground level releases in units of sec/m^3. The value of (X/Q) is the same for all nuclides (i) in the dose calculation, but the value of (X/Q) does vary depending on the Limiting Sector the Control is based on, etc.

Q_i = the number of micro-curies of nuclide (i) released (or projected) during the dose calculation exposure period. (e.g., month, quarter, or year)

2.4 (continued)

From an evaluation of past releases, a single effective gamma air dose factor (M_{eff}) has been derived, which is representative of the radionuclide abundances and corresponding dose contributions typical of past operation. (Refer to Appendix C for a detailed explanation and evaluation of M_{eff}). The value of M_{eff} has been derived from the radioactive noble gas effluents for the years 1978, 1979, and 1980. The value is

$$M_{eff} = 7.4 \ X \ 10^2 \ \frac{mrad/yr}{\mu Ci/m^3}$$

This value may be used in conjunction with the total noble gas releases (Q_i) to simplify the dose evaluation and to verify that the cumulative gamma air dose is within the limits of Control 3.11.2.2. To allow for any unexpected variability in the radionuclide distribution, a conservatism factor of 0.8 is introduced into the calculation. The simplified equation is

$$D_{\gamma-air} = \frac{3.17 \ X \ 10^{-8}}{0.8} \ M_{eff} \ X/Q \sum_i Q_i$$

For purposes of calculations, the appropriate meteorological dispersion (X/Q) from Table M-1 should be used. Control 3.11.2.2 requires that the doses be evaluated once per 31 days, (i.e., monthly). The quarterly dose limit is 5 mrads, which corresponds to a monthly allotment of 1.7 mrads. If the 1 7 mrads is substituted for $D_{\gamma-air}$, a cumulative noble gas monthly release objective can be calculated. This value is 36,000 Ci/month, noble gases.

As long as this value is not exceeded in any month, no additional calculations are needed to verify compliance with the quarterly noble gas release limits of Control 3.11.2.2. Also, the gamma air dose is more limiting than the beta air dose. Therefore, the beta air dose does not need to be calculated per Section 2.5 if the M_{eff} dose factor is used to determine the gamma air dose. Refer to Appendix C for a detailed evaluation and explanation.

The calculations of Section 2.5 may be omitted when this limited analysis approach is used, but should be performed if the radionuclide specific dose analysis is performed. Also, the radionuclide specific calculations will be performed for inclusion in annual reports.

2.4 (continued)

The following steps provide a detailed explanation of how the radionuclide specific dose is calculated. This method is used to evaluate quarterly doses in accordance with Control 3.11.2.2 if the releases of noble gases during any month of the quarter exceed 36,000 Ci.

1. To determine the applicable (X/Q) refer to Table M-1 to obtain the value for the type of dose calculation being performed. (i.e., Quarterly Control or Dose Projection for examples). This value of (X/Q) applies to each nuclide (i).

2. Determine (M_i) the gamma air dose factor for nuclide (i) from Table G-2.

3. Obtain the micro-Curies of nuclide (i) from the in-plant radioactive gaseous waste management logs for the sources under consideration during the time interval.

4. Solve for D_i as follows:

$$D_i = \frac{3.17 \ X\ 10^{-8}\ yr}{sec} \ X \ \frac{M_i \ mrad\text{-}m^3}{\mu Ci\text{-}yr} \ X \ \frac{(X/Q)\ sec}{m^3} \ X \ \frac{Q_i \ \mu Ci}{1}$$

D_i = mrad = the dose from nuclide (i)

5. Perform steps 2.4.2 through 2.4.4 for each nuclide (i) reported during the time interval in the source.

6. The total gamma air dose for the pathway is determined by summing the D_i dose of each nuclide (i) to obtain D_γ-air dose.

$D_{\gamma\text{-air}} = D_1 + D_2 + \_\_\_\_\_ + D_n$ = mrad

NOTE
Compliance with a 1/31 day Control, Quarterly Control, yearly or 12 consecutive months Control can be demonstrated by the limited analysis approach using M_{eff}. Using this method only requires that steps 2.4.2 through 2.4.5 be performed one time, remembering that the dose must be divided by 0.8, the conservatism factor.

7. Refer to in-plant procedures for comparing the calculated dose to any applicable limits that might apply.

2.5 Determining the Beta Air Dose for Radioactive Noble Gas Releases

Discussion - Control 3.11.2.2 limits the quarterly air dose due to beta radiation from noble gases in gaseous effluents to <10 mrads in any calendar quarter and <20 mrads in any calendar year. The following calculation method is provided for determining the beta air dose and is based on Section 5.3.1 of NUREG-0133, November 1978. The dose calculation is independent of any age group. The equation may be used for Control dose calculation, dose calculation for annual reports, or for projecting dose, provided that the appropriate value of (X/Q) is used as outlined in the detailed explanation that follows.

The equation for beta air dose is:

$$D_{B\text{-}air} \sum_{i}^{n} = 3.17 \times 10^{-8} N_i (X/Q)\ Q_i$$

Where:

$D_{B\text{-}air}$ = beta air dose in mrad from radioactive noble gases.

$\sum_{i}^{n}$ = a mathematical symbol to signify the operations to the right side of the symbol are to be performed for each nuclide (i) through (n), and summed to arrive at the total dose, from all nuclides reported during the interval. No units apply.

3.17×10^{-8} = the inverse of the number of seconds per year with units of year/sec.

N_i = the beta air dose factor for radioactive noble gas nuclide (i) in units of $\frac{\text{mrad-m}^3}{\mu\text{Ci-yr}}$

(X/Q) = the long term atmospheric dispersion factor for ground level releases in units of sec/m^3. The value of (X/Q) is the same for all nuclides (i) in the dose calculation, but the value of (X/Q) does vary depending on the Limiting Sector the Control is based on, etc.

Q_i = the number of micro-Curies of nuclide (i) released (or projected) during the dose calculation exposure period

2.5 (continued)

The beta air dose does not have to be evaluated if the noble gas gamma air dose is evaluated by the use of the effective gamma air dose factor (M_{eff}). However, if the nuclide specific dose calculation is used to evaluate compliance with the quarterly gamma air dose limits (Section 2.4), the beta air dose should also be evaluated as outlined below for the purpose of evaluating compliance with the quarterly beta air dose limits of Control 3.11.2.2. The following steps provide a detailed explanation of how the dose is calculated.

1. To determine the applicable (X/Q) refer to Table M-1 to obtain the value for the type of dose calculation being performed (i.e., quarterly Control or Dose projection for examples). This value of (X/Q) applies to each nuclide (i).

2. Determine (N_i) the beta air dose factor for nuclide (i) from Table G-2.

3. Obtain the micro-curies of nuclide (i) from the in-plant radioactive gaseous waste management logs for the source under consideration during the time interval.

4. Solve for D_i as follows:

$$D_i = \frac{3.17 \ X \ 10^{-8} \ yr}{\text{sec}} \ X \ \frac{N_i \ mrad\text{-}m^3}{\mu Ci\text{-}yr} \ X \ \frac{(X/Q) \ \text{sec}}{M^3} \ X \ \frac{Q_i \ \mu Ci}{1}$$

   D_i = mrad = the dose from nuclide (i)

5. Perform steps 2.5.2 through 2.5.4 for each nuclide (i) reported during the time interval in the release source.

6. The total beta air dose for the pathway is determined by summing the D_i dose of each nuclide (i) to obtain $D_{B\text{-}air}$ dose.

   $D_{B\text{-}air} = D_1 + D_2$ ________ $+ D_n$ = mrad

7. Refer to in-plant procedures for comparing the calculated dose to any applicable limits that might apply.

2.6 Determining the Radioiodine and Particulate Dose To Any Organ From Cumulative Releases

Discussion - Control 3.11.2.3 limits the dose to the whole body or any organ resulting from the release of I-131, I-133, tritium, and particulates with half-lives >8 days to ≤7.5 mrem during any calendar quarter and ≤15 mrem during any calendar year. The following calculation method is provided for determining the critical organ dose due to releases of radioiodines and particulates and is based on Section 5.3.1 of NUREG-0133, November 1978. The equation can be used for any age group provided that the appropriate dose factors are used and the total dose reflects only those pathways that are applicable to the age group. The $(X/Q)_D$ symbol represents a DEPLETED-(X/Q) which is different from the Noble Gas (X/Q) in that $(X/Q)_D$ takes into account the loss of I&8DP and H-3 from the plume as the semi-infinite cloud travels over a given distance. The (D/Q) dispersion factor represents the rate of fallout from the cloud that affects a square meter of ground at various distances from the site. The I&8DP and H-3 notations refer to I-131, I-133 Particulates having half-lives >8 days, and Tritium. For ease of calculations, dose from other Iodine nuclides may be included (see 2.1). Tritium calculations are always based on $(X/Q)_D$. The first step is to calculate the I&8DP and H-3 dose for each pathway that applies to a given age group. The total dose to an organ can then be determined by summing the pathways that apply to the receptor in the sector.

The equations are:

For Inhalation Pathway (excluding H-3):

$$D_{I\&8DP_{\tau}} = \sum_{i}^{n} 3.17 \times 10^{-8} R_i \, (X/Q)_D Q_i$$

For Ground Plane or Grass-Cow/Goat-Milk

$$D_{I\&8DP_{\tau}} = \sum_{i}^{n} 3.17 \times 10^{-8} R_i \, (D/Q) Q_i$$

For each pathway above (excluding Ground Plane) For Tritium:

$$D_{H\text{-}3_{\tau}} = 3.17 \times 10^{-8} R_{H\text{-}3\tau} (X/Q)_D Q_i$$

2.6 (continued)

For Total Dose from Particulate Gaseous effluent to organ T of a specified age group:

$$D_T = \sum_Z [D_{I\&8DP} + D_{H\text{-}3}]$$

Where:

T = the organ of interest of a specified age group

z = the applicable pathways for the age group of interest

$D_{I\&8DP}$ = Dose in mrem to the organ T of a specified age group from radioiodines and 8D Particulates

$D_{H\text{-}3}$ = Dose in mrem to the organ T of a specified age group from Tritium

D_T = Total Dose in mrem to the organ T of a specified age group from Gaseous particulate Effluents

$\sum_i^n$ = A mathematical symbol to signify the operations to the right of the symbol are to be performed for each nuclide (i) through (n), and the individual nuclide doses are summed to arrive at the total dose from the pathway of interest to organ T.

$\sum_Z$ = A mathematical symbol to indicate that the total dose D_T to organ T is the sum of each of the pathway doses of I&8DP and H-3 from gaseous particulate effluents.

3.17×10^{-8} = The inverse of the number of seconds per year with units of year/sec.

R_i = The dose factor for nuclide (i) (or H-3) for pathway Z to organ T of the specified age group. The units are either

$$\frac{mrem\text{-}m^3}{yr\text{-}\mu Ci} \text{ for pathways using } (X/Q)_D \quad OR \quad \frac{mrem\text{-}m^2\text{ - sec}}{yr\text{-}\mu Ci} \text{ for pathways using } (D/Q)$$

2.6 (continued)

$(X/Q)_D$ = The depleted-(X/Q) value for a specific location where the receptor is located (see discussion). The units are sec/m^3

(D/Q) = the deposition value for a specific location where the receptor is located (see discussion). The units are 1/m^2 where m=meters.

Q_i = The number of micro-Curies of nuclide (i) released (or projected) during the dose calculation exposure period.

Q_{H-3} = the number of micro-Curies of H-3 released (or projected) during the dose calculation exposure period.

As discussed in Section 2.3, the grass-cow/goat-milk pathway has been identified as the most limiting pathway with the infant's thyroid being the critical organ. This pathway typically contributes >90% of the total dose received by the infant's thyroid and the radioiodine contributes essentially all of this dose. Therefore, it is possible to demonstrate compliance with the dose limit of Control 3.11.2.3 for radioiodines and particulates by only evaluating the infant's thyroid dose due to the release of radioiodines via the grass-cow/goat-milk pathway. The calculation method of Section 2.6.3 is used for this determination.

The dose determined by Section 2.6.3 should be divided by a conservatism factor of 0.8. This added conservatism provides assurance that the dose determined by this limited analysis approach will not be < the dose that would be determined by evaluating all radionuclides and all pathways. If this limited analysis approach is used, the dose calculations for other radioactive particulate matter and other pathways need not be performed. Only the calculations of Section 2.6.3 for the radioiodines are required to demonstrate compliance with the Control dose limit. However, for the dose assessment included in Annual Reports, doses will be evaluated for the infant age groups and all organs via all designated pathways from radioiodines and particulates measured in the gaseous effluents according to the sampling and analyses required in Control Table 4.11-2. The following steps provide a detailed explanation of how the dose is calculated for the given pathways:

2.6 (continued)

1. The Inhalation Dose Pathway Method:

NOTE
The H-3 dose should be calculated as per 2.6.4.

A. Determine the applicable $(X/Q)_D$ from Table M-2 for the location where the receptor is located. This value is common to each nuclide (i)

B. Determine the R_i factor of nuclide (i) for the organ T and age group from Table G-5.

C. Obtain the micro-Curies (Q_i) of nuclide (i) from the radioactive gas waste management logs for the release source(s) under consideration during the time interval.

D. Solve for D_i

$$D_i = 3.17 \times 10^{-8} Ri(X/Q)_D Q_i$$

D_i = ________ mrem from nuclide (i)

E. Perform steps 2.6.1.B through 2.6.1.D for each nuclide (i) reported during the time interval for each organ.

F. The Inhalation dose to organ T of the specified age group is determined by summing the D_i Dose of each nuclide (i)

$$D_{\substack{Inhalation \\ (Age\ Group)}} = D_1 + D_2 + \_\_\_\_ + D_n = \_\_\_\_\_\_ \text{ mrem}$$

Refer to 2.6.5 to determine the total dose to organ T from radioiodines & 8D Particulates

2.6 (continued)

2. The Ground Plane Dose Pathway Method:

> **NOTE**
> Tritium dose via the ground plane is zero. The Whole Body is the only organ considered for the Ground Plane pathway dose.

A. Determine the applicable (D/Q) from Table M-2 for the location where the receptor is located. This (D/Q) value is common to each nuclide (i)

B. Determine the Ri factor of nuclide (i) for the whole body from Table G-4. The ground plane pathway dose is the same for all age groups.

C. Obtain the micro-Curies (Q_i) of nuclide (i) from the radioactive gas waste management logs for the source under consideration.

D. Solve for D_i

$$D_i = 3.17 \times 10^{-8} R_i\,(D/Q)Q_i$$

D_i = mrem for nuclide (i)

E. Perform steps 2.6.2.B through 2.6.2.D for each nuclide (i) reported during the time interval.

F. The Ground Plane dose to the whole body is determined by summing the Di Dose of each nuclide (i)

$D_{Gr.Pl.\text{-}WBody} = D_1 + D_2 + \_\_\_\_ + D_n =$ mrem

Refer to step 2.6.5 to calculate total dose to the Whole Body.

2.6 (continued)

3. The Grass-Cow/Goat-Milk Dose Pathway Method:

NOTE
Tritium dose is calculated as per 2.6.4.

A. A cow, or a goat, will be the controlling animal; (i.e., dose will not be the sum of each animal), as the human receptor is assumed to drink milk from only the most restrictive animal. Refer to Table M-3 to determine which animal is controlling based on its (D/Q).

B. Determine the dose factor R_i for nuclide (i), for organ T, from

1. From Table G-6 for a cow, or;

2. From Table G-7 for a goat.

If the limited analysis approach is being used, limit the calculation to the infant thyroid.

C. Obtain the micro-Curies (Q_i) of nuclide (i) from the radioactive gas waste management logs for the release source under consideration during the time interval.

D. Solve for D_i

$D_i = 3.17 \times 10^{-8} R_i (D/Q) Q_i$

D_i = mrem from nuclide (i)

E. Perform steps 2.6.3.B through 2.6.3.D for each nuclide (i) reported during the time interval. Only the radioiodines need to be included if the limited analysis approach is used.

2.6 (continued)

3. (continued)

F. The Grass-Cow-Milk (or Grass-Goat-Milk) pathway dose to organ T is determined by summing the Di dose of each nuclide(i).

$D_{G\text{-}C\text{-}M}$ (or $D_{G\text{-}G\text{-}M}$) = $D_1 + D_2 +$ ______ $+ D_n =$ m:em

The dose to each organ should be calculated in the same manner with steps 2.6.3.B through 2.6.3.F. Refer to step 2.6.5 to determine the total dose to organ T from radioiodines &8D Particulates. If the limited analysis approach is being used the infant thyroid dose via the grass-cow(goat)-milk pathway is the only dose that needs to be determined. Section 2 6.5 can be omitted.

4. The Gaseous Tritium Dose (Each Pathway) Method:

A. The controlling locations for the pathway(s) has already been determined by:
   - Inhalation - as per 2.6.1.A
   - Ground Plane - not applicable for H-3
   - Grass-Cow/Goat-Milk - as per 2.6.3.A

B. Tritium dose calculations use the depleted $(X/Q)_D$ instead of (D/Q). Table M-2 describes where the $(X/Q)_D$ value should be obtained from.

C. Determine the Pathway Tritium dose factor ($R_{H\text{-}3}$) for the organ T of interest from the Table specified below:

AGE	INHALATION	MILK	
		COW	GOAT
Infant	G-5	G-6	G-7

D. Obtain the micro-Curies (Q) of Tritium from the radioactive gas waste management logs (for projected doses - the micro-Curies of nuclide (i) to be projected) for the release source(s) under consideration during the time interval. The dose can be calculated from a single release source, but the total dose for Control limits or quarterly reports shall be from all gaseous release sources.

2.6 (continued)

4. (continued)

E. Solve for D_{H-3}

$$D_{H-3} = 3.17 \times 10^{-8} R_{H-3}(X/Q)_D Q$$

D_{H-3} = mrem from Tritium in the specified pathway for organ T of the specified age group

5. Determining the Total Organ Dose From Iodines, 8D-Particulates, and H-3 From Cumulative Gaseous Releases

> **NOTE**
> Control dose limits for I&8DP shall consider dose from all release sources from the reactor unit of interest.

A. The following pathways shall be summed to arrive at the total dose to organ T from a release source, or if applicable to Control, from all release sources:

PATHWAY	DOSE (mrem)	STEP # REF.
Inhalation (I&8DP)		2.6.1.F
Ground Plane (I&8DP)	(Whole Body only)	2.6.2.F
Grass-________Milk (I&8DP)		2.6.3.F
Inhalation (H-3)		2.6.4.E
Grass-________-Milk (H-3)		2.6.4.E
$Dose_T$ =	(sum of above)	

B. The dose to each of the INFANT'S ORGANS shall be calculated:

BONE, LIVER, THYROID, KIDNEY, LUNG, WHOLE BODY, & GI-LLI

The INFANT organ receiving the highest exposure relative to its Control Limit is the most critical organ for the radioiodine & 8D Particulates gaseous effluents.

2.7 Projecting Dose for Radioactive Gaseous Effluents

Discussion - Control 3.11.2.4 requires that the waste gas holdup system be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see TS Figure 5-1-1) would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The following calculation method is provided for determining the projected doses. This method is based on using the results of the calculations performed in Sections 2.4 and 2.5.

1. Obtain the latest results of the monthly calculations of the gamma air dose (Section 2.4) and the beta air dose if performed (Section 2.5). These doses can be obtained from the in-plant records.

2. Divide these doses by the number of days the plant was operational during the month.

3. Multiply the quotient by the number of days the plant is projected to be operational during the next month. The product is the projected dose for the next month. The value should be adjusted as needed to account for any changes in failed-fuel or other identifiable operating conditions that could significantly alter the actual releases.

4. If the projected doses are >0.2 mrads gamma air dose or > 0.4 mrads beta air dose, the appropriate subsystems of the waste gas holdup system shall be used.

3.0 40 CFR 190 Dose Evaluation

Discussion - Dose or dose commitment to a real individual from all uranium fuel cycle sources be limited to ≤25 mrem to the whole body or any organ (except thyroid, which is limited to ≤75 mrem) over a period of 12 consecutive months. The following approach should be used to demonstrate compliance with these dose limits. This approach is based on NUREG-0133, Section 3.8.

3.1 Evaluation Bases

Dose evaluations to demonstrate compliance with the above dose limits need only be performed if the quarterly doses calculated in Sections 1.4, 2.4 and 2.6 exceed twice the dose limits of Controls 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2a, 3.1.2.2b, 3.11.2.3a, and 3.11.2.3b respectively; i.e., quarterly doses exceeding 3 mrem to the whole body (liquid releases), 10 mrem to any organ (liquid releases), 10 mrads gamma air dose, 20 mrads beta air dose, or 15 mrem to the thyroid or any organ from radioiodines and particulates (atmospheric releases). Otherwise, no evaluations are required and the remainder of this section can be omitted.

3.2 Doses From Liquid Releases

For the evaluation of doses to real individuals from liquid releases, the same calculation method as employed in Section 1.4 will be used. However, more realistic assumptions will be made concerning the dilution and ingestion of fish and shellfish by individuals who live and fish in the area. Also, the results of the Radiological Environmental Monitoring program will be included in determining more realistic dose to these real people by providing data on actual measured levels of plant related radionuclides in the environment.

3.3 Doses From Atmospheric Releases

For the evaluation of doses to real individuals from the atmospheric releases, the same calculation methods as employed in Section 2.4 and 2.6 will be used. In Section 2.4, the whole body dose factor (K_i) should be substituted for the gamma air dose factor (M_i) to determine the whole body dose. Otherwise the same calculation sequence applies. However, more realistic assumptions will be made concerning the actual location of real individuals, the meteorological conditions, and the consumption of food (e.g., milk). Data obtained from the latest land use census (Control 3.12.2) should be used to determine locations for evaluating doses. Also, the results of the Radiological Environmental Monitoring program will be included in determining more realistic doses to these real people by providing data on actual measured levels of radioactivity and radiation at locations of interest.

4.0 Annual Radioactive Effluent Report

Discussion - The information contained in a annual report shall not apply to any Control. The reported values are based on actual release conditions instead of historical conditions that the Control dose calculations are based on. The Control dose limits are therefore included in item 1 of the report, for information only. The MPC's in item 2 of the report shall be those listed in Tables L-1 and G-1 of this manual. The average energy in item 3 of the report is not applicable to the St. Lucie Plant. The format, order of nuclides, and any values shown as an example in Tables 3.3 through 3.8 are samples only. Other formats are acceptable if they contain equivalent information. A table of contents should also accompany the report. The following format should be used:

RADIOACTIVE EFFLUENTS - SUPPLEMENTAL INFORMATION

1. Regulatory Limits:

1.1 For Radioactive liquid waste effluents:

a. The concentration of radioactive material released from the site (see Figure 5.1-1 in TS-A) 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} µCi/ml total activity.

b. 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 Fig. 5.1-1 in TS-A) shall be limited during any calendar quarter to $\leq$1.5 mrem to the whole body and to $\leq$5 mrem to any organ and $\leq$3 mrem to the whole body and $\leq$10 mrem to any organ during any calendar year.

1.2 For Radioactive Gaseous Waste Effluents:

a. The dose rate in unrestricted areas (see Fig. 5.1-1 in the TS-A) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

The dose rate limit for noble gases shall be $\leq$500 mrem/yr to the whole body and $\leq$3000 mrem/yr to the skin, and

The dose rate limit from I-131, I-133, Tritium, and particulates with half-lives >8 days shall be $\leq$1500 mrem/yr to any organ.

4.0 (continued)

1. (continued)

1.2 (continued)

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

During any calendar quarter, to ≤5 mrad for gamma radiation and ≤10 mrad for beta radiation and during any calendar year to ≤10 mrad for gamma radiation and ≤20 mrad for beta radiation

c. The dose to a MEMBER OF THE PUBLIC from I-131, I-133, Tritium, and all radionuclide in particulate form, with half-lives >8 days in gaseous effluents released from each reactor unit to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1 in the TS-A) shall be limited to the following:

During any calendar quarter to ≤7.5 mrem to any organ, and during any calendar year to ≤15 mrem to any organ.

2. Maximum Permissible Concentrations:

Air - as per attached Table G-1

Water - as per attached Table L-1

3. Average energy of fission and activation gases in gaseous effluents is not applicable to the St. Lucie Plant.

4.0 (continued)

4. Measurements and Approximations of Total Radioactivity:

A summary of liquid effluent accounting methods is described in Table 3.1.

A summary of gaseous effluent accounting methods is described in Table 3.2.

Estimate of Errors:

	LIQUID		GASEOUS	
Error Topic	Avg. %	Max. %	Avg. %	Max. %
Release Point Mixing	2	5	NA	NA
Sampling	1	5	2	5
Sample Preparation	1	5	1	5
Sample Analysis	3	10	3	10
Release Volume	2	5	4	15
Total %	9	30	10	35

(above values are examples only)

The predictability of error for radioactive releases can only be applied to nuclides that are predominant in sample spectrums. Nuclides that are near background relative to the predominant nuclides in a given sample could easily have errors greater than the above listed maximums.

4.0 (continued)

4. (continued)

TABLE 3.1
RADIOACTIVE LIQUID EFFLUENT SAMPLING AND ANALYSIS

LIQUID SOURCE	SAMPLING FREQUENCY	TYPE OF ANALYSIS	METHOD OF ANALYSIS
MONITOR TANK RELEASES[1]	EACH BATCH	PRINCIPAL GAMMA EMITTERS	p.h.a.
	MONTHLY COMPOSITE	TRITIUM	L.S.
		GROSS ALPHA	G.F.P.
	QUARTERLY COMPOSITE	Sr-89, Sr-90, Fe-55	C.S. & L.S.
STEAM GENERATOR BLOWDOWN RELEASES	FOUR PER MONTH	PRINCIPAL GAMMA EMITTERS AND DISSOLVED GASES	p.h.a.
	MONTHLY COMPOSITE	TRITIUM	L.S.
		GROSS ALPHA	G.F.P.
	QUARTERLY COMPOSITE	Sr-89, Sr-90, Fe-55	C.S. & L.S.

TABLE NOTATION:

[1] Boric Acid Evaporator condensate is normally recovered to the Primary Water Storage Tank for recycling into the reactor coolant system and normally does not contribute to liquid waste effluent totals.

p.h.a. - gamma spectrum pulse height analysis using Lithium Germanium detectors. All peaks are identified and quantified.

L.S. - Liquid Scintillation counting

C.S. - Chemical Separation

G.F.P. - Gas Flow Proportional Counting

4.0 (continued)

4. (continued)

TABLE 3.2
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS

GASEOUS SOURCE	SAMPLING FREQUENCY	TYPE OF ANALYSIS	METHOD OF ANALYSIS
Waste Gas Decay Tank Releases	Each Tank	Principal Gamma Emitters	G, p.h.a.
Containment Purge Releases	Each Purge	Principal Gamma Emitters	G, p.h.a.
		H-3	L.S.
Plant Vent	Four per Month	Principal Gamma Emitters	(G, C, P) - p.h.a.
		H-3	L.S.
	Monthly Composite (Particulates)	Gross Alpha	P - G. F. P.
	Quarterly Composite (Particulates)	Sr-90 Sr-89	C.S. & L.S.

G - Gaseous Grab Sample

C - Charcoal Filter Sample

P - Particulate Filter Sample

L.S. - Liquid Scintillation Counting

C.S. - Chemical Separation

p.h.a. - Gamma spectrum pulse height analysis using Lithium Germanium detectors. All peaks are identified and quantified.

G.F.P. - Gas Flow Proportional Counting

4.0 (continued)

5. Batch Releases

A. Liquid

1. Number of batch releases: ______
2. Total time period of batch releases: ______minutes
3. Maximum time period for a batch release: ______minutes
4. Average time period for a batch release: ______minutes
5. Minimum time period for a batch release: ______minutes
6. Average dilution stream flow during the period (see Note 1 on Table 3.3): ______GPM

All liquid releases are summarized in tables

B. Gaseous

1. Number of batch releases: ______
2. Total time period for batch releases: ______minutes
3. Maximum time period for a batch release: ______minutes
4. Average time period for batch releases: ______minutes
5. Minimum time period for a batch release: ______minutes

All gaseous waste releases are summarized in tables

4.0 (continued)

6. Unplanned Releases

   A. Liquid

      1. Number of releases: ______

      2. Total activity releases: ______Curies

   B. Gaseous

      1. Number of releases: ______

      2. Total activity released: ______Curies

   C. See attachments (if applicable) for:

      1. A description of the event and equipment involved.

      2. Cause(s) for the unplanned release.

      3. Actions taken to prevent a recurrence

      4. Consequences of the unplanned release

7. Description of dose assessment of radiation dose from radioactive effluents to the general public due to their activities inside the site are reported on the January annual report.

8. Offsite dose calculation manual revisions initiated during this reporting period. See Control 3.11.2.6 for required attachments to the Annual Report.

9. Solid waste and irradiated fuel shipments as per requirements of Control 3.11.2.6.

4.0 (continued)

10. Process Control Program (PCP) revisions as per requirements of TS 6.13.

11. Major changes to Radioactive Liquid, Gaseous and Solid Waste Treatment Systems as per requirements of Control 3.11.2.5.

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TABLE 3.3: LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES

	UNIT	QUARTER #	QUARTER #
A. Fission and Activation Products			
1. Total Release - (Not including Tritium, Gases, Alpha)	Ci	_____E	_____E
2. Average Diluted Concentration During Period	µCi/ML	_____E	_____E
B. Tritium			
1. Total Release	Ci	_____E	_____E
2. Average Diluted Concentration During Period	µCi/ML	_____E	_____E
C. Dissolved and Entrained Gases			
1. Total Release	Ci	_____E	_____E
2. Average Diluted Concentration During Period	µCi/ML	_____E	_____E
D. Gross Alpha Radioactivity			
1. Total Release	Ci	_____E	_____E

TABLE 3.3: LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES
(continued)

		UNIT	QUARTER #	QUARTER #
E.	Volume of Waste Released (Prior to Dilution)	LITERS	______E	______E
F.	Volume of Dilution Water Used During Period[1]	LITERS	______E	______E

1 - The volume reported should be for the entire interval of the reporting period, not just during release intervals. This volume should also be used to calculate average dilution stream flow during the period.

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TABLE 3.4: LIQUID EFFLUENTS (EXAMPLE FORMAT)

NUCLIDES RELEASED*	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER #	QUARTER #	QUARTER #	QUARTER #
I-131	CI	E	E	E	E
I-133	CI	E	E	E	E
I-135	CI	E	E	E	E
NA-24	CI	E	E	E	E
CR-51	CI	E	E	E	E
MN-54	CI	E	E	E	E
CO-57	CI	E	E	E	E
CO-58	CI	E	E	E	E
FE-59	CI	E	E	E	E
CO-60	CI	E	E	E	E
ZN-65	CI	E	E	E	E
NI-65	CI	E	E	E	E
AG-110	CI	E	E	E	E
SN-113	CI	E	E	E	E
SB-122	CI	E	E	E	E
SB-124	CI	E	E	E	E
W-187	CI	E	E	E	E
NP-239	CI	E	E	E	E
ZR-95	CI	E	E	E	E
MO-99	CI	E	E	E	E
RU-103	CI	E	E	E	E
CS-134	CI	E	E	E	E
CS-136	CI	E	E	E	E
CS-137	CI	E	E	E	E
BA-140	CI	E	E	E	E
CE-141	CI	E	E	E	E
BR-82	CI	E	E	E	E
ZR-97	CI	E	E	E	E
SB-125	CI	E	E	E	E

* All nuclides that were detected should be added to the partial list of the example format.

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TABLE 3.4: LIQUID EFFLUENTS (EXAMPLE FORMAT)
(continued)

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER #	QUARTER #	QUARTER #	QUARTER #
CE-144	CI	E	E	E	E
SR-89	CI	E	E	E	E
SR-90	CI	E	E	E	E
UNIDENTIFIED	CI	E	E	E	E
TOTAL FOR PERIOD (ABOVE)	CI	E	E	E	E

AR-41	CI	E	E	E	E
KR-85	CI	E	E	E	E
XE-131M	CI	E	E	E	E
XE-133	CI	E	E	E	E
XE-133M	CI	E	E	E	E
XE-135	CI	E	E	E	E

ST. LUCIE PLANT
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METHODOLOGY SECTION

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT #_____
TABLE 3.5
LIQUID EFFLUENTS - DOSE SUMMATION

Age Group: Adult Location: Any Adult

Exposure Interval: From_______________ Through________________

Fish & Shellfish Pathway to Organ	CALENDAR YEAR DOSE (mrem)
BONE	
LIVER	
THYROID	
KIDNEY	
LUNG	
GI-LLI	
WHOLE BODY	

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FLORIDA POWER & LIGHT COMPANY

ST. LUCIE UNIT #_____

ANNUAL REPORT - ____/____/____ THROUGH ____/____/____

TABLE 3.6: GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

	UNIT	QUARTER #	QUARTER #
A. Fission and Activation Gases			
1. Total Release	Ci	______E	______E
2. Average Release Rate For Period	µCi/SEC	______E	______E
B. Iodines			
1. Total Iodine-131	Ci	______E	______E
2. Average Release Rate for Period	µCi/SEC	______E	______E
C. Particulates			
1. Particulates T-1/2 > 8 Days	Ci	______E	______E
2. Average Release Rate for Period	µCi/SEC	______E	______E
3. Gross Alpha Radioactivity	Ci	______E	______E
D. Tritium			
1. Total Release	Ci	______E	______E
2. Average Release Rate for Period	µCi/SEC	______E	______E

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ST. LUCIE UNIT #_____

ANNUAL REPORT - ____/____/____ THROUGH ____/____/____

TABLE 3.7 GASEOUS EFFLUENTS - GROUND LEVEL RELEASES
(EXAMPLE FORMAT)

NUCLIDES RELEASED*	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER #	QUARTER #	QUARTER #	QUARTER #
1. Fission Gases					
AR-41	CI	E	E	E	E
KR-85	CI	E	E	E	E
KR-85M	CI	E	E	E	E
KR-87	CI	E	E	E	E
KR-88	CI	E	E	E	E
XE-131M	CI	E	E	E	E
XE-133	CI	E	E	E	E
XE-133M	CI	E	E	E	E
XE-135	CI	E	E	E	E
XE-135M	CI	E	E	E	E
XE-138	CI	E	E	E	E
UNIDENTIFIED	CI	E	E	E	E
TOTAL FOR PERIOD (ABOVE)	CI	E	E	E	E
2. Iodines					
I-131	CI	E	E	E	E
I-133	CI	E	E	E	E
I-135	CI	E	E	E	E
TOTAL FOR PERIOD (ABOVE)	CI	E	E	E	E
3. Particulates					
CO-58	CI	E	E	E	E
SR-89	CI	E	E	E	E
SR-90	CI	E	E	E	E

*All nuclides that were detected should be added to the partial list of the example format.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT #____
TABLE 3.8
GASEOUS EFFLUENTS - DOSE SUMMATION - CALENDAR YEAR
AGE GROUP: INFANT EXPOSURE INTERVAL: FROM________ THROUGH________

PATHWAY	BONE (mrem)	LIVER (mrem)	THYROID (mrem)	KIDNEY (mrem)	LUNG (mrem	GI-LLI (mrem)	WHOLE BODY (mrem)
Ground Plane (A)							
Grass- -Milk (B)							
Inhalation (A)							
TOTAL							

(A)	SECTOR:	RANGE: miles

(B)	COW / GOAT	SECTOR:	RANGE: miles

NOBLE GASES	CALENDAR YEAR (mrad)
Gamma Air Dose	
Beta Air Dose	
Sector:	Range: 0.97 miles

NOTE

The dose values above were calculated using actual meteorological data during the specified time interval with MET data reduced as per Reg. Guide 1.111, March 1976.

APPENDIX A

MPC, DOSE FACTOR

AND

HISTORICAL METEOROLOGICAL TABLES

ST. LUCIE PLANT
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TABLE L-1
MAXIMUM PERMISSIBLE CONCENTRATIONS IN WATER IN UNRESTRICTED AREAS

NOTE
If a nuclide is not listed below, refer to 10 CFR Part 20, Appendix B, Table 2 Effluent Concentrations Column 2, and use the most conservative MPC listed for the nuclide.

Nuclide	MPC (uCi/ml)	Nuclide	MPC (uCi/ml)	Nuclide	MPC (uCi/ml)
H-3	1 E-3	Y-90	7 E-6	Te-129	4 E-4
Na-24	5 E-5	Y-91m	2 E-3	Te-131m	8 E-6
P-32	9 E-6	Y-91	8 E-6	Te-131	8 E-5
Cr-51	5 E-4	Y-92	4 E-5	Te-132	9 E-6
Mn-54	3 E-5	Y-93	2 E-5	I-130	2 E-5
Mn-56	7 E-5	Zr-95	2 E-5	I-131	1 E-6
Fe-55	1 E-4	Zr-97	9 E-6	I-132	1 E-4
Fe-59	1 E-5	Nb-95	3 E-5	I-133	7 E-6
Co-57	6 E-5	Nb-97	3 E-4	I-134	4 E-4
Co-58	2 E-5	Mo-99	2 E-5	I-135	3 E-5
Co-60	3 E-6	Tc-99m	1 E-3	Cs-134	9 E-7
Ni-65	1 E-4	Tc-101	2 E-3	Cs-136	6 E-6
Cu-64	2 E-4	Ru-103	3 E-5	Cs-137	1 E-6
Zn-65	5 E-6	Ru-105	7 E-5	Cs-138	4 E-4
Zn-69	8 E-4	Ru-106	3 E-6	Ba-139	2 E-4
Br-82	4 E-5	Ag-110	6 E-6	Ba-140	8 E-6
Br-83	9 E-4	Sn-113	3 E-5	Ba-141	3 E-4
Br-84	4 E-4	In-113m	7 E-4	Ba-142	7 E-4
Rb-86	7 E-6	Sb-122	1 E-5	La-140	9 E-6
Rb-88	4 E-4	Sb-124	7 E-6	La-142	1 E-4
Rb-89	9 E-4	Sb-125	3 E-5	Ce-141	3 E-5
Sr-89	8 E-6	Te-125m	2 E-5	Ce-143	2 E-5
Sr-90	5 E-7	Te-127m	9 E-6	Ce-144	3 E-6
Sr-91	2 E-5	Te-127	1 E-4	Pr-144	6 E-4
Sr-92	4 E-5	Te-129m	7 E-6	W-187	3 E-5
				Np-239	2 E-5

TABLE L-2
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS FOR LIQUID DISCHARGES
PATHWAY - SALT WATER FISH AND SHELLFISH AGE GROUP - ADULT
ORGAN DOSE FACTOR (MREM/HR PER µCi/ML)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
H-3	0.	3.60E-01	3.60E-01	3.60E-01	3.60E-01	3.60E-01	3.60E-01
NA-24	6.08E-01	6.08E-01	6.08E-01	6.08E-01	6.08E-01	6.08E-01	6.08E-01
P-32	1.67E+07	1.05E+06	0.	0.	0.	1.88E+06	6.47E+05
CR-51	0.	0.	3.34E+00	1.23E+00	7.42E+00	1.41E+03	5.59E+00
MN-54	0.	7.07E+03	0.	2.10E+03	0.	2.17E+04	1.35E+03
MN-56	0.	1.78E+02	0.	2.26E+02	0.	5.68E+03	3.17E+01
FE-55	1.15E+05	5.19E+05	0.	0.	6.01E+05	2.03E+05	1.36E+05
FE-59	8.08E+04	1.92E+05	0.	0.	5.32E+04	6.33E+05	7.29E+04
CO-57	0.	1.42E+02	0.	0.	0.	3.60E+03	2.36E+02
CO-58	0.	6.05E+02	0.	0.	0.	1.22E+04	1.35E+03
CO-60	0.	1.74E+03	0.	0.	0.	3.26E+04	3.83E+03
NI-65	2.02E+02	2.63E+01	0.	0.	0.	6.65E+02	1.20E+01
CU-64	0.	2.15E+02	0.	5.41E+02	0.	1.83E+04	1.01E+02
ZN-65	1.62E+05	5.13E+05	0.	3.43E+05	0.	3.23E+05	2.32E+05
ZN-69	3.43E+02	6.60E+02	0.	4.27E+02	0.	9.87E+01	4.57E+01

Based on 1 µCi/sec release rate of each isotope in discharge flow of 1 cc/sec with no additional dilution

TABLE L-2
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS FOR LIQUID DISCHARGES
PATHWAY - SALT WATER FISH AND SHELLFISH AGE GROUP - ADULT

ORGAN DOSE FACTOR (MREM/HR PER μCi/ML)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
BR-82	0.	0.	0.	0.	0.	4.68E+00	4.08E+00
BR-83	0.	0.	0.	0.	0.	1.05E-01	7.26E-02
BR-84	0.	0.	0.	0.	0.	7.38E-07	9.42E-02
BR-85	0.	0.	0.	0.	0.	0.	3.86E-03
RB-86	0.	6.25E+02	0.	0.	0.	1.23E+02	2.91E+02
RB-88	0.	1.79E+00	0.	0.	0.	0.	9.50E-01
RB-89	0.	1.19E+00	0.	0.	0.	0.	8.38E-01
SR-89	5.01E+03	0.	0.	0.	0.	8.01E+02	1.44E+02
SR-90	1.23E+05	0.	0.	0.	0.	1.65E+03	3.02E+04
SR-91	9.43E+01	0.	0.	0.	0.	4.75E+02	4.15E+00
SR-92	3.50E+01	0.	0.	0.	0.	6.91E+02	1.51E+00
Y-90	6.07E+00	0.	0.	0.	0.	6.43E+04	1.63E-01
Y-91M	5.74E-02	0.	0.	0.	0.	1.68E-01	2.23E-03
Y-91	8.89E+01	0.	0.	0.	0.	4.89E+04	2.38E+00
Y-92	5.34E-01	0.	0.	0.	0.	9.33E+03	1.56E-02

Based on 1 μCi/sec release rate of each isotope in discharge flo*v* of 1 cc/sec with no additional dilution

ST. LUCIE PLANT
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TABLE L-2
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS FOR LIQUID DISCHARGES
PATHWAY - SALT WATER FISH AND SHELLFISH AGE GROUP - ADULT
ORGAN DOSE FACTOR (MREM/HR PER µCi/ML)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
Y-93	1.69E+00	0.	0.	0.	0.	5.36E+04	4.67E-02
ZR-95	1.60E+01	5.13E+00	0.	8.09E+00	0.	1.59E+04	3.47E+00
ZR-97	8.82E-01	1.78E-01	0.	2.69E-01	0.	5.51E+04	8.19E-02
NB-95	4.48E+02	2.49E+02	0.	2.47E+02	0.	1.51E+06	9.79E+01
NB-97	3.76E+00	9.50E-01	0.	1.11E+00	0.	3.51E+03	3.47E-01
MO-99	0.	1.28E+02	0.	2.90E+02	0.	2.97E+02	2.43E+01
TC-99M	1.30E-02	3.67E-02	0.	5.57E-01	1.80E-02	2.17E+01	4.67E-01
TC-101	1.33E-02	1.93E-02	0.	3.47E-01	9.82E-03	0.	1.89E-01
RU-103	1.07E+02	0.	0.	4.09E+02	0.	1.25E+04	4.61E+01
RU-105	8.90E+00	0.	0.	1.15E+02	0.	5.44E+03	3.51E+00
RU-106	1.59E+03	0.	0.	3.08E+03	0.	1.03E+05	2.01E+02
AG-110	1.57E+03	1.45E+03	0.	2.85E+03	0.	5.92E+05	8.62E+02
SB-124	2.78E+02	5.23E+00	6.71E-01	0.	2.15E+02	7.85E+03	1.10E+02
SB-125	2.20E+02	2.37E+00	1.96E-01	0.	2.30E+04	1.95E+03	4.42E+01
TE-125M	2.17E+02	7.89E+01	6.54E+01	8.83E+02	0.	8.67E+02	2.91E+01

Based on 1 µCi/sec release rate of each isotope in discharge flow of 1 cc/sec with no additional dilution

ST. LUCIE PLANT
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TABLE L-2
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS FOR LIQUID DISCHARGES
PATHWAY - SALT WATER FISH AND SHELLFISH AGE GROUP - ADULT
ORGAN DOSE FACTOR (MREM/HR PER µCi/ML)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
TE-127M	5.50E+02	1.92E+02	1.40E+02	2.23E+03	0.	1.84E+03	6.70E+01
TE-127	8.92E+00	3.20E+00	6.61E+00	3.63E+01	0.	7.04E+02	1.93E+00
TE-129M	9.32E+02	3.49E+02	3.20E+02	3.89E+03	0.	4.69E+03	1.48E+02
TE-129	2.55E+00	9.65E-01	1.95E+00	1.07E+01	0.	1.92E+00	6.21E-01
TE-131M	1.41E+02	6.87E+01	1.09E+02	6.95E+02	0.	6.81E+03	5.72E+01
TE-131	1.60E+00	6.68E-01	1.31E+00	7.00E+00	0.	2.39E-01	5.04E-01
TE-132	2.05E+03	1.33E+02	1.46E+02	1.28E+03	0.	6.25E+03	1.24E+02
I-130	3.98E+01	1.18E+02	1.50E+04	1.83E+02	0.	1.01E+02	4.63E+01
I-131	2.18E+02	3.13E+02	1.02E+05	5.36E+02	0.	8.24E+01	1.79E+02
I-132	1.07E+01	2.85E+01	3.76E+03	4.55E+01	0.	5.36E+00	1.01E+01
I-133	7.51E+01	1.30E+02	2.51E+04	2.27E+02	0.	1.15E+02	3.98E+01
I-134	5.57E+00	1.51E+01	1.96E+03	2.41E+01	0.	1.32E-02	5.41E+00
I-135	2.33E+01	6.14E+01	8.03E+03	9.77E+01	0.	6.88E+01	2.25E+01
CS-134	6.85E+03	1.63E+04	0.	5.29E+03	1.75E+03	2.85E+02	1.33E+04
CS-136	7.17E+02	2.83E+03	0.	1.58E+03	2.16E+02	3.22E+02	2.04E+03

Based on 1 µCi/sec release rate of each isotope in discharge flow of 1 cc/sec with no additional dilution

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TABLE L-2
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS FOR LIQUID DISCHARGES
PATHWAY - SALT WATER FISH AND SHELLFISH AGE GROUP - ADULT
ORGAN DOSE FACTOR (MREM/HR PER µCi/ML)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
CS-137	8.79E+03	1.20E+04	0.	4.09E+03	1.36E+03	2.31E+02	7.88E+03
CS-138	6.08E+00	1.20E+01	0.	8.84E+00	8.73E-01	5.12E-05	5.96E+00
BA-139	7.87E+00	5.61E-03	0.	5.24E-03	3.18E-03	1.39E+01	2.30E-01
BA-140	1.65E+03	2.07E+00	0.	7.04E-01	1.18E+00	3.39E+03	1.09E+02
BA-141	0.	2.89E-03	0.	2.68E-03	1.64E-03	1.80E-09	1.29E-01
BA-142	1.73E+00	1.78E-03	0.	1.50E-03	1.01E-03	0.	1.09E-01
LA-140	1.58E+00	7.95E-01	0.	0.	0.	5.83E+04	2.11E-01
LA-142	8.07E-02	3.67E-02	0.	0.	0.	2.68E+02	9.15E-03
CE-141	3.43E+00	2.32E+00	0.	1.08E+00	0.	8.87E+03	2.63E-01
CE-143	6.05E-01	4.47E+02	0.	1.97E-01	0	1.67E+04	4.95E-02
CE-144	1.79E+02	7.48E+01	0.	4.43E+01	0.	6.05E+04	9.60E+00
PR-144	1.91E-02	7.88E-03	0.	4.45E-03	0.	2.73E-09	9.65E-04
W-187	9.17E+00	7.68E+00	0.	0.	0.	2.51E+03	2.69E+00
NP-239	3.56E-02	3.50E-03	0.	1.08E-02	0.	7.12E+02	1.92E-03

Based on 1 µCi/sec release rate of each isotope in discharge flow of 1 cc/sec with no additional dilution

ST. LUCIE PLANT
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TABLE G-1
MAXIMUM PERMISSIBLE CONCENTRATIONS IN AIR IN UNRESTRICTED AREAS

> NOTE
> If a nuclide is not listed below, refer to 10 CFR Part 20, Appendix B, Table 2 Effluent Concentrations Column 1, and use the most conservative MPC listed for the nuclide.

Nuclide	MPC uCi/ml	Nuclide	MPC uCi/ml	Nuclide	MPC uCi/ml
Ar-41	8 E-7	Co-57	9 E-10	Sb-124	3 E-10
Kr-83m	5 E-5	Co-58	1 E-9	Sb-125	7 E-10
Kr-85m	1 E-7	Fe-59	5 E-10	Te-125m	1 E-9
Kr-85	7 E-7	Co-60	5 E-11	Te-127m	4 E-10
Kr-87	2 E-8	Zn-65	4 E-10	Te-129m	3 E-10
Kr-88	9 E-9	Rb-86	1 E-9	I-130	3 E-9
Kr-89	None	Rb-88	9 E-8	I-131	2 E-10
Kr-90	None	Sr-89	2 E-10	I-132	2 E-8
Xe-131m	2 E-6	Sr-90	6 E-12	I-133	1 E-9
Xe-133m	6 E-7	Y-91	2 E-10	I-134	6 E-8
Xe-133	5 E-7	Zr-95	4 E-10	I-135	6 E-9
Xe-135m	4 E-8	Nb-95	2 E-9	Cs-134	2 E-10
Xe-135	7 E-8	Ru-103	9 E-10	Cs-136	9 E-10
Xe-137	None	Ru-106	2 E-11	Cs-137	2 E-10
Xe-138	2 E-8	Ag-110	1 E-10	Ba-140	2 E-9
H-3	1 E-7	Sn-113	8 E-10	La-140	2 E-9
P-32	1 E-9	In-113m	2 E-7	Ce-141	8 E-10
Cr-51	3 E-8	Sn-123	2 E-10	Ce-144	2 E-11
Mn-54	1 E-9	Sn-126	8 E-11		

TABLE G-2
DOSE FACTORS FOR NOBLE GASES*

RADIONUCLIDE	WHOLE BODY DOSE FACTOR K_i (mrem/yr per µCi/m^3)	SKIN DOSE FACTOR L_i (mrem/yr per µCi/m^3)	GAMMA AIR DOSE FACTOR M_i (mrad/yr per µCi/m^3)	BETA AIR DOSE FACTOR N_i (mrad/yr per µCi/m^3)
Kr-83m	7.56E-02**	--------	1.93E+01	2.88E+02
Kr-85m	1.17E+03	1.46E+03	1.23E+03	1.97E+03
Kr-85	1.61E+01	1.34E+03	1.72E+01	1.95E+03
Kr-87	5.92E+03	9.73E+03	6.17E+03	1.03E+04
Kr-88	1.47E+04	2.37E+03	1.52E+04	2.93E+03
Kr-89	1.66E+04	1.01E+04	1.73E+04	1.06E+04
Kr-90	1.56E+04	7.29E+03	1.63E+04	7.83E+03
Xe-131m	9.15E+01	4.76E+02	1.56E+02	1.11E+03
Xe-133m	2.51E+02	9.94E+02	3.27E+02	1.48E+03
Xe-133	2.94E+02	3.06E+02	3.53E+02	1.05E+03
Xe-135m	3.12E+03	7.11E+02	3.36E+03	7.39E+02
Xe-135	1.81E+03	1.86E+03	1.92E+03	2.46E+03
Xe-137	1.42E+03	1.22E+04	1.51E+03	1.27E+04
Xe-138	8.83E+03	4.13E+03	9.21E+03	4.75E+03
Ar-41	8.84E+03	2.69E+03	9.30E+03	3.28E+03

* The listed dose factors are for radionuclides that may be detected in gaseous effluents.

** 7.56E-02 = 7.56 X 10^{-2}

TABLE G-3
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS P (I) FOR GASEOUS DISCHARGES
PATHWAY - GROUND PLANE DEPOSITION AGE GROUP - INFANT

ORGAN DOSE FACTOR (SQ. METER - MREM/YR PER µCi/Sec)

NUCLIDE	WHOLE BODY
H-3	0.
CR-51	6.68E+06
MN-54	1.10E+09
FE-59	3.92E+08
CO-57	1.64E+08
CO-58	5.27E+08
CO-60	4.40E+09
ZN-65	6.87E+08
RB-86	1.29E+07
SR-89	3.07E+04
SR-90	5.94E+05
Y-91	1.53E+06
ZR-95	6.94E+08
NB-95	1.95E+08
RU-103	1.57E+08
RU-106	2.99E+08
AG-110	3.18E+09

Based on 1 µCi/sec release rate of each isotope in and a Value of 1. for X/Q, depleted X/Q and Relative Deposition

TABLE G-3
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS P (I) FOR GASEOUS DISCHARGES
PATHWAY - GROUND PLANE DEPOSITION AGE GROUP - INFANT
ORGAN DOSE FACTOR (SQ. METER - MREM/YR PER µCi/Sec)

NUCLIDE	WHOLE BODY
SN-126	4.80E+09
SB-124	8.42E+08
SB-125	7.56E+08
TE-125M	2.19E+06
TE-127M	1.15E+06
TE-129M	5.49E+07
I-130	7.90E+06
I-131	2.46E+07
I-132	1.78E+06
I-133	3.54E+06
I-134	6.43E+05
I-135	3.66E+06
CS-134	2.82E+09
CS-136	2.13E+08
CS-137	1.15E+09
BA-140	2.39E+08
CE-141	1.95E+07
CE-144	9.52E+07

Based on 1 µCi/sec release rate of each isotope in and a Value of 1. for X/Q, depleted X/Q and Relative Deposition

TABLE G-4

ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS R (I) FOR GASEOUS DISCHARGES

PATHWAY - GROUND PLANE DEPOSITION AGE GROUP - CHILD - TEEN-ADULT & INFANT

ORGAN DOSE FACTOR (SQ. METER - MREM/YR PER µCi/Sec)

NUCLIDE	WHOLE BODY
H-3	0.
CR-51	4.68E+06
MN-54	1.38E+09
FE-59	2.75E+08
CO-57	1.89E+08
CO-58	3.80E+08
CO-60	2.15E+10
ZN-65	7.43E+08
RB-86	9.01E+06
SR-89	2.17E+04
SR-90	5.35E+06
Y-91	1.08E+06
ZR-95	5.01E+08
NB-95	1.36E+08
RU-103	1.10E+08
RU-106	4.19E+08
AG-110	3.58E+09

Based on 1 µCi/sec release rate of each isotope in and a Value of 1. for X/Q, depleted X/Q and Relative Deposition

TABLE G-4
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS R (I) FOR GASEOUS DISCHARGES
PATHWAY - GROUND PLANE DEPOSITION AGE GROUP - CHILD - TEEN-ADULT & INFANT
ORGAN DOSE FACTOR (SQ. METER - MREM/YR PER µCi/Sec)

NUCLIDE	WHOLE BODY
SN-126	5.16E+10
SB-124	5.98E+08
SB-125	2.30E+09
TE-125M	1.55E+06
TE-127M	8.79E+05
TE-129M	3.85E+07
I-130	5.53E+06
I-131	1.72E+07
I-132	1.25E+06
I-133	2.48E+06
I-134	4.50E+05
I-135	2.56E+06
CS-134	6.99E+09
CS-136	1.49E+08
CS-137	1.03E+10
BA-140	1.68E+08
CE-141	1.37E+07
CE-144	1.13E+08

Based on 1 µCi/sec release rate of each isotope in and a Value of 1. for X/Q, depleted X/Q and Relative Deposition

TABLE G-5
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS R(I)/P(I) FOR GASEOUS DISCHARGES
PATHWAY - INHALATION AGE GROUP - INFANT
ORGAN DOSE FACTOR (MREM/YR PER µCi/Cu Meter)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
H-3	0.	4.30E+02	4.30E+02	1.88E+02	4.30E+02	4.30E+02	4.30E+02
P-32	2.31E+05	1.35E+04	0.	0.	0.	1.51E+04	8.78E+03
CR-51	0.	0.	1.40E+01	3.99E+00	2.52E+03	5.81E+02	1.75E+01
MN-54	0.	6.93E+03	0.	1.72E+03	2.45E+05	1.35E+04	1.10E+03
FE-59	2.06E+03	4.86E+06	0.	0.	1.78E+05	3.29E+04	1.85E+03
CO-57	0.	1.21E+02	0.	0.	6.47E+04	5.50E+03	1.18E+02
CO-58	0.	1.18E+02	0.	0.	8.79E+05	1.21E+04	1.68E+02
CO-60	0.	8.40E+02	0.	0.	5.57E+06	3.28E+04	1.17E+03
ZN-65	5.67E+03	1.81E+04	0.	1.21E+04	1.53E+05	9.35E+03	8.15E+03
RB-86	0.	2.37E+04	0.	0.	0.	2.91E+03	1.03E+04
SR-89	4.31E+04	0.	0.	0.	2.31E+06	6.80E+04	1.24E+03
SR-90	1.32E+07	0.	0.	0.	1.53E+07	1.39E+05	8.06E+05
Y-91	5.98E+04	0.	0.	0.	2.63E+06	7.17E+04	1.60E+03
ZR-95	1.08E+04	2.73E+03	0.	9.48E+03	1.81E+06	1.41E+04	1.95E+03
NB-95	1.28E+03	5.75E+02	0.	1.35E+03	4.77E+05	1.21E+04	3.37E+02
RU-103	1.69E+02	0.	0.	1.02E+03	5.66E+05	1.58E+04	5.85E+01
RU-106	9.31E+03	0.	0.	2.34E+04	1.50E+07	1.76E+05	1.14E+03
AG-110	1.89E+03	1.75E+03	0.	3.44E+03	8.12E+05	5.29E+04	1.04E+03

Based on 1 µCi/sec release rate of each isotope in and a Value of 1. for X/Q, depleted X/Q and Relative Deposition

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TABLE G-5
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS R(I)/P(I) FOR GASEOUS DISCHARGES
PATHWAY - INHALATION AGE GROUP - INFANT
ORGAN DOSE FACTOR (MREM/YR PER µCi/Cu Meter)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
SN-123	3.11E+04	6.45E+02	6.45E+02	0.	3.61E+06	5.99E+04	1.02E+03
SN-126	2.21E+05	5.85E+03	1.72E+03	0.	1.64E+06	2.23E+04	8.40E+03
SB-124	5.46E+03	1.03E+02	1.32E+01	0.	4.34E+05	7.11E+04	2.17E+03
SB-125	1.16E+04	1.25E+02	1.03E+01	0.	3.85E+05	1.76E+04	2.32E+03
TE-125M	4.54E+02	1.95E+02	1.53E+02	2.17E+03	4.96E+05	1.36E+04	6.16E+01
TE-127M	2.21E+03	9.83E+02	5.75E+02	8.01E+03	1.68E+05	2.62E+04	2.74E+02
TE-129M	1.32E+03	5.80E+02	5.08E+02	6.40E+03	1.83E+06	7.32E+04	2.03E+02
I-130	8.02E+02	2.35E+03	3.05E+05	3.65E+03	0.	1.35E+03	9.25E+02
I-131	3.63E+04	4.27E+04	1.41E+07	1.07E+04	0.	1.07E+03	2.51E+04
I-132	2.03E+02	5.70E+02	7.67E+04	9.09E+02	0.	7.11E+01	2.03E+02
I-133	1.34E+04	1.93E+04	4.66E+06	4.55E+03	0.	2.28E+03	5.87E+03
I-134	1.13E+02	3.02E+02	4.02E+04	4.82E+02	0.	1.76E-01	1.08E+02
I-135	4.70E+02	1.22E+03	1.64E+05	1.95E+03	0.	9.18E+02	4.51E+02
CS-134	4.80E+05	8.25E+05	0.	5.04E+04	1.01E+05	1.37E+03	7.32E+04
CS-136	6.85E+03	2.56E+04	0.	1.50E+04	2.10E+03	2.04E+03	1.95E+04
CS-137	6.86E+05	7.31E+05	0.	3.89E+04	9.45E+04	1.32E+03	4.41E+04
BA-140	5.70E+03	4.27E+00	0.	2.93E+00	1.64E+06	3.88E+03	2.95E+02
CE-141	2.52E+03	1.55E+03	0.	1.10E+03	5.24E+05	2.06E+04	1.81E+02
CE-144	4.68E+05	1.82E+05	0.	1.48E+05	1.27E+07	1.61E+05	2.49E+04

Based on 1 µCi/sec release rate of each isotope in and a Value of 1. for X/Q, depleted X/Q and Relative Deposition

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TABLE G-6
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS R(I)/P(I) FOR GASEOUS DISCHARGES
PATHWAY - COWS MILK (CONTAMINATED FORAGE) AGE GROUP - INFANT
ORGAN DOSE FACTOR (SQ. METER - MREM/YR PER µCi/Sec)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
H-3	0.	2.37E+03	2.37E+03	1.04E+03	2.37E+03	2.37E+03	2.37E+03
P-32	1.82E+10	1.14E+09	0.	0.	0.	2.05E+09	7.05E+08
CR-51	0.	0.	1.82E+04	6.72E+03	4.04E+04	7.66E+06	3.05E+04
MN-54	0.	8.96E+06	0.	2.67E+06	0.	2.74E+07	1.71E+06
FE-59	3.17E+07	7.52E+07	0.	0.	2.09E+07	2.48E+08	2.86E+07
CO-57	0.	1.36E+06	0.	0.	0.	3.46E+07	2.27E+06
CO-58	0.	2.55E+07	0.	0.	0.	6.60E+07	6.24E+07
CO-60	0.	8.73E+07	0.	0.	0.	2.16E+08	2.09E+08
ZN-65	1.46E+09	4.65E+09	0.	3.11E+09	0.	2.93E+09	2.10E+09
RB-86	0.	2.77E+09	0.	0.	0.	5.45E+08	1.29E+09
SR-89	1.47E+10	0.	0.	0.	0.	2.75E+08	4.22E+08
SR-90	1.65E+11	0.	0.	0.	0.	1.61E+09	4.21E+10
Y-91	8.12E+04	0.	0.	0.	0.	5.37E+06	2.16E+03
ZR-95	2.12E+05	9.41E+04	0.	1.86E+04	0.	7.47E+07	5.56E+04
NB-95	5.49E+05	2.47E+05	0.	4.84E+04	0.	1.98E+08	1.45E+05
RU-103	8.30E+03	0.	0.	4.16E+03	0.	1.04E+05	2.86E+03
RU-106	2.01E+05	0.	0.	4.20E+04	0.	1.56E+06	2.46E+04
AG-110	6.21E+07	5.75E+07	0.	1.13E+08	0.	2.35E+10	3.42E+07

Based on 1 µCi/sec release rate of each isotope in and a Value of 1. for X/Q, depleted X/Q and Relative Deposition
Note: The units for C-14 and H-3 are (MREM/YR Per µCi/Cu. Meter)

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TABLE G-6
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS R(I)/P(I) FOR GASEOUS DISCHARGES

PATHWAY - COWS MILK (CONTAMINATED FORAGE) AGE GROUP - INFANT

ORGAN DOSE FACTOR (SQ. METER - MREM/YR PER µCi/Sec)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
SN-126	1.75E+09	3.48E+07	1.01E+07	0.	4.97E+06	1.16E+09	5.25E+07
SB-124	2.75E+07	5.19E+05	6.64E+04	0.	2.13E+07	7.78E+08	1.09E+07
SB-125	3.59E+07	3.27E+06	2.93E+06	3.96E+06	2.83E+09	2.43E+08	6.62E+06
TE-125M	1.57E+08	5.30E+07	5.18E+07	7.05E+07	0.	7.57E+07	2.10E+07
TE-127M	5.54E+07	1.93E+07	1.79E+07	2.00E+08	0.	3.24E+08	7.38E+06
TE-129M	5.87E+08	2.02E+08	2.21E+08	2.70E+08	0.	3.54E+08	8.95E+07
I-130	4.54E+05	1.35E+06	1.71E+08	2.09E+06	0.	1.15E+06	5.29E+05
I-131	2.59E+09	3.09E+09	9.94E+11	7.24E+08	0.	1.16E+08	1.81E+09
I-132	1.78E-01	4.76E-01	6.26E+01	7.58E-01	0.	8.93E-02	1.69E-01
I-133	3.75E+07	5.48E+07	1.30E+10	1.29E+07	0.	9.74E+06	1.66E+07
I-134	0.	0.	1.06E-09	0.	0.	0.	0.
I-135	1.49E+04	3.94E+04	5.15E+06	6.26E+04	8.07E-02	4.41E+04	1.44E+04
CS-134	4.43E+10	7.97E+10	0.	4.65E+09	9.12E+09	1.90E+08	6.75E+09
CS-136	2.78E+08	1.10E+09	0.	6.11E+08	8.37E+07	1.25E+08	7.90E+08
CS-137	6.44E+10	7.21E+10	0.	3.66E+09	8.69E+09	1.86E+08	4.14E+09
BA-140	2.45E+08	2.47E+05	0.	1.22E+04	1.51E+05	8.13E+06	1.27E+07
CE-141	2.65E+05	1.62E+05	0.	9.72E+03	0.	7.87E+07	1.90E+04
CE-144	2.10E+07	8.29E+06	0.	5.67E+05	0.	8.66E+08	1.13E+06

Based on 1 µCi/sec release rate of each isotope in and a value of 1. for X/Q, depleted X/Q and relative deposition.

Note: The units for C-14 and H-3 are (MREM/YR Per µCi/Cu. Meter)

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TABLE G-7
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS R(I)/P(I) FOR GASEOUS DISCHARGES
PATHWAY - GOATS MILK (CONTAMINATED FORAGE) AGE GROUP - INFANT
ORGAN DOSE FACTOR (SQ. METER - MREM/YR PER µCi/Sec)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
H-3	0.	4.84E+03	4.84E+03	2.11E+03	4.84E+03	4.84E+03	4.84E+03
P-32	2.19E+10	1.37E+09	0.	0.	0.	2.46E+09	8.46E+08
CR-51	0.	0.	2.19E+03	8.07E+02	4.85E+03	9.19E+05	3.66E+03
MN-54	0.	1.08E+06	0.	3.20E+05	0.	3.29E+06	2.05E+05
FE-59	4.12E+05	9.78E+05	0.	0.	2.72E+05	3.23E+06	3.72E+05
CO-57	0.	1.64E+05	0.	0.	0.	4.15E+06	2.72E+05
CO-58	0.	3.06E+06	0.	0.	0.	7.92E+06	7.49E+06
CO-60	0.	1.05E+07	0.	0.	0.	2.59E+07	2.51E+07
ZN-65	1.76E+08	5.57E+08	0.	3.73E+08	0.	3.51E+08	2.52E+08
RB-86	0.	3.32E+08	0.	0.	0.	6.54E+07	1.55E+08
SR-89	3.09E+10	0.	0.	0.	0.	5.77E+08	8.87E+08
SR-90	3.46E+11	0.	0.	0.	0.	3.35E+09	8.83E+10
Y-91	9.74E+03	0.	0.	0.	0.	6.45E+05	2.60E+02
ZR-95	2.54E+04	1.13E+04	0.	2.23E+03	0.	8.95E+06	6.67E+03
NB-95	6.59E+04	2.97E+04	0.	5.81E+03	0.	2.37E+07	1.75E+04
RU-103	9.96E+02	0.	0.	4.99E+02	0.	1.24E+04	3.43E+02
RU-106	2.41E+04	0.	0.	5.04E+03	0.	1.87E+05	2.96E+03
AG-110	7.45E+06	6.90E+06	0.	1.36E+07	0.	2.81E+09	4.10E+06

Based on 1 µCi/sec release rate of each isotope in and a Value of 1. for X/Q, depleted X/Q and Relative Deposition

Note: The units for C-14 and H-3 are 1MREM/Yr per µCi/Cu meter.

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TABLE G-7
ENVIRONMENTAL PATHWAY-DOSE CONVERSION FACTORS R(I)/P(I) FOR GASEOUS DISCHARGES
PATHWAY - GOATS MILK (CONTAMINATED FORAGE)   AGE GROUP - INFANT
ORGAN DOSE FACTOR   (SQ. METER - MREM/YR PER µCi/Sec)

NUCLIDE	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	WHOLE BODY
SN-126	2.10E+08	4.17E+06	1.22E+06	0.	5.97E+05	1.40E+08	6.30E+06
SB-124	3.30E+06	6.22E+04	7.97E+03	0.	2.56E+06	9.33E+07	1.30E+06
SB-125	4.31E+06	3.92E+05	3.52E+05	4.76E+05	3.40E+08	2.92E+07	7.94E+05
TE-125M	1.89E+07	6.36E+06	6.21E+06	8.46E+06	0.	9.09E+06	2.52E+06
TE-127M	6.64E+06	2.31E+06	2.15E+06	2.40E+07	0.	3.88E+07	8.85E+05
TE-129M	7.05E+07	2.42E+07	2.66E+07	3.23E+07	0.	4.25E+07	1.07E+07
I-130	5.45E+05	1.61E+06	2.05E+08	2.51E+06	0.	1.38E+06	6.35E+05
I-131	3.11E+09	3.70E+09	1.19E+12	9.28E+08	0.	1.39E+08	2.17E+09
I-132	2.13E-01	5.71E-01	7.51E+01	9.10E-01	0.	1.07E-01	2.03E-01
I-133	4.50E+07	6.57E+07	1.55E+10	1.55E+07	0.	1.17E+07	1.99E+07
I-134	0.	0.	1.27E-09	0.	0.	0.	0.
I-135	1.79E+04	4.72E+04	6.18E+06	7.51E+04	2.42E-01	5.29E+04	1.73E+04
CS-134	1.33E+11	2.39E+11	0.	1.39E+10	2.74E+10	5.69E+08	2.02E+10
CS-136	8.34E+08	3.29E+09	0.	1.83E+09	2.51E+08	3.74E+08	2.37E+09
CS-137	1.93E+11	2.16E+11	0.	1.10E+10	2.61E+10	5.59E+08	1.24E+10
BA-140	2.95E+07	2.96E+04	0.	1.47E+03	1.81E+04	9.76E+05	1.52E+06
CE-141	3.17E+04	1.95E+04	0.	1.17E+03	0.	9.44+06	2.28E+03
CE-144	2.52E+06	9.95E+05	0.	6.80E+04	0.	1.04E+08	1.36E+05

Based on 1 µCi/sec release rate of each isotope in and a value of 1. for X/Q, depleted X/Q and relative deposition.

Note: The units for C-14 and H-3 are 1MREM/Yr per µCi/Cu meter.

TABLE M-1

Selecting the Appropriate Long Term (X/Q) for Dose Calculations Involving Noble Gases for:

(1) Whole Body dose from instantaneous releases

(2) Skin dose from instantaneous releases

(3) Gamma air dose (cumulative)

(4) Beta air dose (cumulative)

TYPE OF DOSE CALCULATION	LIMITING RANGE (miles)	LIMITING Sector	(X/Q) VALUE sec/m^3
Instantaneous	0.97	NW	1.6 X 10^{-6}
1/31 days	0.97	1. Normally (X/Q) = 1.6 X 10^{-6} sec/m^3 2. May use option of actual meteorological data for time of concern.	
Quarterly Yearly	0.97		
12 Consecutive months	0.97		
Annual Report	0.97	N/A	Note-1

NOTE 1

The (X/Q) has to be calculated based on actual meteorological data that occurred during the period of interest. The sector of interest is N/A because the limiting (X/Q) will be determined from the actual meteorological data and may occur in any sector.

0.97 miles — Corresponds to the minimum site boundary distance in the north direction and 0.97 miles was chosen for all other sectors for ease of calculations when the averaging is done for quarterly reports.

TABLE M-2

Selecting the Appropriate Long Term $(X/Q)_D$ or (D/Q) for Dose

Calculations Involving Radioiodines & 8 D Particulates for:

(1) Inhalation (2) Tritium (All gas pathways) (3) Ground Plane

TYPE OF DOSE CALCULATION	LIMITING RANGE (miles)	LIMITING SECTOR (OL)	$(X/Q)_D$ sec/m^3	(D/Q) 1/m^2
Instantaneous	0.97	NW	B 1.3 X 10^{-6}	
		WNW		8.2 X 10^{-9}
Quarterly for Annual Reports	0.97	A	A, B	
	0.97	A		A
1/31 days, Qtr. yearly, Annual Total Dose	0.97	NW	B 1.3 X 10^{-6}	
	0.97	WNW		8.2 X 10^{-9}

(OL) Over land areas only

(A) To be determined by reduction of actual met data occurring during each quarter

(B) For Tritium in the Milk Animal Pathway, the $(X/Q)_D$ value should be that of the respective controlling sector and range where the Milk Animal is located as per Table M-3. Example: If a cow was located at 4.25 miles in NW sector, use the $(X/Q)_D$ for 4.25 miles NW.

TABLE M-3

Selecting the Appropriate Long Term (D/Q) for Dose Calculations Involving Radioiodines and 8D Particulates for Grass-Cow-Milk or Grass-Goat-Milk:

TYPE OF DOSE CALCULATION	LIMITING RANGE	LIMITING SECTOR	(D/Q) Value $1/m^2$
Release Rate	A	A	A
1/31 Days	B	B	B
Quarterly - Yearly	B	B	B
Annual (Calendar Year)	B	B	B
Annual Report	C	C	C

A. The worst cow or goat as per locations from land census. If no milk animal in any sector, assume a cow at 4.25 miles in the highest (D/Q) sector over land.

B. The historical (D/Q) of all land sectors with the worst cow or goat from each sector as reported in the Land Census. A 4.25 mile cow should be assumed in the worst sector over land when no milk animal is reported.

C. The highest (D/Q) at a milk animal location of all milk animals reported in the Land Census Report. (If no milk animals within 5 miles a 4.25 mile cow should be assumed in the sector having the highest (D/Q) at 4.25 miles over land). Actual Met Data should be used for the selection of the worst case milk animal and for the dose calculations. If both goat and milk animals are reported inside 5 miles, dose calculations should be performed on each animal and the higher dose animal contribution should be used.

The historical wind frequency fractions for each sector are listed in Table M-8.

ST. LUCIE PLANT
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TABLE M-4
TERRAIN CORRECTION FACTORS

Florida Power & Light Company
St. Lucie Unit 1
Hutchinson Island, Florida
Dames and Moore Job No: 4598 - 112

Terrain Correction Factors (PUFF / STRAIGHT LINE)
Period of Record: 8/29/77 to 8/31/78
Base Distance in Miles/Kilometers

AFFECTED SECTOR	DESIGN DISTANCE MILES	.25 .40	.75 1.21	1.25 2.01	1.75 2.82	2.25 3.62	2.75 4.42	3.25 5.23	3.75 6.03	4.25 6.84	4.75 7.64
NNE	0.	1.906	1.576	1.465	1.404	1.338	1.318	1.334	1.386	1.346	1.338
NE	0.	1.887	1.581	1.461	1.391	1.310	1.259	1.164	1.128	1.101	1.116
ENE	0.	1.452	1.230	1.122	1.081	1.047	1.033	.941	.941	.906	.902
E	0.	1.662	1.425	1.277	1.193	1.151	1.123	1.097	1.121	1.123	1.122
ESE	0.	1.690	1.483	1.328	1.260	1.246	1.190	1.134	1.094	1.032	.968
SE	0.	1.818	1.691	1.470	1.427	1.435	1.361	1.366	1.331	1.279	1.239
SSE	0.	1.812	1.586	1.370	1.302	1.270	1.263	1.229	1.193	1.171	1.151
S	0.	1.398	1.321	1.125	1.083	1.108	1.127	1.073	1.063	1.047	1.024
SSW	0.	1.534	1.411	1.296	1.192	1.205	1.132	1.135	1.116	1.077	1.060
SW	0.	1.685	1.492	1.294	1.233	1.200	1.222	1.160	1.160	1.198	1.196
WSW	0.	1.620	1.333	1.210	1.173	1.082	1.091	1.099	1.056	1.034	1.004
W	0.	1.651	1.415	1.290	1.218	1.154	1.099	1.081	1.067	1.093	1.083
WNW	0.	1.720	1.430	1.267	1.185	1.150	1.133	1.125	1.085	1.033	1.045
NW	0.	1.681	1.407	1.257	1.173	1.119	1.078	1.063	.995	.998	.978
NNW	0.	1.739	1.488	1.316	1.212	1.172	1.122	1.135	1.080	1.099	1.091
N	0.	1.816	1.524	1.389	1.285	1.257	1.263	1.285	1.267	1.231	1.213

Note 1: Any interpolations between stated mileages will be done by log-log

ST. LUCIE PLANT
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TABLE M-5
HISTORICAL LONG TERM - (X/Q) (Frequency corrected)

Terrain / Recirculation Adjusted   Program ANNXOQ9 Version - 11/18/76

Florida Power & Light Company
St. Lucie Unit 1
Hutchinson Island, Florida
Dames and Moore Job No: 1.4598 - 112

Average Annual Relative Concentration (sec/cubic meter)
Period of Record: 9/1/76 to 8/31/78
Base Distance in Miles/Kilometers

AFFECTED SECTOR	DESIGN DISTANCE MILES	.25 .40	.75 1.21	1.25 2.01	1.75 2.82	2.25 3.62	2.75 4.42	3.25 5.23	3.75 6.03	4.25 6.84	4.75 7.64
NNE	0.	1.1E-05	1.7E-06	7.8E-07	4.5E-07	3.1E-07	2.2E-07	1.7E-07	1.5E-07	1.2E-07	1.0E-07
NE	0.	1.3E-05	2.1E-06	8.9E-07	5.1E-07	3.4E-07	2.4E-07	1.7E-07	1.4E-07	1.1E-07	9.8E-08
ENE	0.	9.3E-06	1.4E-06	6.2E-07	3.7E-07	2.5E-07	1.9E-07	1.3E-07	1.1E-07	8.8E-08	7.5E-08
E	0.	9.8E-06	1.6E-06	6.5E-07	3.7E-07	2.5E-07	1.8E-07	1.4E-07	1.2E-07	9.9E-08	8.4E-08
ESE	0.	1.2E-05	1.9E-06	8.1E-07	4.8E-07	3.2E-07	2.4E-07	1.8E-07	1.4E-07	1.1E-07	9.0E-08
SE	0.	1.4E-05	2.4E-06	9.7E-07	5.7E-07	4.0E-07	2.9E-07	2.3E-07	1.9E-07	1.4E-07	1.2E-07
SSE	0.	1.1E-05	1.7E-06	7.3E-07	4.3E-07	2.9E-07	2.1E-07	1.6E-07	1.3E-07	1.1E-07	9.1E-08
S	0.	6.2E-06	1.0E-06	4.2E-07	2.5E-07	1.8E-07	1.4E-07	1.0E-07	8.0E-08	6.6E-08	5.5E-08
SSW	0.	5.7E-06	9.0E-07	4.0E-07	2.3E-07	1.6E-07	1.1E-07	8.9E-08	7.0E-08	5.7E-08	4.8E-08
SW	0.	6.1E-06	9.4E-07	3.9E-07	2.2E-07	1.6E-07	1.1E-07	8.6E-08	7.0E-08	6.0E-08	5.1E-08
WSW	0.	7.3E-06	1.1E-06	4.6E-07	2.7E-07	1.7E-07	1.3E-07	1.0E-07	8.0E-08	6.5E-08	5.4E-08
W	0.	7.6E-06	1.2E-06	5.2E-07	2.9E-07	2.0E-07	1.3E-07	1.0E-07	8.4E-08	7.2E-08	6.1E-08
WNW	0.	1.4E-05	2.1E-06	9.1E-07	5.2E-07	3.4E-07	2.6E-07	2.0E-07	1.5E-07	1.2E-07	1.0E-07
NW	0.	1.6E-05	2.4E-06	1.0E-06	5.9E-07	3.9E-07	2.8E-07	2.1E-07	1.7E-07	1.4E-07	1.2E-07
NNW	0.	1.5E-05	2.2E-06	9.6E-07	5.5E-07	3.6E-07	2.6E-07	2.0E-07	1.6E-07	1.3E-07	1.2E-07
N	0.	9.1E-06	1.4E-06	6.3E-07	3.6E-07	2.4E-07	1.8E-07	1.4E-07	1.2E-07	9.4E-08	7.9E-08

Number of Valid Observations = 17135   Number of Calms Lower Level = 95
Number of Invalid Observations = 385   Number of Calms Upper Level = 0
Note 1 - Any interpolations between stated mileages will be done by log-log

ST. LUCIE PLANT
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OFFSITE DOSE CALCULATION MANUAL (ODCM)

TABLE M-6

HISTORICAL LONG TERM DEPLETED - $(X/Q)_D$ (Frequency corrected)

Terrain / Recirculation Adjusted   Program ANNXOQ9 Version - 11/18/76

Florida Power & Light Company
St. Lucie Unit 1
Hutchinson Island, Florida
Dames and Moore Job No: 4598 - 112

Average Annual Relative Concentration Depleted (sec/cubic meter)
Period of Record: 9/1/76 to 8/31/78
Base Distance in Miles/Kilometers

AFFECTED SECTOR	DESIGN DISTANCE MILES	.25 .40	.75 1.21	1.25 2.01	1.75 2.82	2.25 3.62	2.75 4.42	3.25 5.23	3.75 6.03	4.25 6.84	4.75 7.64
NNE	0.	1.1E-05	1.6E-06	6.6E-07	3.8E-07	2.4E-07	1.7E-07	1.3E-07	1.1E-07	9.2E-08	7.6E-08
NE	0.	1.2E-05	1.7E-06	7.6E-07	4.3E-07	2.8E-07	1.9E-07	1.4E-07	1.1E-07	8.6E-08	7.4E-08
ENE	0.	8.9E-06	1.2E-06	5.3E-07	3.0E-07	2.0E-07	1.4E-07	1.0E-07	8.4E-08	6.6E-08	5.6E-08
E	0.	9.1E-06	1.3E-06	5.6E-07	3.1E-07	2.1E-07	1.5E-07	1.1E-07	9.1E-08	7.5E-08	6.3E-08
ESE	0.	1.2E-05	1.6E-06	6.9E-07	3.9E-07	2.6E-07	1.9E-07	1.4E-07	1.1E-07	8.5E-08	6.7E-08
SE	0.	1.3E-05	2.0E-06	8.2E-07	4.7E-07	3.3E-07	2.3E-07	1.8E-07	1.3E-07	1.1E-07	9.0E-08
SSE	0.	1.1E-05	1.6E-06	6.3E-07	3.5E-07	2.4E-07	1.8E-07	1.4E-07	1.0E-07	8.2E-08	6.8E-08
S	0.	5.9E-06	9.1E-07	3.6E-07	2.1E-07	1.4E-07	1.1E-07	7.7E-08	6.2E-08	5.0E-08	4.1E-08
SSW	0.	5.4E-06	8.0E-07	3.4E-07	1.9E-07	1.3E-07	8.9E-08	6.9E-08	5.5E-08	4.3E-08	3.6E-08
SW	0.	5.7E-06	8.4E-07	3.4E-07	1.8E-07	1.2E-07	9.2E-08	6.7E-08	5.3E-08	4.6E-08	3.8E-08
WSW	0.	7.0E-06	9.6E-07	4.0E-07	2.2E-07	1.4E-07	1.0E-07	8.0E-08	6.1E-08	5.0E-08	4.0E-08
W	0.	7.3E-06	1.1E-06	4.4E-07	2.4E-07	1.6E-07	1.1E-07	8.2E-08	6.4E-08	5.5E-08	4.4E-08
WNW	0.	1.3E-05	1.9E-06	7.9E-07	4.4E-07	2.9E-07	2.0E-07	1.6E-07	1.2E-07	9.3E-08	7.8E-08
NW	0.	1.5E-05	2.1E-06	8.9E-07	4.9E-07	3.1E-07	2.3E-07	1.7E-07	1.3E-07	1.0E-07	8.5E-08
NNW	0.	1.4E-05	2.1E-06	8.3E-07	4.5E-07	2.9E-07	2.0E-07	1.6E-07	1.2E-07	1.0E-07	8.6E-08
N	0.	8.7E-06	1.3E-06	5.4E-07	3.0E-07	2.0E-07	1.4E-07	1.1E-07	8.9E-08	7.0E-08	5.8E-08

Number of Valid Observations = 17135   Number of Calms Lower Level = 95
Number of Invalid Observations = 385   Number of Calms Upper Level = 0
Note 1 - Any interpolations between stated mileages will be done by log-log

ST. LUCIE PLANT
CHEMISTRY OPERATING PROCEDURE NO. C-200, REVISION 14
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TABLE M-7
HISTORICAL LONG TERM - (D/Q) (Frequency corrected)
TERRAIN / RECIRCULATION ADJUSTED PROGRAM ANNXOQ9 VERSION - 11/18/76

Florida Power & Light Company
St. Lucie Unit 1
Hutchinson Island, Florida
Dames and Moore Job No: 4598 - 112

Average Annual Relative Deposition Rate (square meter - 1)
Period of Record: 9/1/76 to 8/31/78
Base Distance in Miles/Kilometers

AFFECTED SECTOR	DESIGN DISTANCE MILES	.25 .40	.75 1.21	1.25 2.01	1.75 2.82	2.25 3.62	2.75 4.42	3.25 5.23	3.75 6.03	4.25 6.84	4.75 7.64
NNE	0.	6.5E-08	9.3E-09	3.7E-09	2.1E-09	1.3E-09	9.0E-10	6.8E-10	5.5E-10	4.3E-10	3.5E-10
NE	0.	6.0E-08	8.9E-09	3.5E-09	1.9E-09	1.2E-09	8.1E-10	5.6E-10	4.3E-10	3.3E-10	2.8E-10
ENE	0.	3.2E-08	4.8E-09	1.9E-09	1.0E-09	6.6E-10	4.6E-10	3.2E-10	2.4E-10	1.9E-10	1.5E-10
E	0.	3.0E-08	4.6E-09	1.8E-09	9.5E-10	6.0E-10	4.2E-10	3.1E-10	2.5E-10	2.0E-10	1.6E-10
ESE	0.	3.7E-08	5.8E-09	2.3E-09	1.2E-09	8.0E-10	5.4E-10	3.9E-10	3.0E-10	2.2E-10	1.7E-10
SE	0.	6.4E-08	1.0E-08	4.0E-09	2.1E-09	1.4E-09	9.7E-10	7.2E-10	5.6E-10	4.3E-10	3.5E-10
SSE	0.	6.2E-08	9.5E-09	3.6E-09	2.0E-09	1.2E-09	8.7E-10	6.4E-10	4.9E-10	3.9E-10	3.1E-10
S	0.	4.2E-08	7.0E-09	2.6E-09	1.4E-09	9.5E-10	6.9E-10	4.9E-10	3.8E-10	3.0E-10	2.5E-10
SSW	0.	3.4E-08	5.4E-09	2.2E-09	1.1E-09	7.5E-10	5.0E-10	3.7E-10	2.9E-10	2.3E-10	1.8E-10
SW	0.	4.5E-08	7.0E-09	2.6E-09	1.5E-09	9.0E-10	6.6E-10	4.6E-10	3.6E-10	3.0E-10	2.5E-10
WSW	0.	5.3E-08	7.7E-09	3.0E-09	1.6E-09	1.0E-09	7.3E-10	5.5E-10	4.1E-10	3.3E-10	2.6E-10
W	0.	5.0E-08	7.5E-09	3.0E-09	1.6E-09	9.8E-10	6.7E-10	5.0E-10	3.8E-10	3.2E-10	2.6E-10
WNW	0.	8.8E-08	1.3E-08	4.9E-09	2.6E-09	1.7E-09	1.1E-09	8.7E-10	6.6E-10	5.1E-10	4.2E-10
NW	0.	8.2E-08	1.2E-08	4.7E-09	2.5E-09	1.6E-09	1.1E-09	7.9E-10	5.8E-10	4.7E-10	3.8E-10
NNW	0.	8.2E-08	1.2E-08	4.6E-09	2.4E-09	1.5E-09	1.1E-09	8.1E-10	5.9E-10	4.8E-10	4.0E-10
N	0.	5.1E-08	7.3E-09	2.9E-09	1.5E-09	9.8E-10	7.1E-10	5.4E-10	4.2E-10	3.2E-10	2.7E-10

Number of Valid Observations = 17135 Number of Calms Lower Level = 95
Number of Invalid Observations = 385 Number of Calms Upper Level = 0
Note 1 - Any interpolations between stated mileages will be done by log-log

ST. LUCIE PLANT
CHEMISTRY OPERATING PROCEDURE NO. C-200, REVISION 14
OFFSITE DOSE CALCULATION MANUAL (ODCM)

TABLE M-8

Joint Wind Frequency Distribution   Data Period: September 1, 1976 - August 31, 1978

All Winds   St. Lucie Unit 2
Data Source: On-Site   Hutchinson Island, Florida
Wind Sensor Height 10.00 Meters   Florida Power & Light Co.
Table Generated: 12/05/78. 07.42.18.   Dames and Moore Job No: 4598 - 112 - 27

Wind Speed Categories (Meters per Second)

WIND SECTOR	0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL[1]	MEAN SPEED
NNE	71 .43	206 1.25	318 1.92	71 .43	3 .02	0 0.00	669 4.05	3.32
NE	62 .38	292 1.77	385 2.33	128 .77	0 0.00	0 0.00	867 5.25	3.43
ENE	60 .36	334 2.02	505 3.06	158 .96	0 0.00	0 0.00	1057 6.40	3.51
E	69 .42	355 2.15	510 3.09	76 .46	0 0.00	0 0.00	1010 6.11	3.25
ESE	115 .70	684 4.14	744 4.50	72 .44	1 .01	0 0.00	1616 9.78	3.04
SE	183 1.11	660 3.99	749 4.53	28 .17	0 0.00	0 0.00	1620 9.81	2.88
SSE	129 .78	579 3.50	656 3.97	93 .56	1 .01	0 0.00	1458 8.82	3.10
S	72 .44	310 1.88	407 2.46	99 .60	8 .05	1 .01	897 5.43	3.36
SSW	84 .51	372 2.25	446 2.70	105 .64	33 .20	4 .02	1044 6.32	3.48
SW	129 .78	440 2.66	336 2.03	106 .64	14 .08	0 0.00	1025 6.20	3.10
WSW	155 .94	320 1.94	186 1.13	29 .18	5 .03	0 0.00	695 4.21	2.59
W	174 1.05	267 1.62	119 .72	37 .22	2 .01	0 0.00	599 3.63	2.43
WNW	203 1.23	304 1.84	172 1.04	17 .10	0 0.00	0 0.00	696 4.21	2.34
NW	143 .87	518 3.14	424 2.57	50 .30	0 0.00	0 0.00	1135 6.87	2.85
NNW	85 .51	379 2.29	535 3.24	70 .42	1 .01	0 0.00	1070 6.46	3.22
N	91 .55	194 1.17	531 3.21	148 .90	5 .03	0 0.00	969 5.86	[illegible]
CALM	95 .57						95 .57	
TOTAL	1920 11.62	6214 37.61	7023 42.51	1287 7.79	73 .44	5 .03	16522 100.00	3.10

NUMBER OF VALID OBSERVATIONS 16522 94.30 PCT.
NUMBER OF INVALID OBSERVATIONS 988 5.70 PCT.
TOTAL NUMBER OF OBSERVATIONS 17520 100.00 PCT.

Key: XXX Number of Occurrences; XXX Percent Occurrences

[1] - Totals below are given in hours & percent for wind frequency by sectors

APPENDIX B
LIMITED ANALYSIS DOSE ASSESSMENT FOR LIQUID RADIOACTIVE EFFLUENTS

The radioactive liquid effluents for the years 1978, 1979, and 1980 were evaluated to determine the dose contribution of the radionuclide distribution. This analysis was performed to evaluate the use of a limited dose analysis for determining environmental doses. Limiting the dose calculation to a few selected radionuclides that contribute the majority of the dose provides a simplified method of determining compliance with the dose limits of Control 3.11.1.2.

Tables B-1 and B-2 present the results of this evaluation. Table B-1 presents the fraction of the adult whole body dose contributed by the major radionuclides. Table B-2 presents the same data for the adult GI-LLI dose. The adult whole body and adult GI-LLI were determined to be the limiting doses based on an evaluation of all age groups (adult, teenager, child, and infant) and all organs (bone, liver, kidney, lung, and GI-LLI). As the data in the tables show, the radionuclides Fe-59, Co-58, Co-60, Zn-65, Cs-134, and Cs-137 dominate the whole body dose; the radionuclides, Fe-59, Co-58, Co-60, Zn-65, and Nb-95 dominate the GI-LLI dose. In all but one case (1979-fish, GI-LLI dose) these radionuclides contribute 90% or more of the total dose. If for 1979 the fish and shellfish pathways are combined as is done to determine the total dose, the contribution from these nuclides is 84%[A] of the total GI-LLI dose.

Therefore, the dose commitment due to radioactive material in liquid effluents can be reasonably estimated by limiting the dose calculation to the radionuclides, Fe-59, Co-58, Co-60, Zn-65, Nb-95, Cs-134, and Cs-137, which cumulatively contribute the majority of the total dose calculated by using all radionuclides detected. This limited analysis dose assessment method is a simplified calculation that provides a reasonable evaluation of doses due to liquid radioactive effluents and allows for an estimate of Fe-55 contribution to dose.

Tritium is not included in the limited analysis dose assessment for liquid releases because the potential dose resulting from normal reactor releases is negligible and is essentially independent of radwaste system operation. The amount of tritium releases annually is about 300 curies. At St. Lucie, 300 Ci/yr released to the Atlantic Ocean produces a calculated whole body dose of 5×10^{-7} mrem/yr via the fish and shellfish pathways. This amounts to less than 0.001% of the design objective dose of 3 mrem/yr. Furthermore, the release of tritium is a function of operating time and power level and is essentially unrelated to radwaste system operation.

[A] The dose due to Iron -55 made it necessary to change the conservatism factor from 0.8 to 0.6, which was done on Revision 7 to the ODCM, based on early 1986 data.

TABLE B-1
ADULT WHOLE BODY DOSE CONTRIBUTIONS FRACTION OF TOTAL

RADIONUCLIDE	1978		1979		1980	
	FISH	SHELLFISH	FISH	SHELLFISH	FISH	SHELLFISH
Co-58	0.08	0.27	0.06	0.28	0.02	0.05
Co-60	0.05	0.19	0.03	0.15	0.20	0.44
Fe-59	0.10	0.25	0.04	0.13	0.15	0.22
Zn-65	0.01	0.10	0.02	0.19	0.04	0.20
Cs-134	0.31	0.07	0.46	0.14	0.27	0.04
Cs-137	0.42	0.10	0.38	0.11	0.30	0.04
TOTAL	0.97	0.98	0.99	1.00	0.98	0.99

TABLE B-2
ADULT GI-LLI DOSE CONTRIBUTION FRACTION OF TOTAL

RADIONUCLIDE	1978		1979		1980	
	FISH	SHELLFISH	FISH	SHELLFISH	FISH	SHELLFISH
Co-58	0.03	0.36	0.25	0.44	0.01	0.07
Co-60	0.02	0.23	0.12	0.22	0.05	0.57
Fe-59	0.03	0.31	0.16	0.19	0.04	0.29
Zn-65	0.01	0.02	0.01	0.05	0.01	0.04
Nb-95	0.89	0.01	0.21	0.01	0.88	0.01
TOTAL	0.98	0.92	0.75	0.90	0.97	0.97

APPENDIX C
TECHNICAL BASES FOR EFFECTIVE DOSE FACTORS

Overview

The evaluation of doses due to releases of radioactive material to the atmosphere can be simplified by the use of effective dose transfer factors instead of using dose factors which are radionuclide specific. These effective factors, which are based on the typical radionuclide distribution in the releases, can be applied to the total radioactivity released to approximate the dose in the environment, i.e., instead of having to sum the isotopic distribution multiplied by the isotope specific dose factor only a single multiplication (K_{eff}, M_{eff}, or N_{eff}) times the total quantity of radioactive material released would be needed. This approach provides a reasonable estimate of the actual dose while eliminating the need for a detailed calculational technique.

Determination of Effective Dose Factors

The effective dose transfer factors are based on past operating data. The radioactive effluent distribution for the past years can be used to derive single effective factors by the following equations:

$$K_{eff} = \sum_i K_i \cdot f_i \qquad \text{(C-1)}$$

Where:

K_{eff} = the effective whole body dose factor due to gamma emissions from all noble gases released

K_i = the whole body dose factor due to gamma emissions from each noble gas radionuclide i released

f_i = the fractional abundance of noble gas radionuclide i is of the total noble gas radionuclides

APPENDIX C
TECHNICAL BASES FOR EFFECTIVE DOSE FACTORS
(continued)

$$(L + 1.1\ M)_{eff} = \sum_i (L_i + 1.1\ M_i) \bullet f_i \qquad \text{(C-2)}$$

Where:

$(L+1.1\ M)_{eff}$ = the effective skin dose factor due to beta and gamma emissions from all noble gases released

$(L_i + 1.1\ M_i)$ = the skin dose factor due to beta and gamma emissions from each noble gas radionuclide i released

$$M_{eff} = \sum_i M_i \bullet f_i \qquad \text{(C-3)}$$

Where:

M_{eff} = the effective air dose factor due to gamma emissions from all noble gases released

M_i = the air dose factor due to gamma emissions from each noble gas radionuclide i released

$$N_{eff} = \sum_i N_i \bullet f_i \qquad \text{(C-4)}$$

Where:

N_{eff} = the effective air dose factor due to beta emissions from all noble gases released

N_i = the air dose factor due to beta emissions from each noble gas radionuclide i

APPENDIX C
TECHNICAL BASES FOR EFFECTIVE DOSE FACTORS
(continued)

To determine the appropriate effective factors to be used and to evaluate the degree of variability, the atmospheric radioactive effluents for the past 3 years have been evaluated. Tables C-1 and C-2 present the results of this evaluation.

As can be seen from Tables C-1 and C-2, the effective dose transfer factors varies little from year to year. The maximum observed variability from the average value is 18%. This variability is minor considering other areas of uncertainty and conservatism inherent in the environmental dose calculation models.

To provide an additional degree of conservatism, a factor of 0.8 is introduced into the dose calculation process when the effective dose transfer factor is used. This added conservatism provides additional assurance that the evaluation of doses by the use of a single effective factor will not significantly underestimate any actual doses in the environment.

Reevaluation

The doses due to the gaseous effluents are evaluated by the more detailed calculation methods (i.e., use of nuclide specific dose factors) on a yearly basis. At this time a comparison can be made between the simplified method and the detailed method to assure the overall reasonableness of this limited analysis approach. If this comparison indicates that the radionuclide distribution has changed significantly causing the simplified method to underestimate the doses by more than 20%, the value of the effective factors will need to be reexamined to assure the overall acceptability of this approach. However, this reexamination will only be needed if the doses as calculated by the detailed analysis exceed 50% of the design bases doses (i.e., greater than 5 mrads gamma air dose or 10 mrads beta air dose).

In any case, the appropriateness of the A_{eff} value will be periodically evaluated to assume the applicability of a single effective dose factor for evaluating environmental doses.

TABLE C-1
EFFECTIVE DOSE FACTORS

NOBLE GASES-WHOLE BODY AND SKIN DOSES

YEAR	WHOLE BODY EFFECTIVE DOSE FACTOR K_{eff} $\frac{mrem\text{-}m^3}{\mu Ci\text{-}yr}$	SKIN EFFECTIVE DOSE FACTOR $(L+1.1M)_{eff}$ $\frac{mrem\text{-}m^3}{\mu Ci\text{-}yr}$
1978	7.3×10^2	1.4×10^3
1979	7.4×10^2	1.4×10^3
1980	5.6×10^2	1.2×10^3
AVERAGE	6.8×10^2	1.3×10^3

TABLE C-2
EFFECTIVE DOSE FACTORS

NOBLE GASES - AIR DOSES

YEAR	GAMMA AIR EFFECTIVE DOSE FACTOR M_{eff} $\frac{mrad\text{-}m^3}{\mu Ci\text{-}yr}$	BETA AIR EFFECTIVE DOSE FACTOR N_{eff} $\frac{mrad\text{-}m^3}{\mu Ci\text{-}yr}$
1978	8.0×10^2	1.2×10^3
1979	8.0×10^2	1.2×10^3
1980	6.2×10^2	1.2×10^3
AVERAGE	7.4×10^2	1.2×10^3

APPENDIX D
TECHNICAL BASES FOR ELIMINATING CURIE INVENTORY LIMIT FOR GASEOUS WASTE STORAGE TANKS

The NRC Standard Technical Specifications include a limit for the amount of radioactivity that can be stored in a single waste gas storage tank. This curie inventory limit is established to assure that in the event of a tank failure releasing the radioactivity to the environment the resulting whole body dose at the site boundary would not exceed 0.5 rem.

For St. Lucie, the inventory limit in the waste gas storage tank has been determined to be approximately 285,000 curies (Xe-133, equivalent). An allowable primary coolant radioactivity concentration is established by the Appendix A Technical Specifications which limits the primary coolant radioactivity concentrations to $100/\bar{E}$ with $\bar{E}$ being the average energy of the radioactivity in Mev. This equation yields an upper primary coolant gross activity limit of about 160 µCi/ml. By applying this activity concentration limit to the total liquid volume of the primary system, a total activity limit can be determined. For St. Lucie the primary system volume is about 70,000 gallons, which yields a limiting total inventory of approximately 43,000 Ci.

By assuming a typical radionuclide distribution an equivalent Xe-133 inventory can be determined. Table D-1 provides the typical radionuclide (noble gases) distribution and the Xe-133 equivalent concentration. The equivalent concentration is determined by multiplying the radionuclide concentration by the ratio of the nuclide whole body dose factor to the Xe-133 whole body dose factor. Summing all the individual radionuclide equivalent concentrations provides the overall reactor coolant Xe-133 equivalent concentration. The data show that the equivalent concentration is a factor of 2 larger than the gross concentration (i.e., 24 µCi/gm total versus 47 µCi/gm equivalent). The resulting Xe-133 equivalent curie inventory of the reactor coolant system is approximately 86,000 Ci.

Therefore, even if the total primary system at the maximum Tech. Spec. allowable concentration was degassed to a single waste gas decay tank, the tank curie inventory would be well below the 285,000 Ci limit. Based on this evaluation, the curie inventory limit on a single waste gas storage tank cannot exceed the Technical Specification requirement.

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TABLE D-1
REACTOR COOLANT - XE-133 EFFECTIVE CONCENTRATION

RADIONUCLIDE	REACTOR COOLANT* CONCENTRATION (µCi/gm)	REG. GUIDE 1.109 WHOLE BODY DF (mrem/yr)/(pCi/ml)	RATIO TB DF / Xe-133 DF	Xe-133 EFFECTIVE CONCENTRATION (µCi/gm)
Kr-85m	0.19	1.2×10^{-3}	4.1	0.78
Kr-85	0.83	1.6×10^{-5}	0.06	0.05
Kr-87	0.16	5.9×10^{-3}	20.	3.2
Kr-88	0.31	1.5×10^{-2}	52.	16.
Xe-131m	8.8	9.2×10^{-5}	0.32	2.8
Xe-133m	0.20	2.5×10^{-4}	0.86	0.17
Xe-133	12.	2.9×10^{-4}	1.0	12.
Xe-135m	0.11	3.1×10^{-3}	11.	1.2
Xe-135	1.2	1.8×10^{-3}	6.2	7.4
Xe-137	0.02	1.4×10^{-3}	4.8	0.1
Xe-138	0.12	8.8×10^{-3}	30.	3.6
TOTALS	24.			47.

*Data adapted from the NRC GALE Code

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APPENDIX E
RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE
ST. LUCIE PLANT
Key to Sample Locations

PATHWAY	LOCATION	DESCRIPTION	SAMPLES COLLECTED	SAMPLE COLLECTION FREQUENCY	APPROXIMATE DISTANCE (miles)	DIRECTION SECTOR
Direct Radiation	N-1	North of Blind Creek	TLD	Quarterly	1	N
Direct Radiation	NNW-5	South of Pete Stone Creek	TLD	Quarterly	5	NNW
Direct Radiation	NNW-10	C. G. Station	TLD	Quarterly	9	NNW
Direct Radiation	NW-5	Indian River Drive at Rio Vista Drive	TLD	Quarterly	6	NW
Direct Radiation	NW-10	Intersection of SR 68 and SR 607	TLD	Quarterly	10	NW
Direct Radiation	WNW-2	Cemetery South of 7107 Indian River Drive	TLD	Quarterly	3	WNW
Direct Radiation	WNW-5	US-1 at SR 712	TLD	Quarterly	5	WNW
Direct Radiation	WNW-10	SR 70, West of Turnpike	TLD	Quarterly	10	WNW
Direct Radiation	W-2	7609 Indian River Drive	TLD	Quarterly	2	W
Direct Radiation	W-5	Oleander and Sager Streets	TLD	Quarterly	5	W
Direct Radiation	W-10	I-95 and SR 709	TLD	Quarterly	9	W
Direct Radiation	WSW-2	8503 Indian River Drive	TLD	Quarterly	2	WSW
Direct Radiation	WSW-5	Prima Vista Blvd. at Yacht Club	TLD	Quarterly	5	WSW
Direct Radiation	WSW-10	Del Rio and Davis Streets	TLD	Quarterly	10	WSW
Direct Radiation	SW-2	9207 Indian River Drive	TLD	Quarterly	2	SW
Direct Radiation	SW-5	US 1 and Village Green Drive	TLD	Quarterly	5	SW
Direct Radiation	SW-10	Port St. Lucie Blvd. and Cairo Road	TLD	Quarterly	10	SW
Direct Radiation	SSW-2	10307 Indian River Drive	TLD	Quarterly	3	SSW

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APPENDIX E
RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE
(continued)
ST. LUCIE PLANT
Key to Sample Locations

PATHWAY	LOCATION	DESCRIPTION	SAMPLES COLLECTED	SAMPLE COLLECTION FREQUENCY	APPROXIMATE DISTANCE (miles)	DIRECTION SECTOR
Direct Radiation	SSW-5	Port St. Lucie Blvd. and US 1	TLD	Quarterly	6	SSW
Direct Radiation	SSW-10	Pine Valley and Westmoreland Roads	TLD	Quarterly	8	SSW
Direct Radiation	S-5	13179 Indian River Drive	TLD	Quarterly	5	S
Direct Radiation	S-10	US 1 and SR 714	TLD	Quarterly	10	S
Direct Radiation	S/SSE-10	Indian River Drive and Quail Run Lane	TLD	Quarterly	10	SSE
Direct Radiation	SSE-5	Entrance of Nettles Island	TLD	Quarterly	5	SSE
Direct Radiation	SSE-10	Elliot Museum	TLD	Quarterly	10	SSE
Direct Radiation	SE-1	South of Cooling Canal	TLD	Quarterly	1	SE
Direct Radiation	*H-32	U. of Florida - 1FAS Entomology Lab Vero Beach	TLD	Quarterly	19	NNW
Airborne	H08	FPL Substation - Weatherby Road	Radioiodine & Particulates	Weekly	6	WNW
Airborne	*H12	FPL Substation - SR 76, Stuart	Radioiodine & Particulates	Weekly	12	S
Airborne	H14	Onsite - near south property line	Radioiodine & Particulates	Weekly	1	SE
Airborne	H30	Power Line - 7609 Indian River Drive	Radioiodine & Particulates	Weekly	2	W

*Denotes Control Sample

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APPENDIX E
RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE
(continued)
ST. LUCIE PLANT
Key to Sample Locations

PATHWAY	LOCATION	DESCRIPTION	SAMPLES COLLECTED	SAMPLE COLLECTION FREQUENCY	APPROXIMATE DISTANCE (miles)	DIRECTION SECTOR
Airborne	H34	Onsite - At Meteorological Tower	Radioiodine & Particulates	Weekly	0.5	N
Waterborne	H15	Atlantic Ocean vicinity of public beaches east side of Route A1A	Surface Water (ocean) Sediment from shoreline	Weekly Semi-Annually	< 1	ENE/E/ESE
Waterborne	*H59	Near south end of Hutchinson Island	Surface Water (ocean) Sediment from shoreline	Monthly Semi-Annually	10-20	S/SSE
Food Products	H15	Ocean side vicinity of St. Lucie Plant (NOTE 1)	Crustacea Fish	Semi-Annually Semi-Annually	<1	ENE/E/ESE
Food Products	H51	Offsite near north property line	Broad Leaf vegetation (mangrove)	Monthly (when available)	1	N/NNW

*Denotes control sample

APPENDIX E
RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE
(continued)
ST. LUCIE PLANT
Key to Sample Locations

PATHWAY	LOCATION	DESCRIPTION	SAMPLES COLLECTED	SAMPLE COLLECTION FREQUENCY	APPROXIMATE DISTANCE (miles)	DIRECTION SECTOR
Food Products	H52	Offsite near south property line	Broad leaf vegetation (mangrove)	Monthly (when available)	1	S/SSE
Food Products	*H59	Near south end of Hutchinson Island	Crustacea Fish Broad leaf vegetation (mangrove)	Semi-Annually Semi-Annually Monthly	10-20	S/SSE

*Denotes control sample

It is the policy of Florida Power & Light Company (FPL) that the St. Lucie 1 & 2 Radiological Environmental Monitoring Programs are conducted by the State of Florida Department of Health and Rehabilitative Services (DHRS), pursuant to an Agreement between FPL and DHRS and; that coordination of the Radiological Environmental Monitoring Programs with DHRS and compliance with the Radiological Environmental Monitoring Program Controls are the responsibility of the Nuclear Energy Services Department.

NOTE 1

These samples may be collected from or supplemented by samples collected from the plant intake canal if the required analyses are unable to be performed due to unavailability or inadequate quantity of sample from the ocean side location.

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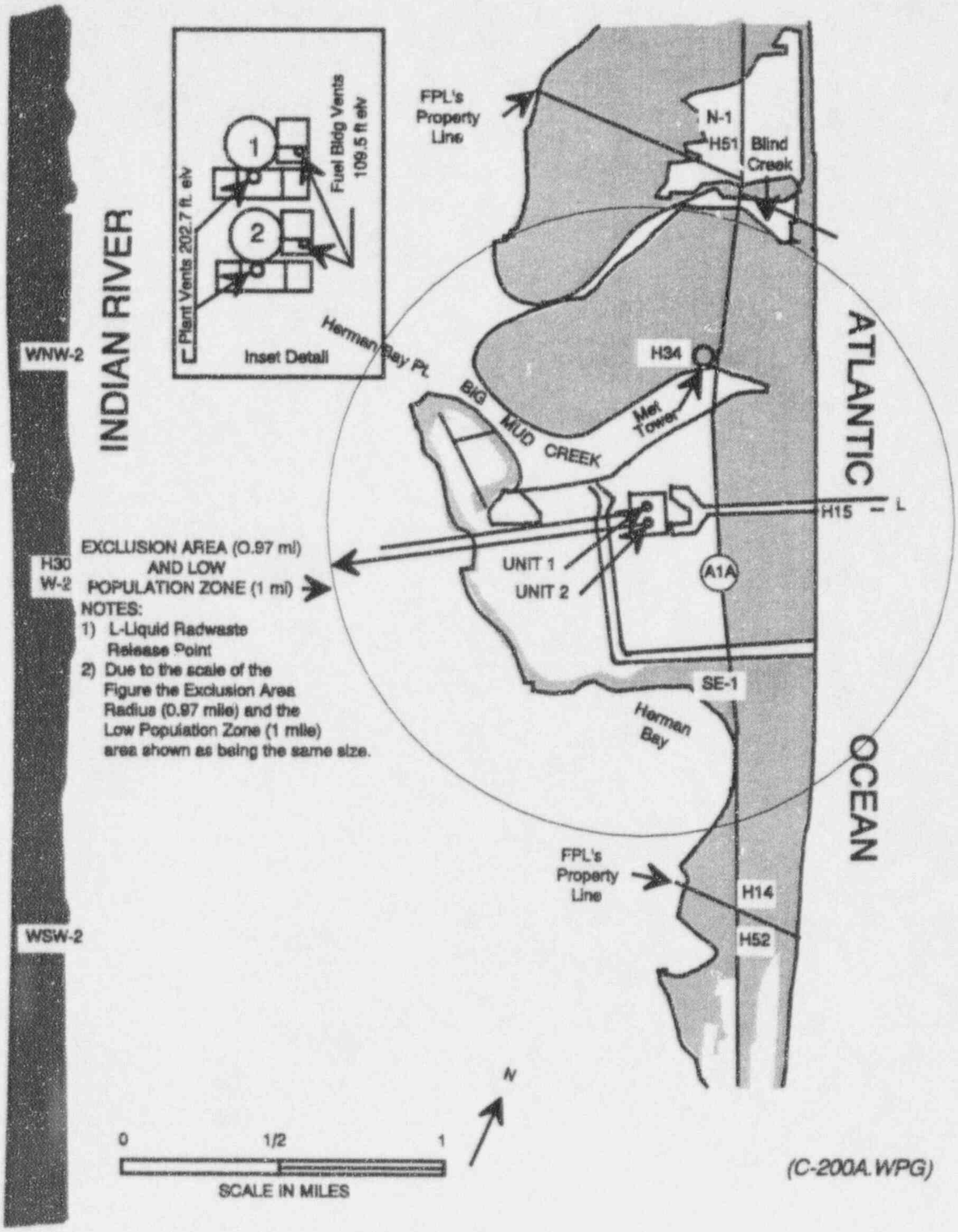
FIGURE 1-1
SITE AREA MAP & ENVIRONMENTAL SAMPLE LOCATIONS



(C-200A.WPG)

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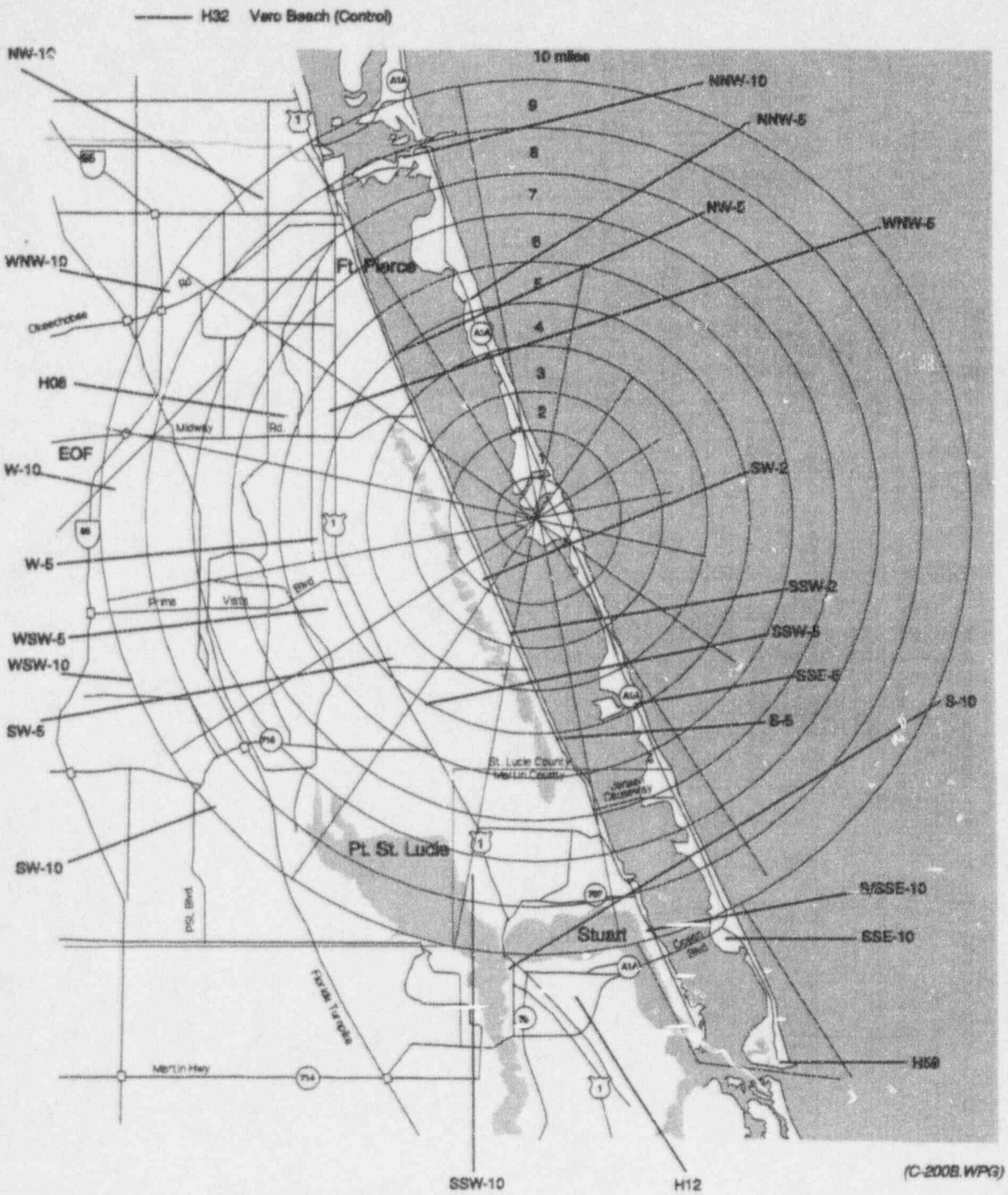
FIGURE 1-2
ENVIRONMENTAL SAMPLE LOCATIONS (10 MILES)



(C-200B.WPG)

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APPENDIX F
METEOROLOGICAL DISPERSION FORMULAS*

For X/Q:

EQ (1)

$$X/Q = \frac{2.032}{\sqrt{3}\ \sigma_z\ (\bar{u}) D} \qquad \text{EQ (2)}$$

Where:

C = .5

V = 207.5 ft. (63.2 meters)

$(\bar{u})$ = a name for one term

X/Q was calculated using each of the above EQ's for each hour. The highest X/Q from EQ (1) or EQ (2) was selected. The total integrated relative concentration at each sector and distance was then divided by the total number of hours in the data base.

* Terrain correction factors given by Table M-4 were also applied to Dispersion Formulas

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APPENDIX F
METEOROLOGICAL DISPERSION FORMULAS*
(continued)

For Depleted X/Q:

$(X/Q)_D$ = (X/Q) X (Depletion factor of Figure 2 of R.G. 1.111-R1)

For Deposition (D/Q):

D/Q = RDep/(2 sin [11.25] X) X (Freq. distribution)

Where:

D/Q	=	Ground deposition rate
X	=	Calculation distance
RDep	=	Relative ground deposition rate from Figure 6 of R.G. 1.111, R1