PROPOSED CHANGES
TO THE

E. I. HATCH NUCLEAR PLANT
OFFSITE DOSE CALCULATION MANUAL

IN ACCORDANCE WITH
NRC GENERIC LETTER 89-01

E. I. HATCH NUCLEAR PLANT
OFFSITE DOSE CALCULATION MANUAL

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REVISION 7 2/7/92

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OFFSITE DOSE CALCULATION MANUAL

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OFFSITE DOSE CALCULATION MANUAL

FOR

GEORGIA POWER COMPANY

EDWIN I. HATCH NUCLEAR PLANT

FEBRUARY 1992

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INTRODUCTION

The Offsite Dose Calculation Manual (ODCM) is a supporting document of the technical specifications which address radiological effluents and radiological environmental monitoring. As such, the ODCM describes the methodology and parameters to be used in the calculation of offsite doses due to radioactive liquid and gaseous effluents and in the calculation of liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints. The ODCM contains a list and graphical description of the specific sampl locations for the radiological environmental monitoring program. Schematic configurations of the liquid and gaseous radwaste treatment systems are also included.

The ODCM will be maintained at the plant for use as a reference guide and training document of accepted methodologies and calculations. Changes in the calculational methods or parameters will be incorporated into the ODCM in order to assure that the ODCM represents the present methodology in all applicable areas. Computer software to perform the described calculations will be maintained current with the ODCM.

As required by the revised definition of the ODCM presented in NRC Generic Letter 89-01, this manual also contains the Radioactive Effluent Controls required by Technical Specification 6.18 and the Radiological Environmental Monitoring Program required by Technical Specification 6.19, along with the methods and parameters for conducting this program. This document also contains descriptions of information that should be included in the Annual Radiological Environmental Operating Report required by Technical Specification 6.9.1.6 and descriptions of information that should be included in the Semiannual Radioactive Effluent Release Report required by Technical Specification 6.9.1.8.

where:

C_{mpc} = the effluent concentration limit (Section 1.7.2) implementing 10 CFR 20 for the site, corresponding to the specific mix of radionuclides in the waste tank being considered for discharge, in μCi/ml.

c = the setpoint, in μCi/ml, of the radioactivity monitor that measures the radioactivity concentration in the effluent line prior to dilution and subsequent release; the setpoint, which is inversely proportional to the volumetric flow of the effluent line and proportional to the volumetric flow of the dilution stream plus the effluent stream, represents a value that, if exceeded, would result in concentrations exceeding the limits of 10 CFR 20 in the UNRESTRICTED AREA.

f = the flow setpoint as determined at the radiation monitor location in volume per unit time, but in the same units as F, below.

F = the dilution water flow setpoint as determined prior to the release point, in volume per unit time.

As stated earlier, at Plant Hatch, each of the two units is served by its own independent liquid radwaste system; the two liquid radwaste systems discharge to separate dilution streams. If additional dilution flow is needed for either dilution stream, it is available from the plant service water system. The two dilution streams release to the Altamaha River.

The sources of liquid radioactive effluents from Unit 1 are:

- Waste sample tank A.
- Waste sample tank B.
- Chemical waste sample tank A.
- Chemical waste sample tank B.

Step 1

The radionuclide concentration for a waste tank to be released is obtained from the sum of measured concentrations as determined by the analyses required in Table 1.7-3:

$$\sum_i C_i = \sum_g C_g + (C_a + C_s + C_f + C_t) \qquad (2)$$

where:

C_g = the concentration of each measured gamma emitter observed by gamma-ray spectroscopy of the particular waste sample.

C_a = the concentration of alpha emitters in liquid waste as measured in the MONTHLY composite sample. (NOTE: Sample is analyzed for gross α.)

C_s = the measured concentrations of Sr-89 and Sr-90 in liquid waste as observed in the QUARTERLY composite sample.

C_f = the measured concentrations of Fe-55 in liquid waste as observed in the QUARTERLY composite sample.

C_t = the measured concentration of H-3 in liquid waste as determined from analysis of the MONTHLY composite sample.

The C_g term will be included in the analysis of each batch; terms for alpha, strontiums, iron, and tritium will be included in accordance with Table 1.7-3, as appropriate.

Step 2

The measured radionuclide concentrations are used to calculate a dilution factor, DF, which is the ratio of total dilution flow rate to tank flow rate

1.2 DOSE CALCULATION FOR LIQUID EFFLUENTS

For liquid release from Plant Hatch to the Altamaha River, the only human exposure pathway is due to consumption of fish taken from the river. No known drinking water is taken from the Altamaha River downstream from Plant Hatch. The methodology for calculating doses to a MEMBER OF THE PUBLIC due to fish consumption is presented in this section.

The dose contribution to the maximum exposed MEMBER OF THE PUBLIC by way of fish consumption from radionuclides identified in liquid effluents released to UNRESTRICTED AREAS will be calculated for the purpose of implementing Section 1.7.3. In accordance with Appendix A of Reference 3, noble gases are excluded from these dose calculations. Doses to a MEMBER OF THE PUBLIC are calculated as follows:

$$D_\tau = \sum_i A_{i\tau} \sum_{\ell=1}^{m} \Delta t_\ell \, C_{i\ell} \, e^{-\lambda_i t_c} \, F_\ell \qquad (9)$$

where:

D_τ = the cumulative dose commitment to the total body or any organ, τ, due to radioactivity in liquid effluents for the total time period

$$\sum_{\ell=1}^{m} \Delta t_\ell, \text{ in mrem. (Reference 1).}$$

Δt_ℓ = the length of the ℓth time period over which $C_{i\ell}$ and F_ℓ are averaged for all liquid releases, in hours.

$C_{i\ell}$ = the average concentration of radionuclide i, in undiluted liquid effluent during time period Δt_ℓ from any liquid release, in μCi/ml.

λ_i = the decay constant for radionuclide i (sec^{-1}).

1.3 DOSE PROJECTIONS FOR LIQUID EFFLUENTS

1.3.1 Monthly Dose Projections

In order to meet the requirements of Section 1.7.4, which pertains to operation of the liquid radwaste treatment systems, dose projections must be made at least monthly during periods in which discharge of untreated liquid effluents containing radioactive materials to UNRESTRICTED AREAS occurs or is expected.

Projected 31-day doses to a MEMBER OF THE PUBLIC due to liquid effluents may be determined as follows:

$$D_{tb(prj)} = \left(\frac{D_{tb(c)}}{t}\right) \times 31$$

$$D_{o(prj)} = \left(\frac{D_{o(c)}}{t}\right) \times 31$$

where:

$D_{tb(c)}$ = the cumulative total-body dose for the elapsed portion of the current quarter plus the release under consideration.

t = the number of days into the current quarter.

$D_{o(c)}$ = the cumulative organ doses for the elapsed portion of the current quarter plus the release under consideration.

If activities planned during the remainder of the quarter are expected to contribute a significant dose and the determination can be reasonably made, this contribution should be included in the equations:

$$D_{tb(prj)} = \left[\left(\frac{D_{tb(c)}}{t}\right) \times 31\right] + D_{PA}$$

$$D_{o(proj)} = \left[\left(\frac{D_{o(c)}}{t}\right) \times 31\right] + D_{PA}$$

where:

D_{PA} = the expected dose due to the particular planned activity.

1.3.2 Dose Projections for Specific Releases

Dose projections may be performed for a particular release by performing a prerelease dose calculation, assuming that the planned release will proceed as anticipated. For individual dose projections due to liquid releases, follow the methodology presented in Section 1.2, using sample analysis values for the source to be released and parametric values expected to exist for the release period.

Term	Definition	Section of Initial Use
C_{il}	= the average concentration of radionuclide i, in undiluted liquid effluent during time period Δt_l, in μCi/ml.	1.2.1
C_{MPC}	= the effluent concentration limit (Section 1.7.2) implementing 10 CFR 20 for the site in μCi/ml.	1.1.1
C_s	= the concentration of Sr-89 or Sr-90 in liquid wastes as determined by analysis of the QUARTERLY composite sample.	1.1.1
C_t	= the measured concentration of H-3 in liquid waste as determined by analysis of the MONTHLY composite.	1.1.1
D_τ	= the cumulative dose commitment to the total body or any organ, τ, from the liquid effluents for the total time period.	1.2.1
DF	= the dilution factor, which is the ratio of the total dilution flow rate to the effluent stream flow rate(s) required to assure that the limiting concentration of 10 CFR, Part 20, Appendix B, Table II, Column 2 are met at the point of discharge to the unrestricted area.	1.1.1
$DF_{i\tau}$	= a dose conversion factor for nuclide i, for adults in preselected organ, τ, in mrem/pCi, from Table 1.2-2.	1.2.1
f	= the flow setpoint as determined for the radiation monitor location. (General expression for Equation 1.)	1.1.1

1.7 LIMITS OF OPERATION

1.7.1 Liquid Effluent Monitoring Instrumentation Control

In accordance with Technical Specification 6.18(1), the radioactive liquid effluent monitoring instrumentation channels shown in Table 1.7-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits specified in Section 1.7.2 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with Section 1.1.

1.7.1.1 Applicability

As shown in Table 1.7-1.

1.7.1.2 Action

1.7.1.2.1 With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above control, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, declare the channel inoperable, or change the setpoint to a conservative value.

1.7.1.2.2 With the number of channels OPERABLE less than the minimum channels required by Table 1.7-1, take the ACTION shown in Table 1.7-1.

1.7.1.2.3 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the unit. Entry into an Operational Mode or other specified condition may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable

1.7.1.3 Surveillance Requirements

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 1.7-2.

1.7.1.4 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 of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methods in Section 1.1 to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation are consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

1.7.2 Liquid Effluent Concentration Control

In accordance with Technical Specifications 6.18(2) and (3), the concentration of radioactive material released at any time from the site to UNRESTRICTED AREAS (Figure 1.7-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentrations shall be limited to 2E-4 μCi/ml total activity.

1.7.2.1 Applicability

At all times.

1.7.2.2 Action

1.7.2.2.1 With the concentration of radioactive material released from the site to UNRESTRICTED AREAS exceeding the above limits, without delay restore concentration within the above limits and provide notification to the Nuclear

Regulatory Commission by including a discussion of the causes and corrective actions taken in the next semiannual radioactive effluent release report, in accordance with Section 6.2.1.

1.7.2.2.2 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the unit. Entry into an Operational Mode or other specified condition may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

1.7.2.3 Surveillance Requirements

1.7.2.3.1 Radioactive liquid effluents shall be sampled and analyzed according to the sampling and analysis program of Table 1.7-3.

1.7.2.3.2 The results of radioactive analyses shall be used in accordance with the methods in Section 1.1 to assure that the concentrations at the point of release are maintained within the limits of Section 1.7.2.

1.7.2.4 Bases

This control 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. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and within the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radionuclide, and its Maximum Permissible Concentration in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

1.7.3 Liquid Effluent Dose Control

In accordance with Technical Specifications 6.18(4) and (5), the dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (Figure 1.7-1) shall be limited to:

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

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

1.7.3.1 Applicability

At all times.

1.7.3.2 Action

1.7.3.2.1 With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the limits specified above in Section 1.7.3, prepare and submit to the Nuclear Regulatory Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases of radioactive materials in liquid effluents to ensure that subsequent releases will be in compliance with the limits specified above in Section 1.7.3. For Unit 2, this report is in lieu of any other report required by Technical Specification 6.9.1. (This report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act.)

1.7.3.2.2 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the unit. Entry into an Operational Mode or other specified condition

may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

1.7.3.3 Surveillance Requirements

Dose Calculations: Cumulative dose contributions from liquid effluents shall be determined at least monthly in accordance with Section 1.2.

1.7.3.4 Bases

This control is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The control implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable" (ALARA). The dose calculations in Section 1.2 implement the requirements in Section III.A of Appendix I, which state that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in Section 1.2 for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with the methodology provided in Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Revision 1, October 1977. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

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

1.7.4 Liquid Radwaste Treatment System Control

In accordance with Technical Specification 6.18(6), the liquid radwaste treatment system, as described in Section 1.5, shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluents, from each unit, to UNRESTRICTED AREAS (Figure 1.7-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ of a MEMBER OF THE PUBLIC in 31 days.

1.7.4.1 Applicability

At all times.

1.7.4.2 Action

1.7.4.2.1 With radioactive liquid waste being discharged without treatment and in excess of the above limits, within 30 days, prepare and submit to the Nuclear Regulatory Commission, pursuant to Technical Specification 6.9.2, a Special Report that includes the following information:

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

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

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

1.7.4.2.2 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the unit. Entry into an Operational Mode or other specified condition may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

1.7.4.3 Surveillance Requirements

Doses due to liquid releases shall be projected at least monthly in accordance with Section 1.3, during periods in which discharge of untreated liquid effluent containing radioactive materials to UNRESTRICTED AREAS occurs or is expected to occur.

1.7.4.4 Bases

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to UNRESTRICTED AREAS. 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 ALARA. This control implements the requirements of 10 CFR Part 50.36(a), General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

TABLE 1.7-1 (SHEET 1 OF 2)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION*

Instrument	Minimum Channels OPERABLE	Applicability	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release			
Liquid Radwaste Effluent Line	1	(a)	100
2. Gross Radioactivity Monitors not Providing Automatic Termination of Release			
Service Water System Effluent Line	1	(b)	101
3. Flowrate Measurement Devices**			
Liquid Radwaste Effluent Line	1	(a)	102
Discharge Canal	1	(a) (b)	102
4. Service Water System to Closed Cooling Water System Differential Pressure	1	At all times	103

*Applies to each unit.

**Pump curves may be utilized to estimate flow; in such cases, ACTION statement 102 is not required.

(a) Whenever the radwaste discharge valves are not locked closed.

(b) Whenever the service water system pressure is below the closed cooling water system pressure or ΔP indication is not available.

TABLE 1.7-1 (SHEET 2 OF 2)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATIONS

ACTION 100 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may be continued, provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Section 1.7.2.3.1.

b. At least two technically qualified individuals independently verify the release rate calculations and discharge valving.

Otherwise, suspend release of radioactive effluents via this pathway. If the channel remains inoperable for over 30 days, an explanation of the circumstances shall be included in the next semiannual radioactive effluent release report.

ACTION 101 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided that once per shift grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a Lower Limit of Detection of at least 1E-7 μCi/ml. If the channel remains inoperable for over 30 days, an explanation of the circumstances shall be included in the next semiannual radioactive effluent release report.

ACTION 102 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided the flowrate is estimated at least once per 4 hours during actual releases. If the channel remains inoperable for over 30 days, an explanation of the circumstances shall be included in the next semiannual radioactive effluent release report.

ACTION 103 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, assure that the service water system effluent monitor is OPERABLE.

TABLE 1.7-2 (SHEET 1 OF 2)

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS*

Instrument	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation				
Liquid Radwaste Effluent Line	D**	P(3)	R	Q(1)
2. Gross Gamma Radioactivity Monitors Providing Alarm but not Providing Automatic Isolation				
Service Water System Effluent Line	D**	M	R	Q(4)
3. Flowrate Measurement Devices				
Liquid Radwaste Effluent Line	D(2)**	NA	R	Q
Discharge Canal	D(2)**	NA	R	Q
4. Service Water System to Closed Cooling Water System Differential Pressure	D	NA	R	NA

TABLE 1.7-2 (SHEET 2 OF 2)

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS*

TABLE NOTATIONS

*Applies to each unit.

**During releases via this pathway.

(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:

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

b. Instrument indicates an isolation on high alarm.

c. Instrument controls are not set in operate mode.

(2) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or batch releases are made.

(3) The SOURCE CHECK prior to release shall consist of verifying that the instrument is reading onscale.

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

a. Instrument indicates measured levels above the alarm setpoint.

b. Instrument indicates a downscale failure.

c. Instrument controls are not set in operate mode.

TABLE 1.7-3 (SHEET 1 OF 2)

RADIOACTIVE LIQUID EFFLUENT SAMPLING AND ANALYSIS PROGRAM*

Liquid Release Type	Sampling Frequency[d]	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection[a] (μCi/ml)
Batch Waste Release Tanks	P Each Batch	P Each Batch	Principal Gamma Emitters[e]	5E-7[b]
			I-131	1E-6
	P One Batch/M	M	Dissolved and Entrained Gases	1E-5
	P Each Batch	M Composite[c]	H-3	1E-5
			Gross Alpha	1E-7
	P Each Batch	Q Composite[c]	Sr-89, Sr-90	5E-8
			Fe-55	2E-6

TABLE 1.7-3 (SHEET 2 OF 2)

RADIOACTIVE LIQUID EFFLUENT SAMPLING AND ANALYSIS PROGRAM*

TABLE NOTATIONS

*Applies to each unit.

a. The Lower Limit of Detection is defined in Table Notation (a) of Table 3.2-3.

b. For certain radionuclides with low gamma yield or low energies or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the Lower Limit of Detection. Under these circumstances, the Lower Limit of Detection may be increased inversely proportional to the magnitude of the gamma yield (i.e., 5E-7/I, where: I = photon abundance expressed as a decimal fraction) but in no case shall the Lower Limit of Detection, as calculated in this manner for a specific radionuclide, be greater than 10 percent of the maximum Permissible Concentration value specified in 10 CFR 20, Appendix B, Table II, Column 2.

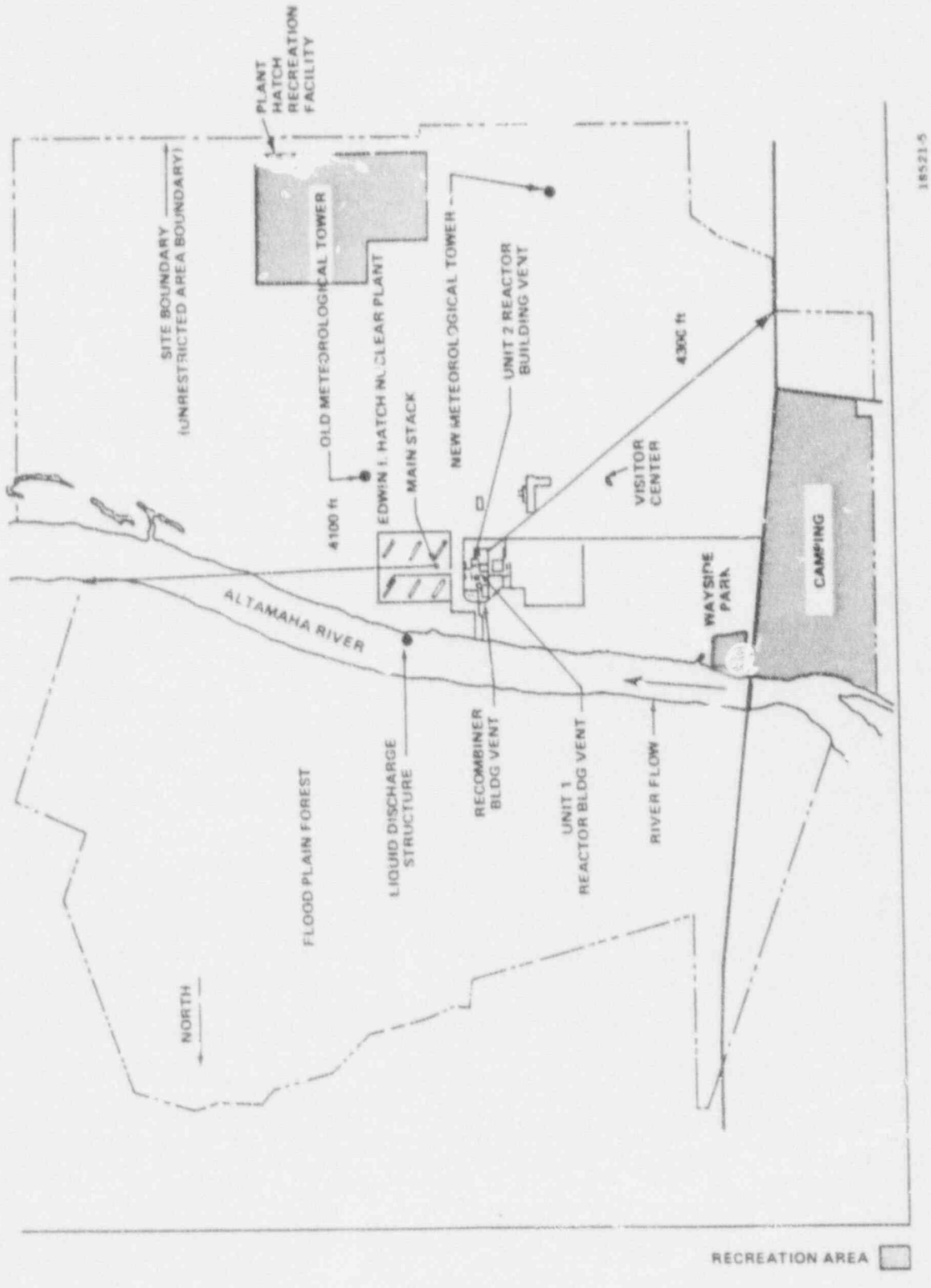
c. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analysis, each batch shall be isolated and then thoroughly mixed, as described in Section 1.6, to assure representative sampling.

e. The principal gamma emitters for which the Lower Limit of Detection requirements will apply are exclusively 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 measurable and identifiable peaks, together with the above nuclides, shall also be identified and reported.

FIGURE 1.7-1

UNRESTRICTED AREA BOUNDARY



2.0 GASEOUS EFFLUENTS

At Plant Hatch there are four points where radioactivity normally is released to the atmosphere in gaseous discharges. These four release points are:

- The main stack which serves both units.
- Unit 1 reactor building vent stack.
- Unit 2 reactor building vent stack.
- Unit 1 recombiner building vent.

In addition to these four release points, releases also may be made from building exhaust augmented ventilation system(s) provided such system(s) has (have) been included in Tables 2.6-1, 2.6-2, and 2.6-3.

The main stack serves as the discharge point for the following release sources from each unit:

- Mechanical vacuum pumps.
- Off-gas system.
- Gland seal exhaust.
- Standby gas treatment system through which drywell purges are discharged.

The waste gas treatment building ventilation also discharges through the main stack.

Each reactor building vent stack serves as the discharge point for the following (of each respective unit):

- Reactor building.
- Refueling floor ventilation.
- Turbine building.
- Radwaste building.

The Unit 1 recombiner building vent discharges directly to the atmosphere.

Gaseous effluent monitor setpoints are required only for noble gas monitors serving the release points; the methodology for calculating noble gas monitor setpoints is presented in Section 2.1. Although setpoint calculations are not required for radioiodine and particulate monitors, the methodology for assuring that the potential organ dose rates due to Iodine-131, Iodine-133, tritium, and particulates with half-lives greater than 8 days in gaseous releases from the site to areas at and beyond the SITE BOUNDARY do not exceed the limits of Section 2.6.2 is presented in the note following Section 2.2.1.b.

2.1 GASEOUS EFFLUENT MONITOR SETPOINTS

The gaseous monitor setpoint values determined in the following sections will be regarded as upper bounds for the actual setpoint adjustments. That is, setpoint adjustments are not required to be performed if the existing setpoint level corresponds to a lower count rate than the calculated value. Setpoints may be established at values lower than the calculated values, if desired.

If no release is planned for a particular pathway, or if there is no detectable activity in the planned release, the monitor setpoint should be established as close to background as practical to prevent spurious alarms and yet alarm should an inadvertent release occur.

If a calculated setpoint is less than the monitor reading associated with the particular release pathway, no release may be made under current conditions. Under such circumstances, the number of simultaneous release pathways may be reduced or contributing source terms may be reduced and the setpoint recalculated.

2.1.1 Unit 1 Reactor Building Vent Stack, Unit 2 Reactor Building Vent Stack, Unit 1 Recombiner Building Vent, and Building Exhaust Augmented Ventilation

Monitors: D11-K619 A and B, 2D11-K636 A and B, D11-P003 A and B
(Monitor identification for building exhaust augmented ventilation will be determined prior to making releases via this pathway.)

For the purpose of implementing Section 2.6.1, the alarm setpoint level for these noble gas monitors will be calculated as follows:

C_s = the monitor reading of the noble gas monitor at the alarm setpoint concentration.

$$(AG \times SF) \times R_{tv} \times D_{TB} \quad (1)$$

C_s = the lesser of or

$$(AG \times SF) \times R_{sv} \times D_{ss} \quad (2)$$

SF = the safety factor; a conservative factor applied to each noble gas monitor to compensate for statistical fluctuations and errors of measurement. (For example, SF = 0.5 corresponds to a 100-percent variation.)

AG = an administrative allocation factor applied to apportion the release setpoints among all gaseous release discharge pathways (normally four) to assure that release limits will not be exceeded by simultaneous releases. The allocation factor for a particular discharge pathway may be assigned any desired value between 0 and 1 under the condition that the sum of the allocation factors for all simultaneous release pathways does not exceed 1. For ease of implementation, AG may be set equal to 1/n, where: n = the number of simultaneous final gaseous release points. For a more exact determination of allocation factors, see Section 2.1.3.

$D_{T.}$ = the dose rate limit to the total body of a MEMBER OF THE PUBLIC, which is 500 mrem/yr.

R_{tv} = the monitor reading per mrem/yr to the total body for vent releases.

$$= C + ((\overline{X/Q})_a \sum_i K_i Q_{iv}). \quad (3)$$

where:

C = the monitor reading of a noble gas monitor corresponding to the grab sample radionuclide concentrations taken in accordance with Table 2.6-3. The monitor response corresponding to the measured concentration is determined from the monitor calibration curve for the particular monitor.

$(\overline{X/Q})_a$ = the highest annual average relative concentration at and beyond the SITE BOUNDARY. (If desired, the annual average relative concentration at and beyond the SITE BOUNDARY for the particular release point may be used.) The release points addressed in this section are ground-level releases.

$(\overline{X/Q})_a$ = 8.37×10^{-6} sec/m^3 in the ENE sector.

K_i = the total-body dose factor due to gamma emissions from radionuclide i (mrem/yr per μCi/m^3) from Table 2.1-1.

Q_{iv} = the rate of release of noble gas radionuclide i (μCi/sec) from the vent release pathway under consideration, which is the product of X_{iv} and F_v, where: X_{iv} = the concentration of radionuclide i for the particular release, and F_v = the maximum expected release flow rate for this release point (X_{iv} in μCi/ml and F_v in ml/sec).

D_{ss} = the dose rate limit to the skin of the body of a MEMBER OF THE PUBLIC in areas at and beyond the SITE BOUNDARY, which is 3000 mrem/year.

R_{sv} = the monitor reading per mrem/yr to the skin.

$$R_{sv} = C \div \left(\left(\overline{X/Q} \right)_o \sum_i \left(L_i + 1.1\, M_i \right) Q_{iv} \right) \quad (4)$$

where:

L_i = the skin dose factor due to beta emissions from radionuclide i (mrem/yr per $\mu Ci/m^3$) from Table 2.1-1.

1.1 = the mrem skin dose per mrad air dose.

M_i = the air dose factor due to gamma emissions from radionuclide i (mrad/yr per $\mu Ci/m^3$) from Table 2.1-1.

2.1.2 Main Stack

Monitor: D11-K600 A and B

For the purpose of implementing Section 2.6.1, the alarm setpoint level for the main stack noble gas monitor will be calculated as follows:

C_s = the monitor reading of the noble gas monitor at the alarm setpoint concentration.

$$(AG \times SF) \times R_{ts} \times D_{TB} \quad (5)$$

= the lesser of or

$$(AG \times SF) \times R_{ss} \times D_{ss} \quad (6)$$

$$R_{ts} = C \div \sum_i V_i\, Q_{is} \quad (7)$$

$$R_{\alpha} = C + \sum_{i} (L_i (\overline{X/Q})_{\varepsilon} + 1.1\ B_i) Q_{i\alpha} \quad (8)$$

where:

V_i = the constant, which includes the dose factor, for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume resulting from the main stack release, in mrem/yr per μCi/sec, from Table 2.1-2.

B_i = the constant, which includes the air dose factor, for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume resulting from the main stack release, in mrad/yr per μCi/sec, from Table 2.1-2.

$Q_{i\alpha}$ = the rate of release of noble gas radionuclide i (μCi/sec) from the main stack, which is equal to the product of $X_{i\alpha}$ and F_{α}, where: $X_{i\alpha}$ = the concentration of radionuclide i for the main stack release, and F_{α} = the maximum expected main stack release flow rate ($X_{i\alpha}$ in μCi/ml and F_{α} in ml/sec).

$(\overline{X/Q})_{\varepsilon}$ = the highest annual average relative concentration in areas at and beyond the SITE BOUNDARY associated with releases from the main stack. The main stack is an elevated release.

= the 4.10×10^{-8} sec/m^3 in the ENE sector.

All other terms were identified previously in Section 2.1.1.

2.1.3 Determination of Allocation Factor, AG

When simultaneous gaseous releases are made to the environment, an (administrative) allocation factor must be applied to each discharge pathway. This is to ensure that simultaneous gaseous releases from the site to areas at and beyond the SITE BOUNDARY will not exceed the dose rate limits specified in Section 2.6.2. For Plant Hatch, final discharge pathways which may be released simultaneously are:

- The main stack.
- Unit 1 reactor building vent stack.
- Unit 2 reactor building vent stack.
- Unit 1 recombiner building vent.
- Building exhaust augmented ventilation.

The allocation factor for each discharge pathway must be between 0 and 1, and the sum of the allocation factors for the simultaneous releases must not exceed 1.

There are three methods by which allocation factors may be determined:

1. The allocation factor for a particular release pathway may be administratively selected based on an estimate of the fraction of the total dose rate (from all simultaneous releases) which is contributed by the particular release pathway.

2. The allocation factor may be calculated using the expression:

   $$AG = 1/n$$

   where: n = the number of release pathways to be released simultaneously.

3. The allocation factor may be determined for a particular discharge pathway by calculating the ratio of the total-body dose rate due to noble gases released from the particular discharge pathway under consideration to the total-body dose rate due to noble gases in all simultaneous releases.

For the main stack:

$$AG = \frac{\sum_i V_i Q_{is}}{\sum_i V_i Q_{is} + \sum_n (\overline{X/Q})_s \sum_i K_i Q_{iv}} \quad (9)$$

where: n = the number of simultaneous releases.

For vent releases:

$$AG = \frac{(\overline{X/Q})_s \sum_i K_i Q_{iv(r)}}{\sum_i V_i Q_{is} + \sum_n (\overline{X/Q})_s \sum_i K_i Q_{iv}} \quad (10)$$

where: n = the number of simultaneous vent releases.

(r) = the particular discharge pathway number for which an allocation factor is being determined.

2.1.4 Unit 1 Condenser Offgas Pretreatment Monitor and Unit 2 Condenser Offgas Pretreatment Monitor

Monitors: 1D11-K601, 2D11-K601, 1D11-K602, and 2D11-K602

For the purpose of implementing Section 2.6.1, the alarm setpoint level for these noble gas monitors will be calculated as follows:

$$C_{ms} = 240{,}000 / (CF_m * F_{cog}) \quad (11)$$

where: 240,000 = the release rate limit for pretreatment condenser offgas specified in Technical Specifications 3.15.2.7 (Unit 1) and 3.11.2.7 (Unit 2) (μCi/sec).

C_{ms} = the monitor reading of the condenser offgas pretreatment monitor at the alarm setpoint (mR/hr).

CF_m = the calibration factor for the condenser offgas pretreatment monitor ((μCi/sec)/((mR/hr)*(cfm))).

F_{cog} = the condenser offgas flow rate (cfm).

2.2 GASEOUS EFFLUENT DOSE RATE AND DOSE CALCULATIONS

2.2.1 SITE BOUNDARY Dose Rates

2.2.1.a Dose Rates Due To Noble Gases

For the purpose of implementing Section 2.6.2.a, the dose rate in areas at and beyond the SITE BOUNDARY due to noble gases shall be calculated as follows:

D_t = the total body dose rate at time of release (mrem/yr).

$$= \left[\sum_v (\overline{X/Q})_a \sum_i K_i Q_{iv}\right] + \left[\sum_v V_i Q_{iv}\right] \quad (9)$$

D_s = the skin dose rate at time of release (mrem/yr).

$$= \left[\sum_v (\overline{X/Q})_a \sum_i (L_i + 1.1 M_i) Q_{iv}\right] + \left[\sum_i (L_i(\overline{X/Q})_\varepsilon + 1.1 B_i) Q_{is}\right] \quad (10)$$

Terms were defined previously in Sections 2.1.1 and 2.1.2.

The dose rate limits are site limits at any point in time; therefore, dose rates are summed over all releases occurring simultaneously. For Plant Hatch the vent releases are:

- Unit 1 reactor building vent stack.
- Unit 2 reactor building vent stack.
- Unit 1 recombiner building vent.
- Building exhaust augmented ventilation.

The only elevated release is the main stack which serves both units. Simultaneous releases may include any combination of these four release points.

2.2.1.b Dose Rates Due to Iodine-131, Iodine-133, Tritium, and Particulates

For the purpose of implementing Section 2.6.2.b, organ dose rates due to Iodine-131, Iodine-133, tritium, and all radioactive materials in particulate form with half-lives greater than 8 days are required to be calculated for the inhalation pathway for the child age group. The child age group would experience the highest potential dose rate via the inhalation pathway. In accordance with Appendix C to Reference 3, noble gases are excluded from these calculations. These dose rates are calculated as follows:

D_o = the organ dose rate at time of release (mrem/yr).

$$= \left[\sum_{v} (\overline{X/Q})_{G} \sum_{i} P_{io} Q'_{iv}\right] + \left[(\overline{X/Q})_{E} \sum_{i} P_{io} Q'_{is}\right] \qquad (11)$$

where:

$(\overline{X/Q})_G$ = defined in Section 2.1.1.

$(\overline{X/Q})_E$ = defined in Section 2.1.2.

Q'_{iv} = the release rate (μCi/sec) of Iodine-131, Iodine-133, tritium, and particulates with half-lives greater than 8 days (required by Section 2.6.2.3.2) from the Unit 1 reactor building vent stack, the Unit 2 reactor building vent stack, the Unit 1 recombiner building vent, and building exhaust augmented ventilation.

Q'_{is} = the release rate (μCi/sec) of Iodine-131, Iodine-133, tritium, and particulates with half-lives greater than 8 days (required by Section 2.6.2.3.2) from the main stack.

P_{io} = the organ dose parameter for organ o and radionuclide i (mrem/yr per $\mu Ci/m^3$) for inhalation determined as follows:

$$P_{io} = K\,(BR)\,DF_{io} \quad (12)$$

and where:

K = the constant of unit conversion, 10^6 pCi/μCi.

BR = the breathing rate for child age group (3700 m^3/year), Table 2.2-10, from Reference 3.

DF_{io} = the inhalation pathway dose factor for child age group for organ o and radionuclide i, Table 2.2-2, from Reference 3.

NOTE: In order to assure that potential dose rates (prerelease) to an organ due to Iodine-131, Iodine-133, tritium, and particulates with half-lives greater than 8 days in simultaneous gaseous releases from the site to areas at and beyond the SITE BOUNDARY do not exceed 1500 mrem/yr as specified in Section 2.6.2.b, the potential organ dose rate D_o must be limited as follows:

$$D_o + (AG \times SF) \leq 1500 \text{ mrem/yr} \quad (13)$$

where: AG and SF are assigned the same values as were used in Section 2.1 for the release source pathway under consideration. To further ensure that dose rate limits were not exceeded, (post-release) dose rates from simultaneous releases shall be summed, as shown above.

2.2.2 Air Dose and Dose to a MEMBER OF THE PUBLIC in Areas at and Beyond the SITE BOUNDARY

2.2.2.a Air Dose in Areas at and Beyond the SITE BOUNDARY

For the purpose of implementing Section 2.6.3, the air dose in areas at and beyond the SITE BOUNDARY shall be determined as follows:

D_γ = the air dose due to gamma emissions from noble gas radionuclides (mrad).

$$= 3.17 \times 10^{-8} \left[\left[\sum_v (\overline{X/Q})_v \sum_i M_i \tilde{Q}_{iv} \right] + \left[\sum_i B_i \tilde{Q}_{is} \right] \right]$$

where:

3.17×10^{-8} = the fraction of 1 yr per 1 sec.

$\tilde{Q}_{iv}$ = the cumulative release of noble gas radionuclide i over the period of interest (μCi) from the vent release under consideration.

$\tilde{Q}_{is}$ = the cumulative release of noble gas radionuclide i over the period of interest (μCi) from the main stack.

M_i = defined previously in Section 2.1.1.

B_i = defined previously in Section 2.1.2.

$(\overline{X/Q})_v$ = defined previously in Section 2.1.1.

D_β = the air dose due to beta emissions from noble gas radionuclides (mrad).

$$= 3.17 \times 10^{-6} \left[\left[\sum_v (\overline{X/Q})_a \sum_i N_i \tilde{Q}_{iv} \right] + \left[(\overline{X/Q})_\varepsilon \sum_i N_i \tilde{Q}_{i\varepsilon} \right] \right]$$

where:

N_i = the air dose factor due to beta emissions from noble gas radionuclide i (mrad/yr per $\mu Ci/m^3$) from Table 2.1-1.

$(\overline{X/Q})_\varepsilon$ = defined previously in Section 2.1.2.

2.2.2.b Dose to a MEMBER OF THE PUBLIC in Areas at and beyond the SITE BOUNDARY

Dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and radioactive material in particulate form with half-lives greater than 8 days will be calculated for the purpose of implementing Section 2.6.4. In accordance with Appendix C of Reference 3, noble gases are excluded from these dose calculations. Doses to a MEMBER OF THE PUBLIC are calculated as follows:

NOTE: For Plant Hatch, the controlling receptor for which doses must be calculated, the dispersion and deposition values at the location of the controlling receptor, and the applicable exposure pathways are presented in Table 2.2-12.

D_j = dose to an organ j of a MEMBER OF THE PUBLIC in age group a from Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days (mrem).

$$= 3.17 \times 10^{-6} \sum_{pi} R_{aipj} \left[W'_{vp} \tilde{Q}'_{iv} + W'_{sp} \tilde{Q}'_{is} \right] \qquad (16)$$

where:

3.17×10^{-8} = the fraction of 1 yr per 1 sec.

W'_{vp} = the pathway-dependent relative dispersion or deposition in the areas at and beyond the SITE BOUNDARY at the location of the controlling receptor, associated with plant vent releases.

W'_{vp} =

- $(\overline{X/Q'})_{vp}$ = the annual average relative dispersion parameter for location of controlling (critical) receptor for plant vent releases. $(\overline{X/Q'})_{vp}$ applies only to inhalation and all tritium pathways. (For all tritium pathways, the $\tilde{Q}'_i$ source term is limited to tritium.) See Table 2.2-12 for value.

- $(\overline{D/Q'})_{vp}$ = the annual average deposition parameter for the location of controlling (critical) receptor for plant vent releases. $(\overline{D/Q'})_{vp}$ applies to ground-plane and all ingestion pathways, with the exception of tritium. See Table 2.2-12 for value.

W'_{sp} = the pathway-dependent relative dispersion or deposition in the area at and beyond the SITE BOUNDARY at the location of the controlling receptor, associated with main stack releases.

W'_{sp} =

- $(\overline{X/Q'})_{sp}$ = the annual average relative dispersion parameter for location of controlling (critical) receptor for main stack releases. $(\overline{X/Q'})_{sp}$ applies only to inhalation and all tritium pathways. (For all tritium pathways, the $\tilde{Q}'_i$ source term is limited to tritium.) See Table 2.2-12 for value.

- $(\overline{D/Q'})_{sp}$ = the annual average deposition parameter for the location of controlling (critical) receptor for main stack releases. $(\overline{D/Q'})_{sp}$ applies to ground-plane and all ingestion pathways, with the exception of tritium. See Table 2.2-12 for value.

The selection of the dispersion or deposition parameter, X/Q or D/Q, is dependent upon the pathway being considered. The dispersion parameter, X/Q, is required for the inhalation pathway. The deposition parameter, D/Q, is required for the ground-plane pathway and all ingestion pathways, with the exception of tritium. Since tritium is taken up by vegetation directly from surrounding air, X/Q is required for tritium contributions from the ingestion pathways.

Q'_{iv} = the cumulative release (μCi), from plant vent releases, of radionuclide i as required by Section 2.6.4.3 over the period of interest. Dose determinations required by Section 2.6.4.3 are on a per reactor basis; therefore, cumulative release quantities must also be reactor-specific. (For dose contribution due to tritium from the ingestion pathways, the $\tilde{Q}'_{iv}$ term is limited to tritium.)

Q'_{is} = the cumulative release (μCi), from the main stack releases, of radionuclide i as required by Section 2.6.4.3 over the period of interest. Dose determinations required by Section 2.6.4.3 are on a per reactor basis; therefore, cumulative release quantities must also be reactor-specific. Since the main stack serves both reactors, release quantities must be apportioned between the two units. In absence of evidence that one reactor contributes a greater quantity of radioactivity than the other over the period of interest, release quantities may be apportioned equally between the two units. (For dose contributions due to tritium from ingestion pathways, the $\tilde{Q}'_{is}$ term is limited to tritium.)

R_{aipj} = the pathway-specific, individual age-specific organ dose factor for radionuclide i, pathway p, organ j, and individual age group a. Routine dose calculations for a MEMBER OF THE PUBLIC address the inhalation, ground-plane, grass-cow (or goat)-milk, grass-cow-meat and garden vegetation pathways. However, the dose pathways actually present at the controlling location, as well as the controlling age group for a MEMBER OF THE PUBLIC are determined through the Land Use Survey for the site. Pathway factors R_{aipj} are determined as shown in the following subsections.

$$R_{a(pj)} = K'K''''F_m Q_F U_{ap} (DFL_{ij})_a [0.75(0.5/H)] \text{ (mrem/yr per } \mu Ci/m^3) \quad (26)$$

where:

K'''' = a constant of unit conversion, 10^3 gm/kg.

H = absolute humidity of the atmosphere, in gm/m^3.

0.75 = the fraction of total feed that is water.

0.5 = the ratio of the specific activity of the feed grass water to the atmospheric water.

Other parameters and values are given above.

2.2.2.c Dose Calculations To Support Other Specific Technical Specifications

In the event radiological impact assessment becomes necessary to implement RETS 6.9.1.12 or 6.9.1.13, dose calculations will be performed using the equations in Section 2.2.2.b, with the substitution of average meteorological parameters for the period of the report and the appropriate pathway receptor dose factors ($R_{a(pj)}$).

For the purpose of implement Section 3.1.2, dose calculations may be performed using the equations in Section 2.2.2.b, substituting the appropriate pathway receptor dose factor ($R_{a(pj)}$) and the appropriate dispersion parameters for the location(s) of interest. Annual average dispersion parameters may be used for these calculations.

The receptor for which dose calculations may be required in order to support Technical Specifications 6.6, 6.9.1, 6.9.2, or Section 3.1.2 may not be the previously identified critical receptor. The receptor age group and exposure pathways present (and applicable) at the location of interest must be determined. The equations for calculating the pathway factors $R_{a(pj)}$ were presented in Section 2.2.2.b. Plant Hatch site-specific values, or appropriate default values, required in the pathway factor determinations are presented in Table 2.2-13.

TABLE 2.2-11

STABLE ELEMENT TRANSFER DATA*

Element	F_m - Milk (Cow)	F_m - Milk (Goat)	F_f - Meat
H	1.0E-02	1.7E-01	1.2E-02
C	1.2E-02	1.0E-01	3.1E-02
Na	4.0E-02	4.0E-02	3.0E-02
P	2.5E-02	2.5E-01	4.6E-02
Cr	2.2E-03	2.2E-03	2.4E-03
Mn	2.5E-04	2.5E-04	8.0E-04
Fe	1.2E-03	1.3E-04	4.0E-02
Co	1.0E-03	1.0E-03	1.3E-02
Ni	6.7E-03	6.7E-03	5.3E-02
Cu	1.4E-02	1.3E-02	8.0E-03
Zn	3.9E-02	3.9E-02	3.0E-02
Rb	3.0E-02	3.0E-02	3.1E-02
Sr	8.0E-04	1.4E-02	6.0E-04
Y	1.0E-05	1.0E-05	4.6E-03
Zr	5.0E-06	5.0E-06	3.4E-02
Nb	2.5E-03	2.5E-03	2.8E-01
Mo	7.5E-03	7.5E-03	8.0E-03
Tc	2.5E-02	2.5E-02	4.0E-01
Ru	1.0E-06	1.0E-06	4.0E-01
Rh	1.0E-02	1.0E-02	1.5E-03
Ag	5.0E-02	5.0E-02	1.7E-02
Te	1.0E-03	1.0E-03	7.7E-02
I	6.0E-03	6.0E-02	2.9E-03
Cs	1.2E-02	3.0E-01	4.0E-03
Ba	4.0E-04	4.0E-04	3.2E-03
La	5.0E-06	5.0E-06	2.0E-04
Ce	1.0E-04	1.0E-04	1.2E-03
Pr	5.0E-06	5.0E-06	4.7E-03
Nd	5.0E-06	5.0E-06	3.3E-03
W	5.0E-04	5.0E-04	1.3E-03
Np	5.0E-06	5.0E-06	2.0E-04

F_m units: days/liter

F_f units: days/kg

*Reference 3, Table E-1.

2.3.1.b Elevated Releases

X/Q = the sector-averaged annual average relative concentration at any distance in the given sector for radionuclides other than noble gases.

$$= 2.032\, K_r \sum_{jk} \delta_k \frac{n_{jk} \exp(-h^2/2\delta_{zk}^2)}{N U_{jr} \delta_{zk}}$$

where:

δ_k = the plume depletion factor taken from Figure 2.3-4. For an elevated release, this factor is stability dependent.

h = effective release height, in meters = $h_s - h_t$.

NOTE: Effective release height may be further adjusted for plume rise in accordance with section E.4.3.2 of Appendix E of Reference 5.

h_s = the height of the main stack (120 m).

h_t = terrain height at location of interest, in meters (obtained from Figure 2.3-12 of Reference 6).

n = the number of hours the wind of wind speed class j is directed into the given sector during the time atmospheric stability category k existed. These values may be obtained from Table 2 of Reference 23.

U_{jk} = the wind speed (mid-point of wind speed class j) at the height of release h (m sec^{-1}) during atmospheric stability k.

N = the total hours of valid meteorological data recorded for all sectors, wind speed classes, and stability categories from Table E.4-7 of Reference 5.

The remaining symbols are the same as those previously defined.

When considering the direct gamma radiation from an elevated finite plume, the constants B_i and V_i defined in Section 2.1.2 for each identified noble gas radionuclide are calculated using the following:

$$B_i = \frac{K}{i} \sum_j \sum_k \sum_\ell \left[\frac{n_{jk} A_{fi} \mu_a E_\ell I_{(r)k\ell}}{N U_j} \right] \frac{\text{mrad/yr}}{\mu\text{Ci/sec}}$$

Term		Definition	Section of Initial Use
D_j	=	the dose to an organ of a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days (mrem).	2.2.2.b
D_o	=	the organ dose rate at time of release (mrem/yr).	2.2.1.b
D_s	=	the skin dose rate at time of release (mrem/yr).	2.2.1.a
D_{ss}	=	the limiting dose rate to the skin of the body of a MEMBER OF THE PUBLIC in areas at and beyond the SITE BOUNDARY, which is 3000 mrem/yr.	2.1.1
D_t	=	the total body dose rate at time of release (mrem/yr).	2.2.1.a
D_{TB}	=	the limiting dose rate to the total body of a MEMBER OF THE PUBLIC in areas at and beyond the SITE BOUNDARY, which is 500 mrem/yr.	2.1.1
D_β	=	the air dose due to beta emissions from noble gases (mrad).	2.2.2.a
D_γ	=	the air dose due to gamma emissions from noble gases (mrad).	2.2.2.a
D/Q	=	the sector-averaged relative deposition for any distance in a given sector.	2.3.2
$(\overline{D/Q}')_{vp}$	=	the annual average deposition parameter for the location of controlling (critical) receptor for plant vent releases.	2.2.2.b

Term		Definition	Section of Initial Use
$Q_{i,v}$	=	the rate of release of noble gas radionuclide i (μCi/sec) from the vent release pathway under consideration.	2.1.1
$Q_{i,s}$	=	the rate of release of noble gas radionuclide i (μCi/sec) from the main stack.	2.1.2
$\tilde{Q}_{i,v}$	=	the cumulative release of noble gas radionuclide i over the period of interest (μCi) from the vent release under consideration.	2.2.2.a
$\tilde{Q}_{i,s}$	=	the cumulative release of noble gas radionuclide i over the period of interest (μCi) from the main stack.	2.2.2.a
$\tilde{Q}'_{i,v}$	=	the cumulative release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days from plant vent releases over the period of interest (μCi).	2.2.2.b
$\tilde{Q}'_{i,s}$	=	the cumulative release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days from the main stack over the period of interest (μCi).	2.2.2.b
r	=	the distance from the point of release to the receptor of interest for dispersion calculations (meters).	2.3.1
$R_{a,i,p,j}$	=	the pathway-specific, individual age-specific organ dose factor for radionuclide i, pathway p, organ j, and age group a (mrem/yr per μCi/m^3) or (m^2-mrem/yr per μCi/sec).	2.2.2.b

Term		Definition	Section of Initial Use
μ	=	the total absorbtion coefficient for air (m^{-1}).	2.3.1
μ_e	=	the energy absorbtion coefficient for air (m^{-1}).	2.3.1
μ_T	=	the tissue energy absorbtion coefficient for photons of energy E_ℓ ($cm^3\ gm^{-1}$).	2.3.1
u_j	=	the wind speed (midpoint of windspeed class j) at the height of release, h.	2.3.1
u_{jk}	=	the wind speed (midpoint of windspeed class j) at ground level (m/sec) during atmospheric stability class k.	2.3.1
U_{jk}	=	the wind speed (midpoint of windspeed clas j) at the height of release, h, of an elevated release during atmospheric stability class k.	2.3.1
V_ℓ	=	a constant, which includes the dose factor, for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume resulting from the main stack release (mrem/yr per μCi/sec) from Table 2.1-2.	2.1.2
W'_{vp}	=	the pathway-dependent relative dispersion or deposition in areas at and beyond the SITE BOUNDARY at the location of the controlling receptor associated with plant vent releases.	2.2.2.b

Term		Definition	Section of Initial Use
W'_{sp}	=	the pathway-dependent relative dispersion or deposition in areas at and beyond the SITE BOUNDARY at the location of the controlling receptor associated with stack releases.	2.2.2.b
X/Q	=	the sector-averaged annual average relative concentration at any distance in the given sector.	2.3.1
$\overline{X/Q}_G$	=	the highest annual average relative concentration in areas at and beyond the SITE BOUNDARY associated with ground-level releases.	2.1.1
$\overline{X/Q}_E$	=	the highest annual average relative concentration in areas at and beyond the SITE BOUNDARY associated with releases from the main stack.	2.1.1
$(\overline{X/Q'})_{vp}$	=	the annual average relative dispersion parameter for the location of the controlling receptor for plant vent releases.	2.2.2.b
$(\overline{X/Q'})_{sp}$	=	the annual average relative dispersion parameter for the location of the controlling receptor for main stack releases.	2.2.2.b

Term		Definition	Section of Initial Use
z	=	the vertical distance from a ground-level receptor to the volume element considered as a point souce in the evaluation of $I_{(r)kl}$.	2.3.1
C_{ms}	=	the monitor reading of the condenser offgas pretreatment monitor at the alarm setpoint (mR/hr).	2.1.4
CF_m	=	the calibration factor for the condenser offgas pretreatment monitor ((μCi/sec)/((mR/hr)*(cfm))).	2.1.4
F_{cog}	=	the condenser offgas flow rate (cfm).	2.1.4

2.5 GASEOUS RADWASTE TREATMENT SYSTEM

Figure 2.5-1 is a schematic of the condenser offgas treatment system showing the release points to areas at and beyond the SITE BOUNDARY. This schematic is representative of Unit 1 and Unit 2.

2.6 LIMITS OF OPERATION

2.6.1 Gaseous Effluent Monitoring Instrumentation Control

In accordance with Technical Specification 6.18(1), the radioactive gaseous effluent monitoring instrumentation channels shown in Table 2.6-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Section 2.6.2 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with Section 2.1.

2.6.1.1 Applicability

As shown in Table 2.6-1.

2.6.1.2 Action

2.6.1.2.1 With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value that will ensure that the limits of Section 2.6.2 are met, without delay restore the setpoint to a value that will ensure that the limits of Section 2.6.2 are met or declare the channel inoperable.

2.6.1.2.2 With the number of channels OPERABLE less than the minimum channels required by Table 2.6-1, take the ACTION shown in Table 2.6-1.

2.6.1.2.3 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the unit. Entry into an Operational Mode or other specified condition may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

2.6.1.3. Surveillance Requirements

Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 2.6-2.

2.6.1.4 Bases

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methods in Section 2.1 to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation are consistent with the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

2.6.2 Gaseous Effluent Dose Rate Control

In accordance with Technical Specifications 6.18(3) and (7), the dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (Figure 1.7-1) shall be limited to the following:

a. The dose rate limit for noble gases shall be less than or equal to 500 mrem/year to the total body and less than or equal to 3000 mrem/year to the skin.

b. The dose rate limit for I-131, I-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days shall be less than or equal to 1500 mrem/year to any organ.

2.6.2.1 Applicability

At all times.

2.6.2.2 Action

With the dose rate(s) exceeding the above limits, without delay decrease the release rate to comply with the limit(s) stated in Section 2.6.2.

2.6.2.3 Surveillance Requirements

2.6.2.3.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with methods and procedures described in Sections 2.1 and 2.2.1.a.

2.6.2.3.2 The dose rate due to radioactive materials other than noble gases in gaseous effluents shall be determined to be within limits stated in Section 2.6.2 in accordance with the methods and procedures described in Section 2.2.1.b by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 2.6-3.

2.6.2.4 Bases

This control is provided to ensure that at all times the dose rate at and beyond the SITE BOUNDARY from gaseous effluents from all onsite units 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 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20

(10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrem/year.

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

2.6.3 Gaseous Effluent Air Dose Control

In accordance with Technical Specifications 6.18(5) and (8), the air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (Figure 1.7-1) shall be limited to the following:

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

b. During any calendar year, to less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

2.6.3.1 Applicability

At all times.

2.6.3.2 Action

2.6.3.2.1 With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Nuclear Regulatory Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report identifying the cause(s) for exceeding the limit(s) and defining the corrective actions taken to reduce the releases and proposed corrective actions to be taken to assure that subsequent releases will be in compliance with Section 2.6.3.

2.6.3.2.2 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the unit. Entry into an Operational Mode or other specified condition may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

2.6.3.3 Surveillance Requirements

Dose Calculations: Cumulative air dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined at least monthly in accordance with Section 2.2.2.a.

2.6.3.4 Bases

This control is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The control implements the guides set forth in Sectin II.B of Appendix I; the ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I, assuring that the releases of

radioactive material in gaseous effluents will be kept ALARA. The Surveillance Requirements implement the requirements in Section III.A of Appendix I, which state 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 a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in Section 2.2.2.a for calculating the doses due to the actual releases of radioactive noble gases in gaseous effluents will be consistent with the methodology provided in Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Revision 1, October 1977; and Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, Revision 1, July 1977. The equations in Section 2.2.2.a provided for determining the air doses at and beyond the SITE BOUNDARY will be based upon historical annual average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

2.6.4 Gaseous Effluent Dose to a MEMBER OF THE PUBLIC Control

In accordance with Technical Specifications 6.18(v) and (ix), the dose to any organ of a MEMBER OF THE PUBLIC from I-131, I-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (Figure 1.7-1) shall be limited to the following:

a. During any calendar quarter to less than or equal to 7.5 mrem to any organ.

b. During any calendar year to less than or equal to 15 mrem to any organ.

2.6.4.1 Applicability

At all times.

2.6.4.2 Action

2.6.4.2.1 With the calculated dose from the release of I-131, I-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Nuclear Regulatory Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report identifying the cause(s) for exceeding the limits and defining the corrective actions taken to reduce releases and proposed corrective actions to be taken to assure that subsequent releases will be in compliance with Section 2.6.4.

2.6.4.2.2 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the unit. Entry into an Operational Mode or other specified condition may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

2.6.4.3 Surveillance Requirements

Cumulative organ dose contributions to a MEMBER OF THE PUBLIC from I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at and beyond the SITE BOUNDARY, from each unit, for the current calendar quarter and the current calendar year shall be determined at least monthly in accordance with Section 2.2.2.b.

2.6.4.4 Bases

This control is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I, 10 CFR Part 50. The statements in Section 2.6.4.2 provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I, assuring that the releases of radioactive materials in gaseous effluents will be kept ALARA. The calculational methods specified in Section 2.6.4.3 implement the requirements of Section III.A of Appendix I, which state that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The calculational methods approved by the NRC for calculating the doses due to the actual releases of the subject materials are required to be consistent with Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Revision 1, October 1977; and Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, Revision 1, July 1977. These equations also provide for determining the actual doses based upon historical annual average atmospheric conditions. The release requirements for the radionuclides stated in the control are dependent upon the existing radionuclide pathways to man in areas at and beyond the SITE BOUNDARY. The pathways examined in the development of these calculations are:

a. Individual inhalation of airborne radionuclides.

b. Deposition of radionuclides onto green, leafy vegetation with subsequent consumption by man.

c. Deposition onto grassy areas where MILK ANIMALS and meat-producing animals graze with consumption of the milk and meat by man.

d. Deposition on the ground with subsequent exposure of man.

2.6.5 GASEOUS RADWASTE TREATMENT SYSTEM Control

In accordance with Technical Specification 6.18(6), the GASEOUS RADWASTE TREATMENT SYSTEM as described in Section 2.5 shall be in operation.

2.6.5.1 Applicability

Whenever the main condenser air ejector system is in operation.

2.6.5.2 Action

2.6.5.2.1 With the GASEOUS RADWASTE TREATMENT SYSTEM inoperable for more than 7 days, prepare and submit to the Nuclear Regulatory Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report which includes the following information:

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

b. Action(s) taken to restore the inoperable equipment to OPERABLE status.

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

2.6.5.2.2 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the unit. Entry into an Operational Mode or other specified condition may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

2.6.5.3 Surveillance Requirements

GASEOUS RADWASTE TREATMENT SYSTEM operability shall be demonstrated by administrative controls which assure that the offgas treatment system is not bypassed.

2.6.5.4 Bases

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides reasonable assurance that the releases of radicactive materials in gaseous effluents will be kept ALARA. This control implements the requirements of 10 CFR Part 50.36(a), General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the system were specified as a suitable fraction of the guides set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

TABLE 2.6-1 (SHEET 1 OF 4)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels OPERABLE	Applicability	Parameter	Action
1. Reactor Building Vent Stack Monitoring System (Each Unit)				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	105
b. Iodine Sampler Cartridge	1	*	Verify presence of Cartridge	107
c. Particulate Sampler Filter	1	*	Verify presence of Filter	107
d. Effluent System Flowrate Measurement Device	1	*	System Flowrate Measurement	104
e. Sampler Flowrate Measurement Device	1	*	Sampler Flowrate Measurement	104
2. Recombiner Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	105
b. Iodine Sampler Cartridge	1	*	Verify Presence of of Cartridge	107
c. Particulate Sampler Filter	1	*	Verify Presence of Filter	107
d. Sampler Flowrate Measurement Device	1	*	Sampler Flowrate Measurement	104

TABLE 2.6-1 (SHEET 2 OF 4)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels OPERABLE	Applicability	Parameter	Action
3. Main Stack Monitoring System				
a. Noble Gas Activity Monitor	1	*	Radioactivity Rate Measurement	105
b. Iodine Sampler Cartridge	1	*	Verify Presence of Cartridge	107
c. Particulate Sampler Filter	1	*	Verify Presence of Filter	107
d. Effluent System Flowrate Measurement Device	1	*	System Flowrate Measurement	104
e. Sampler Flowrate Measurement Device	1	*	Sampler Flowrate Measurement	104
4. Condenser Offgas Pretreatment Monitor (Each Unit)				
Noble Gas Activity Monitor	1	**	Radioactivity Rate Measurement	108

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATIONS

+Monitor must be capable of responding to a Lower Limit of Detection of 1E-4 μCi/ml.

*During releases via this pathway.

**During operation of the main condenser air ejector.

ACTION 104 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided the flowrate is estimated at least once per 4 hours.

If the number of channels OPERABLE remains less than required by the Minimum Channels OPERABLE requirement for over 30 days, an explanation of the circumstances shall be included in the next Semiannual Radioactive Effluent Release Report.

ACTION 105 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are taken daily and analyzed daily for gross activity within 24 hours. With the number of main stack monitoring system channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, without delay suspend drywell purge.

If the number of channels OPERABLE remains less than required by the Minimum Channels OPERABLE requirement for over 30 days, an explanation of the circumstances shall be included in the next Semiannual Radioactive Effluent Release Report.

TABLE 2.6-1 (SHEET 4 OF 4)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATIONS

ACTION 107 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided samples are continuously collected with auxiliary sampling equipment for periods on the order of 7 days and analyzed within 48 hours after the end of the sampling period.

If the number of channels OPERABLE remains less than required by the Minimum Channels OPERABLE requirement for over 30 days, an explanation of the circumstances shall be included in the next Semiannual Radioactive Effluent Release Report.

ACTION 108 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, release to the environment may continue for up to 72 hours provided:

a. The offgas system is not bypassed, and

b. The offgas post-treatment monitor (D11-K615) or the main stack monitor (D11-K600) is OPERABLE.

Otherwise, be in at least HOT STANDBY within 12 hours.

If the number of channels OPERABLE remains less than required by the Minimum Channels OPERABLE requirement for over 30 days, an explanation of the circumstances shall be included in the next Semiannual Radioactive Effluent Release Report.

TABLE 2.6-? (SHEET 1 OF 3)

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Reactor Building Vent Stack Monitoring System (Each Unit)				
a. Noble Gas Activity Monitor	D*	M	R	Q(1)
b. Iodine Sampler Cartridge	W*(3)	NA	NA	NA
c. Particulate Sampler Filter	W*(3)	NA	NA	NA
d. Effluent System Flowrate Measuring Device	D*	NA	R	Q
e. Sampler Flowrate Measuring Device	D*	NA	R	Q
2. Recombiner Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	D*	M	R	Q(1)
b. Iodine Sampler Cartridge	W*(3)	NA	NA	NA
c. Particulate Sampler Filter	W*(3)	NA	NA	NA
d. Sampler Flowrate Measuring Device	D*	NA	R	Q

TABLE 2.6-2 (SHEET 2 OF 3)

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
3. Main Stack Monitoring System				
a. Noble Gas Activity Monitor	D*	M	R	Q(1)
b. Iodine Sampler	W*(3)	NA	NA	NA
c. Particulate Sampler	W*(3)	NA	NA	NA
d. Flowrate Monitor	D*	NA	R	Q
e. Sampler Flowrate Monitor	D*	NA	R	Q
4. Condenser Offgas Pretreatment Monitor (Each Unit)				
Noble Gas Activity Monitor	D**	M	R	Q(1)

TABLE 2.6-2 (SHEET 3 OF 3)

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

*During releases via this pathway.

**During operation of the main condenser air ejector.

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

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

b. Circuit failure occurs.

c. Instrument indicates a downscale failure.

(3) The CHANNEL CHECK shall consist of verifying the presence of a filter element and sampler flow at the weekly filter changeout.

TABLE 2.6-3 (SHEET 1 OF 3)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection[a] (μCi/ml)
A. Environmental Release Points	M[c] Grab Sample	M[c]	Principal Gamma Emitters[f]	1E-4[b]
1. Main Stack			H-3	1E-6
2. Reactor Building Vent (Each Unit)				
3. Recombiner Building Vent				
B. All Release Types (as listed in A above)	Continuous[e]	W[d] Charcoal Sample	I-131	1E-12
			I-133	1E-10
	Continuous[e]	W[d] Particulate Sample	Principal Gamma Emitters[f] (I-131, Others)	1E-11
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1E-11
	Continuous[e]	Q Composite Particulate Sample	Sr-89, Sr-90	1E-11

TABLE 2.6-3 (SHEET 2 OF 3)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATIONS

a. Lower Limit of Detection is defined in Table Notation (a) of Table 3.2-3.

b. For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the Lower Limit of Detection. Under these circumstances, the Lower Limit of Detection may be increased inversely proportional to the magnitude of the gamma yield (i.e., 1E-4/I, where I = photon abundance expressed as a decimal fraction), but in no case shall the Lower Limit of Detection, as calculated in this manner for a specified radionuclide, be greater than 10 percent of the Maximum Permissible Concentration value specified in 10 CFR 20, Appendix B, Table II, Column 1.

c. Sampling and analyses for principal gamma emitters shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a 1-hour period if analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant and the Main Stack Noble Gas Activity Monitor reading have increased by a factor of 3.

d. Sampling shall be performed weekly, and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed once per 24 hours for 7 days following each shutdown, startup, or THERMAL POWER change exceeding 15 percent RATED THERMAL POWER in 1 hour and analyses completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding Lower Limits of Detection may be increased by a factor of 10. The more frequent sampling and analysis requirement applies only if analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant and the Main Stack Noble Gas Activity Monitor reading have increased by a factor of 3.

TABLE 2.6-3 (SHEET 3 OF 3)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATIONS

e. The ratio of the sample flowrate to the sampled stream flowrate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Sections 2.6.2, 2.6.3, and 2.6.4.

f. The principal gamma emitters for which the Lower Limit of Detection requirement will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, Xe-138 for noble gas releases; and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate releases. This list does not mean that only these nuclides are to be detected and reported. Other measurable and identifiable peaks, together with the above nuclides, shall also be identified and reported. Nuclides below the Lower Limit of Detection for the analyses should not be reported as being present at the Lower Limit of Detection level for that nuclide. When unusual circumstances result in a Lower Limit of Detection higher than required, the reasons shall be documented in the Semiannual Radioactive Effluent Release Report.

3.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Sampling locations, as required by Section 3.1.1, are described in Table 3.0-1 and shown on maps in Figures 3.0-1 through 3.0-3.

There are no known drinking water users downstream of Plant Hatch. Therefore, the LLD for I-131 in water need not be as stringent as that for milk.

The census of MILK ANIMALS is based on the requirement in Appendix I to 10 CFR Part 50 that the licensee "Identify changes in the use of UNRESTRICTED AREAS (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure." The consumption of milk from animals grazing on contaminated pasture and the consumption of vegetation contaminated by airborne radioiodine are major potential sources of exposure. Samples from MILK ANIMALS are considered a better indicator of radioiodine in the environment than vegetation. Because sufficient milk samples frequently are not available within 5 miles, vegetation samples will be collected also.

Grass is available almost year-round, whereas leafy vegetation is available only for 8 months of the year at best. The sampling stations for grass are located near the SITE BOUNDARY in two sectors with high offsite D/Q values where it might be practical to establish a vegetation plot. The highest offsite D/Q for each individual sector occurs approximately at the SITE BOUNDARY.

Although either fish or clam samples may be collected from the river, fish samples are preferred, because the maximum dose commitment to a MEMBER OF THE PUBLIC as a result of liquid effluents is through the fish consumption pathway.

Sediment will be collected annually, because shoreline recreational areas are under water and, therefore, not in use approximately half of the year.

Allowing deviations from the sampling schedule is based on the recognition of unavoidable practical difficulties which, in the absence of the allowed deviations, would result in violation of Technical Specification 6.19(1).

3.1 LIMITS OF OPERATION

3.1.1 Radiological Environmental Monitoring Program

In accordance with Technical Specification 6.19(1), the Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.1-1.

3.1.1.1 Applicability

At all times.

3.1.1.2 Action

3.1.1.2.1 Any deviations in conducting the Radiological Environmental Monitoring Program[a] from that as specified in Table 3.1-1 shall be documented in the Annual Radiological Environmental Surveillance Report; the reasons for these deviations and any appropriate plans for preventing a recurrence shall be stated. (Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, unavailability, inclement weather, malfunction of equipment, or other just reasons. If deviations are due to equipment malfunction, strenuous efforts shall be made to complete corrective action prior to the end of the next sampling period.)

3.1.1.2.2 With the confirmed[b], measured level of radioactivity as a result of plant effluents in an environmental sampling medium as specified in Table 3.1-1 exceeding the reporting levels of Table 3.1-2 when averaged over any calendar quarter, submit to the Nuclear Regulatory Commission within 30 days or after confirmation, whichever is later, pursuant to Technical Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the

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[a] The requirements for radiological environmental monitoring are the same for both units at the site. Thus, a single program including monitoring, land use census, and quality assurance serves both units.

[b] Defined as a confirmatory reanalysis of the original, a duplicate, or a new sample, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis.

3.1.1.2.2 (Continued)

limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose[a] to a MEMBER OF THE PUBLIC is less than the calendar year limits specified in Sections 1.7.3, 2.6.3, and 2.6.4. When more than one of the radionuclides in Table 3.1-2 are detected in the sampling medium, this report shall be submitted if:

[concentration(1)/limit level(1)] + [concentration(2)/limit level(2)] +...>1.0

When radionuclides other than those in Table 3.1-2 are detected and are the result of plant effluents, this report shall be submitted if the calculated annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the annual limits stated in Sections 1.7.3, 2.6.3, and 2.6.4. 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 described in the Annual Radiological Environmental S rveillance Report. The levels of naturally occurring radionuclides need not be reported.

3.1.1.2.3 If adequate samples of milk, grass or leafy vegetation (during the growing season) from any of the sample locations required by Table 3.1-1 can no longer be obtained or the availability is frequently or persistently wanting, efforts shall be made to find replacement locations. The cause of the unavailability and identification of the affected locations and the locations (if any) for obtaining replacement samples shall be submitted to the Nuclear Regulatory Commission in the next Semiannual Radioactive Effluent Release Report. The locations from which samples became unavailable may be deleted; however, any locations from which suitable replacement samples are available shall be added to the program.

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

3.1.1.3 Surveillance Requirements

The radiological environmental monitoring samples shall be collected, pursuant to Table 3.1-1, from the locations described in Section 3.0, and shall be analyzed pursuant to the requirements of Tables 3.1-1 and 3.1-3.

3.1.1.4 Bases

The Radiological Environmental Monitoring Program required by this control provides measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides leading to the highest potential radiation exposures of MEMBERS OF THE PUBLIC, resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by measuring concentrations of radioactive materials and levels of radiation that may then be compared with those expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

3.1.2 Land Use Census

In accordance with Technical Specification 5.1.(2), a land use census shall be conducted to identify the location of the nearest MILK ANIMAL and the nearest permanent residence in each of the 16 meteorological sectors within a distance of 5 miles and the locations of all MILK ANIMALS within a distance of 3 miles.

3.1.2.1 Applicability

At all times.

3.1.2.2 Action

3.1.2.2.1 With a land use census identifying a location(s) yielding a calculated thyroid dose or dose commitment greater than the values currently being calculated in accordance with Section 2.6.4, submit the new location(s) to the Nuclear Regulatory Commission in the next Semiannual Radioactive Effluent Release Report.

3.1.2.2.2 With a land use census identifying a location(s) yielding a calculated thyroid dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained, add the new location(s) to the program within 30 days if samples are available. The sampling location having the lower calculated thyroid dose may then be deleted from the program.

3.1.2.3 Surveillance Requirements

The land use census shall be conducted once per 12 months by door-to-door survey, by visual survey from automobile or aircraft, by consulting local agriculture authorities, or by a combination of these methods as feasible using the information to provide a good census. Results of the annual census, as well as any changes in sampling locations, shall be discussed in the Annual Radiological Environmental Surveillance Report.

3.1.2.4 Bases

This control is provided to ensure 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. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50.

3.1.3 Interlaboratory Comparison Program

In accordance with Technical Specification 6.19(3), analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Nuclear Regulatory Commission. Analyses need to be performed only where the type analysis and sample are the same as that required in Table 3.1-1.

3.1.3.1 Applicability

At all times.

3.1.3.2 Action

With analyses not being performed as required above, report the corrective actions taken (to prevent a recurrence) in the Annual Radiological Environmental Surveillance Report.

3.1.3.3 Surveillance Requirements

A summary of results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Surveillance Report.

3.1.3.4 Bases

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

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TABLE 3.1-1 (SHEET 1 OF 4)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Approximate Number of Sample Locations[a]	Sampling and Collection Frequency	Type of Analysis and Frequency
1. Airborne			
a. Radioiodine and Particulates	5	Continuous operation of sampler with sample collection weekly.	Radioiodine canister. I-131 weekly.
			Particulate sampler. Analyze for gross beta radioactivity not less than 24 hours following filter change and analyze for I-131 weekly. Perform gamma isotopic analysis on affected sample when gross beta activity is 10 times the yearly mean of control samples. Composite (by location) for gamma isotopic analysis quarterly.
2. Direct Radiation	35	Quarterly	Gamma dose quarterly.

TABLE 3.1-1 (SHEET 2 OF 4)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Approximate Number of Sample Locations[a]	Sampling and Collection Frequency	Type of Analysis and Frequency
3. Ingestion			
a. Milk	4[b]	Bi-weekly	Gamma isotopic and I-131 analyses bi-weekly.
b. Fish[c] or clams	2	Semi-annually	Gamma isotopic analysis on edible portions semi-annually.
c. Grass or Leafy Vegetation	3	Monthly during growing season.	Gamma isotopic analysis monthly.[d]
4. Waterborne			
a. Surface	2	Composite[e] sample collected monthly.	Gamma isotopic analysis monthly. Composite (by location) for tritium analysis quarterly.
b. Sediment	1	Yearly	Gamma isotopic analysis sample yearly.

TABLE 3.1-1 (SHEET 3 OF 4)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Approximate Number of Sample Locations[a]	Sampling and Collection Frequency	Type of Analysis and Frequency
4. Waterborne (Continued)			
c. Drinking Water[f,g]	1* 1**	River Water collected near the intake will be a composite sample; the finished water will be a grab sample. These samples will be collected monthly unless the calculated dose due to consumption of the water is greater than 1 mrem/year; then the collection will be bi-weekly. The collection may revert to monthly should the calculated doses become less than 1 mrem/year.	I-131 analysis on each sample when bi-weekly collections are required. Gross beta and gamma isotopic analyses on each sample; composite (by location) for tritium quarterly.

*One sample of river water near the intake.

**One sample of finished water from each of one to three of nearest water supplies which could be affected by HNP discharge.

TABLE 3.1-1 (SHEET 4 OF 4)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

TABLE NOTATIONS

a. Sample locations are shown in Table 3.0-1 and Figures 3.0-1 through 3.0-4.

b. Up to three sampling locations within 5 miles and in different sectors will be used as available. In addition, one or more control locations beyond 10 miles will be used.

c. Commercially or recreationally important fish may be sampled. Clams will be sampled if difficulties are encountered in obtaining sufficient fish samples.

d. If gamma isotopic analysis is not sensitive enough to meet the Lower Limit of Detection, a separate analysis for I-131 may be performed.

e. Composite samples shall be collected by collecting an aliquot at intervals not exceeding a few hours.

f. If it is found that river water downstream of HNP is used for drinking, water samples will be collected and analyzed as specified herein.

g. A survey shall be conducted annually at least 50 river miles downstream of HNP to identify those who use Altamaha River water for drinking.

TABLE 3.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pci/l)	Grass (pCi/kg wet)
H-3	3E4*				
Mn-54	1E3		3E4		
Fe-59	4E2		1E4		
Co-58	1E3		3E4		
Co-60	3E2		1E4		
Zn-65	3E2		2E4		
Zr-95	4E2				
Nb-95	6E2				
I-131	2E0	9E-1		3E0	1E2
Cs-134	3E1	1E1	1E3	6E1	1E3
Cs-137	5E1	2E1	2E3	7E1	2E3
Ba-140	2E2			3E2	
La-140	2E2			4E2	

* For drinking water samples, the reporting level is 2E4 pCi/liter.

TABLE 3.1-3 (SHEET 1 OF 3)

LOWER LIMIT OF DETECTION[a] [b]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/l)	Grass (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4E0	1E-2				
H-3	2E3[d]					
Mn-54	2E0		1E2			
Fe-59	3E1		3E2			
Co-58	2E1		1E2			
Co-60	2E1		1E2			
Zn-65	3E1		3E2			
Zr-95	3E1					
Nb-95	2E1					
I-131[c]	1E0	7E-2		1E0	6E1	
Cs-134	2E1	5E-2	1E2	2E1	6E1	2E2
Cs-137	2E1	6E-2	2E2	2E1	8E1	2E2
Ba-140	6E1			6E1		
La-140	2E1			2E1		

LOWER LIMIT OF DETECTION

TABLE NOTATIONS

a. The Lower Limit of Detection (LLD) is the smallest concentration of radioactive material in a sample that will be detected with 95 percent probability with 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = (4.66 * s_b) / (E * V * 2.22 * Y * \exp(-\lambda\Delta T))$$

where:

LLD = the "a priori" Lower Limit of Detection (defined as pCi per unit mass or volume).

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

E = the counting efficiency (as counts per disintegration).

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

2.22 = the number of disintegrations per minute per picocurie.

Y = the fractional radiochemical yield (when applicable).

λ = the radioactive decay constant for the particular radionuclide.

ΔT = the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

ΔT = the elapsed time between midpoint of sample collection period and time of counting (for plant effluents).

TABLE 3.1-3 (SHEET 3 OF 3)

LOWER LIMIT OF DETECTION

TABLE NOTATIONS

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and ΔT should be used in the evaluation.

b. This does not mean that only the radionuclides in Table 3.1-3 are to be detected and reported. Other measurable and identifiable peaks, together with the above nuclides, shall be identified and reported. Only manmade radionuclides need be reported.

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

d. If no drinking water pathway exists, a value of 3E3 pCi/liter may be used.

4.0 TOTAL DOSE DETERMINATIONS

For the purpose of implementing Section 4.1, total dose determinations will be made by:

Calculating doses due to liquid effluents in accordance with Section 1.7.3.

Calculating doses due to gaseous effluents in accordance with Section 2.C.4.

Combining direct radiation doses based on direct radiation measurements with these effluent doses to determine total dose to a MEMBER OF THE PUBLIC.

The methodology for calculating doses to a MEMBER OF THE PUBLIC due to liquid effluents is presented in Section 1.2, and the methodology for calculating doses to a MEMBER OF THE PUBLIC due to gaseous effluents is presented in Section 2.2.2.b.

4.1 LIMITS OF OPERATION

4.1.1 Total Dose Control

In accordance with Technical Specification 6.18(10), the annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

4.1.1.1 Applicability

At all times.

4.1.1.2 Actions

4.1.1.2.1 With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Sections 1.7.3, 2.6.3, or 2.6.4, calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the limits stated above in Section 4.1.1 have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Nuclear Regulatory Commission within 30 days, pursuant to Technical Specification 5.9.2, a Special Report that defines the corrective action(s) to be taken to reduce subsequent releases to prevent recurrance of exceeding the above limits and include the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis estimating the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 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.

4.1.1.2.2 For Unit 1: When the ACTION statement or other requirements of this control cannot be met, steps need not be taken to change the Operational Mode of the Unit. Entry into an Operational Mode or other specified condition may be made if, as a minimum, the requirements of the ACTION statement are satisfied.

For Unit 2: The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

4.1.1.3 Surveillance Requirements

4.1.1.3.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Sections 1.7.3.3, 2.6.3.3, and 2.6.4.3 and in accordance with the methodology and parameters described in Sections 1.2, 2.2.2.a, and 2.2.2.b.

4.1.1.3.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters described in Section 4.0. This requirement is applicable only under conditions set forth above in Section 4.1.1.2.1.

4.1.1.4 Bases

This control is provided to meet the reporting requirements of 40 CFR 190.

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5.0 POTENTIAL DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

For the purpose of implementing Section 6.2.1.2, an assessment of potential doses to MEMBERS OF THE PUBLIC due to their activities within the SITE BOUNDARY will be performed if circumstances have changed such that any of the limits of Sections 2.6.3 or 2.6.4 are exceeded. The locations of concern within the SITE BOUNDARY are the roadside park, the camping area, the recreation area, and the Visitors Center. Historical annual average dispersion and deposition values, and elevated plume dose factors for these locations are presented in Tables 5.0-1 through 5.0-4, along with the estimated occupancy factors. (Estimated occupancy factors are for a MEMBER OF THE PUBLIC during a year.)

In the event that any limit of Section 2.6.3 is exceeded, an assessment will be performed considering direct radiation dose to a MEMBER OF THE PUBLIC resulting from noble gases in the plume. This assessment will take into consideration the annual average dispersion parameters and the estimated occupancy factors stated in Tables 5.0-1 through 5.0-4, or more precise values if available, for the locations of interest.

In the event that any limit of Section 2.6.4 is exceeded, an assessment will be performed considering the dose to a MEMBER OF THE PUBLIC due to inhalation of airborne radioactive materials suspended in the plume and due to radioactive material deposited on the ground. This assessment will take into consideration the annual average dispersion and deposition parameters and the estimated occupancy factors stated in Tables 5.0-1 through 5.0-4, or more precise values if available, for the locations of interest.

If none of the limits discussed above is exceeded, potential annual doses to a MEMBER OF THE PUBLIC at locations of concern within the SITE BOUNDARY are as follows:

Potential doses to a MEMBER OF THE PUBLIC at the Visitors Center are not expected to exceed 0.03 mrem due to inhalation and ground-plane exposure, and 0.02 mrem due to direct radiation from the plume.

Potential doses to a MEMBER OF THE PUBLIC at the roadside park are not expected to exceed 0.005 mrem due to inhalation and ground-plane exposure, and 0.003 mrem due to direct radiation from the plume.

Potential doses to a MEMBER OF THE PUBLIC at the camping area are not expected to exceed 0.04 mrem due to inhalation and ground-plane exposure, and 0.06 mrem due to direct radiation from the plume.

Potential doses to a MEMBER OF THE PUBLIC at the recreation area are not expected to exceed 0.6 mrem due to inhalation and ground-plane exposure, and 0.3 mrem due to direct radiation from the plume.

These values are based on annual average dispersion and deposition parameters and the estimated occupancy factors referenced above. Estimated occupancy factors for the recreation area, Visitors Center, the roadside park, and the camping area are based on activities observed at these locations over the last several years and continued anticipated usage of these areas.

SECTION 6.0

REPORTS

6.1 ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT

In accordance with Technical Specification 6.9.1.6, the Annual Radiological Environmental Surveillance Report covering the radiological environmental surveillance activities related to the plant during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in this ODCM and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

6.1.1 Report Contents

6.1.1.1 Summaries, interpretations, and statistical evaluations of the results of the radiological environmental surveillance activities for the reporting period, including (as appropriate) a comparison with the preoperational studies, operational controls, previous environmental surveillance reports, and an assessment of any observed impacts of the plant operation on the environment shall be included in the report. The report shall also include the results of the land use census required by Section 3.1.2 and the results of licensee participation in the Interlaboratory Comparison Program required by Section 3.1.3

6.1.1.2 The report shall include summarized and tabulated results in the format of Table 6.1-1 of all radiological environmental samples taken during the report period, with the exception of naturally occurring radionuclides which need not be reported. 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 practicable in a supplementary report.

6.1.1.3 Also to be included in the report are the following: a summary description of the Radiological Environmental Monitoring Program, a map of all sampling locations as keyed to a table indicating distances and directions from the main stack, and results of the licensee participation in the Interlaboratory Comparison Program.

6.1.1.4 Any deviations in conducting the Radiological Environmental Monitoring Program from that specified in Table 3.1-1 shall be documented in the report, in accordance with Section 3.1.1.2.1.

6.1.1.5 If the measured level of radioactivity in an environmental sampling medium is not the result of plant effluents, the condition shall be reported as required by Section 3.1.1.2.2.

6.1.1.6 In addition to the radionuclides listed in Table 3.1-3, other measurable and identifiable peaks shall be identified and reported. Only manmade radionuclides need be reported.

6.1.1.7 If Interlaboratory Comparison Program analyses are not performed as required by Section 3.1.3, the corrective actions taken to prevent a recurrence must be included in the report.

TABLE 6.1-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility Edwin I. Hatch Nuclear Plant Docket No. 50-321, 50-366

Location of Facility Appling County, Georgia Reporting Period _____

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection[a]	All Indicator Locations Mean Range	Name, Distance, and Direction	Mean Range[b]	Control Locations Mean Range	Number of REPORTABLE EVENTS

[a] Lower Limit of Detection is defined in Table Notation a of Table 3.1-3.

[b] Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses.

6.2 SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

In accordance with Technical Specification 6.9.1.8, the Semiannual Radioactive Effluent Release Report covering the operation of the plant during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in this ODCM and the Process Control Program (PCP) and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

6.2.1 Report Contents

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

6.2.1.2 The report to be submitted 60 days after January 1 of each year shall include and annual summary of meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, atmospheric stability and precipitation (if measured) on magnetic tape or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall include an assessment of the radiation doses from liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 1.7-1) during the reporting period if circumstances have changed such that the potential doses are significantly greater than expected at onsite locations as discussed in Section 5.0. All assumptions used in making these assessments, i.e., specific

activity, exposure time, and location, shall be included in these reports. Historical annual average meteorological conditions or meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with Sections 1.7.3, 2.2.2, and 5.0.

6.2.1.3 For each type of solid waste shipped offsite during the report period, the following information shall be included in the report:

a. Container volume,

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

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

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

e. Type of container, e.g., LSA, type A, type B, large quantity.

f. Solidification agent, e.g., cement.

6.2.1.4 The report shall include (on a quarterly basis) unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents that were in excess of 1 Ci, excluding dissolved and entrained gases and tritium for liquid effluents, or those in excess of 150 Ci of noble gases or 0.02 Ci of radioiodines for gaseous releases.

6.2.1.5 Any changes to the ODCM made during the reporting period shall be included in the report.

6.2.1.6 If a radioactive liquid effluent monitoring instrumentation channel required by Table 1.7-1 remains inoperable for over 30 days, an explanation of the circumstances shall be included in the next report. This requirement does not include the Service Water System to Closed Cooling Water System Differential Pressure channel.

6.2.1.7 If the concentration of radioactive material released from the site to UNRESTRICTED AREAS exceeds the limits of Section 1.7.2, a discussion of the causes and corrective actions taken must be included in the next report.

6.2.1.8 If a radioactive gaseous effluent monitoring instrumentation channel required by Table 2.6-1 remains inoperable for over 30 days, an explanation of the circumstances shall be included in the next report.

6.2.1.9 When unusual circumstances result in a Lower Limit of Detection higher than required by Table 2.6-3 (which addresses radioactive gaseous sample analysis), the reasons shall be documented in the report.

6.2.1.10 If adequate samples of milk, grass or leafy vegetation (during the growing season) from any of the sample locations required by Table 3.1-1 are unavailable, the cause of the unavailability and identification of the affected location(s) and the location(s), if any, for obtaining replacement samples shall be submitted in the next report.

6.2.1.11 With a land use census identifying a location(s) yielding a calculated thyroid dose or dose commitment greater than the values currently being calculated in accordance with Section 2.6.4, submit the new location(s) in the next report.

6.2.1.12 If the contents within any outside temporary tank exceed the limits of Technical Specification 3.15.1.4 (Unit 1) or 3.11.1.4 (Unit 2), notification shall be included in the report.

6.3 MONTHLY OPERATING REPORT

A report of any major changes to the radioactive waste treatment systems shall be submitted to the Nuclear Regulatory Commission with the Monthly Operating Report (which must be submitted in accordance with Technical Specification 6.9.1.10) for the period in which the evaluation was reviewed and accepted by the Plant Review Board.

6.4 SPECIAL REPORTS

Special reports shall be submitted to the Nuclear Regulatory Commission in accordance with Technical Specification 6.9.2 as required by Sections 1.7.3.2.1, 1.7.4.2.1, 2.6.3.2.1, 2.6.4.2.1, 2.6.5.2.1, 3.1.1.2.2, and 4.1.1.2.1.

This section addresses only reporting requirements included in the ODCM; special reports also may be required under circumstances described in Plant Hatch Technical Specifications.

SECTION 7.0

GENERAL DEFINITIONS

7.1 TERM TRANSFERRED TO THE ODCM FROM TECHNICAL SPECIFICATIONS

The term defined in this section was transferred from the Technical Specifications to the ODCM in accordance with NRC Generic Letter 89-01. Wherever this term appears in the text of the Limits of Operation Sections of the ODCM, it is presented in all capital letters to indicate that it is specifically defined.

7.1.1 GASEOUS RADWASTE TREATMENT SYSTEM

The GASEOUS RADWASTE TREATMENT SYSTEM is the offgas holdup 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.

7.2 TERMS DEFINED IN TECHNICAL SPECIFICATIONS

The following definitions are contained in the Technical Specifications Section 1.0 "Definitions." Because these terms are used extensively throughout the ODCM, they are also included in this section for convenience. Throughout the text of the Limits of Operation Sections of the ODCM, these terms are presented in all capital letters to indicate that they are specifically defined. In some cases, the definition of a particular term is not exactly the same for both units; in that case, both definitions are presented along with an indication of the unit to which the definition applies.

7.2.1 ACTION

For Unit 1: ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

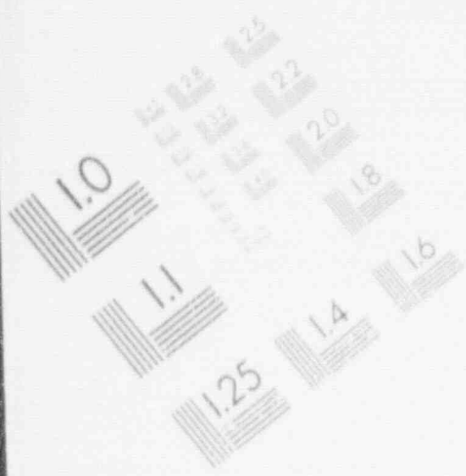
For Unit 2: ACTIONS shall be those additional requirements specified as corollary statements to each specification and shall be part of the specifications.

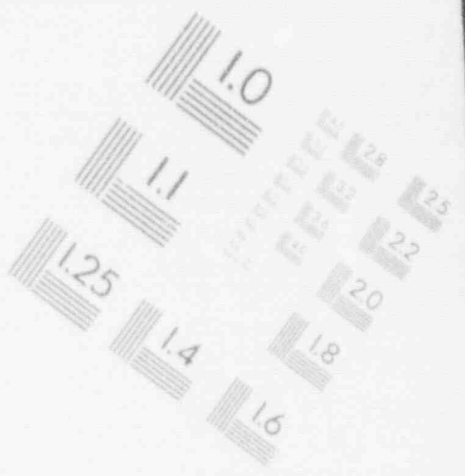
7.2.2 CHANNEL CALIBRATION

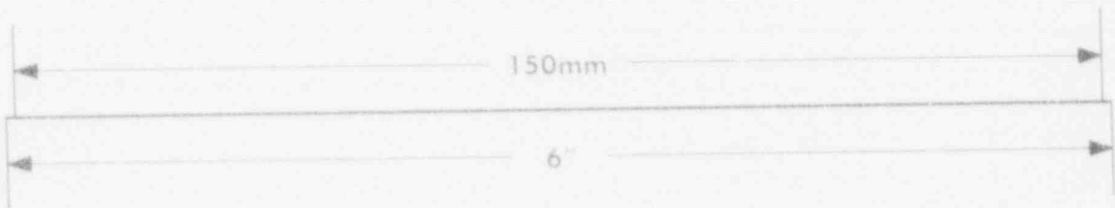
A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip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

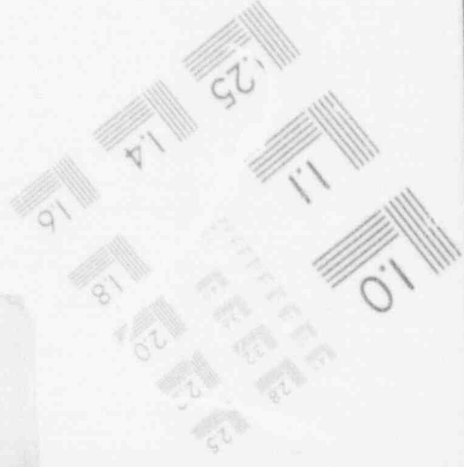
1

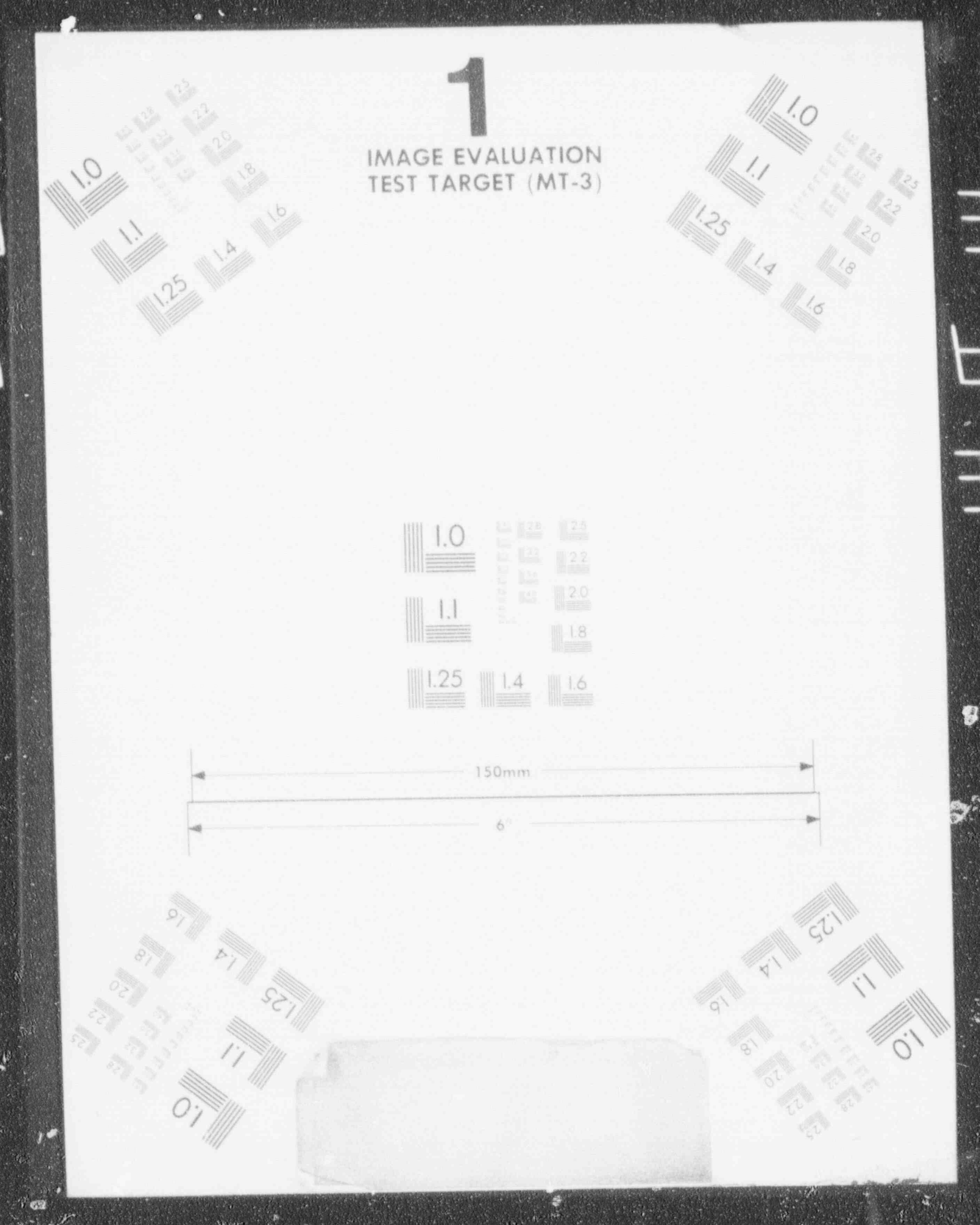
IMAGE EVALUATION
TEST TARGET (MT-3)





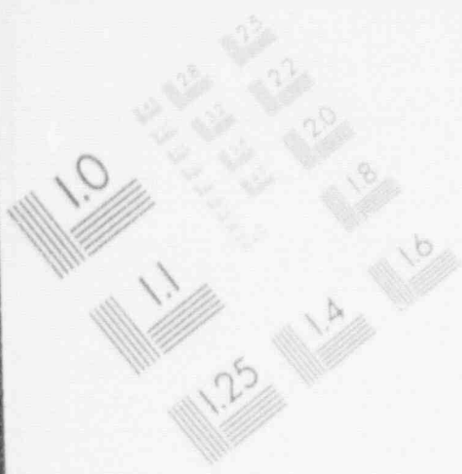


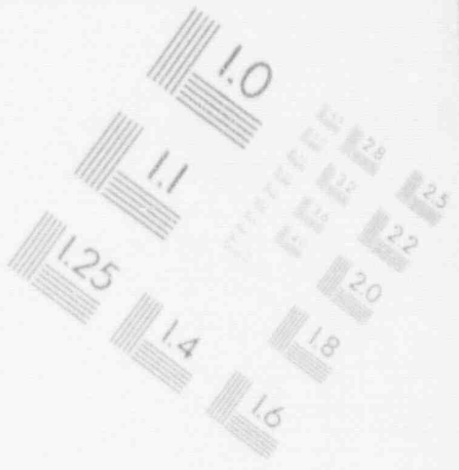


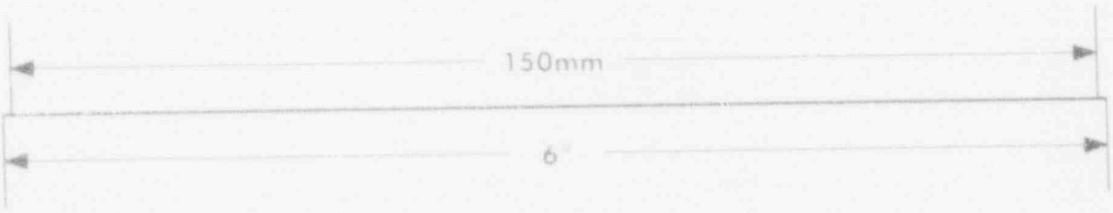

1
IMAGE EVALUATION
TEST TARGET (MT-3)
1.0
1.1
1.25
1.4
1.6
1.8
2.0
2.2
2.5
150mm
6"

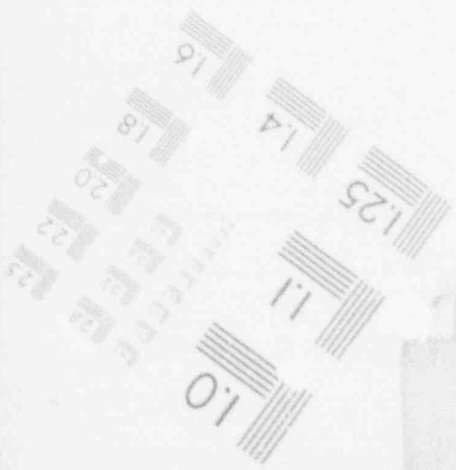
1

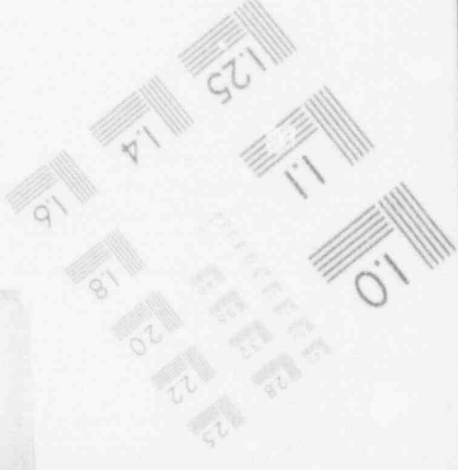
IMAGE EVALUATION
TEST TARGET (MT-3)





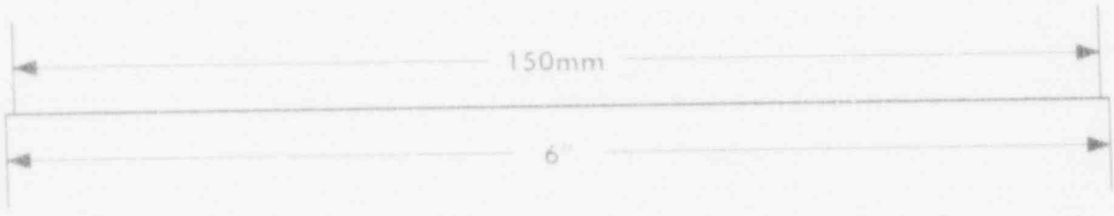






1

IMAGE EVALUATION
TEST TARGET (MT-3)



7.2.3 CHANNEL CHECK

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

7.2.4 CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

For Unit 1:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

For Unit 2:
The definition is the same as for Unit 1 with the exception that "and channel failure trips" is added to item a.

7.2.5 DOSE EQUIVALENT IODINE

The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram), which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844 or those in NRC Regulatory Guide 1.109, Revision 1, October 1977.

7.2.6 FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals as follows:

NOTATION	DEFINITION	FREQUENCY
S	Once per shift	Once per 12 hours
D	Daily	Once per 24 hours
W	Weekly	Once per 7 days
M	Monthly	Once per 31 days
Q	Quarterly	Once per 92 days
SA	Semi-annually	Once per 184 days
R	REFUELING	Once per 18 months
S/U	STARTUP	Prior to each reactor startup
P	Prior	Completed prior to each release
NA	Not applicable	Not applicable

7.2.7 MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the 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.

7.2.8 MILK ANIMAL

A MILK ANIMAL is a cow or goat that is producing milk for human consumption.

7.2.9 OPERATIONAL CONDITION

For Unit 1: This term is not specifically defined. See the definition of REACTOR MODE below.

For Unit 2: An OPERATIONAL CONDITION shall be any one inclusive combination of mode switch position and average reactor coolant temperature indicated as follows:

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any Temperature
2. STARTUP	Startup/Hot Standby	Any Temperature
3. HOT SHUTDOWN	Shutdown	> 212° F***
4. COLD SHUTDOWN	Shutdown	< or = 212° F***
5. REFUELING*	Refuel**	< or = 212° F

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

** See Special Test Exception 3.10.3.

*** During the performance of inservice hydrostatic or leak testing with all control rods fully inserted and reactor coolant temperatures above 212° F, the reactor may be considered to be in the COLD SHUTDOWN condition for the purpose of determining Limiting Condition for Operation applicability. However, compliance with an ACTION requiring COLD SHUTDOWN shall require a reactor coolant temperature < or = 212° F. In addition, compliance with the following specifications is required when performing the hydrostatic and leak testing under the identified conditions: 3.6.5.1, 3.6.5.2, 3.6.6.1, and 3.7.1.1.

7.2.10 OPERABLE - OPERABILITY

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

7.2.11 REACTOR MODE

For Unit 1: The reactor mode is established by the Mode Switch position. The switch positions are REFUEL, SHUTDOWN, START & HOT STANDBY and RUN; thus the four possible reactor modes are: Refuel Mode, Shutdown Mode, Start & Hot Standby Mode, and Run Mode. (See Unit 1 Technical Specifications Section 1.0 "Definitions" for definitions of these terms.)

For Unit 2: This term is not specifically defined. See the definition of OPERATIONAL CONDITION above.

7.2.12 RATED THERMAL POWER

For Unit 1: Rated thermal power means the reactor is operating, at a steady state power of 2436 megawatts thermal. This is also referred to as 100 percent thermal power.

For Unit 2: RATED THERMAL POWER shall be a total reactor core heat transfer to the reactor coolant of 2436 MW.

7.2.13 SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by Georgia Power Company, as shown in Figure 1.7-1.

7.2.14 SOURCE CHECK

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

7.2.15 THERMAL POWER

For Unit 1: This term is not defined.

For Unit 2: THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

7.2.16 UNRESTRICTED AREA

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

REFERENCES (Continued)

14. Hatch Nuclear Plant Land Use Survey - 1987, Georgia Power Company, February 1987.

15. Letter to Georgia Power Company from Pickard, Lowe, and Garrick, Inc., Washington, D.C., May 11, 1987.

16. Letter to Georgia Power Company from Pickard, Lowe, and Garrick, Inc., Washington, D.C., June 3, 1987.

17. Letter to Georgia Power Company from Pickard, Lowe, and Garrick, Inc., Washington, D.C., June 11, 1987.

18. Internal Memorandum, W.H. Ollinger to D.M. Hopper, Georgia Power Company, June 9, 1987.

19. Letter to Georgia Power Company from Quantum Technology, Inc., Marietta, Georgia, June 17, 1987.

20. Hatch Nuclear Plant Land Use Survey, Georgia Power Company, November 1987.

21. Letter to Georgia Power Company from Pickard, Lowe, and Garrick, Inc., Washington, D.C., November 30, 1987.

22. Hatch Nuclear Plant Land Use Survey, Georgia Power Company, November 16, 1988, and December 20, 1988.

23. Letter to Georgia Power Company from J. H. Davis, Health Physics Consultant, Lilburn, Georgia, September 17, 1990.

24. Meinke, W.W. and T.H. Essig, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors," Generic Letter 89-01, Supplement No. 1, NUREG-1302, April 1991.