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OCONEE NUCLEAR STATION
PROPOSED TECHNICAL SPECIFICATION REVISION

RADIOLOGICAL EFFLUENT CONTROL

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1.8 RADIOLOGICAL EFFLUENT CONTROL

1.8.1 Source Check

A Source Check is the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.8.2 Offsite Dose Calculation Manual (ODCM)

The Offsite Dose Calculation Manual is a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints and in the conduct of environmental radiological monitoring.

1.8.3 Process Control Program (PCP)

The Process Control Program is a procedure that shall contain the sampling, analysis, and formulation determination by which solidification of radioactive liquid waste is assured.

1.8.4 Solidification

Solidification shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the waste type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the Process Control Program (PCP).

1.8.5 Gaseous Radwaste Treatment System

A Gaseous Radwaste Treatment System is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

1.8.6 Ventilation Exhaust Treatment System

A Ventilation Exhaust Treatment System is 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. Engineered Safety Features (ESF) atmospheric cleanup systems are not considered to be Ventilation Exhaust Treatment System components.

1.8.7 Purge-Purging

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

1.8.8 Venting

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during Venting. Vent, used in system names, does not imply a venting process.

1.8.9 Member(s) Of The Public

Members(s) Of The Public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its 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.

1.8.10 Unrestricted Area

An Unrestricted Area shall be any area at or beyond the site boundary to which access 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 industrial, commerical institutional and/or recreational purposes.

3.5.5 Radioactive Effluent Monitoring Instrumentation

Applicability

Applies to radioactive liquid effluent, gaseous effluent, and gaseous process monitoring instrumentation.

Specifications

3.5.5.1 Liquid Effluents

a. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.5.5-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.1 are not exceeded.

b. If a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, without delay 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.

c. In the event that the number of operable radioactive liquid effluent monitoring instrumentation channels falls below the limit given under Table 3.5.5-1, Column A, action shall be as shown in Column B. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.5.2 Gaseous Process and Effluents

a. The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.5.5-2 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.10.1 are not exceeded.

b. If a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, without delay 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.

c. In the event that the number of radioactive gaseous process or effluent monitoring instrumentation channels falls below the limit given under Table 3.5.5-2, Column A, action shall be taken as shown in Column B. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.5.3 Setpoints

The setpoints shall be determined in accordance with the methodology described in the ODCM and shall be recorded. Setpoint correction may be permitted without declaring the channel inoperable.

3.5.5.4 The provisions of Technical Specification 3.0 do not apply.

Bases

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure 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.

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

Table 3.5.5-1
LIQUID EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS

	INSTRUMENT	A MINIMUM OPERABLE CHANNELS	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
1.	Monitors Providing Automatic Termination of Release			
	Liquid Radwaste Effluent Line Monitors 1 RIA-33	1	*	(a)
	Turbine Building Sump			
	1 RIA-54 (Units 1 & 2)	1	*	(b)
	3 RIA-54 (Unit 3)	1	*	(b)
2.	Monitors not Providing Automatic Termination of Release			
	Low Pressure Service Water			
	1 RIA-35	1	*	(d)
	2 RIA-35	1	*	(d)
	3 RIA-35	1	*	(d)
3.	Flow Rate Measuring Devices			
	Liquid Radwaste Effluent Line	1	*	(c)
	Keowee Hydroelectric Station Tailrace Discharge **	NA	NA	NA
4.	Continuous Composite Sampler			
	#3 Chemical Treatment Pond Composite Sampler and Sampler Flow Monitor (Turbine Building Sumps Effluent)	1	*	(d)

*At all times.
**Flow determined from number of hydro units operating; if hydro is not operating, leakage flow, which is measured periodically, is used.

Table 3.5.5-1 NOTES

(a) Effluent releases may continue provided that prior to initiating a release:

1. Two independent samples are analyzed in accordance with Specification 3.9 and;

2. Two independent data entry checks for release rate calculations and valve lineups of the effluent pathway are conducted.

Otherwise, suspend release of radioactive effluents by this pathway.

(b) Effluent releases may continue provided that prior to each discrete release of the sump, grab samples are collected and analyzed for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml.

(c) Effluent releases may continue provided flow rate is estimated at least once per four hours during actual releases.

(d) Effluent releases may continue provided that grab samples are collected and analyzed for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml every 12 hours.

Table 3.5.5-2
GASEOUS PROCESS AND EFFLUENT
MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
1. Waste Gas Holdup Tanks			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination Of Release (RIA-37, - 38)	1	**	(a)
b. Effluent Flow Rate Monitor (Waste Gas Discharge Flow)	1	**	(b)
2. Unit Vent Monitoring System			
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Containment Purge Release (RIA - 45)	1	*	(a)
b. Iodine Sampler	1	*	(d)
c. Particulate Sampler	1	*	(d)
d. Effluent Flow Rate Monitor (Unit Vent Flow)	1	*	(b)
e. Sampler Flow Rate Monitor	1	*	(e)
f. Effluent Flow Rate Monitor (Containment Purge)	1	**	(b)
3. Interim Radwaste Building Ventilation Monitoring System			
a. Noble Gas Activity Monitor (RIA - 53)	1	*	(c)
b. Iodine Sampler#	1	*	(d)
c. Particulate Sampler#	1	*	(d)

Table 3.5.5-2 (Cont'd)
GASEOUS PROCESS AND EFFLUENT
MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
d. Effluent Flow Rate Monitor (Interim Radwaste Exhaust)#	1	*	(b)
e. Sampler Flow Rate Monitor#	1	*	(e)
4. Hot Machine Shop Ventilation Monitoring System			
a. Iodine Sampler#	1	*	(d)
b. Particulate Sampler#	1	*	(d)
c. Effluent Flow Rate Monitor (Hot Machine Shop Exhaust)#	1	*	(b)
d. Sampler Flow Rate Monitor#	1	*	(e)

* At all times.
** During waste gas holdup tank releases and/or containment purge operation.
\# Effective upon installation of equipment.

Table 3.5.5-2 NOTES

(a) Effluent releases from waste gas tanks or containment purges may continue provided that prior to initiating a release:

1. Two independent samples are analyzed and;

2. Two independent data entry checks for release rate calculations and valve lineups of the effluent pathway are conducted and;

Effluent release from ventilation system or condenser air ejectors may continue provided that grab samples are taken once per 8 hours and these samples are analayzed for gross activity (beta and/or gamma) within 24 hours, or continuously monitor through the unit vent. Otherwise, suspend release of radioactive effluents via this pathway.

(b) Effluent releases may continue provided the flow rate is estimated at least once per 4 hours.

(c) Effluent releases may continue provided grab samples are taken once per 8 hours and these samples are analayzed for gross activity (beta and/or gamma) within 24 hours.

(d) Effluent releases may continue provided samples are continuously collected with auxiliary sampling equipment for periods not to exceed 7 days and analyzed within 48 hours of the end of sample collection.

(e) Alarms indicating low flow may be substituted for flow measuring devices.

3.9 RADIOACTIVE LIQUID EFFLUENTS

Applicability

Applies at all times to the controlled release of all liquid waste discharged from the site which may contain radioactive materials, except as noted. Appendix I dose limits for radioactive liquid effluent releases (T.S. 3.9.2) are applicable only during normal operating conditions which include expected operational occurrences, and are not applicable during unusual operating conditions that result in activation of the Oconee Emergency Plan.

Objective

To establish conditions for the controlled release of radioactive liquid effluents. To implement the requirements of 10 CFR 20, 10 CFR 50.36a, Appendix A to 10 CFR 50, Appendix I to 10 CFR 50, 40 CFR 141 and 40 CFR 190.

Specification

3.9.1 Concentration

a. The concentration of radioactive material released at anytime from the site boundary for liquid effluents to Unrestricted Areas (denoted in Figure 2.1-4(a) of the Oconee Nuclear Station Final Safety Analysis Report) shall be limited to the concentration 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. If the concentration of radioactive material released in liquid effluents to Unrestricted Areas exceeds the above Specified limits, without delay restore the concentration to within the above limits.

3.9.2 Dose

a. The dose or dose commitment to a Member Of The Public from radioactive materials in liquid effluents to Unrestricted Areas shall be limited to:

1) during any calendar quarter:

$\leq$ 4.5 mrem to the total body

$\leq$ 15 mrem to any organ and;

2) during any calendar year:

$\leq$ 9 mrem to the total body

$\leq$ 30 mrem to any organ.

b. If the calculated dose from the release of radioactive materials in liquid effluents exceeds any of the above limits, except during unusual operating conditions that result in activation of the Oconee Emergency Plan, and in lieu of any other report required by Section 6.6.2, a report shall be submitted within 30 days from the end of the quarter during which the release occurred, to the regional NRC Office of Inspection and Enforcement which includes the following:

1. Cause(s) for exceeding the limit(s)

2. A description of the program of corrective action initiated to: reduce the releases of radioactive materials in liquid effluents, and to keep these levels of radioactive materials in liquid effluents in compliance with the above limits, or as low as reasonably achievable.

3. Results of radiological analyses of the drinking water source and the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141.

3.9.3 Liquid Waste Treatment

a. The appropriate subsystems of the liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid waste prior to their discharge, if the projected dose due to liquid effluent releases to unrestricted areas, when averaged over 31 days would exceed 0.18 mrem to the total body or 0.6 mrem to any organ.

b. If radioactive liquid waste is discharged without treatment and in excess of the above limit, a report shall be submitted within 30 days to the regional NRC Office of Inspection and Enforcement which includes the following:

1. Cause of equipment or subsystem inoperability.

2. Corrective action to restore equipment and prevent recurrence.

3.9.4 Chemical Treatment Ponds (CTP 1 and 2)

a. The quantity of radioactive material in the Chemical Treatment Ponds (CTP) shall be limited so that, for all radionuclides identified, excluding noble gases and tritium, the sum of the ratios of activity (in curies) to the limits in 10 CFR 20, Appendix B, Table II, Column 2 shall not exceed 1.7×10^5.

$$\sum_j \frac{Aj}{Cj} < 1.7 \times 10^5$$

where A_j = pond inventory limit for single radionuclide 'j' (curies)

C_j = 10 CFR 20, Appendix B, Table II, Column 2, concentration for single radionuclide 'j' (curies)

b. After a primary to secondary leak is detected, the initial batch of used Powdex resin shall not be transferred to the CTP. No batch of used powdex resin shall be transferred to the CTP unless the sum of the ratios of the activity of the radionuclides identified in the preceeding batch from any powdex cell in the same unit is less than 0.1% of the limit identified in 3.9.4.a.

$$\sum_j \frac{Q_j}{A_j} < 1.0 \times 10^{-3}$$

where Q_j = radionuclide activity in the batch

A_j = pond inventory limit for radionuclide 'j'

c. The radionuclide inventory per batch of used powdex resin transferred, averaged over the transfers of the previous 13 weeks, shall not exceed 0.01% of the pond radionuclide inventory limit. If this average exceeds 0.01% of the pond radionuclide inventory limit, then a report will be submitted within 30 days to the Regional NRC Office of Inspection and Enforcement describing the reason or reasons for exceeding the objective and plans for future operation. Decay of radionuclides may be taken into account in determining inventory levels.

$$\frac{Qj_1 + Qj_2 + \ldots Qj_{(n-1)} + Qjn}{n} \leq .01\% \times Aj$$

where Q_j = activity or radionuclide 'j' in the batch

n = number of batches transferred to the chemical treatment ponds during the previous 13-week period.

3.9.5 Liquid Holdup Tanks

a. The quantity of radioactive material contained in each outside temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

b. The quantity of radioactive material contained in each of the outside temporary tanks shall be determined to be within the above limit by analyzing a representative sample of the tanks contents at least once per 7 days when radioactive materials are being added to the tank.

c. If the quantity of radioactive material in any outside temporary tank exceeds the above limit, suspend all additions to radioactive material to the tank without delay.

3.9.6 The provisions of Technical Specification 3.0 do not apply.

Bases

The concentration specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. The concentration limit for 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.

The dose specification is provided to assure that the release of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies which 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 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I. that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated.

Section IV of Appendix I of 10 CFR 50 states that the licensee is permitted the flexibility of operation during unusual operating conditions, to assure the public is provided with a dependable source of power when compatible with considerations of health and safety of the public. Section I of Appendix I of 10 CFR 50 states that this appendix provides specific numerical guides for design objectives and limiting conditions for operation, to assist holders of licenses for light-water-cooled nuclear power reactors in meeting the requirements to keep releases of radioactive material to unrestricted areas as low as practical, and reasonably achievable, <u>during normal reactor operations, including expected operational occurrences.</u> Using the flexibility granted during unusual operating conditions, and the stated applicability of the design objectives for the Oconee Nuclear Station, Appendix I dose limits for radioactive liquid effluent releases (T.S. 3.9.2), are concluded to be not applicable during unusual operating conditions that result in the activation of the Oconee Emergency Plan.

For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix A to 10 CFR Part 50.

The inventory limits of the chemical treatment ponds are based on limiting the consequences of an uncontrolled release of the pond inventory. The short term rate limit (2 mrem/hr) of 10CFR20.105 is applied to 10CFR20.106 in the following expression:

$$\frac{\dfrac{Aj}{1.3 \times 10^6 \text{ gal}} \times \dfrac{10^6 \ \mu\text{Ci}}{\text{curie}} \times \dfrac{\text{gal}}{3785 \text{ ml}}}{Cj} \leq \frac{2 \text{ mrem/hr}}{500 \text{ mrem/yr}} \times \frac{8760 \text{ hr}}{\text{yr}}$$

$$\frac{Aj}{Cj} \leq 1.7 \times 10^5$$

where Aj = pond inventory limit for radionuclide 'j' (curies)

Cj = 10CFR20 Appendix B, Table II, Column 2 concentration for radionuclide 'j'

1.3×10^6 gal = estimated volume of smaller chemical treatment pond

The batch limits provide assurance that activity input to the CTP will be minimized.

3.10 RADIOACTIVE GASEOUS EFFLUENTS

Applicability

Applies at all times to the controlled release of all gaseous waste discharged from the station which may contain radioactive materials.

Objective

To establish conditions for the controlled release of radioactive gaseous effluents. To implement the requirements of 10CFR20, 10CFR50.36a, Appendix A to 10CFR50, Appendix I to 10CFR50, and 40CFR190.

Specifications

3.10.1 Dose Rate

a. The instantaneous dose rate at the site (exclusion area) boundary for gaseous effluents (Figure 2.1-4(a) of the Oconee Nuclear Station Final Safety Analysis Report) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

   1. The dose rate limit for noble gases shall be:

      $\leq$ 500 mrem/yr to the total body

      $\leq$ 3000 mrem/yr to the skin and;

   2. The dose rate limit for all radioiodines and for all radioactive materials in particulate form and radionuclides other than noble gases with half-lives greater than 8 days shall be $\leq$ 1500 mrem/yr to any organ.

b. If the dose rate exceeds the above limits, without delay decrease the release rate to within the above limits.

3.10.2 Dose

a. The air dose due to noble gases released in gaseous effluent from the site shall be limited to the following:

   1. During any calendar quarter:

      $\leq$ 15 mrad for gamma radiation
      $\leq$ 30 mrad for beta radiation

   2. During any calendar year:

      $\leq$ 30 mrad for gamma radiation
      $\leq$ 60 mrad for beta radiation

b. The dose to a Member Of The Public from radioiodines, tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released from the site, shall be limited to the following:

1. During any calendar quarter:

   $\leq$ 22.5 mrem to any organ

2. During any calendar year:

   $\leq$ 45 mrem to any organ.

c. If the calculated dose from these gaseous effluents exceeds any of above limits in lieu of any other report required by Specification 6.6.2, a report shall be submitted within 30 days from the end of the quarter during which the release occured to the regional NRC Office of Inspection and Enforcement which includes the following:

1. Cause(s) for exceeding the limit(s);

2. A description of the program of corrective action initiated to: reduce the releases of radioactive materials in gaseous effluents, and to keep these levels of radioactive materials in gaseous effluents in compliance with the above limits or as low as reasonably achievable.

3.10.3 Gaseous Radwaste Treatment

a. The Gaseous Radwaste Treatment System shall be used to reduce the noble gases in gaseous wastes prior to their discharge, if the projected gaseous effluent air dose due to gaseous effluent releases from the site, when averaged over 31 days exceeds 0.6 mrad for gamma radiation and 1.2 mrad for beta radiation.

b. The Ventilation Treatment Exhaust System shall be used to reduce radioactive materials other than noble gases in gaseous waste prior to their discharge when the projected doses due to effluent releases to unrestricted areas when averaged over 31 days would exceed 0.9 mrem to any organ. This does not apply to the Auxiliary Building Exhaust System since it is not "treated" prior to release.

c. If radioactive gaseous waste is discharged without treatment for more than 31 days and in excess of the above limits in lieu of any other report required by specification 6.6.2, a report shall be submitted within 30 days to the regional NRC Office of Inspection and Enforcement which includes the following:

1. Cause of equipment or subsystems inoperability

2. Corrective action to restore equipment and prevent recurrence

3.10.4 Waste Gas Holdup Tanks

a. The quantity of radioactivity contained in each waste gas holdup tank shall be limited to $\leq$ 3.8E+05 curies noble gases (considered as Xe-133).

b. Daily, when radioactive materials are being added to a waste gas holdup tank, the quantity of radioactive material contained in the tank being filled shall be determined.

c. If the quantity of radioactive material in any waste gas holdup tank exceeds the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours, reduce the tank contents to within the above limit.

3.10.5 Used Oil Incineration

Used oil, contaminated by radioactivity, may be incinerated in the Station auxiliary boiler provided releases do not exceed one-tenth of one percent (0.1%) of the limits in Technical Specification 3.10.2.b.2.

3.10.6 The provisions of Technical Specifications 3.0 do not apply.

Bases

Specification 3.10.1 is provided to assure that the dose rate at anytime at the exclusion area boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactivity material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the exclusion area boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the exclusion area boundary to $\leq$ 500 mrem/year to the total body or to $\leq$ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid rate above background to an infant via the milk animal-milk-infant pathway to $\leq$ 1500 mrem/year for the nearest milk animal to the plant.

Specification 3.10.3 is provided to implement the requirements of Appendix I, 10 CFR Part 50. The specification provides the required operating flexibility and at the same time implement the guides set forth in Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." Surveillance requirements are implemented to meet the requirements of Appendix I. Calculational procedures based on models and data show that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated.

The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating 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 I, 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."

Equations in the ODCM are provided for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which are examined in the development of these calculations are: 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.

The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the release of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section IID of Appendix I to 10 CFR Part 50.

Restricting the quantity of radioactivity contained in each waste gas holdup tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem.

3.11 SOLID RADIOACTIVE WASTE

Applicability

Applies to the processing and packaging of radioactive solid waste prior to shipment from the site.

Specification

3.11.1 Solid Radioactive Waste

a. The Solid Radwaste System shall be used in accordance with a Process Control Program, for the solidification of wet radioactive wastes. Prior to the shipment of containers of radioactive wastes from the site, radioactive wastes shall be processed and packaged to ensure meeting the requirements of 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations governing the disposal of radioactive wastes.

b. If the requirements of 10CFR Part 20 and/or 10CFR Part 71 are not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

3.11.2 Process Control Program

The Process Control Program shall be used to verify the Solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste to be solidified.

1. Solidification

a. If any test specimen fails to verify Solidification, the Solidification of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative Solidification parameters can be determined in accordance with the Process Control Program, and a subsequent test verifies Solidification. Soldification of the batch may then be resumed using the alternative Solidification parameters determined by the Process Control Program.

2. Process Control Program

b. If the initial test specimen from a batch of waste fails to verify Solidification, the Process Control Program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate Solidification. The Process Control Program shall be modified as required to assure Solidification of subsequent batches of waste.

3.11.3 The provisions of Technical Specification 3.0 do not apply.

Bases

The solid radwaste system will be used whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification

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

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

4.1.2 The frequency and type of surveillance required for selected equipment shall be as stated in Table 4.1-2.

4.1.3 Required sampling should be performed as detailed in Table 4.1-3.

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in Table 4.1-4.

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days.

Bases

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system

instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Substantial calibration shifts within a channel (essentially a channel failure) are revealed during routine checking and testing procedures. Thus, the minimum calibration frequencies set forth are considered acceptable.

Periodic use of the Incore Instrumentation System for power mapping is sufficient to assure that axial and radial power peaks and the peak locations are controlled in accordance with the provisions of the Technical Specifications.

REFERENCE

(1) FSAR, Section 7.2.3.4.

TABLE 4.1-1 (Continued)

Channel Description	Check	Test	Calibrate	Remarks
20. Reactor Building Spray System Logic	NA	MO	NA	
21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	RF	
22. Pressurizer Temperature	ES	NA	RF	
23. Control Rod Absolute Position	ES(1)	NA	RF(2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.
24. Control Rod Relative Position	ES(1)	NA	RF(2)	(1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel.
25. Core Flood Tanks				
a. Pressure	ES	NA	RF	
b. Level	ES	NA	RF	
26. Pressurizer Level	ES	NA	RF	
27. Letdown Storage Tank	DA	NA	RF	
28. Delete				
29. High and Low Pressure Injection Systems Flow Channels	NA	NA	RF	

TABLE 4.1-3

Minimum Sampling Frequency And Analysis Program

Item	Check	Frequency
1. Reactor Coolant	a. Gamma Isotopic Analysis	a. 3 times/week*
	b. Radiochemical Analysis for Sr 89, 90	b. Monthly*
	c. Tritium	c. Monthly*
	d. Gross Beta Activity (1)	d. 3 times/week*
	e. Chemistry (Cl, F and O_2)	e. 5 times/week*
	f. Boron Concentration	f. 2 times/week**
	g. Gross Alpha Activity	g. Monthly*
	h. $\bar{E}$ Determination (2)	h. Semi-annually
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly* and after each makeup
3. Core Flooding Tank	Boron Concentration	Monthly* and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly*** and after each makeup
5. OTSG or Final Feedwater	a. Gross Beta Activity	a. Weekly*
	b. Gamma Isotopic Analysis (3)	
6. Concentrated Boric Acid Tank	Boron Concentration	Twice weekly*

*Not applicable if reactor is in a cold shutdown condition for a period exceeding the sampling frequency.
**Applicable only when fuel is in the reactor.
***Applicable only when fuel is in wet storage in the spent fuel pool.

TABLE 4.1-3 Continued

Minimum Sampling Frequency And Analysis Program

Item	Check	Frequency	Lower Limit of Detection[5] of Lab Analysis for Waste
7. Condensate Test Tank, Condensate Monitoring Tank, Laundry-Hot Shower Tank	a. Principal Gamma Emitters[6] including Dissolved Noble Gases	a. Composite Grab Sample prior to release of each batch[11]	a. Ce-144 and Mo-99 $<5x10^{-6}$ μCi/ml Other Gamma Nuclides $<5x10^{-7}$ μCi/m Dissolved Gases $<10^{-5}$ μCi/ml I-131 $<10^{-6}$ μCi/ml
	b. Radiochemical Analysis Sr 89, 90, Fe-55	b. Quarterly from all composited batches[9]	b. $<5x10^{-8}$ μCi/ml for Sr's $<10^{-6}$ μCi/ml for Fe-55
	c. Tritium	c. Monthly Composite	c. $<10^{-5}$ μCi/ml
	d. Gross Alpha Activity	d. Monthly Composite	d. $<10^{-7}$ μCi/ml
8. Unit Vent Sampling (Includes Waste Gas Decay Tanks, Reactor Building Purges, Auxiliary Building Ventilation, Spent Fuel Pool Ventilation, Air Ejectors)	a. Iodine Spectrum[4]	a. Continuous monitor, weekly sample[8]	a. $<10^{-10}$ μCi/cc (I-133) $<10^{-12}$ μCi/cc (I-131)
	b. Particulates[4]		
	(1) Ce-144 and Mo-99	(1) Weekly Composite[8]	(1) $<5x10^{-9}$ μCi/cc
	(2) Other Principal Gamma Emitters[7]	(2) Weekly Composite[8]	(2) $<10^{-10}$ μCi/cc
	(3) Gross Alpha Activity	(3) Monthly, using composite samples of one week	(3) $<10^{-11}$ μCi/cc
	(4) Radiochemical Analysis Sr 89, 90	(4) Quarterly Composite	(4) $<10^{-11}$ μCi/cc
	c. Gases by Principal Gamma Emitters[7]	c. Weekly Grab Sample	c. $<10^{-4}$ μCi/cc
	d. Tritium	d. Weekly Grab Sample	d. $<10^{-6}$ μCi/cc

TABLE 4.1-3 Continued

Minimum Sampling Frequency And Analysis Program

	Item	Check	Frequency	Lower Limit of Detection[5] of Lab Analysis for Waste
8a.	Waste Gas Decay Tank	a. Principal Gamma Emitters[7]	a. Grab Sample prior to release of each batch	a. $<10^{-4}$ µCi/cc (gases) $<10^{-10}$ µCi/cc (particulates and iodines)
		b. Tritium	b. Grab sample prior to release of each batch	b. $<10^{-6}$ µCi/cc
8b.	Reactor Building	a. Principal Gamma Emitters[7]	a. Grab Sample each purge	a. $<10^{-4}$ µCi/cc (gases) $<10^{-10}$ µCi/cc (particulates and iodines)
		b. Tritium	b. Grab Sample each purge	b. $<10^{-6}$ µCi/cc
9.	Keowee Hydro Dam Dilution Flow	Measure Leakage Flow Rate	Annually	
10.	Delete			
11.	Backwash Receiving Tanks	Principle Gamma Emitters including dissolved noble gases	Grab Sample prior to release of each batch	
12.	#3 Chemical Treatment Pond Effluent	a. Principal Gamma Emitters[6]	a. Monthly from composite sample[10]	a. Ce-144 and Mo-99 $<5\times10^{-6}$ µCi/ml Other Gamma Nuclides $<5\times10^{-7}$ µCi/ml Dissolved Gases $<10^{-5}$ µCi/ml I-131 $<10^{-6}$ µCi/ml
		b. Radiochemical Analysis Sr-89, Sr-90, Fe-55	b. Quarterly from composite sample[9]	b. $<5\times10^{-8}$ µCi/ml for Sr's $<10^{-6}$ µCi/ml for Fe-55
		c. Tritium	c. Monthly from composite sample[10]	c. $<10^{-5}$ µCi/ml
		d. Gross Alpha Activity	d. Monthly from composite sample[10]	d. $<10^{-7}$ µCi/ml

TABLE 4.1-3 NOTES

(1) When radioactivity level is greater than 10 percent of the limits of Specification 3.1.4, the sampling frequency shall be increased to a minimum of once each day.

(2) $\bar{E}$ determination will be started when gross gamma activity analysis indicates greater than 10 μCi/ml and will be redetermined for each 10 μCi/ml increase in gross gamma activity analysis thereafter. A radiochemical analysis for this purpose shall consist of a quantitative measurement of 95 percent of the radionuclides in the reactor coolant with half lives greater than 30 minutes. This is expected to consist of gamma isotopic analysis of the primary coolant, including dissolved gaseous activities, radiochemical analysis for Sr-89 and Sr-90, and tritium analysis.

(3) When gross beta activity increases by a factor of two above background, iodine concentrations will be determined by gamma isotopic analysis and performed thereafter when the gross beta activity increases by 10 percent.

(4) Samples shall be changed at least once per 24 hours and analyses shall be completed within 48 hours after changing (on after removal from sampler).

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

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

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

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

E is the counting efficiency (as counts per disintegration),

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

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

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

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

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.

(6) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

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

(8) 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 Specification 3.10.1, 3.10.2.a and 3.10.2.b.

(9) 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 which is representative of the liquids released.

(10) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

TABLE 4.1-3 NOTES (Continued)

(11) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, to assure representative sampling.

TABLE 4.1-4

RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL RESPONSE CHECK(4)	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Liquid Radwaste Effluent Line				
a. Effluent Line Monitor (1 RIA-33)	*	DA	AN	QU(1)
b. Effluent Flow Rate Monitor	*	NA	AN	NA
c. Minimum Flow Device	*	NA	AN	NA
2. Turbine Building Sump				
a. Sump Monitor (RIA-54)	DA	MO	AN(3)	QU(2)
b. Minimum Flow Device	*	NA	AN	NA
3. Low Pressure Service Water				
a. Effluent Line Monitor (RIA-35)	DA	MO	AN(3)	QU(1)
b. Minimum Flow Device	*	NA	AN	NA
4. #3 Chemical Treatment Pond Composite Sampler	DA	NA	AN	NA
5. Waste Gas Holdup System				
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RIA-37, -38)	*	DA	AN(3)	QU(1)
b. Effluent Flow Rate Monitor (Waste Gas Discharge Flow)	*	NA	AN	NA
6. Unit Vent Monitoring				
a. Noble Gas Activity Monitor (RIA-45)	DA	MO	AN(3)	QU(2)
b. Iodine Sampler	DA	NA	NA	NA
c. Particulate Sampler	DA	NA	NA	NA
d. Effluent Flow Rate Monitor (Unit Vent Flow)	DA	NA	AN	NA
e. Minimum Flow Device	DA	NA	AN	NA

TABLE 4.1-4 (CONTINUED)

RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL RESPONSE CHECK(4)	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
7. Interim Radwaste Building Ventilation Monitoring				
a. Noble Gas Activity Monitor (RIA-53)	DA	MO	AN(3)	QU(2)
b. Iodine Sampler#	DA	NA	NA	NA
c. Particulate Sampler#	DA	NA	NA	NA
d. Effluent Flow Rate Monitor (Interim Radwaste Exhaust)#	DA	NA	AN	NA
e. Minimum Flow Device#	DA	NA	AN	NA
8. Hot Machine Shop				
a. Iodine Sampler#	DA	NA	NA	NA
b. Particulate Sampler#	DA	NA	NA	NA
c. Effluent Flow Rate Monitor (Hot Machine Shop Exhaust)#	DA	NA	AN	NA
d. Minimum Flow Device#	DA	NA	AN	NA

*During each release via this pathway.
#Effective upon installation of equipment.

Frequency Notation

DA -Daily
QU - Quarterly
MO - Monthly
AN - Annually
PR - Completed prior to each release
NA - Not Applicable

New Page

TABLE 4.1-4 (Continued)

TABLE NOTATION

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

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure (downscale only).

(2) The Channel Functional Test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists.

1. Instrument indicates measured levels above the alarm setpoint.

2. Circuit failure (downscale only).

(3) The initial Channel Calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained form suppliers that participate in measurement assurance activities with NBS. The standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent Channel Calibration sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for this requirement).

(4) The Channel Response Check shall consist of verifying indications during periods of release. Channel Response Check shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

4.11 RADIOLOGICAL ENVIRONMENTAL MONITORING

Applicability

Applies to the surveillance of the station environ for radiation and radioactive materials attributable to station operation and effluent releases.

Specification

4.11.1 Radiological Environmental Monitoring Program

a. The radiological environmental monitoring samples shall be collected in accordance with Table 4.11-1 and shall be analyzed pursuant to the requirements of Tables 4.11-1, 4.11-2 and 4.11-3.

b. If the radiological environmental monitoring program is not conducted as required, a description of the reason for not conducting the program as required and plans to prevent a recurrence shall be included in the Annual Radiological Environmental Operating Report. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

c. If samples become permanently unavailable from any of the required sample locations, the locations from which samples were unavailable may then be deleted from the program provided replacement samples were obtained and added to the environmental monitoring program, if available. These new locations will be identified in the next semi-annual report.

4.11.2 Land Use Census

a. A land use census shall be conducted and shall identify the location of the nearest milk animal and the nearest residence in each of the 16 meterological sectors within a distance of five miles. Broad leaf vegetation sampling shall be performed at the site boundary in the direction sector with the highest $\overline{D/Q}$ in lieu of the garden census.

b. If a land use census identifies a location which yields a calculated dose or dose commitment (via the same exposure pathway) greater than a location from which samples are currently being obtained pursuant to Specification 4.11.1, then the new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location having the lowest calculated dose or dose commitment (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. These new locations will be identified in the next semi-annual report.

c. 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.

4.11.3 Interlaboratory Comparison Program

a. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the NRC.

b. If these analyses are not performed as required, report corrective actions in the Annual Radiological Environmental Operating Report.

c. A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the methodology and parameters in the ODCM shall be included in the Annual Radiological Environmental Operating Report.

4.11.4 The provisions of Technical Specification 3.0 do not apply.

Bases

The environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program 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 modeling of program will be effective for at least the first three years of commerical operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.11-1 are considered optimum for routine environmental measurements in industrial laboratories. The specified lower limits of detection correspond to less than the 10CFR50, Appendix I, design objective dose-equivalent to 45 mrem/year for atmospheric releases to the most sensitive organ and individual.

The land use census specification is provided to assure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census.

The requirements for participation in an Interlaboratory Comparison Program is provided to assure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

TABLE 4.11-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis***
1. AIRBORNE			
a. Radioiodine and Particulates	5 Locations	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.	Radioiodine canister. Gamma isotopic analysis for I-131 or each sample. Particulate sampler. Gamma isotopic analysis on each sample.
2. DIRECTION RADIATION			
	40 Locations	Continuous integration with collection at least once per 92 days.	Gamma dose on each dosimeter.
3. WATERBORNE			
a. Surface	2 Locations	Composite* sample collected over a period of ≤ 31 days.	Gamma isotopic analysis of each composite sample by location. Tritium analyses of composite sample at least once per 92 days.
b. Drinking	3 Locations	Composite* sample collected over a period of ≤ 31 days.	Gross beta and gamma isotopic analysis of each composite sample. Tritium analysis of composite sample at least once per 92 days.

*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.
**Sample locations are identified in the ODCM.
***Frequency of analysis stated only if different from collection frequency.

TABLE 4.11-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis***
c. Sediment from Shoreline	2 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.
4. INGESTION			
a. Milk	3 Locations	At least once per 15 days when animals are on pasture; at least once per 31 days at other times.	Gamma isotopic and I-131 analysis of each sample.
b. Fish	2 Locations	At least once per 184 days. One sample of each of the following species: 1. Bass 2. Catfish	Gamma isotopic analysis on edible portion.
c. Broad-leaf Vegetation	2 Locations	At least once per 31 days.	Gamma isotopic analysis.

*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.
**Sample locations are identified in the ODCM.
***Frequency of analysis stated only if different from collection frequency.

TABLE 4.11-2

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m³)	Fish (pCi/kg,wet)	Milk (pCi/l)	Broadleaf Vegetation (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4					
^{3}H	2000					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr	30					
^{95}Nb	15					
^{131}I	15[b]	7×10^{-2}		1	60	
$^{134, 137}Cs$	15,18	$5,6 \times 10^{-2}$	130,150	15,18	60,80	150,180
^{140}Ba	60			60		
^{140}La	15			15		

TABLE 4.11-2 (Continued)

TABLE NOTATION

a - The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample with 95% probability of detection and with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

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

where

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume)

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

E is the counting efficiency (as counts per disintegration)

V is the sample size (in units of mass or volume)

2.22 is the number of disintegrations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide.

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting

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

The LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as _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, unavoidably small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances, may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b - LLD for gamma isotopic analysis for I-131 in drinking water samples. Low level I-131 analysis on drinking water will not be routinely performed because the calculated dose from I-131 in drinking water at all locations is less than 1 mrem per year. Low level I-131 analyses will be performed if abnormal releases occur which could reasonably result in $\geq$ 1 pCi/liter of I-131 in drinking water. For low level analyses of I-131 an LLD of 1 pCi/liter will be achieved.

c - Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.11-2, shall be identified and reported.

TABLE 4.11-3

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)	Broad Leaf Vegetation (pCi/Kg,wet)
H-3	2×10^4*				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2				
I-131	2**	1.0		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2			3×10^2	

*For drinking water samples. This is 40 CFR Part 141 value.
**If low level I-131 analyses are performed.

4.21 DOSE CALCULATIONS

Applicability

Applies to the projected and cummulative dose contributions from all radioactive liquid and gaseous effluents.

Specification

4.21.1 Dose From All Sources

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 total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.9.2.a, 3.10.2.a, or 3.10.2.b, calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 4.21.1 have been exceeded. If such is the case in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6.3, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a Member Of The Public from uranium fuel cycle sources, (including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe the levels of radiation and concentration 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 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Technical Specification 3.0 do not apply.

4.21.2 Dose Due to Liquid Effluents

a. Monthly, cummulative dose contributions from liquid effluents shall be determined in accordance with the Offsite Dose Calculation Manual.

4.21.3 Dose Due to Gaseous Effluents

a. Monthly, cummulative dose contributions from gaseous effluents shall be determined in accordance with the Offsite Dose Calculation Manual.

5 DESIGN FEATURES

5.1 SITE

5.1.1 The Oconee Nuclear Station is approximately eight miles northeast of Seneca, South Carolina. Figure 2-3 of the Oconee FSAR shows the plan of the site. The minimum distance from the reactor center line to the boundary of the exclusion area and to the outer boundary of the low population zone as defined in 10 CFR 100.3, shall be one mile and six miles respectively.

5.1.2 For the purposes of satisfying 10 CFR Part 20, the "Restricted Area," for gaseous release purposes only, is the same as the exclusion area as defined above.

REFERENCE

(1) FSAR, Chapter 2

(2) Technical Specification 3.10.

years of the remaining five years of experience may be fulfilled by academic training, or related technical training on a one-for-one time basis. The Operating Engineer shall hold a Senior Reactor Operator license.

6.1.1.5 Retraining and replacement of station personnel shall be in accordance with Section 5.5 of the ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."

6.1.1.6 A training program for the fire brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions may be held quarterly.

6.1.1.7 The two functions of the Shift Technical Advisor, namely accident assessment and operating experience assessment, are fulfilled in the following manner:

a. An experienced SRO, who has been instructed in additional academic subjects, will be assigned on-shift to provide the accident assessment capability.

b. The operating experience assessment function will be provided by the Station Safety Review Group.

6.1.2 Technical Review and Control

6.1.2.1 Activities

a. Procedures required by Technical Specification 6.4 and other procedures which affect station nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared by a qualified individual/organization. Each such procedure, or procedure change, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or procedure change, but who may be from the same organization as the individual/group which prepared the procedure, or procedure change. Such procedures and procedure changes may be approved for temporary use by two members of the station staff, at least one of whom holds a Senior Reactor Operator's License on the unit(s) affected. Procedures and procedure changes shall be approved prior to use or within seven days of receiving temporary approval for use by the station Manager; or by the Operating Superintendent, the Technical Services Superintendent or the Maintenance Superintendent, as previously designated by the Station Manager.

b. Proposed changes to the Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager.

c. Proposed modifications to station nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to station nuclear safety-related structures, systems and components shall be approved prior to implementation by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent, or the Maintenance Superintendent, as previously designated by the Station Manager.

d. Individuals responsible for reviews performed in accordance with 6.1.2.1.a, 6.1.2.1.b, and 6.1.2.1.c shall be members of the station supervisory staff, previously designated by the Station Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated station review personnel.

e. Proposed tests and experiements which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent or the Maintenance Superintendent, as previously designated by the Station Manager.

f. Incidents reportable pursuant to Technical Specification 6.6.2.1 and violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the station Manager and transmitted to the Vice President, Nuclear Production Department, or his designee; and to the Director of the Nuclear Safety Review Board.

g. The Station Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President, Nuclear Production Department.

h. The station security program, and implementing procedures, shall be reviewed at least annually. Changes determined to be necessary as a result of such review shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production Department, or his designee; and to the Director of the Nuclear Safety Review Board.

i. The station emergency plan, and implementing procedures, shall be reviewed at least annually. Changes determined to be necessary as a result of such review shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production Department, or his designee; and the Director of the Nuclear Safety Review Board.

j. The Station Manager shall assure that an independent fire protection and loss prevention inspection and audit shall be performed annually utilizing qualified off-site personnel and that an inspection and audit by a qualified fire consultant shall be performed at intervals no greater than three years.

k. Unplanned onsite releases of radioactive material to the environs shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production Department, or his designee; and to the Director of the Nuclear Safety Review Board.

l. Proposed changes to the Offsite Dose Calculation Manual (ODCM) shall be prepared by a qualified individual/organization. Each proposed change shall be reviewed by an individual/group other than the individual group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the ODCM shall be approved by the Station Manager prior to implementation.

6.1.2.2 Records

Records of the above activities shall be maintained.

6.1.3 Nuclear Safety Review Board

6.1.3.1 Function

The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations
b. Nuclear Engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Administrative control and quality assurance practices

6.1.3.2 Organization

a. The Director, members and alternate members of the NSRB shall be formally appointed by the Vice President, Nuclear Production Department, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in one or more areas given in 6.1.3.1.

b. The NSRB shall be composed of at least five members, including the Director, Members of the NSRB may be from the Nuclear Production Department, from other departments within the Company or from external to the Company. A maximum of one member of the NSRB may be from the Oconee Nuclear Station staff.

c. Consultants may be utilized by the NSRB to provide expert advice to the NSRB, as determined necessary by the Director of the NSRB.

d. Staff assistance may be provided to the NSRB in order to promote the proper, timely and expenditious performance of its functions.

e. The NSRB shall meet at least once per six months. The period between such meetings shall not exceed eight months.

f. A quorum of the NSRB shall consist of the Director, or his designated alternate, and at least two other NSRB members or alternate members. No more than a minority of the quorum shall have line responsibility for operation of Oconee Nuclear Station.

New Page

6.1.3.3 Subjects Requiring Review

The following subjects shall be reported to and reviewed by the NSRB:

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

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

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

d. Proposed changes in Technical Specifications or the Facility Operating Licenses.

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

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

g. Incidents that are the subject of non-routine reports submitted to the Commission.

h. Quality Assurance Department audits relating to station operations and actions taken in response to these audits.

6.1.3.4 Audits

Audits of station activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of station operation to provisions contained within the Technical Specifications and applicable facility operating license conditions at least once per year.

b. The performance, training and qualifications of the station staff at least once per year.

c. The results of actions taken to correct deficiencies occurring in equipment, structures, systems or methods of operation that affect nuclear safety at least once per six months.

d. The performance of activities required by the quality assurance program to meet the criteria of Appendix B to 10 CFR 50 at least once per two years.

e. The station emergency plan and implementing procedures at least once per 12 months.

f. The station security plan and implementing procedures at least once per 12 months.

g. Any other area of station operation considered appropriate by the NSRB or the Vice President, Nuclear Production Department.

h. The station fire protection program and implementing procedures at least once per 24 months.

i. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

j. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months.

k. The Process Control Program and implementing procedures for solidification of radioactive wastes at least once per 24 months.

l. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 1.21 Revision 1, June 1974 and Regulatory Guide 4.1 Revision 1, April 1975 at least once per 12 months.

6.1.3.5 Responsibilities and Authorities

a. The NSRB shall report to and advise the Vice President, Nuclear Production Department on those areas of responsibility specified in Specifications 6.1.3.3 and 6.1.3.4.

b. Minutes shall be prepared and forwarded to the Vice President, Nuclear Production Department, and to the Executive Vice President, Power Operations, within 14 days following each formal meeting of the NSRB.

c. Records of activities performed in accordance with Specifications 6.1.3.3 and 6.1.3.4 shall be maintained.

d. Audit reports encompassed by Section 6.1.3.4 shall be forwarded to the Vice President, Nuclear Production Department, and to the Executive Vice President, Power Operations and to the management position responsible for the areas audited within 30 days of completion of each audit.

6.4 STATION OPERATING PROCEDURES

Specification

6.4.1

The station shall be operated and maintained in accordance with approved procedures. Written procedures with appropriate check-off lists and instructions shall be provided for the following conditions:

a. Normal startup, operation, and shutdown of the complete facility and of all systems and components involving nuclear safety of the facility.

b. Refueling operations.

c. Actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety and radiation levels, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

d. Emergency procedures involving potential or actual release of radioactivity.

e. Preventive or corrective maintenance which could affect nuclear safety or radiation exposure to personnel.

f. Station survey following an earthquake.

g. Personnel radiation protection procedures.

h. Operation of radioactive waste management systems.

i. Control of pH in recirculated coolant after loss-of-coolant accident. Procedure shall state that pH will be measured and the addition of appropriate caustic to coolant will commence within 30 minutes after switchover to recirculation mode of core cooling to adjust the pH to a range of 7.0 to 8.0 within 24 hours.

j. Nuclear safety-related periodic test procedures.

k. Long-term emergency core cooling systems. Procedures shall include provision for remote or local operation of system components necessary to establish high and low pressure injection within 15 minutes after a line break.

l. Fire Protection Program implementation.

m. Offsite Dose Calculation Manual implementation.

n. Process Control Program implementation.

6.4.2

A respiratory protective program approved by the Commission shall be in force.

h. By-product material inventory records.

i. Minutes of Nuclear Safety Review Board Meetings.

j. Training records.

k. Test results, in units of microcuries, for leak tests performed pursuant to Specification 4.16.

l. Radioactive liquid effluent, gaseous effluent, and gaseous process monitoring instrumentation alarm/trip setpoints.

6.6 STATION REPORTING REQUIRMENTS

6.6.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Office of Inspection and Enforcement, Region II unless otherwise noted.

6.6.1.1 Startup Report

A summary report of unit startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the facility license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly alered the nuclear, thermal, or hydraulic performance of the unit. Startup reports shall be submitted (1) within 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) nine months following initial criticality, whichever occurs first. If a startup report does not cover all three events, i.e., initial criticality, completion of the startup test program and resumption or commencement of commercial power operation supplementary reports shall be submitted at least every three months until all three events are completed.

6.6.1.2 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

6.6.1.3 Personnel Exposure and Monitoring Report

Prior to March 1 of each year, a tabulation shall be submitted to the NRC of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total body dose received from external sources shall be assigned to specific major work functions.

6.6.1.4 Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operating of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the station.

The Radioactive Effluent Release Reports shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter.

The Radioactive Effluent Release Reports shall include an assessment of the radiation doses from radioactive effluents to individuals due to their activities inside the unrestricted area boundary during the report period. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports.

The Radioactive Effluent Release Reports shall include the following information for all unplanned releases to unrestricted areas of radioactive materials in gaseous and liquid effluents:

a. A description of the event and equipment involved.

b. Cause(s) for the unplanned release.

c. Actions taken to prevent recurrence.

d. Consequences of the unplanned release.

The Radioactive Effluent Release Reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the station during each calendar quarter. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The annual average meteorological conditions shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods are acceptable. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual.

The Radioactive Effluent Release Reports shall include the following information for each type of solid waste shipped offsite during the report period:

a. container volume,

b. total curie quantity (determined by measurement or estimate),

c. principal radionuclides (determined by measurement or estimate),

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

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

f. solidification agent (e.g., cement, or other approved agents (media)).

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

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

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 and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Methods for calculating the dose contribution from liquid and gaseous effluents are given in the ODCM.

6.6.1.5 Radiological Environmental Monitoring

Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and 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 the land use censuses required by Specification 4.11. If harmful effects are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The Annual Radiological Environment Operating Report shall include a summary of the results obtained as part of the required Interlaboratory Comparison Program and in accordance with the ODCM. Alternatively, participants in the EPA cross-check program shall provide the EPA program code designation for the unit.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the radiological environmental samples required by Specification 4.11 taken during the report period. In the event that some 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 practical in a supplementary report.

The initial report shall also include the following: a summary description of the radiological environmental monitoring program including sampling methods for each sample type, size and physical characteristics of each sample type, sample preparation methods, analytical methods, and measuring equipment used; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the result of land use censuses required by Specification 4.11. Subsequent reports shall describe all substantial changes in these aspects.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

b. Thirty-Day Written Reports

The types of events listed below shall be the subject of written reports to the Regional Administrator, Office of Inspection and Enforcement, Region II, within 30 days of discovery of the event. (Copy to the Director, Office of Management Information and Program Control.)

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or shutdown required by a limiting condition for operation.

(3) Observed inadequacies in the implementation of administrative or procedural controls during operation of a unit which could cause reduction of degree of redundancy provided in the Reactor Protective System or Engineered Safety Feature Systems.

(4) Occurrence of radioactive material contained in liquid or gaseous holdup tanks in excess of that permitted by the limiting condition for operation established in the technical specifications.

(5) An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1. A description of the event and equipment involved.

2. Cause(s) for the unplanned release.

3. Action taken to prevent recurrence.

4. Consequences of the unplanned release.

(6) Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 4.11-3 when averaged over any calendar quarter sampling period. When more than one of the radionuclides in Table 4.11-3 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 4.11-3 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year objectives of Specifications 3.9 and 3.10. 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.

6.6.2.2 Environmental Monitoring

a. If individual milk samples show I-131 concentrations of 10 picocuries per liter or greater, a plan shall be submitted within one week advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 45 mrem/yr to the thyroid of any individual.

b. If milk samples collected over a calendar quarter show average concentrations of 4.8 picocuries per liter or greater, a plan shall be submitted within 30 days advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 45 mrem/yr to the thyroid of any individual.

6.6.3 Special Reports

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

a. Single Loop Restrictions, Specification 3.1.8

b. Auxiliary Electrical Systems, Specification 3.7

c. Radioactive Liquid Effluents,
   Dose, Specification 3.9.2
   Liquid Waste Treatment, Specification 3.9.3
   Chemical Treatment Ponds, Specification 3.9.4

d. Radioactive Gaseous Effluents,
   Dose, Specification 3.10.2
   Gaseous Radwaste Treatment, Specification 3.10.3

e. Fire Protection and Detection Systems, Specification 3.17

f. Reactor Coolant System Surveillance,
   Inservice Inspection, Specification 4.2.1
   Reactor Vessel Specimen, Specification 4.2.4

g. Reactor Building Surveillance,
   Containment Leakage Tests, Specification 4.4.1

h. Structural Integrity Surveillance,
   Tendon Surveillance, Specification 4.4.2.2

i. Radiological Environmental Monitoring
   Program, Specification 4.11.1
   Land Use Census, Specification 4.11.2

j. Dose Calculations (40 CFR 190), Specification 4.21

6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.8.1

The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications.

The ODCM shall be submitted to the Commission at the time of proposed Radiological Effluent Technical Specifications and shall be subject to review and approval by the Commission prior to implementation.

6.8.2 Any changes to the ODCM shall be made by the following method:

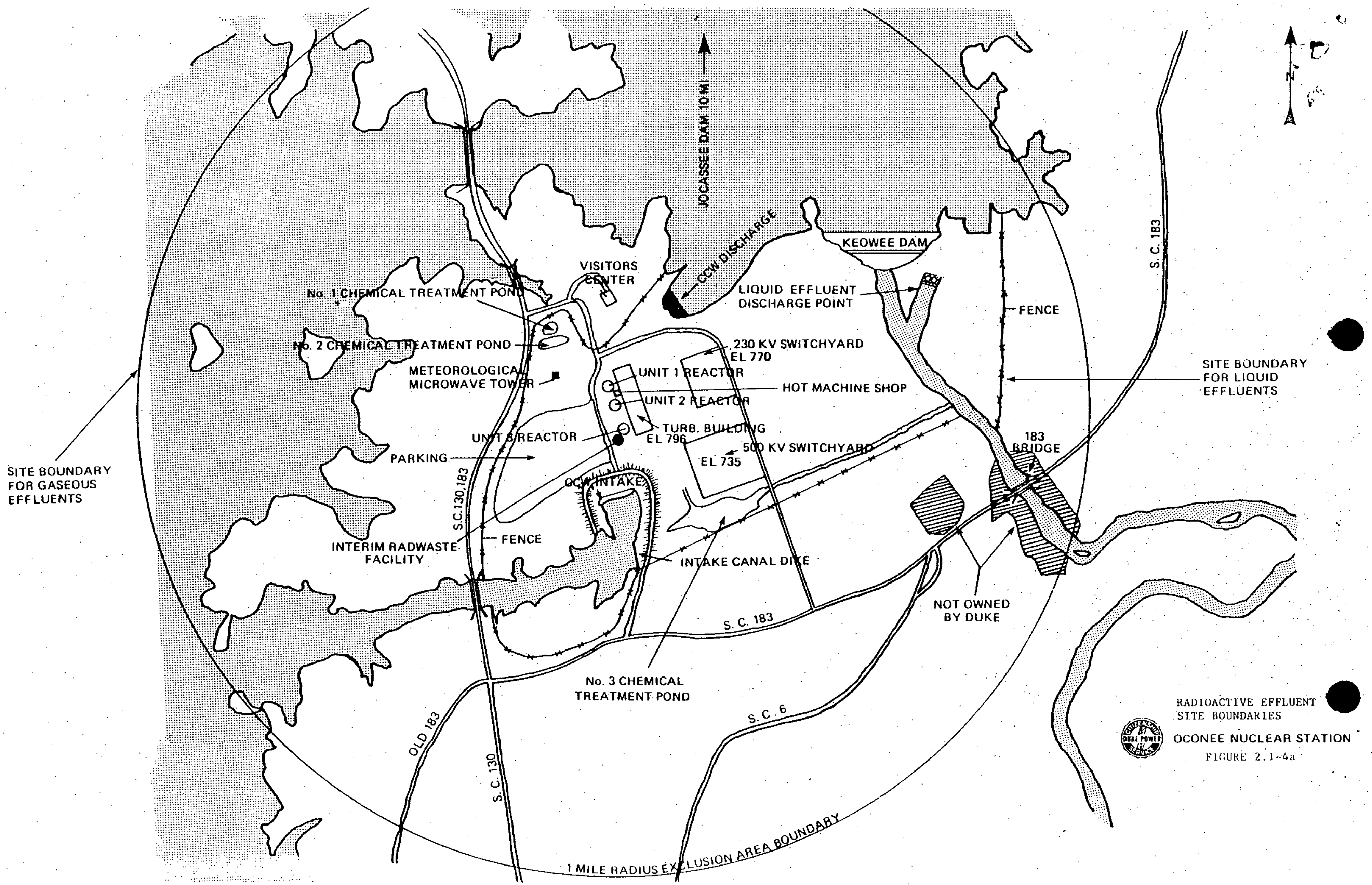
1. Shall be submitted to the Commission by inclusion in the semi-annual Effluent Release Report for the period in which the change(s) was made and shall contain:

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

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

    c. documentation of the fact that the change has been reviewed in accordance with Technical Specification 6.1.2.1.(1) and found acceptable by the Station Manager.

2. Shall become effective upon review and acceptance by the Station Manager after confirmation of receipt unless otherwise acted upon by the Commission through written notification to the licensee.



RADIOACTIVE EFFLUENT SITE BOUNDARIES

OCONEE NUCLEAR STATION

FIGURE 2.1-4a