6.0 OFF-SITE DOSE CALCULATION METHODS

Chapter 6 provides the basis for plant procedures required to meet the 10CFR50, Appendix I, ALARA dose objectives, and the 40CFR190 total dose limits to members of the public in unrestricted areas, as stated in the Radiological Effluent Controls (implementing the requirements of Technical Specification 6.7.D). A simple, conservative method (called Method I) is listed in Tables 1.1.2 to 1.1.7 for each of the Control requirements. Each of the Method I equations is presented, along with their bases in Sections 6.2 through 6.9 and Section 6.11. In addition, reference is provided to more sophisticated but still conservative methods (called Method II) for use when more accurate results are needed. This chapter provides the methods, data, and reference material with which the operator can calculate the needed doses and dose rates. Setpoint methods for effluent monitor alarms are described in Chapter 8.

Demonstration of compliance with the dose limits of 40CFR190 is considered to be a demonstration of compliance with the 0.1 rem limit of 10CFR20.1301(a)(1) for members of the public in unrestricted areas (Reference 56 FR23374, 3rd column).

6.1 Introductory Concepts

The Radiological Effluent Controls Program (Technical Specifications 6.7.D) either limit dose or dose rate. The term "Dose" for ingested or inhaled radioactivity means the dose commitment, measured in mrem, which results from the exposure to radioactive materials that, because of uptake and deposition in the body, will continue to expose the body to radiation for some period of time after the source of radioactivity is stopped. The time frame over which the dose commitment is evaluated is 50 years. The phrases "annual Dose" or "Dose in one year" then refers to the fifty-year dose commitment from one year's worth of releases. "Dose in a quarter" similarly means a fifty-year dose commitment from one quarter's releases. The term "Dose," with respect to external exposures, such as to noble gas clouds, refer only to the doses received during the actual time period of exposure to the radioactivity released from the plant. Once the source of the radioactivity is removed, there is no longer any additional accumulation to the dose commitment.

Gaseous effluents from the plant are also controlled such that the maximum "dose rates" at the site boundary at any time are limited to 500 mrem/year. This instantaneous dose rate limit allows for operational flexibility when off normal occurrences may temporarily increase gaseous effluent release rates from the plant, while still providing controls to ensure that licensees meet the dose objectives of Appendix I to 10CFR50.

It should also be noted that a dose rate due to noble gases that exceeds for a short time period (less than one hour in duration) the equivalent 500 mrem/year dose rate limit stated in Control 3.3.1.a, does not necessarily, by itself, constitute a Licensee Event Report (LER) under 10CFR Part 50.73 unless it is determined that the air concentration of radioactive effluents in unrestricted areas has also exceeded 20 times applicable concentration limits specified in Appendix B to 20.1001 - 20.2401, Table 2, Column 1 (four-hour notification per 10CFR50.72, and 30-day LER per 10CFR50.73).

The quantities D and $\dot{R}$ are introduced to provide calculable quantities, related to off-site dose, or dose rate which demonstrates compliance with the effluent controls.

The dose D is the quantity calculated by the Chapter 6 dose equations. The D calculated by "Method I" equations is not necessarily the actual dose received by a real individual but usually provides an upper bound for a given release because of the conservative margin built into the dose factors and the

selection and definition of critical receptors. The radioisotope specific dose factors in each "Method I" dose equation represent the greatest dose to any organ of any age group accounting for existing or potential pathways of exposure. The critical receptor assumed by "Method I" equations is typically a hypothetical individual whose behavior - in terms of location and intake - results in a dose which is expected to be higher than any real individual. The Method I equations employ five-year historical average atmospheric dispersion factors to define receptors of maximum impact. Method II allows for a more exact dose calculation for real individuals, if necessary, by considering only existing pathways of exposure, or actual concurrent meteorology with the recorded release. Maximum receptor doses determined using quarterly meteorology may be greater than doses calculated with Method I due to short time period variability of meteorological conditions from the long-term average. Quarterly average dispersion values for maximum receptors have been observed to differ from five-year average values by as much as 54%.

$\dot{R}$ is the quantity calculated in the Chapter 6 dose rate equations. It is calculated using the plant's effluent monitoring system reading and an annual average or long-term atmospheric dispersion factor. Dispersion factors based on actual concurrent meteorology during effluent releases can also be used via Method II, if necessary, to demonstrate compliance with off-site dose rate limits.

Each of the methods to calculate dose or dose rate are presented in separate sections of Chapter 6, and are summarized in Tables 1.1.1 to 1.1.7. Each method has two levels of complexity and are called Method I and Method II. Method I is the simplest; generally a linear equation. Method II is a more detailed analysis which allows for use of site-specific factors and variable parameters to be selected to best fit the actual release conditions, within the bounds of the guidance provided.

The plant has both elevated and ground level gaseous release points: the main vent stack (elevated release), and the North Warehouse waste oil burner (ground level release). Therefore, total dose calculations for skin, whole body, and the critical organ from gaseous releases will be the sum of the elevated and ground level doses. Appendix D provides an assessment of the surveillance needs for waste oil to ensure that off-site doses from its incineration is maintained within the ALARA limits of the effluent Controls.

6.2 Method to Calculate the Total Body Dose from Liquid Releases

Effluent Control 3.2.2 limits the total body dose commitment to a Member of the Public from radioactive material in liquid effluents to 1.5 mrem per quarter and 3 mrem per year. Control 3.2.3 requires liquid radwaste treatment when the total body dose estimate exceeds 0.06 mrem in any month. Control 3.4.1 limits the total body dose commitment to any real member of the public from all station sources (including liquids) to 25 mrem in a year. Dose evaluation is required at least once per month. If the liquid radwaste treatment system is not being used, dose evaluation is required before each release.

Use Method I first to calculate the maximum total body dose from a liquid release to the Connecticut River as it is simpler to execute and more conservative than Method II.

Use Method II if a more accurate calculation of total body dose is needed (i.e., Method I indicates the dose is greater than the limit), or if Method I cannot be applied.

If the radwaste system is not operating, the total body dose must be estimated prior to a release (Control 3.2.3). To evaluate the total body dose, use Equation 6.1 to estimate the dose from the planned release and add this to the total body dose accumulated from prior releases during the month.

6.2.1 Method I

The increment in total body dose from a liquid release is:

$$D_{tb} = \sum_i Q_i \, DFL_{itb} \qquad (6\text{-}1)$$

$$(\text{mrem}) \quad (\text{Ci}) \left(\frac{\text{mrem}}{\text{Ci}} \right)$$

where:

DFL_{itb} = Site-specific total body dose factor (mrem/Ci) for a liquid release. See Table 1.1.11.

Q_i = Total activity (Ci) released for radionuclide "i". (For strontiums and Fe-55, use the most recent measurement available.)

Equation 6-1 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Normal operations (not emergency event),

2. Liquid releases were to the Connecticut River, and

3. Any continuous or batch release over any time period.

6.2.2 Basis for Method I

This section serves three purposes: (1) to document that Method I complies with appropriate NRC regulations, (2) to provide background and training information to Method I users, and (3) to provide an introductory user's guide to Method II.

Method I may be used to show that the effluent Controls which limit off-site total body dose from liquids (3.2.2 and 3.2.3) have been met for releases over the appropriate periods. Control 3.2.2 is based on the ALARA design objectives in 10CFR50, Appendix I Subsection II A. Control 3.2.3 is an "appropriate fraction", determined by the NRC, of that design objective (hereafter called the Objective). Control 3.4.1 is based on Environmental Standards for Uranium Fuel Cycle in 40CFR190 (hereafter called the Standard) which applies to direct radiation as well as liquid and gaseous effluents.

Exceeding the Objective or the Standard does not immediately limit plant operation but requires a report to the NRC within 30 days. In addition, a waiver may be required.

Method I was developed such that "the actual exposure of an individual ... is unlikely to be substantially underestimated" (10CFR50, Appendix I). The definition, below, of a single "critical receptor" (a hypothetical individual whose behavior results in an unrealistically high dose) provides part of the conservative margin to the calculation of total body dose in Method I. Method II allows that actual individuals, with real behaviors, be taken into account for any given release. In fact, Method I was based on a Method II analysis for the critical receptor with maximum exposure

conditions instead of any real individual. That analysis was called the "base case"; it was then reduced to form Method I.

The steps performed in the Method I derivation follow. First, in the base case, the dose impact to the critical receptor (in the form of dose factors DFL_{itb}, mrem/Ci) for a 1 curie release of each radioisotope in liquid effluents was derived. The base case analysis uses the methods, data and assumptions in Regulatory Guide 1.109 (Equations A-2, A-3, A-7, A-13 and A-16, Reference A). The liquid pathways identified as contributing to an individual's dose are the consumption of fish from the Connecticut River, the ingestion of vegetables and leafy vegetation which were irrigated by river water, the consumption of milk and meat from cows and beef cattle who had river water available for drinking as well as having feed grown on irrigated land, and the direct exposure from the ground plane associated with activity deposited by the water pathway. A plant discharge flow rate of 44.6 ft^3/sec was used with a mixing ratio of 0.0356 which corresponds to a minimum regulated river flow of 1250 cfs at the Vernon Dam just below the plant discharge outfall.* Tables 6.2.1 and 6.2.2 outline human consumption and environmental parameters used in the analysis. The resulting, site-specific, total body dose factors appear in Table 1.1.11.

For any liquid release, during any period, the increment in annual average total body dose from radionuclide "i" is:

(6-2)

$$\Delta D_{tb} = Q_i \, DFL_{itb}$$

$$(\text{mrem})\,(\text{Ci})\left(\frac{\text{mrem}}{\text{Ci}}\right)$$

where:

DFL_{itb} = Site-specific total body dose factor (mrem/Ci) for a liquid release. See Table 1.1.11.

Q_i = Total activity (Ci) released from radionuclide "i".

* An Mp equal to 1.0 for the fish pathway is assumed between the discharge structure and the dam.

Method I is conservative because it is based on dose factors DFL_{itb} which were chosen from the base case to be the highest of the four age groups for each radionuclide, as well as assuming minimum river dilution flow.

6.2.3 Method II

If Method I cannot be applied, or if the Method I dose exceeds the limit or if a more exact calculation is required, then Method II should be applied. Method II consists of the models, input data and assumptions in Regulatory Guide 1.109, Rev. 1 (Reference A), except where site-specific models, data or assumptions are more applicable, such as the use of actual river flow at the time of actual discharge as opposed to the minimum river flow of 1,260 cfs that is assumed in the Method I dose factors (except for the fish pathway). The base case analysis, documented above, is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

TABLE 6.2.1

Environmental Parameters for Liquid Effluents at Vermont Yankee
(Derived from Reference A)

VARIABLE			POTABLE WATER	AQUATIC FOOD	SHORELINE ACTIVITY	FOOD GROWN WITH CONTAMINATED WATER: VEGETABLES	LEAFY VEG.	MEAT	COW MILK
MP	Mixing Ratio		-	1.0	0.0356	0.0356	0.0356	0.0356	0.0356
TP	Transit Time	(HRS)	-	24.0	0.000	0.0000	0.0000	480.0	48.0
YV	Agricultural Productivity	(KG/M^2)				2.0	2.0	2.0	2.0
P	Soil Surface Density	(KG/M^2)				240.0	240.0	240.0	240.0
IRR	Irrigation Rate	($L/M^2/HR$)				0.152	0.152	0.152	0.152
TE	Crop Exposure Time	(HRS)				1440.0	1440.0	1440.0	1440.0
TH	Holdup Time	(HRS)				1440.0	24.0	2160.0	2160.0
QAW	Water Uptake Rate for Animal	(L/D)						50.0	60.0
QF	Feed Uptake Rate for Animal	(KG/D)						50.0	50.0
FI	Fraction of Year Crops Irrigated					0.5	0.5	0.5	0.5
	Location of Critical Receptor		Connecticut River Below Vernon Dam						

TABLE 6.2.2

Usage Factors for Various Liquid Pathways at Vermont Yankee
(From Reference A, Table E-5. Zero Where No Pathway Exists)

AGE	VEG.	LEAFY VEG.	MILK	MEAT	FISH	INVERT.	POTABLE WATER	SHORELINE
	(KG/YR)	(KG/YR)	(LITER/YR)	(KG/YR)	(KG/YR)	(KG/YR)	(LITER/YR)	(HR/YR)
Adult	520.00	64.00	310.00	110.00	21.00	0.00	0.00	12.00
Teen	630.00	42.00	400.00	65.00	16.00	0.00	0.00	67.00
Child	520.00	26.00	330.00	41.00	6.90	0.00	0.00	14.00
Infant	0.00	0.00	330.00	0.00	0.00	0.00	0.00	0.00

6.3 Method to Calculate Maximum Organ Dose from Liquid Releases

Effluent Control 3.2.2 limits the maximum organ dose commitment to a Member of the Public from radioactive material in liquid effluents to 5 mrem per quarter and 10 mrem per year. Control 3.2.3 requires liquid radwaste treatment when the maximum organ dose estimate exceeds 0.2 mrem in any month. Control 3.4.1 limits the maximum organ dose commitment to any real member of the public from all station sources (including liquids) to 25 mrem in a year except for the thyroid, which is limited to 75 mrem in a year. Dose evaluation is required at least once per month if releases have occurred. If the liquid radwaste treatment system is not being used, dose evaluation is required before each release.

Use Method I first to calculate the maximum organ dose from a liquid release to the Connecticut River as it is simpler to execute and more conservative than Method II.

Use Method II if a more accurate calculation of organ dose is needed (i.e., Method I indicates the dose is greater than the limit), or if Method I cannot be applied.

If the radwaste system is not operating, the maximum organ dose must be estimated prior to a release (Control 3.2.3). To evaluate the maximum organ dose, use Equation 6-3 to estimate the dose from the planned release and add this to the maximum organ dose accumulated from prior releases during the month.

6.3.1 Method I

The increment in maximum organ dose from a liquid release is:

$$D_{mo} = \sum_i Q_i \, DFL_{imo} \qquad (6\text{-}3)$$

(mrem) (Ci) $\left(\frac{\text{mrem}}{\text{Ci}}\right)$

where:

DFL_{imo} = Site-specific maximum organ dose factor (mrem/Ci) for a liquid release. See Table 1.1.11.

Q_i = Total activity (Ci) released for radionuclide "i". (For strontiums and Fe-55, use the most recent measurement available.)

Equation 6-3 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Normal operations (not emergency event),
2. Liquid releases were to the Connecticut River, and
3. Any continuous or batch release over any time period.

6.3.2 Basis for Method I

This section serves three purposes: (1) to document that Method I complies with appropriate NRC regulations, (2) to provide background and training information to Method I users, and (3) to provide an introductory user's guide to Method II. The methods to calculate maximum organ dose parallel the total body dose methods (see Section 6.2.2). Only the differences are presented here.

For each radionuclide, a dose factor (mrem/Ci) was determined for each of seven organs and four age groups. The largest of these was chosen to be the maximum organ dose factor (DFL_{imo}) for that radionuclide.

For any liquid release, during any period, the increment in annual average dose from radionuclide "i" to the maximum organ is:

(6-4)

$$\Delta D_{mo} = Q_i \, DFL_{imo}$$

$$\text{(mrem)} \quad \text{(Ci)} \left(\frac{\text{mrem}}{\text{Ci}} \right)$$

where:

DFL_{imo} = Site-specific maximum organ dose factor (mrem/Ci) for a liquid release. See Table 1.1.11.

Q_i = Total activity (Ci) released for radionuclide "i". (For strontiums and Fe-55, use the most recent measurement available.)

Equation 6-3 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Normal operations (not emergency event),
2. Liquid releases were to the Connecticut River, and
3. Any continuous or batch release over any time period.

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For each radionuclide, a dose factor (mrem/Ci) was determined for each of seven organs and four age groups. The largest of these was chosen to be the maximum organ dose factor (DFL_{imo}) for that radionuclide.

For any liquid release, during any period, the increment in annual average dose from radionuclide "i" to the maximum organ is:

(6-4)

$$\Delta D_{mo} = Q_i \, DFL_{imo}$$

$$\text{(mrem)} \quad \text{(Ci)} \left(\frac{\text{mrem}}{\text{Ci}} \right)$$

where:

DFL_{imo} = Site-specific maximum organ dose factor (mrem/Ci) for a liquid release. See Table 1.1.11.

Q_i = Total activity (Ci) released for radionuclide "i".

Because of the assumptions about receptors, environment, and radionuclides; and because of the low Objective and Standard, the lack of immediate restriction on plant operation, and the adherence to 10CFR20 concentrations (which limit public health consequences) a failure of Method I (i.e., the exposure of a real individual being underestimated) is improbable and the consequences of a failure are minimal.

6.3.3 Method II

If Method I cannot be applied, or if the Method I dose exceeds the limit or if a more exact calculation is required, then Method II should be applied. Method II consists of the models, input data and assumptions in Regulatory Guide 1.109, Rev. 1 (Reference A), except where site-specific models, data or assumptions are more applicable. The base case analysis, documented above, is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

6.4 Method to Calculate the Total Body Dose Rate From Noble Gases

Effluent Control 3.3.1 limits the instantaneous dose rate at any time to the total body from all release sources of noble gases at any location at or beyond the site boundary equal to or less than 500 mrem/year.

Use Method I first to calculate the Total Body Dose Rate from the peak release rate via both elevated and ground level release points. The dose rate limit of Control 3.3.1.a is the total contribution from both ground and elevated releases occurring during the period of interest.

Use Method II if Method I predicts a dose rate greater than the Control limit (i.e., use of actual meteorology over the period of interest) to determine if, in fact, Control 3.3.1 had actually been exceeded during a short time interval.

Compliance with the dose rate limits for noble gases are continuously demonstrated when effluent release rates are below the plant stack noble gas activity monitor alarm setpoint by virtue of the fact that the alarm setpoint is based on a value which corresponds to the off-site dose rate limit of Control 3.3.1, or a value below it, taking into account the potential contribution of releases from all ground level sources.

Determinations of dose rates for compliance with Control (3.3.1) are performed when the effluent monitor alarm setpoint is exceeded and the corrective action required by Control 3.3.1 is unsuccessful, or as required by the notations to Control Table 3.1.2 when the stack noble gas monitor is inoperable.

6.4.1 Method I

The Total Body Dose Rate due to noble gases can be determined by multiplying the individual radionuclide release rates by their respective dose factors, summing all the products together, and then multiplying this total by a conversion constant (0.61), as seen in the following Equation 6-5:

$$\dot{R}_{tbs} = 0.61 \sum_i \dot{Q}_i^{ST} \; DFB_i \qquad (6\text{-}5)$$

$$\left(\frac{mrem}{yr}\right)\left(\frac{pCi-sec}{\mu Ci-m^3}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem-m^3}{pCi-yr}\right)$$

where:

$\dot{Q}_i^{ST}$ = In the case of noble gases, the release rate from the plant stack (μCi/sec) for each radionuclide, "i", identified. The release rate at the plant stack is based on measured radionuclide concentrations and distributions in periodic grab samples taken at the stack. As an alternative method, the radionuclide distribution in the off-gas at the Steam Jet Air Ejector (SJAE) can be used during plant operations, along with the Stack Gas Monitor effluent count rate, to estimate stack radionuclide releases. The release rate at the stack when using SJAE samples can be stated as follows:

$$\dot{Q}_i^{ST} = \frac{\dot{Q}_i^{SJAE}}{\sum_i \dot{Q}_i^{SJAE}} \; M \; \frac{1}{S} \; F \qquad (6\text{-}28)$$

$$\frac{uCi}{sec} = (cpm)\left(\frac{\mu Ci/cc}{cpm}\right)\frac{(cc)}{sec}$$

M = Plant Stack Gas Monitor I or II count rate (cpm).

S_g = Appropriate or conservative plant stack monitor detector counting efficiency for the given nuclide mix (cpm/(μCi/cc)).

F = Stack flow rate (cc/sec).

$\dot{Q}_i^{SJAE}$ = The last measured release rate at the steam jet air ejector of noble gas i (μCi/sec).

DFB_i = Total body gamma dose factor (see Table 1.1.10).

For ground level noble gas releases, the total body dose rate is calculated as follows:

$$\dot{R}_{tbg} = 6.4 \sum_i \dot{Q}_i^{GL} DFB_i \left(\frac{pCi - sec}{\mu Ci - m^3}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem - m^3}{pCi - yr}\right) \quad (6\text{-}39)$$

where:

$\dot{Q}_i^{GL}$ = Ground level release rate (μCi/sec) of noble gas.

The total body dose rate for the site is equal to $\dot{R}_{tbs} + \dot{R}_{tbg}$.

During periods (beyond the first five days) when the plant is shutdown and no radioactivity release rates can be measured at the SJAE, Xe-133 may be used in place of the last SJAE measured mix as the referenced radionuclide to determine off-site dose rate and monitor setpoints. In this case, the ratio of each Q_i^{SJAE} to the sum of all Q_i^{SJAE} in Equation 6-28 above is assumed to reduce to a value of 1, and the total body gamma dose factor DFB_i for Xe-133 (2.94 E-04 mrem-m^3/ρCi-yr) is used in Equation 6-5. Alternately, a relative radionuclide "i" mix fraction (f_i) may be taken from Table 8.2.1 as a function of time after shutdown, and substituted in place of the ratio of Q_i^{SJAE} to the sum of all Q_i^{SJAE} in Equation (6-28) above to determine the relative fraction of each noble gas potentially available for release to the total. Just prior to plant startup before a SJAE sample can be taken and analyzed, the monitor alarm setpoints should be based on Xe-138 as representing the most prevalent high dose factor noble gas expected to be present shortly after the plant returns to power. Monitor alarm setpoints which have been determined to be conservative under any plant conditions may be utilized at any time in lieu of the above assumptions.

Equations 6-5 and 6-39 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Normal operations (not emergency event), and

2. Noble gas releases via either elevated or ground level vents to the atmosphere.

6.4.2 Basis for Method I

Method I may be used to show that the Control limit for total body dose rate from noble gases released to the atmosphere has been met for the peak noble gas release rate.

Method I for stack releases was derived from Regulatory Guide 1.109 as follows:

$$\dot{R}_{tbs} = 1E+06\ S_F\ [X/Q]^{\gamma}_{s} \sum_i \dot{Q}_i^{ST}\ DFB_i \qquad (6\text{-}6)$$

$$\left(\frac{mrem}{yr}\right) = \left(\frac{pCi}{\mu Ci}\right)(\#)\left(\frac{sec}{m^3}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem - m^3}{pCi - yr}\right)$$

where:

S_F = Shielding factor = 1.0 for dose rate determination.

$[X/Q]^{\gamma}_{s}$ = Maximum annual average gamma atmospheric dispersion factor for stack (elevated) releases; = 6.11E-07 (sec/m^3).

$\dot{Q}_i^{ST}$ = Release rate from the plant stack of noble gas "i" ($\mu Ci/sec$).

DFB_i = Gamma total body dose factor, $\left(\frac{mrem - m^3}{pCi - yr}\right)$. See Table 1.1.10.

Equation 6-6 reduces to:

$$\dot{R}_{tbs} = 0.61 \sum_i \dot{Q}_i^{ST}\ DFB_i \qquad (6\text{-}5)$$

$$\left(\frac{mrem}{yr}\right) = \left(\frac{pCi - sec}{\mu Ci - m^3}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem - m^3}{pCi - yr}\right)$$

For ground level releases, the ground level maximum long-term average gamma atmospheric dispersion factor = 6.42E-06 sec/m^3, thus leading to:

$$\dot{R}_{tbg} = 1E+06 * 6.42E-06 \sum_i \dot{Q}_i^{GL}\ DFB_i \qquad (6\text{-}39)$$

or

$$\dot{R}_{tbg} = 6.4 \sum_i \dot{Q}_i^{GL}\ DFB_i$$

The selection of critical receptor, outlined in Section 6.10, is inherent in Method I, as are the maximum expected off-site annual or long-term average atmospheric dispersion factors. Due to the holdup and decay of gases allowed in the AOG, off-gas concentrations at the plant stack during routine plant operations are usually too low for determination of the radionuclide mix at the plant stack. It is then conservatively assumed that most of the noble gas activity at the plant stack is the result of in-plant steam leaks which are removed to the plant stack by building ventilation air flow, and that this air flow has an isotopic distribution consistent with that routinely measured at the SJAE.

The calculation of ground level release dispersion parameters are based on the location of the North Warehouse with respect to the site boundary that would experience the highest exposure. The North Warehouse contains a waste oil burner that can be used for the incineration of low level contaminated waste oil, and is designated as a ground level release point to the atmosphere. Due to differences in building cross sectional areas and resulting building wake effects, the North Warehouse atmospheric dispersion factors are conservative in comparison to those associated with the main plant facilities, such as the Turbine Building. As a consequence, any potential or unexpected ground level release from the Turbine Building or adjoining structures can utilize the above ground release dose assessment equations.

In the case of noble gas dose rates, Method II cannot provide much extra realism because R_{tbs} and R_{tbg} are already based on several factors which make use of current plant parameters. However, should it be needed, the dose rate analysis for critical receptor can be performed making use of current meteorology during the time interval of recorded peak release rate in place of the default atmospheric dispersion factor used in Method I.

6.4.3 Method II

If Method I cannot be applied, or if the Method I dose exceeds the limit, then Method II may be applied. Method II consists of the models, input data and assumptions in Regulatory Guide 1.109, Rev. 1 (Reference A), except where site-specific models, data or assumptions are more applicable. The base case analysis, documented above, is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

6.5 Method to Calculate the Skin Dose Rate from Noble Gases

Effluent Control 3.3.1 limits the instantaneous dose rate at any time to the skin from all release sources of noble gases at any location at or beyond the site boundary to 3,000 mrem/year.

Use Method I first to calculate the Skin Dose Rate from both elevated and ground level release points to the atmosphere. The dose rate limit of Control 3.3.1.a is the total contribution from both ground and elevated releases occurring during the period of interest. Method I applies at all release rates.

Use Method II if Method I predicts a dose rate greater than the Control limits (i.e., use of actual meteorology over the period of interest) to determine if, in fact, Control 3.3.1 had actually been exceeded during a short time interval.

Compliance with the dose rate limits for noble gases are continuously demonstrated when effluent release rates are below the plant stack noble gas activity monitor alarm setpoint by virtue of the fact that the alarm setpoint is based on a value which corresponds to the off-site Control dose rate limit, or a value below it, taking into account the potential contribution releases from all ground level sources.

Determinations of dose rate for compliance with Control (3.3.1) are performed when the effluent monitor alarm setpoint is exceeded and the corrective action required by Control 3.3.1 is unsuccessful, or as required by the notations to Control Table 3.1.2 when the stack noble gas monitor is inoperable.

6.5.1 Method I

The skin dose rate due to noble gases is determined by multiplying the individual radionuclide release rates by their respective dose factors, and summing all the products together as seen in the following Equation 6-7:

$$\dot{R}_{skins} = \sum_i \dot{Q}_i^{ST} \quad DF'_{is} \qquad (6\text{-}7)$$

$$\left(\frac{mrem}{yr}\right) \quad \left(\frac{\mu Ci}{sec}\right) \left(\frac{mrem\text{-}\ sec}{\mu Ci\text{-}\ yr}\right)$$

where:

$\dot{Q}_i^{ST}$ = In the case of noble gases, the noble gas release rate from the plant stack (μCi/sec) for each radionuclide, "i", identified. The release rate at the plant stack is based on measured radionuclide concentrations and distributions in periodic grab samples taken at the stack. As an alternative method, the radionuclide distribution in the off-gas at the Steam Jet Air Ejector (SJAE) can be used during plant operations, along with the Stack Gas Monitor effluent count rate, to estimate stack radionuclide releases. The release rate at the stack when using SJAE samples can be stated as follows:

$$\dot{Q}_i^{ST} = \frac{\dot{Q}_i^{SJAE}}{\sum_i \dot{Q}_i^{SJAE}} \quad M \quad \frac{1}{S_g} \quad F \qquad (6\text{-}28)$$

$$\frac{\mu Ci}{sec} = (cpm) \quad \frac{(\mu Ci/cc)}{cpm} \quad \frac{(cc)}{sec}$$

M = Plant stack gas monitor I or II count rate (cpm).

S_g = Appropriate or conservative plant stack monitor detector counting efficiency for the given nuclide mix (cpm/(μCi/cc)).

F = Stack flow rate (cc/sec).

$\dot{Q}_i^{SJAE}$ = The last measured release rate at the steam jet air ejector of noble gas i (μCi/sec).

DF'_{is} = combined skin dose factor (see Table 1.1.10) for stack release.

For ground level releases, the skin dose rate from noble gases is calculated by Equation 6-38:

$$\dot{R}_{sking} = \sum_i \dot{Q}_i^{GL} DF'_{ig} \qquad (6\text{-}38)$$

where:

Q_i^{GL} = The noble gas release rate from ground level (μCi/sec) for each radionuclide "i" identified.

DF'_{ig} = Combined skin dose factor for a ground level release [see Table 1.1.10A].

The skin dose rate for the site is equal to $R_{skins} + R_{sking}$.

During periods (beyond the first five days) when the plant is shutdown and no radioactivity release rates can be measured at the SJAE, Xe-133 may be used in place of the last SJAE measured mix as the referenced radionuclide to determine off-site dose rate and monitor setpoints. In this case, the ratio each of $\dot{Q}_i^{SJAE}$ to the sum of all $\dot{Q}_i^{SJAE}$ in Equation 6-28 above is assumed to reduce to a value of 1, and the combined skin dose factor DF'_{is} for Xe-133 (5.58 E-04 mrem-sec/μCi-year) is used in Equation 6-7. Alternately, a relative radionuclide "i" mix fraction (f_i) may be taken from Table 8.2.1 as a function of time after shutdown, and substituted in place of the ratio of each $\dot{Q}_i^{SJAE}$ to the sum of all $\dot{Q}_i^{SJAE}$ in Equation 6-28 above to determine the relative fraction of each noble gas potentially available for release to the total. Just prior to plant startup before a SJAE sample can be taken and analyzed, the monitor alarm setpoints should be based on Xe-138 as representing the most prevalent high dose factor noble gas expected to be present shortly after the plant returns to power. Monitor alarm setpoints which have been determined to be conservative under any plant conditions may be utilized at any time in lieu of the above assumptions.

Equations 6-7 and 6-38 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Normal operations (not emergency event), and

2. Noble gas releases via both elevated and ground level vents to the atmosphere.

6.5.2 Basis For Method I

The methods to calculate skin dose rate parallel the total body dose rate methods in Section 6.4.3. Only the differences are presented here.

Method I may be used to show that the Control limit for skin dose rate from noble gases released to the atmosphere (Control 3.3.1) has been met for the peak noble gas release rate.

Method I was derived from Regulatory Guide 1.109 as follows:

$$D^S = 1.11 \; S_F \; D^{\gamma}_{air} + 3.17E+04 \sum_i Q_i \; [X/Q]_S \; DFS_i \qquad (6\text{-}8)$$

$$\left(\frac{mrem}{yr}\right) = \left(\frac{mrem}{mrad}\right) (\#) \left(\frac{mrad}{yr}\right) \left(\frac{pCi-yr}{Ci-sec}\right) \left(\frac{Ci}{yr}\right) \left(\frac{sec}{m^3}\right) \left(\frac{mrem-m^3}{pCi-yr}\right)$$

where:

1.11 = Average ratio of tissue to air absorption coefficients will convert mrad in air to mrem in tissue.

$$D^{\gamma}_{air} = 3.17E+04 \sum_i Q_i \; [X/Q]_s \; DF^{\gamma}_i \qquad (6\text{-}9)$$

$$\left(\frac{mrad}{yr}\right) \left(\frac{pCi-yr}{Ci-sec}\right) \left(\frac{Ci}{yr}\right) \left(\frac{sec}{m^3}\right) \left(\frac{mrad-m^3}{pCi-yr}\right)$$

$$\text{now} \quad D^{\gamma}_{finite} = D^{\gamma}_{air} \; [X/Q]^{\gamma}_s \; / \; [X/Q]_s \qquad (6\text{-}10)$$

$$\left(\frac{mrad}{yr}\right) \left(\frac{mrad}{yr}\right) \left(\frac{sec}{m^3}\right) \left(\frac{m^3}{sec}\right)$$

and $Q_i = 31.54 \quad \dot{Q}_i^{ST}$ (6-11)

$$\left(\frac{Ci}{yr}\right) \quad \left(\frac{Ci-sec}{\mu Ci-yr}\right)\left(\frac{\mu Ci}{sec}\right)$$

so $\dot{R}_{skins} = 1.11 \quad S_F \; 1E+06 \; [X/Q]_s^{\gamma} \sum_i \dot{Q}_i^{ST} \quad DF_i^{\gamma}$ (6-12)

$$\left(\frac{mrem}{yr}\right)\left(\frac{mrem}{mrad}\right)(\#)\left(\frac{pCi}{\mu Ci}\right)\left(\frac{sec}{m^3}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrad-m^3}{pCi-yr}\right)$$

$+1E+06 \quad X/Q_s \sum_i \dot{Q}_i^{ST} \quad DFS_i$

$$\left(\frac{pCi}{Ci}\right) \quad \frac{sec}{m^3} \quad \frac{\mu Ci}{sec}\left(\frac{mrem-m^3}{pCi-yr}\right)$$

substituting

$[X/Q]_s^{\gamma}$ = 6.11E-07 sec/m^3
X/Q_s = 1.04E-06 sec/m^3
S_F = Shielding factor = 1.0 for dose rate determinations

gives

$$\dot{R}_{skins} = 0.68 \sum_i \dot{Q}_i^{ST} \quad DF_i^{\gamma} + 1.04 \sum_i \dot{Q}_i^{ST} \quad DFS_i \qquad (6\text{-}13)$$

$$\left(\frac{mrem}{yr}\right)\left(\frac{pCi-sec-mrem}{\mu Ci-m^3-mrad}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrad-m^3}{pCi-yr}\right)\left(\frac{pCi-sec}{\mu Ci-m^3}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem-m^3}{pCi-yr}\right)$$

$$= \sum_i \dot{Q}_i^{ST} [0.68\ DF_i^{\gamma} + 1.04\ DFS_i] \quad (6\text{-}14)$$

define

$$DF'_{is} = 0.68\ DF_i^{\gamma} + 1.04\ DFS_i \quad (6\text{-}15)$$

then

$$\dot{R}_{skins} = \sum_i \dot{Q}_i^{ST} \quad DF'_{is} \quad (6\text{-}7)$$

$$\left(\frac{mrem}{yr}\right) \quad \left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem-sec}{\mu Ci-yr}\right)$$

For determining combined skin doses for ground level releases, a $[X/Q]_g^{\gamma}$ = 6.42E-06 sec/m^3 and an undepleted X/Q_g = 3.52E-05 sec/m^3 have been substituted into Equation 6-12 to give:

$$\dot{R}_{sking} = \sum_i \dot{Q}_i^{GL} (7.13\ DF_i^{\gamma} + 35.2\ DF_i)$$

$$\text{then} \quad DF'_{ig} = 7.13\ DF_i^{\gamma} + 35.2\ DFS_i \quad (6\text{-}37)$$

$$\text{and} \quad \dot{R}_{sking} = \sum_i Q_i^{GL}\ DF'_{ig} \quad (6\text{-}38)$$

where:

$\dot{Q}_i^{GL}$ = The noble gas release rate from ground level release points (μCi/sec) for each radionuclide "i" identified.

DF'_{ig} = Combined skin dose factor for a ground level release [see Table 1.1.10A].

The selection of critical receptor, outlined in Section 6.10 is inherent in Method I, as it determined the maximum expected off-site atmospheric dispersion factors based on past long-term site-specific meteorology.

The calculation of ground level release dispersion parameters are based on the location of the North Warehouse with respect to the site boundary that would experience the highest exposure. The North Warehouse contains a waste oil burner that can be used for the incineration of low level contaminated waste oil, and is designated as a ground level release point to the atmosphere. Due to differences in building cross sectional areas and resulting building wake effects, the North Warehouse atmospheric dispersion factors are conservative in comparison to those associated with the main plant facilities, such as the Turbine Building. As a consequence, any potential or unexpected ground level release from the Turbine Building or adjoining structures can utilize the above ground release dose assessment equations.

6.5.3 Method II

If Method I cannot be applied, or if the Method I dose exceeds the limit, then Method II may be applied. Method II consists of the models, input data and assumptions in Regulatory Guide 1.109, Rev. 1 (Reference A), except where site-specific models, data or assumptions are more applicable. The base case analysis, documented above, is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

6.6 Method to Calculate the Critical Organ Dose Rate from Iodines, Tritium and Particulates with $T_{1/2}$ Greater Than 8 Days

Effluent Control 3.3.1.b limits the dose rate to any organ, denoted $\dot{R}_{co}$, from all release sources of I-131, I-133, H-3, and radionuclides in particulate form with half lives greater than 8 days to 1500 mrem/year to any organ. The peak release rate averaging time in the case of iodines and particulates is commensurate with the time the iodine and particulate samplers are in service between changeouts (typically a week).

Use Method I first to calculate the critical organ dose rate from both elevated and ground level release points to the atmosphere. The dose rate limit of Control 3.3.1.b is the total contribution from both ground and elevated releases occurring during the period of interest. Method I applies at all release rates.

Use Method II if Method I predicts a dose rate greater than the Control limits (i.e., use of actual meteorology over the period of interest) to determine if, in fact, Control 3.3.1.b had actually been exceeded during the sampling period.

6.6.1 Method I

The critical organ dose rate from stack releases can be determined by multiplying the individual radionuclide release rates by their respective dose factors and summing all their products together, as seen in the following Equation 6-16:

$$\dot{R}_{COS} = \sum_i \dot{Q}_i^{STP} \quad DFG'_{sico} \qquad (6\text{-}16)$$

$$\left(\frac{\text{mrem}}{\text{yr}}\right) \quad \left(\frac{\mu\text{Ci}}{\text{sec}}\right) \left(\frac{\text{mrem - sec}}{\mu\text{Ci - yr}}\right)$$

where:

$\dot{Q}_i^{STP}$ = Stack activity release rate determination of radionuclide "i" (Iodine-131, Iodine-133, particulates with half-lives greater than 8 days, and tritium), in μCi/sec. For i = Sr89, Sr90 or tritium, use the best estimates (such as most recent measurements).

DFG'_{sico} = Site specific critical organ dose rate factor $\left(\frac{\text{mrem - sec}}{\mu\text{Ci - yr}}\right)$ for a ground level gaseous release. See Table 1.1.12.

For ground releases (North Warehouse waste oil burner) the critical organ dose rate from Iodine, Tritium, and Particulates with T 1/2 greater than 8 days is calculated as follows:

$$\dot{R}_{cog} = \sum_i \dot{Q}_i^{GLP} \quad DFG'_{gico} \qquad (6\text{-}40)$$

where:

$\dot{Q}_i^{GLP}$ = Ground activity release rate determination of radionuclide "i" (Iodine-131, Iodine-133, particulates with half-lives greater than 8 days, and tritium), in μCi/sec. For i = Sr89, Sr90, Fe-55, or tritium, use the best estimates (such as most recent measurements). For waste oil, the release rate is the total activity by radionuclide divided by the estimated burn time. (See Appendix D for surveillance criteria on waste oil burning.

DFG'_{gico} = Site specific critical organ dose rate factor $\left(\frac{\text{mrem - sec}}{\mu\text{Ci - yr}}\right)$ for a ground level gaseous release. See Table 1.1.12.

The critical organ dose rate for the site is equal to $\dot{R}_{cos} + \dot{R}_{cog}$.

Equations 6-16 and 6-40 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Normal operations (not emergency event), and

2. Tritium, iodine, and particulate releases via either elevated or ground level vents to the atmosphere.

6.6.2 Basis for Method I

The methods to calculate critical organ dose rate parallel the total body dose rate methods in Section 6.4.3. Only the differences are presented here.

Method I may be used to show that the Control limit for organ dose rate from iodines, tritium and radionuclides in particulate form with half lives greater than 8 days (hereafter called Iodines and Particulates or "I+P") released to the atmosphere (Control 3.3.1.b) has been met for the peak I + P release rate.

The equation for $\dot{R}_{cos}$ and $\dot{R}_{cog}$ is derived by modifying Equation 6-25 from Section 6.9 as follows:

$$\dot{D}_{cos} = \sum_i Q_i \quad DFG_{ico} \qquad (6\text{-}17)$$

(mrem) (Ci) $\left(\frac{\text{mrem}}{\text{Ci}}\right)$

applying the conversion factor, 31.54 (Ci-sec/μCi-yr) and converting Q to $\dot{Q}$ in μCi/sec as it applies to the plant stack yields:

$$\dot{R}_{cos} = 31.54 \sum_i \dot{Q}_i^{STP} \quad DFG_{sico} \qquad (6\text{-}18)$$

$\left(\frac{\text{mrem}}{\text{yr}}\right)$ $\left(\frac{\text{Ci-sec}}{\mu\text{Ci-yr}}\right)$ $\left(\frac{\mu\text{Ci}}{\text{sec}}\right)$ $\left(\frac{\text{mrem}}{\text{Ci}}\right)$

Equation 6.8 is written in the form:

$$\dot{R} = 31.54 \sum_i \dot{Q}_i^{STP} \quad DFG_{sico} \qquad (6\text{-}19)$$

$\left(\frac{\text{mrem}}{\text{yr}}\right)$ $\left(\frac{\text{Ci-sec}}{\mu\text{Ci-yr}}\right)$ $\left(\frac{\mu\text{Ci}}{\text{sec}}\right)$ $\left(\frac{\text{mrem}}{\text{Ci}}\right)$

DFG'_{sico} and DFG'_{gico} (North Warehouse waste oil burner vent releases) incorporates the conversion constant of 31.54 and has assumed that the shielding factor (S_F) applied to the direct exposure pathway from radionuclides deposited on the ground plane is equal to 1.0 in place of the S_F value of 0.7 assumed in the determination of DFG_{sico} and DFG_{gico} for the integrated doses over time.

The selection of critical receptor (based on the combination of exposure pathways which include direct dose from the ground plane, inhalation and ingestion of vegetables, meat, and milk) which is outlined in Section 6.10 is inherent in Method I, as are the maximum expected off-site atmospheric dispersion factors based on past long-term site-specific meteorology.

The calculation of ground level release dispersion parameters are based on the location of the North Warehouse with respect to the site boundary that would experience the highest exposure. The North Warehouse contains a waste oil burner that can be used for the incineration of low level contaminated waste oil, and is designated as a ground level release point to the atmosphere. Due to differences in building cross sectional areas and resulting building wake effects, the North Warehouse atmospheric dispersion factors are conservative in comparison to those associated with the main plant facilities, such as the Turbine Building. As a consequence, any potential or unexpected ground level release from the Turbine Building or adjoining structures can utilize the above ground release dose assessment equations.

Should Method II be needed, the analysis for critical receptor critical pathway(s) and atmospheric dispersion factors may be performed with actual meteorologic and latest land use census data to identify the location of those pathways which are most impacted by these type of releases.

6.6.3 Method II

If Method I cannot be applied, or if the Method I dose exceeds the limit, then Method II may be applied. Method II consists of the models, input data and assumptions in Regulatory Guide 1.109, Rev. 1 (Reference A), except where site-specific models, data or assumptions are more applicable. The base case analysis, documented above, is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

6.7 Method to Calculate the Gamma Air Dose from Noble Gases

Effluent Control 3.3.2 limits the gamma dose to air from all release sources of noble gases at any location at or beyond the site boundary to 5 mrad in any quarter and 10 mrad in any year. Dose evaluation is required at least once per month.

Use Method I first to calculate the gamma air dose for elevated and ground level vent releases during the period. The total gamma air dose limit of Control 3.3.2 is the total contribution from both ground and elevated releases occurring during the period of interest.

Use Method II if a more accurate calculation is needed.

6.7.1 Method I

The gamma air dose from plant stack releases is:

$$D^{\gamma}_{airs} = 0.019 \sum_i Q_i^{ST} \quad D_i^{\gamma} \qquad (6\text{-}21)$$

$$(\text{mrad}) \quad \left(\frac{\text{pCi-yr}}{\text{Ci-m}^3}\right) \quad (\text{Ci}) \left(\frac{\text{mrad-m}^3}{\text{pCi-yr}}\right)$$

where:

Q_i^{ST} = total noble gas activity (Curies) released to the atmosphere via the plant stack of each radionuclide "i" during the period of interest.

DF_i^{γ} = gamma dose factor to air for radionuclide "i". See Table 1.1.10.

For ground level noble gas releases, the gamma air dose is calculated as follows:

$$D^{\gamma}_{airg} = 0.20 \sum_i Q_i^{GL} \, DF^{\gamma}_i \qquad (6\text{-}41)$$

where:

Q_i^{GL} = Total noble gas activity (curies) released to the atmosphere via ground level vents of each radionuclide, "i", during the period of interest.

The gamma air dose for the site is equal to $D_{airs}^{\gamma} + D_{airg}^{\gamma}$.

Equations 6-21 and 6-41 can be applied under the following conditions (otherwise justify Method I or consider Method II):

1. Normal operations (not emergency event), and

2. Noble gas releases via either elevated or ground level vents to the atmosphere.

6.7.2 Basis for Method I

Method I may be used to show that the Control limit for off-site gamma air dose from gaseous effluents (3.3.2) has been met for releases over appropriate periods. This Control is based on the Objective in 10CFR50, Appendix I, Subsection B.1, which limits the estimated annual gamma air dose at unrestricted area locations.

Exceeding the Objective does not immediately limit plant operation but requires a report to the NRC.

For any noble gas release, in any period, the dose is taken from Equations B-4 and B-5 of Regulatory Guide 1.109 with the added assumption that $D_{finite}^{\gamma} = D^{\gamma}\ [X/Q]^{\gamma}/[X/Q]$:

$$D_{airs}^{\gamma} = 3.17E+04\ [X/Q]_s^{\gamma} \quad \sum_i Q_i^{ST} DF_i^{\gamma} \qquad (6\text{-}22)$$

$$(\text{mrad})\left(\frac{\text{pCi-yr}}{\text{Ci-sec}}\right)(\text{sec/m}^3)\,(\text{Ci})\left(\frac{\text{mrad-m}^3}{\text{yr-pCi}}\right)$$

where:

$[X/Q]_s^{\gamma}$ = maximum long term average gamma atmospheric dispersion factor for a stack release.

= 6.11E-07 (sec/m^3)

Q_i^{ST} = number of curies of noble gas "i" released from the plant stack

which leads to:

$$D_{airs}^{\gamma} = 0.019 \sum_i Q_i^{ST} DF_i^{\gamma} \quad \text{(6-21)}$$

$$\text{(mrad)} \left(\frac{\text{pCi} - \text{yr}}{\text{Ci} - \text{m}^3}\right) \text{(Ci)} \left(\frac{\text{mrad} - \text{m}^3}{\text{pCi} - \text{yr}}\right)$$

For the ground level release:

$$D_{airg}^{\gamma} = 3.17E+04\ [X/Q]_g^{\gamma} \sum_i Q_i^{GL} DF_i^{\gamma} \quad \text{(6-42)}$$

where:

$(X/Q)_g^{\gamma}$ = Maximum long-term average gamma atmospheric dispersion factor for a ground level release

= 6.42E-06 sec/m^3

leading to:

$$D_{airg}^{\gamma} = 0.20 \sum_i Q_i^{GL} DF_i^{\gamma} \quad \text{(6-41)}$$

The calculation of ground level release dispersion parameters are based on the location of the North Warehouse with respect to the site boundary that would experience the highest exposure. The North Warehouse contains a waste oil burner that can be used for the incineration of low level contaminated waste oil, and is designated as a ground level release point to the atmosphere. Due to differences in building cross sectional areas and resulting building wake effects, the North Warehouse atmospheric dispersion factors are conservative in comparison to those associated with the main plant facilities, such as the Turbine Building. As a consequence, any potential or unexpected ground level release from the Turbine Building or adjoining structures can utilize the above ground release dose assessment equations.

The main difference between Method I and Method II is that Method II would allow the use of actual meteorology to determine $[X/Q]^{\gamma}$ rather than use the maximum long-term average value obtained for the years 1981 to 1985.

6.7.3 Method II

If the Method I dose determination indicates that the Control limit may be exceeded, or if a more exact calculation is required, then Method II may be applied. Method II consists of the models, input data and assumptions in Regulatory Guide 1.109, Rev. 1 (Reference A), except where site-specific models, data or assumptions are more applicable.

6.8 Method to Calculate the Beta Air Dose from Noble Gases

Effluent Control 3.3.2 limits the beta dose to air from all release sources of noble gases at any location at or beyond the site boundary to 10 mrad in any quarter and 20 mrad in any year. Dose evaluation is required at least once per month.

Use Method I first to calculate the beta air dose for elevated and ground level vent releases during the period. The total beta air dose limit of Control 3.3.2 is the total contribution from both ground and elevated releases occurring during the period of interest.

Use Method II if a more accurate calculation is needed or if Method I cannot be applied.

6.8.1 Method I

The beta air dose from plant vent stack releases is:

$$D_{airs}^{\beta} = 0.033 \sum_{i} Q_i^{ST} \, DF_i^{\beta} \qquad (6\text{-}23)$$

$$(\text{mrad}) \left(\frac{\text{pCi-yr}}{\text{Ci-m}^3} \right) \quad (\text{Ci}) \left(\frac{\text{mrad-m}^3}{\text{pCi-yr}} \right)$$

where:

DF_i^{β} = beta dose factor to air for radionuclide "i". See Table 1.1.10.

Q_i^{ST} = total noble gas activity (curies) released to the atmosphere via the plant stack of each radionuclide "i" during the period of interest.

For ground level noble gas releases, the beta air dose is calculated as follows:

$$D^{\beta}_{air_g} = 1.12 \sum_i Q_i^{GL} \quad DF_i^{\beta} \qquad (6\text{-}43)$$

where:

Q_i^{GL} = Total noble gas activity (curies) released to the atmosphere via the ground level vents of each radionuclide "i" during the period of interest.

The beta air dose for the site is equal to $D^{\beta}_{airs} + D^{\beta}_{airg}$.

Equations 6-23 and 6-43 can be applied under the following conditions (otherwise justify Method I or consider Method II):

1. Normal operations (not emergency event), and

2. Noble gas releases via either elevated or ground level vents to the atmosphere.

6.8.2 Basis for Method I

This section serves three purposes: (1) to document that Method I complies with appropriate NRC regulations, (2) to provide background and training information to Method I users, and (3) to provide an introductory user's guide to Method II. The methods to calculate beta air dose parallel the gamma air dose methods in Section 6.7.3. Only the differences are presented here.

Method I may be used to show that the Control limit for off-site beta air dose from gaseous effluents (3.3.2) has been met for releases over appropriate periods. This Control is based on the Objective in 10CFR50, Appendix I, Subsection B.1, which limits the estimated annual beta air dose at unrestricted area locations.

Exceeding the Objective does not immediately limit plant operation but requires a report to the NRC within 30 days.

For any noble gas release, in any period, the dose is taken from Equations B-4 and B-5 of Regulatory Guide 1.109:

$$D^{\beta}_{airs} = 3.17E+04 \quad X/Q_s \sum_i Q_i^{ST} \quad DF_i^{\beta} \qquad (6\text{-}24)$$

$$(\text{mrad}) \quad \left(\frac{\text{pCi - yr}}{\text{Ci - sec}}\right)\left(\frac{\text{sec}}{\text{m}^3}\right) \quad (\text{Ci}) \quad \left(\frac{\text{mrad - m}^3}{\text{pCi - yr}}\right)$$

substituting

X/Q_s = Maximum long term average undepleted atmospheric dispersion factor for a stack release.

= 1.04E-06 sec/m^3

We have (6-23)

$$D^{\beta}_{airs} = 0.033 \sum_i Q_i^{ST} \quad DF_i^{\beta}$$

$$(\text{mrad}) \quad \left(\frac{\text{pCi - yr}}{\text{Ci - m}^3}\right) \quad (\text{Ci}) \quad \left(\frac{\text{mrad - m}^3}{\text{pCi - yr}}\right)$$

For the ground level release:

$$D^{\beta}_{airg} = 3.17E+04 \ (X/Q)_g \sum_i Q_i^{GL} \ DF_i^{\beta} \qquad (6\text{-}44)$$

where:

$(X/Q)_g$ = Maximum long-term average undepleted atmospheric dispersion factor for a ground level release.

$$= \quad 3.52E\text{-}05 \text{ sec/m}^3$$

leading to:

$$D^{\beta}_{airg} = 1.12 \sum_{i} Q^{GL}_{i} \quad DF^{\beta}_{i} \qquad (6\text{-}43)$$

The calculation of ground level release dispersion parameters are based on the location of the North Warehouse with respect to the site boundary that would experience the highest exposure. The North Warehouse contains a waste oil burner that can be used for the incineration of low level contaminated waste oil, and is designated as a ground level release point to the atmosphere. Due to differences in building cross sectional areas and resulting building wake effects, the North Warehouse atmospheric dispersion factors are conservative in comparison to those associated with the main plant facilities, such as the Turbine Building. As a consequence, any potential or unexpected ground level release from the Turbine Building or adjoining structures can utilize the above ground release dose assessment equations.

6.8.3 Method II

If Method I cannot be applied, or if the Method I dose determination indicates that the Control limit may be exceeded, or if a more exact calculation is required, then Method II may be applied. Method II consists of the models, input data and assumptions in Regulatory Guide 1.109, Rev. 1 (Reference A), except where site-specific models, data or assumptions are more applicable.

6.9 Method to Calculate the Critical Organ Dose from Iodines, Tritium and Particulates

Effluent Control 3.3.3 limits the critical organ dose to a Member of the Public from all release sources of I-131, I-133, Tritium, and particulates with half-lives greater than 8 days (hereafter called "I+P") in gaseous effluents to 7.5 mrem per quarter and 15 mrem per year.

Use Method I first to calculate the critical organ dose from both elevated and ground level vent releases. The total critical organ dose limit of Control 3.3.3 is the total contribution from both ground level and elevated releases occurring during the period of interest.

Use Method II if a more accurate calculation of critical organ dose is needed (i.e., Method I indicates the dose is greater than the limit).

6.9.1 Method I

$$D_{cos} = \sum_i Q_i^{STP} \; DFG_{sico} \qquad (6\text{-}25)$$

(mrem) (Ci) $\left(\frac{\text{mrem}}{\text{Ci}}\right)$

Q_i^{STP} = Total activity (Ci) released from the stack to the atmosphere of radionuclide "i" during the period of interest. For strontiums and tritium, use the most recent measurement.

DFG_{sico} = Site-specific critical organ dose factor for a stack gaseous release of radionuclide "i" (mrem/Ci). For each radionuclide it is the age group and organ with the largest dose factor. See Table 1.1.12.

The critical organ dose is calculated for ground level releases as follows:

$$D_{cog} = \sum_i Q_i^{GLP} \; DFG_{gico} \qquad (6\text{-}44)$$

(mrem) (Ci) $\left(\frac{\text{mrem}}{\text{Ci}}\right)$

Q_i^{GLP} = Total activity (Ci) released from ground level vents to the atmosphere of radionuclide "i" during the period of interest. For tritium, strontiums, and Fe-55 use the most recent measure.

DFG_{gico} = Site-specific critical organ dose factor for a ground level release of nuclide "i" (mrem/Ci). For each radionuclide it is the age group and organ with the largest dose factor. See Table 1.1.12.

The critical organ dose for the site is equal to $D_{cos} + D_{cog}$.

Equations 6-25 and 6-44 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Normal operations (not emergency event),

2. I+P releases via the plant stack, Turbine Building, and waste oil burner (see Appendix D for surveillance criteria on waste oil burning), to the atmosphere, and

3. Any continuous or batch release over any time period.

6.9.2 Basis for Method I

This section serves three purposes: (1) to document that Method I complies with appropriate NRC regulations, (2) to provide background and training information to Method I users, and (3) to provide an introductory user's guide to Method II.

Method I may be used to show that the Control limit for off-site organ dose from gases (3.3.3) has been met for releases over the appropriate periods.

Method I was developed such that "the actual exposure of an individual ... is unlikely to be substantially underestimated" (10CFR50, Appendix I). The use below of a single "critical receptor" provides part of the conservative margin to the calculation of critical organ dose in Method I. Method II allows that actual individuals, with real behaviors, be taken into account for any given release. In fact, Method I was based on a Method II analysis of the critical receptor for the annual average conditions. For purposes of complying with the Control 3.3.3, maximum annual average atmospheric dispersion factors are appropriate for batch and continuous releases. That analysis was called the "base case"; it was then reduced to form Method I. The base case, the method of reduction, and the assumptions and data used are presented below.

The steps performed in the Method I derivation follow. First, in the base case, the dose impact to the critical receptor in the form of dose factors (mrem/Ci) of 1 curie release of each I+P radionuclide to gaseous effluents was derived. Then Method I was determined using simplifying and further conservative assumptions. The base case analysis uses the methods, data and assumptions in Regulatory Guide 1.109 (Equations C-2, C-4 and C-13 in Reference A). Tables 6.9.1 and 6.9.2 outline human consumption and environmental parameters used in the analysis. It is conservatively assumed that the critical receptor lives at the "maximum off-site atmospheric dispersion factor location" as defined in Section 6.10. However, he is exposed, conservatively, to all pathways (see Section 6.10). The resulting site-specific dose factors are for the maximum organ and the age group with the highest dose factor for that organ. These critical organ, critical age dose factors are given in Table 1.1.12.

For any gas release, during any period, the increment in annual average dose from radionuclide "i" is:

$$\Delta D_{ico} = Q_i DFG_{ico} \qquad (6\text{-}26)$$

where DFG_{ico} is the critical dose factor for radionuclide "i" and Q_i is the activity of radionuclide "i" released in curies.

Method I is more conservative than Method II in the region of the effluent dose Control limit because it is based on the following reduction

of the base case. The dose factors DFG_{ico} used in Method I were chosen from the base case to be the highest of the set for that radionuclide. In effect each radionuclide is conservatively represented by its own critical age group and critical organ.

6.9.3 METHOD II

If Method I cannot be applied, or if the Method I dose exceeds the Control limit or if a more exact calculation is required, then Method II should be applied. Method II consists of the models, input data and assumptions in Regulatory Guide 1.109, Rev. 1 (Reference A), except where site-specific models, data or assumptions are more applicable. The base case analysis, documented above, is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

TABLE 6.9.1

Environmental Parameters for Gaseous Effluents at Vermont Yankee
(Derived from Reference A)*

Variable			Vegetables		Cow Milk		Goat Milk		Meat	
			Stored	Leafy	Pasture	Stored	Pasture	Stored	Pasture	Stored
YV	Agricultural Productivity	(Kg/m^2)	2	2	0.70	2	0.70	2	0.70	2
P	Soil Surface Density	(Kg/m^2)	240	240	240	240	240	240	240	240
T	Transport Time to User[5]	(Hrs)			48	48	48	48	480	480
TB	Soil Exposure Time[1]	(Hrs)	131400	131400	131400	131400	131400	131400	131400	131400
TE	Crop Exposure Time to Plume	(Hrs)	1440	1440	720	1440	720	1440	720	1440
TH	Holdup After Harvest	(Hrs)	1440	24	0	2160	0	2160	0	2160
QF	Animals Daily Feed	(Kg/Day)			50	50	6	6	50	50
FP	Fraction of Year on Pasture[2]				0.50		0.50		0.50	
FS	Fraction Pasture When on Pasture[3]				1		1		1	
FG	Fraction of Stored Veg. Grown in Garden		0.76							
FL	Fraction of Leafy Veg. Grown in Garden			1						
FI	Fraction Elemental Iodine = 0.5									
A	Absolute Humidity = 5.6 (gm/m^3)[4]									

*Regulatory Guide 1.109, Revision 1.

TABLE 6.9.1
(Continued)

Notes:

(1) For Method II dose/dose rate analyses of identified radioactivity releases of less than one year, the soil exposure time for that release may be set at 8760 hours (1 year) for all pathways.

(2) For Method II dose/dose rate analyses performed for releases occurring during the first or fourth calendar quarters, the fraction of time animals are assumed to be on pasture is zero (nongrowing season). For the second and third calendar quarters, the fraction of time on pasture (FP) will be set at 1.0. FP may also be adjusted for specific farm locations if this information is so identified and reported as part of the land use census.

(3) For Method II analyses, the fraction of pasture feed while on pasture may be set to less than 1.0 for specific farm locations if this information is so identified and reported as part of the land use census.

(4) For all Method II analyses, an absolute humidity value equal to 5.6 (gm/m^3) shall be used to reflect conditions in the Northeast (Reference: Health Physics Journal, Vol. 39 (August), 1980; Page 318-320, Pergammon Press).

(5) Variable T is a combination of variables TF and TS in Regulatory Guide 1.109, Revision 1.

TABLE 6.9.2

Usage Factors for Various Gaseous Pathways at Vermont Yankee
(from Regulatory Guide 1.109, Table E-5)

Age Group	Vegetables (kg/yr)	Leafy Vegetables (kg/yr)	Milk (l/yr)	Meat (kg/yr)	Inhalation (m^3/yr)
Adult	520.00	64.00	310.00	110.00	8000.00
Teen	630.00	42.00	400.00	65.00	8000.00
Child	520.00	26.00	330.00	41.00	3700.00
Infant	0.00	0.00	330.00	0.00	1400.00

6.10 Receptor Point and Long-Term Average Atmospheric Dispersion Factors for Important Exposure Pathways

The gaseous effluent dose methods have been simplified by assuming an individual whose behavior and living habits inevitably lead to a higher dose than anyone else. The following exposure pathways to gaseous effluents listed in Regulatory Guide 1.109 (Reference A) have been considered for radioiodines, tritium, and particulates with half lives greater than 8 days:

1. Direct exposure to contaminated ground;
2. Inhalation of air;
3 Ingestion of vegetables;
4. Ingestion of cow's milk; and
5. Ingestion of meat.

Beta and gamma air doses have also been considered for noble gases in plant effluents along with whole body and skin dose rate calculations.

Section 6.10.1 details the selection of important off-site locations and receptors. Section 6.10.2 describes the atmospheric model used to convert meteorological data into atmospheric dispersion factors. Section 6.10.3 presents the maximum atmospheric dispersion factors calculated at each of the off-site receptor locations.

6.10.1 Receptor Locations

Distances to the site boundary from the two evaluated gaseous release pathways (the Stack and North Warehouse) are provided in Table 6.10.2. Four important off-site receptor locations are considered in the dose and dose rate equations for gaseous radioactive effluents from these two release pathways. They are:

1. The point of maximum gamma exposure (maximum gamma X/Q) from an overhead noble gas cloud for determining skin and whole body dose rates and gamma air doses;

2. The point of maximum ground level air concentration (maximum undepleted X/Q) of noble gases for determining skin and beta air dose rates and doses;

3. The point of maximum ground level air concentration (maximum depleted X/Q) of radioiodines and other particulates for determining critical organ dose from inhalation; and

4. The point of maximum deposition (maximum D/Q) of radioiodines and other particulates for determining critical organ dose from ingestion.

The Stack release pathway was evaluated as an elevated release assuming a constant nominal Stack flow rate of 175,000 cfm. The point of maximum gamma exposure from Stack releases (SSE sector, 750 meters) was determined by finding the maximum five-year average gamma X/Q at any off-site location. The location of the maximum ground level air concentration and deposition of radioiodines and other particulates (NW sector, 2700 meters) was determined by finding the maximum five-year average depleted X/Q and D/Q at any off-site location. For the purposes of determining the Method I dose factors for radioiodines, tritium, and particulates, a milk animal was assumed to exist at the location of highest calculated ground level air concentration and deposition of radioiodines and other particulates as noted above. This location then conservatively bounds the deposition of radionuclides at all real milk animal locations.

The North Warehouse release pathway was evaluated as a ground level release using the same meteorological period-of-record as the stack. The highest long-term atmospheric dispersion factors at the site boundary were determined (see Table 6.10.1) and doses and dose rates to the critical off-site receptor were calculated assuming the highest site boundary atmospheric dispersion factors all occurred at the same location.

6.10.2 Vermont Yankee Atmospheric Dispersion Model

The long-term average atmospheric dispersion factors are computed for routine releases using AEOLUS-2 Computer Code (Reference B). AEOLUS-2 is based, in part, on the constant mean wind direction model discussed in Regulatory Guide 1.111 (Reference C). Since AEOLUS-2 is a straight-line steady-state model, site-specific recirculation correction factors were developed for each release pathway to adjust the AEOLUS-2 results to account for temporal variations of atmospheric transport and diffusion conditions. The applicable recirculation correction factors are listed in Table 6.10.3.

AEOLUS-2 produces the following average atmospheric dispersion factors for each location:

1. Undepleted X/Q dispersion factors for evaluating ground level concentrations of noble gases;

2. Depleted X/Q dispersion factors for evaluating ground level concentrations of radioiodines and other particulates;

3. Gamma X/Q dispersion factors for evaluating gamma dose rates from a sector averaged finite cloud (undepleted source); and

4. D/Q deposition factors for evaluating dry deposition of elemental radioiodines and other particulates.

The North Warehouse depleted X/Q and D/Q factors were derived using the plume depletion and deposition curves provided in Regulatory Guide 1.111. However, because the Regulatory Guide 1.111 depletion and deposition curves are limited to an effective release height of 100 meters or less and the Vermont Yankee Stack effective release height (stack height plus plume rise) can exceed 100 meters, the Stack depleted X/Q and D/Q factors were derived using the deposition velocity concept presented in "Meteorology and Atomic Energy - 1968" (Reference E, Section 5-3.2) , assuming a constant deposition velocity of 1 cm/sec.

Gamma dose rate is calculated throughout this ODCM using the finite cloud model presented in "Meteorology and Atomic Energy - 1968 (Reference E, Section 7 5.2.5). That model is implemented through the definition of an effective gamma atmospheric dispersion factor, $[X/Q^{\gamma}]$ (Reference B, Section 4), and the replacement of X/Q in infinite cloud dose equations by the $[X/Q^{\gamma}]$.

6.10.3 Long-Term Average Atmospheric Dispersion Factors for Receptors

Actual measured meteorological data for the five-year period, 1988 through 1992, were analyzed to determine all the values and locations of the maximum off-site long-term average atmospheric dispersion factors. Each dose and dose rate calculation incorporates the maximum applicable off-site long-term average atmospheric dispersion factor. The values used and their locations are summarized in Table 6.10.1. Table 6.10.1 also indicates which atmospheric dispersion factors are used to calculate the various doses or dose rates of interest.

TABLE 6.10.1

Atmospheric Dispersion Factors

Release Pathway	Dispersion Factor	Dose to Individual			Dose to Air	
		Total Body	Skin	Critical Organ	Gamma	Beta
Stack	X/Q Depleted (sec/m^3)	-	-	9.40E-07 (2700m NW)	-	-
	X/Q Undepleted (sec/m^3)	-	1.04E-06 (2200m WNW)	-	-	1.04E-06 (2200m WNW)
	D/Q ($1/m^2$)	-	-	9.40E-09 (2700m NW)	-	-
	X/Q^{γ} (sec/m^3)	6.11E-07 (750m SSE)	6.11E-07 (750m SSE)	-	6.11E-07 (750m SSE)	-
North Warehouse	X/Q Depleted (sec/m^3)	-	-	3.32E-05 (417m NE)	-	-
	X/Q Undepleted (sec/m^3)	-	3.52E-05 (417m NE)	-	-	3.52E-05 (417m NE)
	D/Q ($1/m^2$)	-	-	5.97E-08 (357m S)	-	-
	X/Q^{γ} (sec/m^3)	6.42E-06 (417m NE)	6.42E-06 (417m NE)	-	6.42E-06 (417m NE)	-

TABLE 6.10.2

Site Boundary Distances

Downwind Sector	Stack Releases	North Warehouse Releases
N	400 m	459 m
NNE	350 m	417 m
NE	350 m	417 m
ENE	400 m	451 m
E	500 m	570 m
ESE	700 m	561 m
SE	750 m	612 m
SSE	850 m	663 m
S	385 m	357 m
SSW	300 m	238 m
SW	250 m	213 m
WSW	250 m	213 m
W	300 m	221 m
WNW	400 m	281 m
NW	550 m	697 m
NNW	550 m	680 m

TABLE 6.10.3

Recirculation Correction Factors

A. Stack Releases

Sector	0.5 Mi	1.5 Mi	2.5 Mi	3.5 Mi	4.5 Mi	7.5 Mi
N	1.4	1.4	1.2	1.1	1.0	1.0
NNE	1.8	1.8	1.4	1.2	1.0	1.0
NE	1.8	1.8	1.3	1.1	1.0	1.0
ENE	2.1	2.1	1.4	1.2	1.0	1.0
E	1.7	1.7	1.2	1.0	1.0	1.0
ESE	1.5	1.5	1.3	1.1	1.0	1.0
SE	1.8	1.8	1.3	1.2	1.1	1.0
SSE	1.4	1.4	1.2	1.2	1.2	1.2
S	1.3	1.3	1.1	1.1	1.2	1.2
SSW	1.8	1.8	1.5	1.4	1.4	1.2
SW	2.1	2.1	1.7	1.6	1.4	1.1
WSW	2.4	2.4	1.9	1.6	1.5	1.1
W	1.8	1.8	1.5	1.4	1.3	1.0
WNW	1.8	1.8	1.7	1.5	1.4	1.3
NW	1.5	1.5	1.3	1.3	1.3	1.1
NNW	1.5	1.5	1.2	1.2	1.1	1.1

B. North Warehouse Release

Sector	0.5 Mi	1.5 Mi	2.5 Mi	3.5 Mi	4.5 Mi	7.5 Mi
N	1.1	1.1	1.1	1.1	1.1	1.0
NNE	1.2	1.2	1.2	1.1	1.1	1.0
NE	1.1	1.2	1.1	1.1	1.0	1.0
ENE	1.2	1.3	1.4	1.4	1.4	1.3
E	1.1	1.3	1.4	1.4	1.4	1.2
ESE	1.1	1.1	1.2	1.1	1.1	1.0
SE	1.0	1.1	1.1	1.1	1.1	1.1
SSE	1.2	1.2	1.2	1.2	1.2	1.2
S	1.0	1.0	1.0	1.0	1.0	1.0
SSW	1.0	1.1	1.0	1.0	1.0	1.0
SW	1.2	1.3	1.2	1.0	1.0	1.0
WSW	1.1	1.1	1.0	1.0	1.0	1.0
W	1.2	1.2	1.1	1.0	1.0	1.0
WNW	1.2	1.4	1.3	1.2	1.2	1.0
NW	1.1	1.1	1.0	1.0	1.0	1.0
NNW	1.1	1.2	1.2	1.2	1.2	1.1

6.11 Method to Calculate Direct Dose From Plant Operation

Effluent Control 3.4.1 (40CFR190) restricts the dose to the whole body or any organ to any member of the public from all station sources (including direct radiation from fixed sources on-site) to 25 mrem in a calendar year (except the thyroid, which is limited to 75 mrem).

6.11.1 Turbine Building

The maximum contribution of direct dose to the whole body or to any organ due to N-16 decay from the turbine is:

$$D_d = K_{N16}(L) \times E \qquad (6\text{-}27)$$

$$(\text{mrem}) \qquad \frac{(\text{mrem})}{MW_e h} \qquad (MW_e h)$$

where:

D_d = The dose contribution from N-16 decay at either the site boundary of maximum impact (west site boundary) or maximum off-site residence - (mrem).

E = Gross electric output over the period of interest ($MW_e h$).

$K_{N16}(L)$ = The N-16 dose conversion factor for (L) equal to either:

(1) 3.39E-06 for the maximum west site boundary; or

(2) 2.63E-06 for the maximum residence (i.e., SW site boundary with respect to Turbine Hall)(mrem/$MW_e h$). The maximum resident dose may also be corrected for occupancy time (i.e., multiply the dose by the fraction of time typically spent by the resident at the location during the period of interest) if documented.

6.11.2 North Warehouse

Radioactive materials and low level waste can be stored in the North Warehouse. The maximum annual dose contributions to off-site receptors (west site boundary line) from sources in the shielded (east) end and the unshielded (west) end of the North Warehouse are:

$$\underset{\left(\frac{\text{mrem}}{\text{yr}}\right)}{D_S} = \underset{\left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right)}{0.25} \times \underset{\left(\frac{\text{mrem}}{\text{hr}}\right)}{\dot{R}_S} \text{ for the shielded end} \qquad (6\text{-}28)$$

and

$$\underset{\left(\frac{\text{mrem}}{\text{yr}}\right)}{D_U} = \underset{\left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right)}{0.53} \times \underset{\left(\frac{\text{mrem}}{\text{hr}}\right)}{\dot{R}_U} \text{ for the unshielded end} \qquad (6\text{-}29)$$

where:

D_s = The annual dose contribution at the maximum site boundary location from fixed sources of radiation stored in the shielded east end of the North Warehouse $\left(\frac{\text{mrem}}{\text{yr}}\right)$.

D_U = The annual dose contribution at the maximum site boundary location from fixed sources of radiation stored in the unshielded west end of the North Warehouse $\left(\frac{\text{mrem}}{\text{yr}}\right)$.

$\dot{R}_s$ = Dose rate measured at 1 meter from the source in the shielded end of the north warehouse $\left(\frac{\text{mrem}}{\text{hr}}\right)$.

$\dot{R}_U$ = Dose rate measured at 1 meter from the source in the unshielded end of the north warehouse $\left(\frac{\text{mrem}}{\text{hr}}\right)$.

0.25 = Dose rate to dose conversion factor which relates mrem/yr at the west site boundary per mrem/hr measured at 1 meter from

the source in the shielded end of the warehouse assuming it is full to capacity for one year $\left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right)$.

0.53 = Dose rate to dose conversion factor which relates mrem/yr at the west site boundary per mrem/hr measured at 1 meter from the source in the unshielded end of the warehouse assuming it is full to capacity for one year $\left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right)$.

6.11.3 Low Level Waste Storage Pad

Interim storage of packaged Dry Active Waste (DAW) and spent ion exchange and filter media is permitted in modular concrete storage overpacks on the LLW storage pad facility adjacent to the north warehouse. The arrangement of the storage modules is such that DAW is placed in modules which shield higher activity ion exchange media from the west site boundary. The dose at the maximum site boundary receptor from both direct radiation and skyshine scatter can be calculated as follows:

(a) Direct Dose (line of sight)

$$D_{dE} = 0.28 \times \dot{R}_d \times f_d \quad \left(\frac{\text{mrem}}{\text{yr- module}}\right) \quad \left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right) \quad \left(\frac{\text{mrem}}{\text{hr}}\right) \quad (\#) \qquad (6\text{-}30)$$

or

$$D_{dS} = 0.39 \times \dot{R}_d \; f_d \quad \left(\frac{\text{mrem}}{\text{yr- module}}\right) \quad \left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right) \quad \left(\frac{\text{mrem}}{\text{hr}}\right) \; (\#) \qquad (6\text{-}31)$$

where:

D_{dE} = The annual direct dose contribution at the maximum site boundary from a single rectangular storage module which has an unobstructed short end surface (not shielded by other modules) orientated toward the west site boundary $\left(\frac{\text{mrem}}{\text{yr- module}}\right)$.

D_{dS} = The annual direct dose contribution at the maximum site boundary from a single rectangular storage module which has an unobstructed long side surface (not shielded by other modules) orientated toward the west site boundary $\left(\frac{\text{mrem}}{\text{yr- module}}\right)$.

$\dot{R}_d$ = Maximum dose rate measured at 3 feet from the side of the storage module whose unobstructed face (i.e., a side or end surface which is not shielded by other waste modules) is toward the west site boundary.

f_d = The fraction of a year that a storage module is in use on the storage pad.

0.28 = Dose rate to dose conversion factor which relates mrem/yr at the west site boundary per mrem/hr measured at 3 feet from the narrow end of the rectangular storage module when that face is orientated toward the west boundary.

0.39 = Dose rate to dose conversion factor which relates mrem/yr at the west site boundary per mrem/hr measured at 3 feet from the long side of the rectangular storage module when that face is orientated toward the west boundary.

(b) Scatter From Skyshine

$$D_{SKR} = 0.016 \times \dot{R}_{SKR} \times f_{sk} \quad (6\text{-}32)$$

$$\left(\frac{\text{mrem}}{\text{yr- liner}}\right) \quad \left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right) \quad \left(\frac{\text{mrem}}{\text{hr}}\right) \quad (\#)$$

and

$$D_{SKD} = 0.015 \times \dot{R}_{SKD} \times f_{SK} \quad (6\text{-}33)$$

$$\left(\frac{\text{mrem}}{\text{yr- module}}\right) \quad \left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right) \quad \left(\frac{\text{mrem}}{\text{hr}}\right) \quad (\#)$$

where:

R_{SKR} = The annual skyshine scatter contribution to the dose at the maximum site boundary from a single spent ion exchange media liner in a storage module whose top surface is not obstructed due to stacking of modules $\left(\frac{\text{mrem}}{\text{yr- liner}}\right)$.

R_{SKD} = The annual skyshine scatter contribution to the dose at the maximum site boundary from a rectangular storage module containing DAW whose top surface is not obstructed due to stacking of modules $\left(\frac{\text{mrem}}{\text{yr- module}}\right)$.

$\dot{R}_{SKR}$ = For Resins, the maximum dose rate measured at 3 feet over the top of each liner in a storage module (mrem/hr).

$\dot{R}_{SKD}$ = For DAW, the maximum dose rate measured at 3 feet over the top surface of a storage module with DAW (mrem/hr).

f_{SK} = The fraction of a year that a storage module is in use on the storage pad.

0.016 = Dose rate to dose conversion factor for the scatter dose from each resin liner source in storage which relates mrem/yr at the west site boundary per mrem/hour at 3 feet from the top of the module.

0.015 = Dose rate to dose conversion factor for the scatter dose from DAW boxes in storage which relates mrem/yr at the west site boundary per mrem/hr at 3 feet from the top of the module.

(c) Dose From Resin Liners During Transfer

During the movement of resin liners from transfer casks to the storage modules, the liners will be unshielded in the storage pad area for a short period of time. The maximum dose contribution at the site boundary during the unshielded movement of resin liners can be calculated from:

$$D_{trans} = 0.0025 \times \dot{R}_{tran} \times T_{trans} \quad \text{(6-34)}$$

$$\text{(mrem)} \quad \left(\frac{\text{mrem/hr}}{\text{rad/hr}}\right) \quad \left(\frac{\text{rad}}{\text{hr}}\right) \quad \text{(hr)}$$

where:

D_{trans} = The dose contribution to maximum site boundary resulting from the unshielded movement of resin liners between a transfer cask and a storage module (mrem).

$\dot{R}_{trans}$ = Dose rate measured at contact (2") from the unshielded top surface of the resin liner in R/hr.

T_{tran} = The time (in hours) that an unshielded resin liner is exposed in the storage pad area.

0.0025 = The dose rate to dose conversion factor for an unshielded resin liner which relates mrem/hour at the west site boundary per rad/hr at contact (2") from the unshielded surface of the liner.

(d) Intermodular Gap Dose

In addition to the above methods for determining doses at the west site boundary from the LLW storage pad, another dose assessment model has been included to address the possible condition of spaces or gaps existing between the placement of the DAW storage modules situated along the west facing side of the pad. This could result in a radiation streaming condition existing if ion exchange resin liners were placed in storage directly behind the gap. The direct dose equations (6-30 and 6-31) consider that the storage modules situated on the outside of the pad area provide a uniform shield to storage modules placed behind them. The intermodular gap dose equation (6-35) accounts for any physical spacing between the outside storage modules which have not been covered by additional external shielding.

(6-35)

$$D_{Gap} = 2.44E\text{-}2 \times W_{Gap} \times A_{RL} \times f_{Gap}$$

$$\left(\frac{mrem}{yr}\right) \quad \left(\frac{mrem}{yr\text{-}in\text{-}Ci}\right) \quad (in) \quad (Ci) \quad (\#)$$

where:

D_{Gap} = The annual dose contribution at the maximum site boundary (west) from radiation streaming through the intermodular gap between DAW storage modules used to shield resin modules from direct radiation (mrem/yr).

W_{Gap} = The intermodular gap width (inches) between adjacent DAW storage modules facing the west site boundary.

A_{RL} = The total gamma activity contained in a condensate resin liner stored directly in line with the intermodular gap adjacent DAW modules (Ci).

f_{Gap} = The fraction of a year that the intermodular gap is not shielded.

2.44E-2 = The activity to site boundary dose conversion factor for a one-inch wide intermodular gap $\left(\frac{\text{mrem}}{\text{yr-in-Ci}}\right)$.

The site boundary dose from waste materials placed into storage on the Low Level Waste Storage Pad Facility is determined by combining the dose contribution due to direct radiation (line of sight) from Part (a) above with the skyshine scatter dose from Part (b), resin liner transfer dose from Part (c), and any intermodular gap dose from Part (d).

6.11.4 Total Direct Dose Summary

The dose contributions from the N-16 source in the Turbine Building, fixed sources in the North Warehouse, and fixed sources on the Low Level Waste Storage Pad Facility, shall be combined to obtain the estimate of total off-site dose to any member of the public from all fixed sources of radiation located on-site.

6.11.5 Other Fixed Sources

In addition to the fixed sources noted above (Turbine Building, North Warehouse, and LLW Storage Pad), other identified temporary or fixed sources that are created due to plant operations will be included in the total direct summary of 6.11.4 if the projected annual dose contribution would add any notable addition to the reported total (i.e., $\geq$ 0.1 mrem/yr).

In 1995, turbine rotors and casings were replaced in the Turbine Hall with the old rotors and casings placed in storage sheds located on site west of the switchyard along the railroad spur. Radiation surveys (December 1995) of low level contamination (principally Co-60) on the components led to a projected maximum west site boundary dose of 0.2 mrem/yr. This contribution will be added to the maximum site boundary total dose until the contribution is less than 0.1 mrem/yr, or the components are removed from storage location.

6.12 Cumulative Doses

Cumulative Doses for a calendar quarter and a calendar year must be maintained to demonstrate a compliance with Controls 3.2.2, 3.3.2, and 3.3.3 (10CFR50, Appendix I dose objectives). In addition, if the requirements of the Action Statement of Control 3.4.1 dictate, cumulative doses over a calendar year must be determined (demonstration of compliance with total dose, including direct radiation per requirements of 40CFR190). To ensure the limits are not exceeded, a running total must be kept for each release.

Demonstration of compliance with the dose limits of 40CFR190 is considered as demonstrating compliance with the 0.1 rem limit of 10CFR20.1301(a)(1) for members of the public in unrestricted areas.

7.0 ENVIRONMENTAL MONITORING PROGRAM

The radiological environmental monitoring stations are listed in Table 7.1. The locations of the stations with respect to the Vermont Yankee plant are shown on the maps in Figures 7-1 to 7-6.

7.1 Intercomparison Program

All routine radiological analyses for environmental samples are performed at a contracted environmental laboratory. The contracted laboratory participates in several commercial inter-comparison programs in addition to an internal QC sample analysis program and the analysis of client-introduced QC sample programs. The external programs include the Department of Energy – Environmental Measurements Laboratory Quality Assessment Program (EMLQAP), Department of Energy – Mixed Analyte Performance Evaluation Program (MAPEP), Analytics Cross-Check Program - Environmental Inter-laboratory Cross-Check Program, and Environmental Resources Association - Environmental Radioactivity Performance Evaluation Program.

7.2 Airborne Pathway Monitoring

The environmental sampling program is designed to achieve several major objectives, including sampling air in predominant up-valley and down-valley wind directions, and sampling air in nearby communities and at a proper control location, while maintaining continuity with two years of preoperational data and all subsequent years of operational data (post 1972.) The chosen air sampling locations are discussed below.

To assure that an unnecessarily frequent relocation of samplers will not be required due to short-term or annual fluctuations in meteorology, thus incurring needless expense and destroying the continuity of the program, long term, site specific ground level D/Qs (five-year averages - 1978 through 1982) were evaluated in comparison to the existing air monitoring locations to determine their adequacy in meeting the above-stated objectives of the program and the intent of the NRC general guidance. The long-term average meteorological data base precludes the need for an annual re-evaluation of air sampling locations based on a single year's meteorological history.

The Connecticut River Valley in the vicinity of the Vermont Yankee plant has a pronounced up- and down-valley wind flow. Based on five years of meteorological data, wind blows into the 3 "up-valley" sectors (N, NNW, and NW) 27 percent of the time, and the 4 "down-valley" sectors (S, SSE, SE, and ESE) 40 percent of the time, for a total "in-valley" time of 67 percent.

Station AP/CF-12 (NNW, 3.6 km) in North Hinsdale, New Hampshire, monitors the up-valley sectors. It is located in the sector that ranks fourth overall in terms of wind frequency (i.e., in terms of how often the wind blows into that sector), and is approximately 0.5 miles from the location of the calculated maximum ground level D/Q (i.e., for any location in any sector, for the entire Vermont Yankee environs). This station provides a second function by its location in that it also monitors North Hinsdale, New Hampshire, the community with the second highest ground level D/Q for surrounding communities, and it has been in operation since the preoperational period.

The down-valley direction is monitored by two stations - at River Station Number 3.3 (AP/CF-11, SSE, 1.9 km) and at Northfield, Massachusetts (AP/CF-14, SSE, 11.6 km). They both reside in the sector with the maximum wind frequency and they bound the down-valley point of calculated maximum ground level D/Q (the second highest overall ground level D/Q for any location in any sector). Station AP/CF-11 is approximately one mile from this point, between it and the plant. Station AP/CF-14 also serves as a community monitor for Northfield, Massachusetts. Both stations have been in operation since the preoperational period.

In addition to the up- and down-valley locations, two communities have been chosen for community sampling locations. The four nearest population groups with the highest long-term average D/Q values, in decreasing order, are Northfield, Massachusetts, North Hinsdale, New Hampshire, Brattleboro, Vermont, and Hinsdale, New Hampshire. The community sampler for Northfield is at Station AP/CF-14 (mentioned above). North Hinsdale is already monitored by the up-valley station (AP/CF-12, NNW, 3.6 km), which also indirectly monitors the city of Brattleboro, located further out in the same sector. The second sampler specifically designated for a community is at Hinsdale Substation (AP/CF-13, E, 3.1 km) in Hinsdale.

The control air sampler was located at Spofford Lake (AP/CF-21, NNE, 16.4 km) due to its distance from the plant and the low frequency for wind blowing in that direction based on the long-term (five-year) meteorological history. Sectors in the general west to southwest direction, which would otherwise have been preferable due to lower wind frequencies, were not chosen since they approached the region surrounding the Yankee Atomic plant in Rowe, Massachusetts.

An additional air sampler is maintained at the Tyler Hill site (AP/CF-15, WNW, 3.1 km), which is along the western side of the valley in general proximity of historical dairy operations. (The sixth location is not a specific Program requirement as detailed in Table 3.5.1 .)

7.3 Distances and Directions to Monitoring Stations

It should be noted that the distances and directions for direct radiation monitoring locations in Table 7.1, as well as the sectors shown in Figures 7-5 and 7-6, are keyed to the center of the Turbine Building due to the critical nature of the Turbine Building-to-TLD distance for close-in stations. For simplicity, all other radiological environmental sampling locations use the plant stack as the origin.

Control Table 3.5.1, Footnote a, specifies that in the Annual Radiological Environmental Operating Report and ODCM, the reactor shall be used as the origin for all distances and directions to sampling locations. Vermont Yankee interprets "the reactor" to mean the reactor site which includes the plant stack and the Turbine Building. The distances to the plant stack and Turbine Building will, therefore, be used in the Annual Radiological Environmental Operating Reports and ODCM for the sampling and TLD monitoring stations, respectively.

Table 7.1

Radiological Environmental Monitoring Stations[1]

Exposure Pathway and/or Sample Direction[5]		Sample Location and Designated Code[2]	Distance (km)[5]	
1. AIRBORNE (Radioiodine and Particulate)				
	AP/CF-11	River Station No. 3-3	1.88	SSE
	AP/CF-12	N. Hinsdale, NH	3.61	NNW
	AP/CF-13	Hinsdale Substation	3.05	E
	AP/CF-14	Northfield, MA	11.61	SSE
	AP/CF-15	Tyler Hill Road[4]	3.14	WNW
	AP/CF-21	Spofford Lake	16.36	NNE
2. WATERBORNE				
a. Surface	WR-11	River Station No. 3-3	1.88	Downriver
	WR-21	Rt. 9 Bridge	11.83	Upriver
b. Ground	WG-11	Plant Well	0.24	On-Site
	WG-12	Vernon Nursing Well	2.13	SSE
	WG-13	COB Well[4]	0.26	On-Site
	WG-14	Plant Support Bldg Well[4]	0.27	On-Site
	WG-22	Skibniowsky Well	13.73	N
c. Sediment From Shoreline	SE-11	Shoreline Downriver	0.57	SSE
	SE-12	North Storm Drain Outfall[3]	0.13	E
3. INGESTION				
a. Milk [8]	TM-11	Miller Farm	0.82	W
	TM-14	Brown Farm	2.22	S
	TM-18	Blodgett Farm	3.60	SE
	TM-22	Franklin Farm [4]	9.73	WSW
	TM-24	County Farm	21.64	N
	TM-25	Downey-Spencer[4]	6.90	W
	TM-26	Cheney Hill Farm	7.53	WNW
b. Mixed Grasses	TG-11	River Station No. 3-3	1.88	SSE
	TG-12	N. Hinsdale, NH	3.61	NNW
	TG-13	Hinsdale Substation	3.05	E
	TG-14	Northfield, MA	11.61	SSE
	TG-15	Tyler Hill Rd. [4]	3.07	WNW
	TG-21	Spofford Lake	16.36	NNE

Table 7.1

(Continued)

Radiological Environmental Monitoring Stations[1]

Exposure Pathway and/or Sample Direction[5]	Sample Location and Designated Code[2]		Distance (km)[5]	
c. Silage	TC-11	Miller Farm	0.82	W
	TC-14	Brown Farm	2.22	S
	TC-18	Blodgett Farm	3.60	SE
	TC-22	Franklin Farm[4]	9.73	WSW
	TC-24	County Farm	21.64	N
	TM-25	Downey-Spencer[4]	6.90	W
	TM-26	Cheney Hill Farm	7.53	WNW
d. Fish	FH-11	Vernon Pond	(6)	(6)
	FH-21	Rt. 9 Bridge	11.83	Upriver
4. DIRECT RADIATION				
	DR-1	River Station No. 3-3	1.61	SSE
	DR-2	N. Hinsdale, NH	3.88	NNW
	DR-3	Hinsdale Substation	2.98	E
	DR-4	Northfield, MA	11.34	SSE
	DR-5	Spofford Lake	16.53	NNE
	DR-6	Vernon School	0.52	WSW
	DR-7	Site Boundary[7]	0.28	W
	DR-8	Site Boundary	0.25	SSW
	DR-9	Inner Ring	1.72	N
	DR-10	Outer Ring	4.49	N
	DR-11	Inner Ring	1.65	NNE
	DR-12	Outer Ring	3.58	NNE
	DR-13	Inner Ring	1.23	NE
	DR-14	Outer Ring	3.88	NE
	DR-15	Inner Ring	1.46	ENE
	DR-16	Outer Ring	2.84	ENE
	DR-17	Inner Ring	1.24	E
	DR-18	Outer Ring	2.97	E
	DR-19	Inner Ring	3.65	ESE
	DR-20	Outer Ring	5.33	ESE
	DR-21	Inner Ring	1.82	SE
	DR-22	Outer Ring	3.28	SE
	DR-23	Inner Ring	1.96	SSE
	DR-24	Outer Ring	3.89	SSE
	DR-25	Inner Ring	1.91	S

Table 7.1

(Continued)

Radiological Environmental Monitoring Stations[1]

Exposure Pathway and/or Sample Direction[5]	Sample Location and Designated Code[2]		Distance (km)[5]	
	DR-26	Outer Ring	3.77	S
	DR-27	Inner Ring	1.10	SSW
	DR-28	Outer Ring	2.23	SSW
	DR-29	Inner Ring	0.92	SW
	DR-30	Outer Ring	2.36	SW
	DR-31	Inner Ring	0.71	WSW
	DR-32	Outer Ring	5.09	WSW
	DR-33	Inner Ring	0.66	WNW
	DR-34	Outer Ring	4.61	W
	DR-35	Inner Ring	1.30	WNW
	DR-36	Outer Ring	4.43	WNW
	DR-37	Inner Ring	2.76	NW
	DR-38	Outer Ring	7.34	NW
	DR-39	Inner Ring	3.13	NNW
	DR-40	Outer Ring	5.05	NNW

(1) Sample locations are shown on Figures 7.1 to 7.6.

(2) Station Nos. 10 through 19 are indicator stations. Station Nos. 20 through 29 are control stations (for all except milk, silage and the direct radiation stations).

(3) To be sampled and analyzed semiannually.

(4) Non-required Control station.

(5) Distance and direction from the center of the Turbine Building for direct radiation monitors; from the plant stack for all others.

(6) Fish samples are collected from anywhere in Vernon Pond, which is adjacent to the plant (see Figure 7-1).

(7) DR-7 satisfies Control Table 3.5.1 for an inner ring direct radiation monitoring location. However, it is averaged as a Site Boundary TLD due to its close proximity to the plant.

(8) In accordance with Control Table 3.5.1, notation a, samples will be collected on the required schedule as availability of milk permits. All deviations from the sample schedule will be reported in the Annual Radiological Environmental Operating Report.

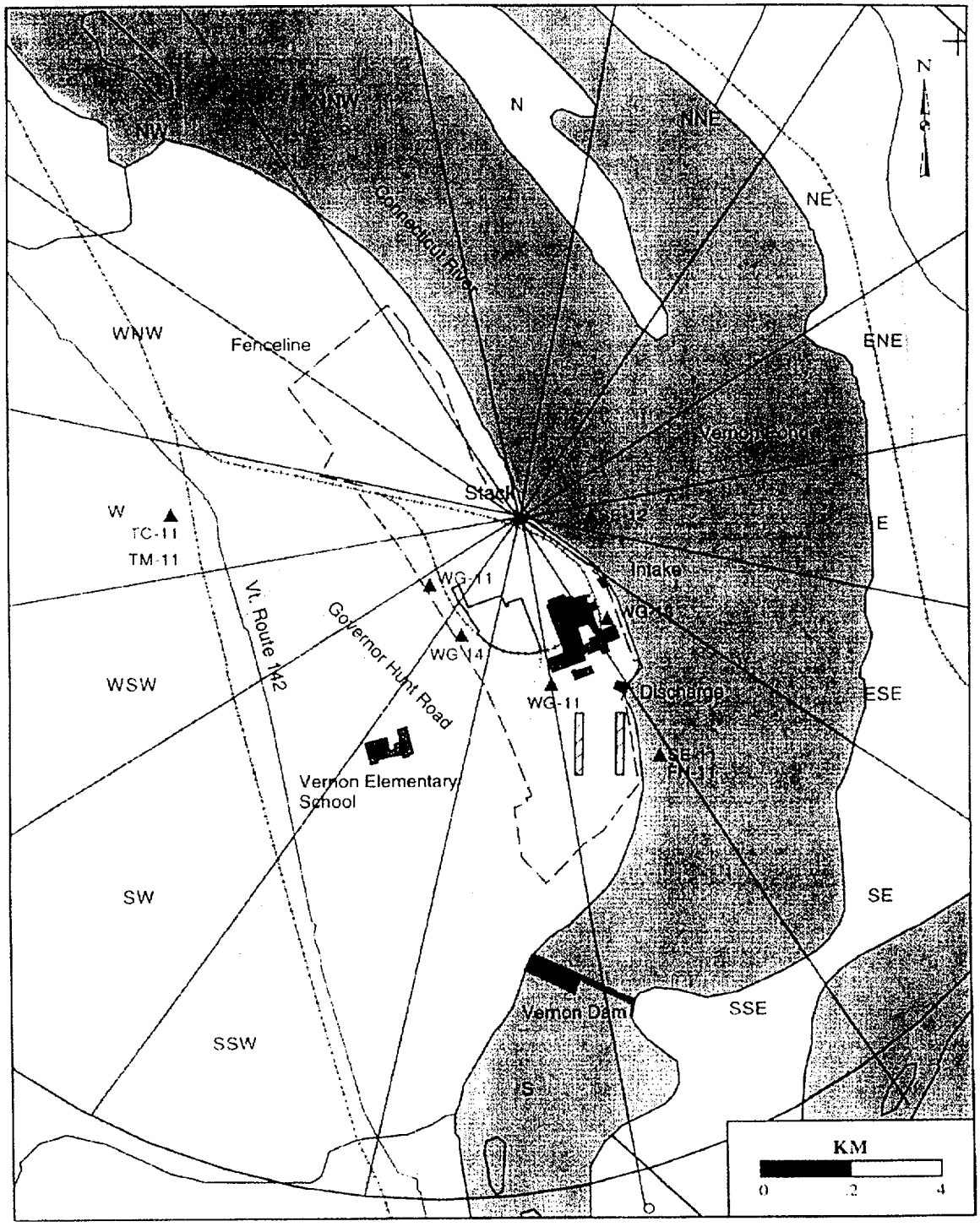

Figure 7-1 Environmental Sampling Locations in Close Proximity to the Plant

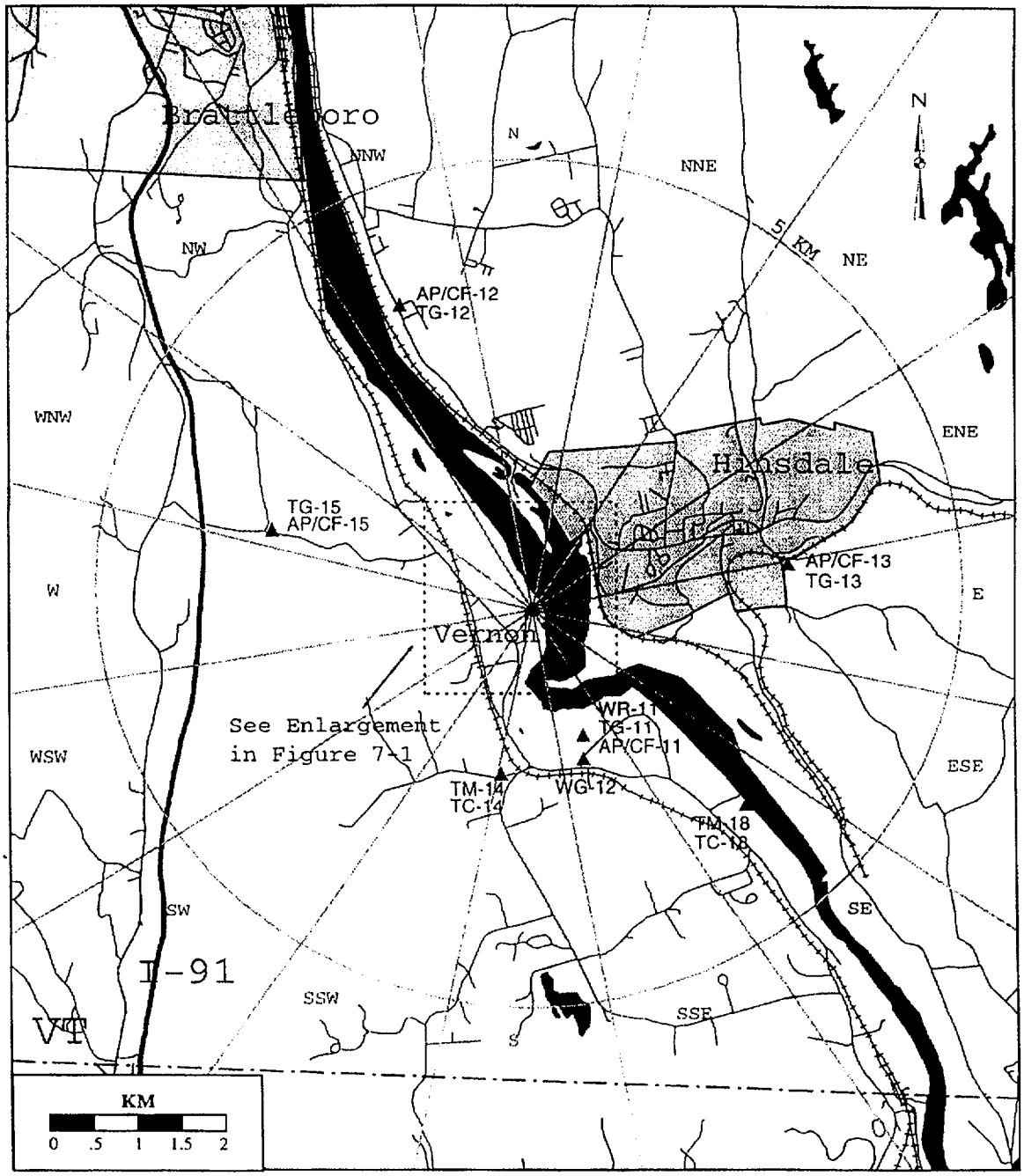

Figure 7-2 Environmental Sampling Locations Within 5 Km of Plant

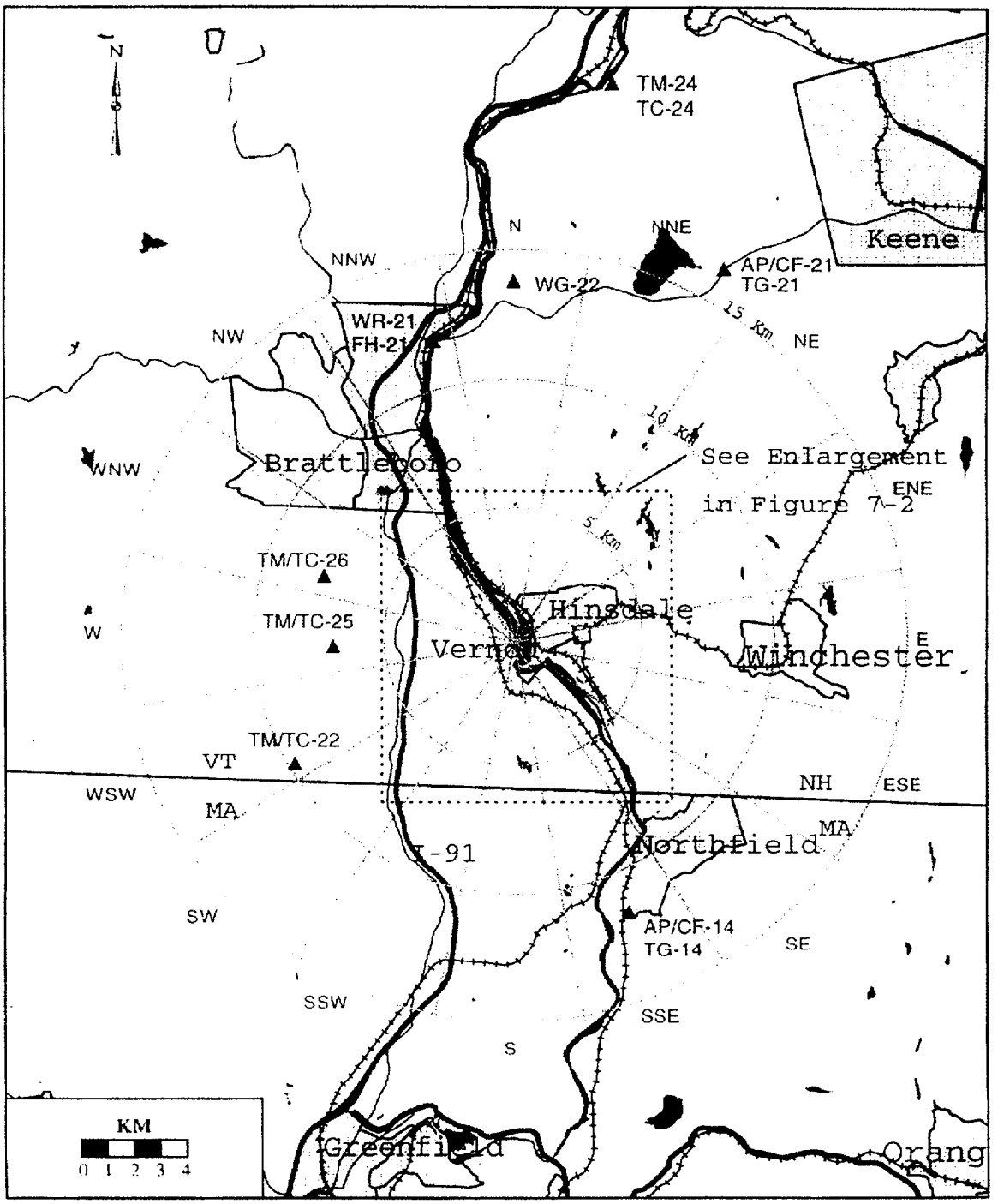

Figure 7-3 Environmental Sampling Locations Greater than 5 Km from Plant

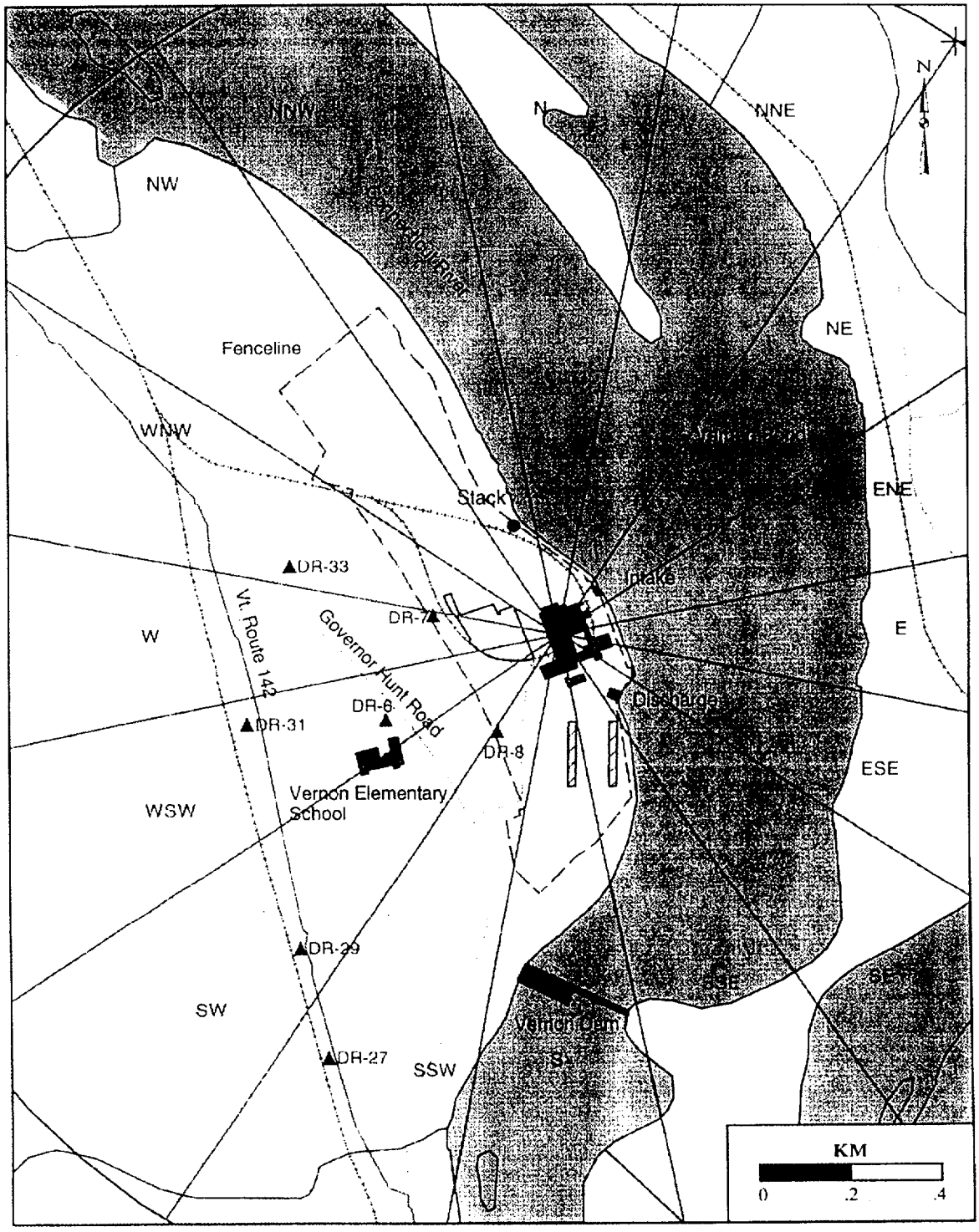

Figure 7-4 TLD Locations in Close Proximity to Plant

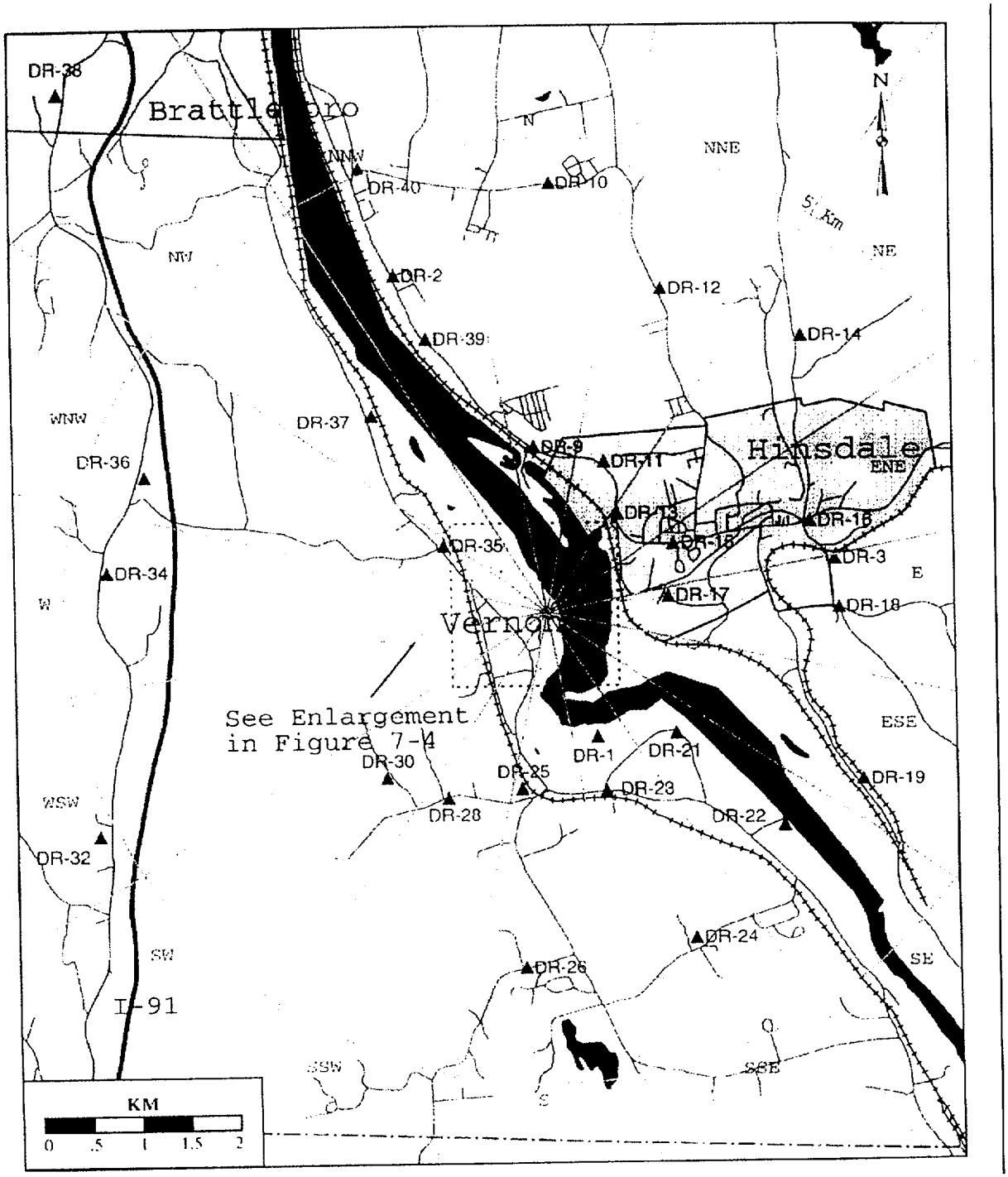

Figure 7-5 TLD Locations Within 5 Km of Plant

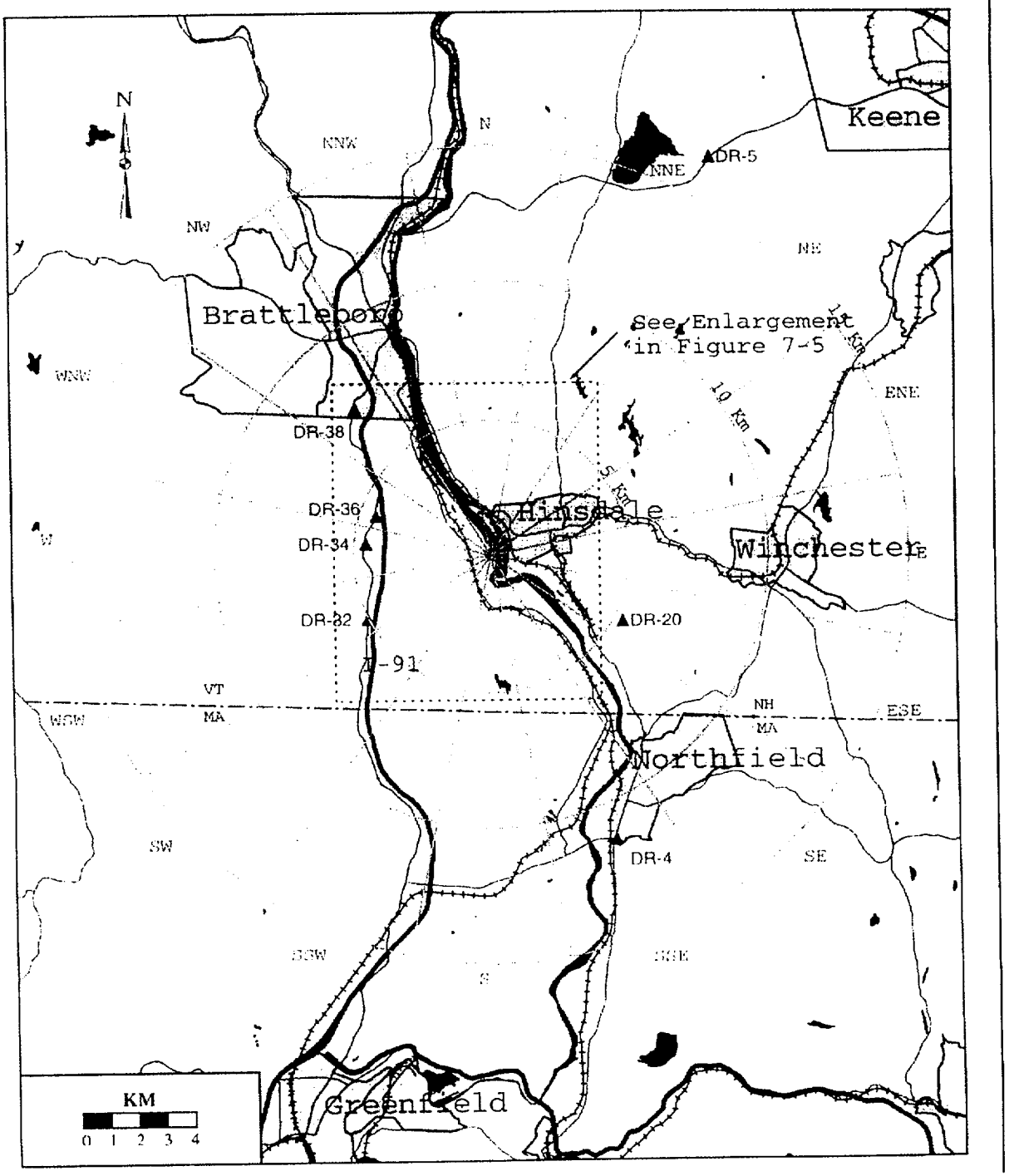

Figure 7-6 TLD Locations Greater Than 5 Km from Plant

8.0 SETPOINT DETERMINATIONS

Chapter 8 contains the basis for plant procedures used to meet the setpoint requirements of the Radioactive Effluent Instrumentation Controls. They are Control 3.1.1 for liquids and Control 3.1.2 for gases. Each outlines the instrumentation channels and the basis for each setpoint.

8.1 Liquid Effluent Instrumentation Setpoints

Control 3.1.1.1 requires that the radioactive liquid effluent instrumentation in Control Table 3.1.1 have alarm setpoints in order to ensure that Control 3.2.1 is not exceeded. Control 3.2.1 limits the activity concentration at any time in liquid effluents to ten items or less the effluent concentration values in Appendix B, Table 2, Column 2 of 10CFR20.1001 through 20.2401, and a total noble gas concentration limit of 2E-04 μCi/ml.

8.1.1 Liquid Radwaste Discharge Monitor (RM-17-350)

The sample tank pathways shown on Figure 9-1 are monitored by the liquid radwaste discharge monitor (RM-17-350). Periodic batch releases may be made from the waste sample tanks, detergent waste tank or floor drain sample tank.

8.1.1.1 Method to Determine the Setpoint of the Liquid Radwaste Discharge Monitor (RM-17-350)

The instrument response (in counts per second) for the limiting concentration at the point of discharge is the setpoint, denoted $R_{setpoint}$, and is determined as follows:

(8-1)

$$R_{setpoint} = \frac{DF}{DF_{min}} \; S_1 \; \sum_i \; C_{mi}$$

$$(cps) \quad (\#) \quad \left(\frac{cps - ml}{\mu Ci}\right) \quad \left(\frac{\mu Ci}{ml}\right)$$

Where:

$DF = \frac{Fd}{Fm}$ = Dilution factor (as a conservative measure, a DF of at least 1000 is used) (dimensionless) (8-2)

F_m = Flow rate past monitor (gpm)

F_d = Flow rate out of discharge canal (gpm)

DF_{min} = Minimum allowable dilution factor (dimensionless)

$$= 0.1 \sum_i \frac{C_{mi}}{ECL_i} \qquad (8\text{-}3)$$

ECL_i	=	Effluent concentration values for radionuclide "i" from 10CFR20.1001-20.2401, Appendix B, Table 2, Column 2 (μCi/ml)
C_{mi}	=	Activity concentration of radionuclide "i" in mixture at the monitor (μCi/ml)
S_l	=	Detector counting efficiency from the most recent liquid radwaste discharge monitor calibration curve (cps/(μCi/ml))

8.1.1.2 Liquid Radwaste Discharge Monitor Setpoint Example

The following alarm setpoint example is for a discharge of the floor drain sample tank. The liquid radwaste discharge monitor has a typical counting efficiency, S_l, of 4.9E+06 cps per 1 μCi/ml of gamma emitters which emit one photon per disintegration.

The activity concentration of each radionuclide, C_{mi}, in the floor drain sample tank is determined by analysis of a representative grab sample obtained at the radwaste sample sink. This setpoint example is based on the following data:

i	C_{mi} (μCi/ml)	ECL_i (μCi/ml)
Cs-134	2.15E-05	9E-07
Cs-137	7.48E-05	1E-06
Co-60	2.56E-05	3E-06

$$\sum_i C_{mi} = 2.15E-05 + 7.48E-05 + 2.56E-05$$

$$\left(\frac{\mu Ci}{ml}\right) \quad \left(\frac{\mu Ci}{ml}\right) \quad \left(\frac{\mu Ci}{ml}\right) \quad \left(\frac{\mu Ci}{ml}\right)$$

$$= 1.22E-04$$

$$\left(\frac{\mu Ci}{ml}\right)$$

(8-3)

$$DF_{min} = 0.1 \sum_i \frac{C_{mi}}{ECL_i}$$

$$\left(\frac{\mu Ci - ml}{ml - \mu Ci}\right)$$

$$= 0.1\left[\frac{2.15E-05}{9E-07} + \frac{7.48E-05}{1E-06} + \frac{2.56E-05}{3E-06}\right]$$

$$\left(\frac{\mu Ci - ml}{ml - \mu Ci}\right) \quad \left(\frac{\mu Ci - ml}{ml - \mu Ci}\right) \quad \left(\frac{\mu Ci - ml}{ml - \mu Ci}\right)$$

$$= 10.7$$

The minimum dilution factor, DF_{min}, needed to discharge the mixture of radionuclides in this example is 10.7. As a conservative measure, an actual dilution factor, DF, of 1,000 is usually used. The release rate of the floor drain sample tank may be adjusted from 0 to 50 gpm and the dilution pumps can supply up to 20,000 gpm of dilution water. With the dilution flow taken as 18,000 gpm, the release rate from the floor drain sample tank may be determined as follows:

$$F_m = \frac{Fd}{DF} \qquad (8\text{-}4)$$

(gpm) (gpm)

$$\frac{18{,}000\ gpm}{1{,}000} = 18\ gpm$$

Under these conditions, the setpoint of the liquid radwaste discharge monitor is:

(8-1)

$$R_{setpoint} = \frac{DF}{DF_{min}} \quad S_i \quad \sum_i \quad C_{mi}$$

$$(cps) \qquad (\#) \qquad \left(\frac{cps - ml}{\mu Ci}\right) \quad \left(\frac{\mu Ci}{ml}\right)$$

$$= \frac{1{,}000}{10.7} \; 4.9E+06 \quad 1.22E-04$$

$$(cps) \qquad (\#) \qquad \left(\frac{cps - ml}{\mu Ci}\right) \quad \left(\frac{\mu Ci}{ml}\right)$$

$$= 55{,}869 \text{ cps}$$

In this example, the calculated limiting count rate alarm point for the liquid radwaste discharge monitor would be 55,869 cps above background. Plant procedures apply administrative limits below the calculated limiting count rate to account for such elements as instrument uncertainty and early alarm warning before exceeding Control limits.

8.1.1.3 Basis for the Liquid Radwaste Discharge Monitor Setpoint

The liquid radwaste discharge monitor setpoint must ensure that Control 3.2.1 is not exceeded for the appropriate in-plant pathways. The liquid radwaste discharge monitor is placed upstream of the major source of dilution flow and responds to the concentration of radioactivity discharged in batch releases as follows:

(8-5)

$$R = \sum_i \quad C_{mi} \quad S_{li}$$

$$(cps) \qquad \left(\frac{\mu Ci}{ml}\right) \quad \left(\frac{cps - ml}{\mu Ci}\right)$$

Where:

R = Response of the monitor (cps)

S_{li} = Detector counting efficiency for radionuclide "i" (cps/(μCi/ml))

C_{mi} = Activity concentration of radionuclide "i" in mixture at the monitor (μCi/ml)

The detector calibration procedure establishes a counting efficiency for a given mix of nuclides seen by the detector. Therefore, in Equation 8-5 one may substitute S_I for S_{Ii}, where S_I represents the counting efficiency determined for the current mix of nuclides. If the mix of nuclides changes significantly, a new counting efficiency should be determined for calculating the setpoint.

(8-6)

$$R = S_I \sum_i C_{mi}$$

$$(cps) \quad \left(\frac{cps - ml}{\mu Ci}\right) \quad \left(\frac{\mu Ci}{ml}\right)$$

The effluent concentration for a given radionuclide must not exceed 10 times the 10 CFR Part 20 ECL at the point of discharge to an unrestricted area at any time. When a mixture of radionuclides is present, the concentration at the point of discharge to an unrestricted area shall be limited as follows:

(8-7)

$$\sum_i \frac{C_{di}}{ECL_i} \leq 10$$

$$\left(\frac{\mu Ci - ml}{ml - \mu Ci}\right)$$

Where:

C_{di} = Activity concentration of radionuclide "i" in the mixture at the point of discharge to an unrestricted area (μCi/ml)

ECL_i = Effluent concentration limit for radionuclide "i" from 10CFR20.1001-20.2401, Appendix B, Table 2, Column 2 (μCi/ml)

The activity concentration of radionuclide "i" at the point of discharge is related to the activity concentration of radionuclide "i" at the monitor as follows:

(8-8)

$$C_{di} = C_{mi} \frac{F_m}{F_d}$$

$$\left(\frac{\mu Ci}{ml}\right) \left(\frac{\mu Ci}{ml}\right) \left(\frac{gpm}{gpm}\right)$$

Where:

C_{di} = Activity concentration of radionuclide "i" in the mixture at the point of discharge (μCi/ml)

F_m = Flow rate past monitor (gpm)

F_d = Flow rate out of discharge canal (gpm)

Substituting the right half of Equation 8-8 for C_{di} in Equation 8-7 and solving for F_d/F_m yields the minimum dilution factor needed to comply with Equation 8-7:

(8-3)

$$DF_{min} \leq \frac{F_d}{F_m} \geq \sum_i \frac{C_{mi}}{ECL_i * 10}$$

$$\left(\frac{gpm}{gpm}\right) \qquad \left(\frac{\mu Ci - ml}{ml - \mu Ci}\right)$$

Where:

F_d = Flow rate out of discharge canal (gpm)

F_m = Flow rate past monitor (gpm)

C_{mi} = Activity concentration of radionuclide "i" in mixture at the monitor (μCi/ml)

ECL_i = Effluent concentration limit for radionuclide "i" from 10CFR20.1001-20.2401, Appendix B, Table 2, Column 2 (μCi/ml)

10 = The instantaneous concentration multiplier allowed by Control 3.2.1

If F_d/F_m is less than DF_{min}, then the tank may not be discharged until either F_d or F_m or both are adjusted such that:

(8-3)

$$\frac{F_d}{F_m} \geq DF_{min}$$

$$\left(\frac{gpm}{gpm}\right)$$

Usually F_d/F_m is greater than DF_{min} (i.e., there is more dilution than necessary to comply with Equation 8-7). The response of the liquid radwaste discharge monitor at the setpoint is therefore:

(8-1)

$$R_{setpoint} = \frac{DF}{DF_{min}} \quad S_1 \quad \sum_i \quad C_{mi}$$

$$(cps) \qquad (\#) \qquad \left(\frac{cps - ml}{\mu Ci}\right) \quad \left(\frac{\mu Ci}{ml}\right)$$

8.1.2 Service Water Discharge Monitor (RM-17-351)

The service water pathway shown on Figure 9-1 is continuously monitored by the service water discharge monitor (RM-17-351). The water in this line is not radioactive under normal operating conditions. The alarm setpoint on the Service Water Monitor (SWM) is set in accordance with the monitor's ability to detect dilute concentrations of radionuclide mixes that are based on measured nuclide distributions in reactor coolant. From routine coolant sample gamma isotopic analyses, a Composite Maximum Permissible Concentration (CMPC) is calculated as follows:

$$C(f_1/MPC_1 + f_2/MPC_2 \ldots) = C/CMPC$$

or

$$CMPC = 1/(f_1/MPC_1 + f_2/MPC_2 \ldots) \qquad (8\text{-}22)$$

where:

C = Total concentration of detected radioactivity in reactor coolant sample (μCi/ml)

f_i = Fraction of total radionuclide concentration represented by the ith radionuclide in the mix

MPC_i = Maximum Permissible Concentration limit for radionuclide "i" as listed in 10CFR20.106, Appendix B, Table II, Column 2 (μCi/ml)

The Composite Effluent Concentration Limit (CECL) is also calculated using the equation above by substituting the appropriate ECL value from 10CFR20.1001-20.2401, Appendix B, Table 2, Column 2, for MPC.

If the SWM's minimum achievable alarm setpoint is higher than the required CMPC equivalent count rate (or the CECL equivalent count rate if it is lower than the CMPC count rate), the monitor is declared inoperable, and daily SWM grab samples are collected and analyzed until the calculated coolant CMPC (or CECL) equivalent count rate is above the SWM's alarm setpoint.

For example, if the reactor coolant radionuclide mix distribution is as listed below, then the corresponding CMPC is calculated as follows:

Nuclides	Conc (μCi/ml)	f_I ($conc_i$/total conc)	10CFR20 MPC_i (μCi/ml)	f_i/MPC_i (ml/μCi)
I-131	6.00E-6	6.59E-2	3.0E-7	2.20E+5
I-133	5.00E-6	5.49E-2	1.0E-6	5.49E+4
Co-60	8.00E-5	8.79E-1	3.0E-5	2.93E+4
Totals	9.10E-5	1.00		3.04E+5

CMPC = 1/3.04E+5 = 3.29E-6 (μCi/ml)

The CECL is also calculated by using the above methodology and substituting the appropriate ECL listed in 10CFR20.1001-20.2401, Appendix B, Table 2, Column 2, for MPC values. For this example, the calculated CECL is equal to 2.73E-6 μCi/ml.)

If the SWM alarm is set at 5 CPS (300 CPM) above background, and the current calibration factor for this monitor is 1.17E+8 CPM/μCi/ml, then the SWM will alarm if a concentration as low as 2.56E-6 μCi/ml above background passes by the monitor. Since the most limiting CMPC or CECL (calculated above to be 2.73E-6 μCi/ml) is above the alarm setpoint (equal to 2.56E-6 μCi/ml), the SWM will be capable of alarming if radioactivity in excess of limiting concentration values for release to unrestricted areas passes by the monitor. However, if the composite concentration (CMPC or CECL) for the service water was found to be less than the SWM alarm setpoint of 2.56E-6 μCi/ml, then daily service water grab samples would have to be collected and analyzed until the composite concentration becomes greater than the concentration corresponding to the SWM's alarm setpoint.

Also, service water is sampled if the monitor is out of service or if the alarm sounds.

Under normal operating conditions, the concentration of radionuclides at the point of discharge to an unrestricted area from the service water effluent pathway will not exceed the effluent concentration limits specified in 10CFR20.1001-20.2401, Appendix B, Table 2, Column 2.

8.2 Gaseous Effluent Instrumentation Setpoints

Control 3.1.2 requires that the radioactive gaseous effluent instrumentation in Control Table 3.1.2 have their alarm setpoints set to ensure that Technical Specification 3.8.K.1 and Control 3.3.1 are not exceeded. Technical Specification 3.8.K.1 (and Control 3.3.7) limits the gross radioactivity release rate at the steam jet air ejector (SJAE) to 0.16 Ci/sec.

8.2.1 Plant Stack Noble Gas Activity Monitors (RM-17-156 and RM-17-157) and Augmented Off-Gas System Noble Gas Activity Monitors (RAN-OG-3127 and RAN-OG-3128)

The plant stack and AOG noble gas activity monitors are shown on Figure 9-2.

8.2.1.1 Method to Determine the Setpoint of the Plant Stack Noble Gas Activity Monitors (RM-17-156 and RM-17-157) and the Augmented Off-Gas System Noble Gas Activity Monitors (RAN-OG-3127 and RAN-OG-3128)

The setpoints of the plant stack and AOG system noble gas activity monitors are determined in the same manner. The plant stack or AOG system noble gas activity monitor response in counts per minute at the limiting off-site noble gas dose rate to the total body or to the skin is the setpoint, denoted R_{spt}. R_{spt} is the lesser of:

(8-9)

$$R_{spt}^{tb} = 818 \; S_g \; \frac{1}{F} \; \frac{1}{DFB_c}$$

$$(cpm) \left(\frac{mrem-\mu Ci-m^3}{yr-pCi-sec}\right)\left(\frac{cpm-cm^3}{\mu Ci}\right)\left(\frac{sec}{cm^3}\right)\left(\frac{pCi-yr}{mrem-m^3}\right)$$

and: (8-10)

$$R_{spt}^{skin} = 3{,}000 \; S_g \; \frac{1}{F} \; \frac{1}{DF_c'}$$

$$(cpm) \left(\frac{mrem}{yr}\right)\left(\frac{cpm-cm^3}{\mu Ci}\right)\left(\frac{sec}{cm^3}\right)\left(\frac{\mu Ci-yr}{mrem-sec}\right)$$

where:

R_{spt}^{tb} = Response of the monitor at the limiting total body dose rate (cpm)

$$818 = \frac{500}{(1E+06)\ (6.11E-07)} \left(\frac{mrem - \mu Ci - m^3}{yr - pCi - sec} \right)$$

500 = Limiting total body dose rate (mrem/yr)

1E+06 = Number of pCi per μCi (pCi/μCi)

6.11E-07 = $[X/Q]^{\gamma}$, maximum five-year average gamma atmospheric dispersion factor (sec/m3)

S_g = Appropriate (plant stack or AOG system) detector counting efficiency from the most recent calibration (cpm/(μCi/cc))

F = Appropriate (plant stack or AOG system) flow rate (cm3 /sec)

DFB_c = Composite total body dose factor (mrem-m3/pCi-yr)

(8-11)

$$= \frac{\sum_i \dot{Q}_i \, DFB_i}{\sum_i \dot{Q}_i}$$

$\dot{Q}_i$ = The relative release rate of noble gas "i" in the mixture at the monitor (either the stack $\dot{Q}^{ST}$ or the AOG, $\dot{Q}^{AOG}$) for noble gases identified (μCi/sec)

DFB_i = Total body dose factor (see Table 1.1-10) (mrem-m3/pCi-yr)

R_{spt}^{skin} = Response of the monitor at the limiting skin dose rate (cpm)

3,000 = Limiting skin dose rate (mrem/yr)

DF'_c = Composite skin dose factor (mrem-sec/μCi-yr)

(8-12)

$$= \frac{\sum_i \dot{Q}_i \, DF'_{is}}{\sum_i \dot{Q}_i}$$

DF'_{is} = Combined skin dose factor (see Table 1.1.10) (mrem-sec/μCi-yr)

8.2.1.2 Plant Stack Noble Gas Activity Monitor Setpoint Example

The following setpoint example for the plant stack noble gas activity monitors demonstrates the use of Equations 8-9 and 8-10 for determining setpoints.

The plant stack noble gas activity monitors, referred to as "Stack Gas I" (RM-17-156) and "Stack Gas II" (RM-17-157), consist of beta sensitive scintillation detectors, electronics, a ratemeter readout, and a digital scaler which counts the detector output pulses. A strip chart recorder provides a permanent record of the ratemeter output. The monitors have typical calibration factors, S_g, of about 3E+07 cpm per μCi/cc of noble gas. The nominal plant stack flow is 7.32E+07 cc/sec ((155,000 cfm x 28,300 cc/ft^3)/60 sec/min).

When monitor responses indicate that activity levels are below the LLDs at the stack (or AOG) monitors, the relative contribution of each noble gas radionuclide can conservatively be approximated by analysis of a sample of off-gas obtained during plant operations at the steam jet air ejector (SJAE). This setpoint example is based on the following data (see Table 1.1.10 for DFB_i and DF'_i):

i	$\dot{Q}_i^{SJAE}$ $\left(\frac{\mu Ci}{sec}\right)$	DFB_i $\left(\frac{mrem - m^3}{pCi - yr}\right)$	DF'_{is} $\left(\frac{mrem - sec}{\mu Ci - yr}\right)$
Xe-138	1.03E+04	8.83E-03	1.06E-02
Kr-87	4.73E+02	5.92E-03	1.43E-02
Kr-88	2.57E+02	1.47E-02	1.28E-02
Kr-85m	1.20E+02	1.17E-03	2.35E-03
Xe-135	3.70E+-2	1.81E-03	3.24E-03
Xe-133	1.97E+01	2.94E-04	5.58E-04

(8-11)

$$DFB_c = \frac{\sum_i \dot{Q}_i^{SJAE} DFB_i}{\sum_i \dot{Q}_i^{SJAE}}$$

$$\sum_i \dot{Q}_i^{SJAE} DFB_i = (1.03E+04)(8.83E-03) + (4.73E-02)(5.92E-03)$$

$$+ (2.57E+02)(1.47E-02) + (1.20E+02)(1.17E-03)$$

$$+ (3.70E+02)(1.81E-03) + (1.97E+01)(2.94E-04)$$

$$= 9.83E+01 \ (\mu Ci\text{-}mrem\text{-}m^3/sec\text{-}pCi\text{-}yr)$$

$$\sum_i \dot{Q}_i^{SJAE} = 1.03E+04 + 4.73+02 + 2.57E+02$$

$$+ 1.20E+02 + 3.70E+02 + 1.97E+01$$

$$= 1.15E+04 \ \mu Ci/sec$$

$$DFB_C = \frac{9.83E+01}{1.15E+04}$$

$$= 8.52E\text{-}03 \ (mrem\text{-}m^3/pCi\text{-}yr)$$

$$R_{spt}^{tb} = 818 \ S_g \ \frac{1}{F} \ \frac{1}{DFB_c}$$

$$= (818)(3E+07)\frac{1}{(7.32E+07)}\frac{1}{(8.52E-03)}$$

$$= 39.348 \text{ cpm}$$

Next:

(8-11)

$$DF_c' = \frac{\sum_i \dot{Q}_i^{SJAE} DF_{is}'}{\sum_i \dot{Q}_i^{SJAE}}$$

$$\sum_i \dot{Q}_i^{SJAE} DF'_{is} = (1.03E+04)\,(1.06E-02) + (4.73E-02)\,(1.43E-02)$$

$$+ (2.57E+02)\,(1.28E-02) + (1.20E+02)\,(2.35E-03)$$

$$+ (3.70E+02)\,(3.24E-03) + (1.97E+01)\,(5.58E-04)$$

$$= 1.14E+02 \text{ (}\mu\text{Ci-mrem-sec/sec-}\mu\text{Ci-yr)}$$

$$DF'_c = \frac{1.14E+02}{1.15E+04}$$

$$= 9.91E\text{-}03 \text{ (mrem-sec/}\mu\text{Ci-yr)}$$

$$R_{spt}^{skin} = 3{,}000 \quad S_g \quad \frac{1}{F} \quad \frac{1}{DF'_c}$$

$$= (3{,}000)\ (3E+07)\ \frac{1}{(7.32E+07)}\ \frac{1}{(9.91E-03)}$$

$$= 124{,}067 \text{ cpm}$$

The setpoint, R_{spt}, is the lesser of R_{spt}^{tb} and R_{spt}^{skin}. For the noble gas mixture in this example R_{spt}^{tb} is less than R_{spt}^{skin}, indicating that the total body dose rate is more restrictive. Therefore, in this example the "Stack Gas I" and "Stack Gas II" noble gas activity monitors should each be set at some administrative value below 39,348 cpm above background to provide conservatism for such issues as instrument uncertainty and secondary releases from other locations. As an example, a conservative value might be based on controlling release rates from the plant in order to maintain off-site air concentrations below 20 x ECL when averaged over an hour, or to account for other minor releases from the waste oil burner. For example, if an administrative limit of 70 percent of the Control whole body dose limit 500 mrem/yr (39,348 cpm) is chosen, then the noble gas monitor alarms should be set at no more than 27,543 cpm above background (0.7 x 39,348 = 27,543).

8.2.1.3 Basis for the Plant Stack and AOG System Noble Gas Activity Monitor Setpoints

The setpoints of the plant stack and AOG system noble gas activity monitors must ensure that Control 3.3.1.a is not exceeded. Sections 6.4 and 6.5 show that Equations 6-5 and 6-7 are acceptable methods for determining compliance with the Control limits. Which equation (i.e., dose to total body or skin) is more limiting depends on the noble gas mixture. Therefore,

each equation must be considered separately. The derivations of Equations 8-9 and 8-10 begin with the general equation for the response R of a radiation monitor:

(8-13)

$$R = \sum_i \quad S_{gi} \quad C_{mi}$$

$$(cpm) \qquad \left(\frac{cpm - cm^3}{\mu Ci}\right) \left(\frac{\mu Ci}{cm^3}\right)$$

where:

R = Response of the instrument (cpm)

S_{gi} = Detector counting efficiency for noble gas "i" (cpm/(μCi/cm^3))

C_{mi} = Activity concentration of noble gas "i" in the mixture at the noble gas activity monitor (μCi/cm^3)

The relative release rate of each noble gas ($\dot{Q}_i$ $\mu Ci/cm^3$), in the total release rate is normally determined by analysis of a sample of off-gas obtained at the Steam Jet Air Ejector (SJAE). Noble gas release rates at the plant stack and the AOG discharge are usually so low that the activity concentration is below the Lower Limit of Detection (LLD) for sample analysis. As a result, the release rate mix ratios measured at the SJAE are used to represent any radioactivity being discharged from the stack, such as may have resulted from plant steam leaks that have been collected by building ventilation. For the AOG monitor downstream of the charcoal delay beds, this leads to a conservative setpoint since several short-lived (high dose factor) noble gas radionuclides are then assumed to be present at the monitor, which in reality, would not be expected to be present in the system at that point. During periods when the plant is shutdown (after five days), and no radioactivity release rates can be measured at the SJAE, Xe-133 is the dominant long-lived noble gas and may be used as the referenced radionuclide to determine off-site dose rates and monitor setpoints. Alternately, a relative radionuclide, "i", mix fraction, (f_i), may be taken from Table 8.2.1 as a function of time after shutdown (including periods shorter than five days) to determine the relative fraction of each noble gas potentially available for release to the total. However, prior to plant startup before a SJAE sample can be taken and analyzed, the monitor alarm setpoints should be based on Xe-138 as representing the most prevalent high dose factor noble gas expected to be present shortly after the plant returns to power. Monitor alarm setpoints which have been determined to be conservative under any plant conditions may be utilized at any time in lieu of the above assumptions. C_{mi}, the activity

concentration of noble gas "i" at the noble gas activity monitor, may be expressed in terms of $\dot{Q}_i$ by dividing by F, the appropriate flow rate. In the case of the plant stack noble gas activity monitors the appropriate flow rate is the plant stack flow rate and for the AOG noble gas activity monitors the appropriate flow rate is the AOG system flow rate.

(8-14)

$$C_{mi} = \dot{Q}_i \frac{1}{F}$$

$$\left(\frac{\mu Ci}{cm^3}\right) \left(\frac{\mu Ci}{sec}\right) \left(\frac{sec}{cm^3}\right)$$

where:

$\dot{Q}_i$ = The release rate of noble gas "i" in the mixture for each noble gas identified (μCi/sec).

F = Appropriate flow rate (cm^3/sec)

Substituting the right half of Equation 8-14 into Equation 8-13 for C_{mi} yields:

(8-15)

$$R = \sum_i S_{gi} \dot{Q}_i \frac{1}{F}$$

$$(cpm) \left(\frac{cpm - cm^3}{\mu Ci}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{sec}{cm^3}\right)$$

The detector calibration procedure establishes a counting efficiency for a reference radionuclide, Xe-133 (half life 5.24 days). For routine conditions where offgas is processed through the AOG, all short lived gases are decayed away before discharge leaving only long lived radionuclides as the significant contributors to the monitor response. In this case, Xe-133 as the reference radionuclide for the detector counting efficiency is representative of the expected release conditions. For off normal conditions that might lead to inclusion of short lived radioactivity in the gas stream being released, Xe-133 as the reference radionuclide is expected to lead to a conservative response factor for the detectors since the short lived noble gases tend to have higher energies that can cause them to over respond. Therefore, in Equation 8-15, one may substitute S_g for S_{gi}, where S_g represents the detector counting efficiency determined from the Xe-133 calibration. If necessary, the actual concentration and discharge rate of individual

gases being released from the stack (or AOG) can be determined by direct grab sample and laboratory analysis during specific periods of interest.

(8-16)

$$R = S_g \frac{1}{F} \sum_i \dot{Q}_i$$

$$(cpm)\left(\frac{cpm - cm^3}{\mu Ci}\right)\left(\frac{sec}{cm^3}\right)\left(\frac{\mu Ci}{sec}\right)$$

The total body dose rate due to noble gases is determined with Equation 8-5:

(8-5)

$$\dot{R}_{tbs} = 0.61 \sum_i \dot{Q}_i DFB_i$$

$$\left(\frac{mrem}{yr}\right)\left(\frac{pCi - sec}{\mu Ci - m^3}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem - m^3}{pCi - yr}\right)$$

Where:

$\dot{R}_{tbs}$ = total body dose rate (mrem/yr) due to noble gases from stack release

0.61 = (1.0E+06) x (6.11E-07) (pCi-sec/μCi-m^3)

1E+06 = number of pCi per μCi (pCi/μCi)

6.11E-07 = $[X/Q]^{\gamma}$, maximum long term average gamma atmospheric dispersion factor (sec/m^3)

$\dot{Q}_i$ = the release rate of noble gas "i" in the mixture for each noble gas identified (μCi/sec) (Equivalent to $\dot{Q}_i^{ST}$ for noble gases released at the plant stack.)

DFB_i = total body dose factor (see Table 1.1.10) (mrem-m^3/pCi-yr)

A composite total body gamma dose factor, DFB_c, may be defined such that:

(8-17)

$$DFB_c = \sum_i \dot{Q}_i = \sum_i \dot{Q}_i \quad DFB_i$$

$$\left(\frac{mrem - m^3}{pCi - yr}\right) \quad \left(\frac{\mu Ci}{sec}\right) \quad \left(\frac{\mu Ci}{sec}\right) \quad \left(\frac{mrem - m^3}{pCi - yr}\right)$$

Solving Equation 8-23 for DFB_c yields:

(8-11)

$$DFB_c = \frac{\sum_i \dot{Q}_i \, DFB_i}{\sum_i \dot{Q}_i}$$

Control 3.3.1.1.a limits the dose rate to the total body from noble gases at any location at or beyond the site boundary to 500 mrem/yr. By setting equal to 500 mrem/yr and substituting DFB_c for DFB_i in Equation 8-5, one may solve for $\sum_i \dot{Q}_i$ at the limiting whole body noble gas dose rate:

(8-18)

$$\sum_i \dot{Q}_i = 818 \quad \frac{1}{DFB_c}$$

$$\left(\frac{\mu Ci}{sec}\right) \left(\frac{mrem - \mu Ci - m^3}{yr - pCi - sec}\right) \left(\frac{pCi - yr}{mrem - m^3}\right)$$

Substituting this result for $\sum_i \dot{Q}_i$ in Equation 8-16 yields R_{spt}^{tb}, the response of the monitor at the limiting noble gas total body dose rate:

(8-9)

$$R_{spt}^{tb} = 818 \quad S_g \quad \frac{1}{F} \quad \frac{1}{DFB_c}$$

$$(cpm) \left(\frac{mrem - \mu Ci - m^3}{yr - pCi - sec}\right) \left(\frac{cpm - cm^3}{\mu Ci}\right) \left(\frac{sec}{cm^3}\right) \left(\frac{pCi - yr}{mrem - m^3}\right)$$

The skin dose rate due to noble gases is determined with Equation 6-7:

(6-7)

$$\dot{R}_{skin} = \sum_i \dot{Q}_i \quad DF'_{is}$$

$$\left(\frac{mrem}{yr}\right) \quad \left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem-sec}{\mu Ci-yr}\right)$$

Where:

$\dot{R}_{skin}$ = Skin dose rate (mrem/yr)

$\dot{Q}_i$ = The release rate of noble gas "i" in the mixture for each noble gas identified (μCi/sec) equivalent to $\dot{Q}_i^{ST}$ for noble gases released at the plant stack).

DF'_{is} = Combined skin dose factor (see Table 1.1.10) (mrem-sec/μCi-yr).

A composite combined skin dose factor, DF'_c, may be defined such that:

(8-19)

$$DF'_c \sum_i \dot{Q}_i = \sum_i \dot{Q}_i \quad DF_{is}$$

$$\left(\frac{mrem-sec}{\mu Ci-yr}\right) \quad \left(\frac{\mu Ci}{sec}\right)\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem-sec}{\mu Ci-yr}\right)$$

Solving Equation 8-19 for DF′ yields:

$$DF'_c = \frac{\sum_i \dot{Q}_i DF'_{is}}{\sum_i \dot{Q}_i}$$

Control 3.3.1.a limits the dose rate to the skin from noble gases at any location at or beyond the site boundary to 3,000 mrem/yr. By setting $\dot{R}_{skin}$ equal to 3,000 mrem/yr and substituting DF_c' for DF_i' in Equation 6-7 one may solve for $\sum_i \dot{Q}_i$ at the limiting skin noble gas dose rate:

$$\sum_i \dot{Q}_i = 3{,}000 \quad \frac{1}{DF_c'}$$

$$\left(\frac{\mu Ci}{sec}\right) \left(\frac{mrem}{yr}\right) \left(\frac{\mu Ci - yr}{mrem - sec}\right)$$

Substituting this result for $\sum_i \dot{Q}_i$ in Equation 8-16 yields R_{spt}^{skin}, the response of the monitor at the limiting noble gas skin dose rate:

(8-10)

$$R_{spt}^{skin} = 3{,}000 \quad S_g \quad \frac{1}{F} \quad \frac{1}{DF_c'}$$

$$(cpm) \left(\frac{mrem}{yr}\right) \left(\frac{cpm - cm^3}{\mu Ci}\right) \left(\frac{sec}{cm^3}\right) \left(\frac{\mu Ci - yr}{mrem - sec}\right)$$

TABLE 8.2.1

Relative Fractions of Core Inventory
Noble Gases After Shutdown

Time	Kr-83m	Kr-85m	Kr-85	Kr-87	Kr-88	Xe-131m	Xe-133m	Xe-133	Xe-135m	Xe-135	Xe-138
$t < 24$ h	.02	.043	.001	.083	.118	.002	.010	.306	.061	.093	.263
24 hr $\leq t < 48$ h	---	.003	.004	---	.001	.004	.022	.758	.010	.198	---
48 h $\leq t < 5$ d	---	---	.005	---	---	.006	.024	.907	.001	.058	---
5 d $\leq t < 10$ d	---	---	.007	---	---	.008	.016	.969	---	---	---
10 d $\leq t < 15$ d	---	---	.014	---	---	.014	.006	.966	---	---	---
15 d $\leq t < 20$ d	---	---	.026	---	---	.022	.002	.950	---	---	---
20 d $\leq t < 30$ d	---	---	.048	---	---	.034	.001	.917	---	---	---
30 d $\leq t < 40$ d	---	---	.152	---	---	.070	---	.777	---	---	---
40 d $\leq t < 50$ d	---	---	.378	---	---	.105	---	.517	---	---	---
50 d $\leq t < 60$ d	---	---	.652	---	---	.108	---	.240	---	---	---
60 d $\leq t < 70$ d	---	---	.835	---	---	.083	---	.082	---	---	---
$t \geq 70$ d	---	---	.920	---	---	.055	---	.024	---	---	---

8.2.2 Steam Jet Air Ejector (SJAE) Noble Gas Activity Monitors (RM-17-150A and RM-17-150B)

The SJAE noble gas activity monitors are shown in Figure 9-2.

8.2.2.1 Method to Determine the Setpoints of the Steam Jet Air Ejector Offgas Activity Monitors (RM-17-150A and RM-17-150B)

The SJAE noble gas activity monitor response in mR/hr at the limiting release rate is the setpoint, denoted, and is determined as follows:

(8-21)

$$R_{spt}^{SJAE} = 1.6E+05 \quad S_g \quad \frac{1}{F}$$

$$(mR/hr) \quad \left(\frac{\mu Ci}{sec}\right) \quad \left(\frac{mR-cc}{hr-\mu Ci}\right) \quad \left(\frac{sec}{cc}\right)$$

R_{spt}^{SJAE} = Response of the monitor at the limiting release rate (mR/hr)

1.6E+05 = Limiting release rate for the SJAE specified in Technical Specification 3.8.K.1 (μCi/sec)

S_g = Detector counting efficiency from the most recent calibration ((mR/hr)/(μCi/cc))

F = SJAE gaseous discharge flow (cc/sec)

8.2.2.2 Basis for the SJAE Noble Gas Activity Monitor Setpoint

The SJAE noble gas activity monitor setpoint must ensure that Technical Specification 3.8.K.1 is not exceeded. The derivation of Equation 8-21 is straightforward. Simply taking Equation 8-16 and substituting the limiting release rate at the SJAE for $\dot{Q}$ yields Equation 8-21, the setpoint equation for the SJAE noble gas activity monitor.

9.0 LIQUID AND GASEOUS EFFLUENT STREAMS, RADIATION MONITORS AND RADWASTE TREATMENT SYSTEMS

Figure 9-1 shows the normal (design) radioactive liquid effluent streams, radiation monitors, and the appropriate Liquid Radwaste Treatment System. Figure 9-2 shows the normal (design) gaseous effluent systems, radiation monitors, and the appropriate Gaseous Radwaste Treatment System.

9.1 In-Plant Radioactive Liquid Effluent Pathways

The Liquid Radwaste System collects, processes, stores, and disposes of all radioactive liquid wastes. Except for the cleanup phase separator equipment, the condensate backwash receiving tank and pump and waste sample tanks, floor drain sample tank and waste surge tank, the entire Radwaste System is located in the Radwaste Building. The Radwaste System is controlled from a panel in the Radwaste Building Control Room.

The Liquid Radwaste System consists of the following components:

1. Floor and equipment drain system for handling potentially radioactive wastes.

2. Tanks, piping, pumps, process equipment, instrumentation and auxiliaries necessary to collect, process, store, and dispose of potentially radioactive wastes.

The liquid radwastes are classified, collected, and treated as either high purity, low purity, chemical or detergent wastes. "High" purity and "low" purity mean that the wastes have low conductivity and high conductivity, respectively. The purity designation is not a measure of the amount of radioactivity in the wastes.

High purity liquid wastes are collected in the 25,000-gallon waste collector tank. They originate from the following sources:

1. Drywell equipment drains.
2. Reactor Building equipment drains.
3. Radwaste Building equipment drains.
4. Turbine Building equipment drains.
5. Decanted liquids from cleanup phase separators.
6. Decanted liquids from condensate phase separators.
7. Resin rinse.

Low purity liquid wastes are collected in the 25,000-gallon floor drain collector tank. They originate from the following sources:

1. Drywell floor drains.
2. Reactor Building floor drains.
3. Radwaste Building floor drains.
4. Turbine Building floor drains.
5. Other floor drains in RCA (e.g., AOG and Service Building, stack, etc.).

Chemical wastes are collected in the 4,000-gallon chemical waste tank and then pumped to the floor drain collector tank. Chemical wastes arise from the chemical laboratory sinks, the laboratory drains and sample sinks. Radioactive decontamination solutions are classified as detergent waste and collected in the 1,000-gallon detergent waste tank.

Once the wastes are collected in their respective waste tanks, they are processed in the most efficient manner and discharged or reused in the nuclear system. From the waste collector tank, the high purity wastes are processed in one of three alternative filter demineralizers and then, if needed, in one "polishing" demineralizer. After processing, the liquid is pumped to a waste sample tank for testing and then recycled for additional processing, transferred to the condensate storage tank for reuse in the nuclear system or discharged.

The low purity liquid wastes are normally processed through the floor drain filter demineralizer and collected in the floor drain sample tank for discharge or they are combined with high purity wastes and processed as high purity wastes.

Chemical wastes are neutralized and combined with low purity wastes for processing as low purity wastes.

Although there is only one discharge pathway from the Radwaste System to the river, there are three locations within the Radwaste System from which releases can be made. They are: the detergent waste tank (detergent wastes), the floor drain sample tank (chemical and low purity wastes), and waste sample tank (high purity wastes). The contents of any of these tanks can be released directly to the river.

The liquid wastes collected in the tanks are handled on a batch basis. The tanks are sampled from the radwaste sample sink and the contents analyzed for radioactivity and water purity. A release is allowed once it is determined that the activity in the liquid wastes will not exceed Control release limits.

A discharge from any of the tanks is accomplished by first starting the sample pumps, opening the necessary valves, and positioning the flow controller. The release rate in the discharge line is set between 0 and 50 gpm. The dilution pumps which supply 20,000 gpm of dilution water are then started. An interlock does not allow discharge to the river when dilution water is unavailable.

The effluent monitor (No. 17/350) in the discharge line provides an additional check during the release. The alarm or trip setpoint on the monitor is set according to the effluent Control limits and an analysis of the contents of the tank. The monitor warns the operator if the activity of the liquid waste approaches regulatory limits. In response to a warning signal from the monitor, the operator may reduce the flow rate or stop the discharge.

9.2 In-Plant Radioactive Gaseous Effluent Pathways

The gaseous radwaste system includes subsystems that dispose of gases from the main condenser air ejectors, the startup vacuum pump, the gland seal condenser, the standby gas treatment system and station ventilation exhausts.

The processed gases are routed to the plant stack for dilution and elevated release to the atmosphere.

The plant stack provides an elevated release point for the release of waste gases. Stack drainage is routed to the liquid radwaste collection system through loop seals.

The air ejector Advanced Off-Gas Subsystem (AOG) reduces the ejector radioactive gaseous release rates to the atmosphere. The AOG System consists of a hydrogen dilution and recombiner subsystem, a dual moisture removal/dryer subsystem, a single charcoal absorber subsystem, and dual vacuum pumps. Equipment is located in shielded compartments to minimize the exposure of maintenance personnel.

Radioactive releases from the air ejector off-gas system consist of fission product noble gases, activation product gases, halogens, and particulate daughter products from the noble gases. The particulates and halogens are effectively removed by the charcoal beds and high efficiency particulate filters in the AOG System. The activation product gases that are generated in significant quantities have very short half-lives and will decay to low levels in the holdup pipe, as well as in the absorber beds. The noble gases, therefore, are expected to provide the only significant contribution to off-site dose. The charcoal off-gas system is designed to provide holdup of 24 hours for krypton and 16.6 days for xenon at a condenser air inleakage rate of 30 scfm.

Steam dilution, process control, and instrumentation systems are designed to prevent an explosive mixture of hydrogen from propagating beyond the air ejector stages. An explosive mixture of hydrogen should never exist in the recombiner subsystem, "30-minute" delay pipe, condenser/dryer, or charcoal absorber beds. To prevent a hydrogen explosion in the recombiner/preheater and upstream lines during shutdown, the residual off-gas steam mixture containing hydrogen is purged with steam or air. Starting procedures insure sufficient steam is introduced upstream of the preheater to dilute any hydrogen entering the AOG System as the air

ejector line is prepared for operation. To prevent operating unsafely, instrumentation is used to detect an explosive mixture.

Hydrogen control is accomplished by providing redundant hydrogen analyzers on the outlet from the Recombiner System. These analyzers initiate recombiner system shutdown and switchover if the hydrogen concentration at the system outlet exceeds 2% by volume. During an automatic shutdown, two main air process valves close to isolate the recombiner system. Additionally, the recombiner bed temperatures and recombiner outlet temperature provide information about recombiner performance to insure that inflammable hydrogen mixtures do not go beyond the recombiner.

Should a number of unlikely events occur, it would be hypothetically possible for a hydrogen explosion to occur in the off-gas system. Such an explosion within the recombiner system could propagate into the large "30-minute" delay pipe, through the condenser/dryer subsystem, and into the charcoal absorber tanks. However, the recombiner/adsorber subsystems, piping, and vessels are designed to withstand hydrogen detonation pressures of 500 psi at a minimum so that no loss of integrity would result. Furthermore, the seven tanks of charcoal would significantly attenuate a detonation shock wave and prevent damage to the downstream equipment.

During normal operation, the dryer/adsorber subsystem may be bypassed if it becomes unavailable provided the releases are within effluent Control limits. With the dryer/adsorber subsystem bypassed, the air ejector off-gas exhausts through the recombiner/condenser subsystems, and the 30-minute delay pipe.

The off-gas mixture combines with steam at the air ejector stage to prevent an inflammable hydrogen mixture of 4% by volume from entering the downstream hydrogen recombiners. Approximately 6,400 lb/hr of steam introduced at the second stage air ejector reduces the concentration of hydrogen to less than 3% by volume.

The recombiner subsystem consists of a single path leading from the hydrogen dilution steam jet ejectors to two parallel flow paths for hydrogen recombination. Each recombination subsystem is capable of operating independently of the other and each is capable of handling the condenser off-gas at a startup design flow of 1,600 lb/hr air and the normal off-gas design flow rate of 370 lb/hr. The major components of each recombiner flow path are a preheater, a hydrogen-oxygen recombiner, and a desuperheating condenser.

The preheater assures that the vapor entering the hydrogen-oxygen recombiner is heated to approximately 300°F. At this temperature, the water vapor in the stream becomes superheated steam, thereby, protecting the recombiner catalyst.

During passages through the recombiner, the recombination of H_2 and O_2 in an exothermic reaction increases the stream temperature to approximately 520°F. This recombination results in a maximum effluent H_2 concentration of 0.1% by volume.

The desuperheating condenser is designed to remove the heat of recombination and condense the steam from the remaining off-gas. The condensers discharge the off-gas through moisture separators into the initial portion of an underground 24-inch diameter delay pipe which allows for 40% of the total system holdup volume. The pipe slopes away from the off-gas particulate (HEPA) filters in both directions for drainage purposes. Loop seals prevent gas escaping through drainage connections. Shorter lived radionuclides undergo a substantial decrease in activity in this section of the system. The preheaters/recombiners operate at pressures slightly above atmospheric; the condenser and the subsystems that follow operate at subatmospheric pressures.

Particulate (HEPA) filters with flame suppressant prefilters are located at the exit side of the delay pipe ahead of the moisture removal subsystem to remove radioactive particulates generated in the delay pipe.

In the moisture removal/dryer subsystem, the moisture of the gas is reduced to increase the effectiveness of the charcoal absorber beds downstream. The subsystem consists of two parallel cooling condensers and gas dryer units. Each condenser is cooled by a mechanical glycol/water refrigeration system that cools the off-gas to -40°F as it removes bulk moisture. The dryer is designed to remove the remaining moisture by a molecular sieve desiccant to a dew point of less than -40°F (1% RH). One of the dryers absorbs moisture from the off-gas; the other desorbs moisture by circulating heated air through the bed in closed cycle.

The mixed refrigerant/dryer concept improves the reliability of the system. If the refrigerant system fails, the two dryer beds operate in parallel to remove the moisture and maintain the off-gas near the design dew point (-40°F). If the dryer fails, the -40°F dew point air leaving the mechanical system can enter the guard bed for over 6 hours without affecting the performance of the charcoal beds downstream.

The charcoal absorber subsystem consists of seven tanks of charcoal preceded by a smaller charcoal guard bed upstream. The guard bed protects the seven main tanks from excessive radioactivity levels or moisture in the event of a malfunction upstream in the moisture removal subsystem. The guard bed also removes compounds which might hinder noble gas delay. The seven tanks hold a minimum of approximately 90,000 pounds of charcoal.

The first two main tanks can be bypassed and used for storing a "batch of high activity" gas for static decay. The remaining five are all in series with no bypassing features so that the off-gas to the stack must be delayed.

Redundant particulate (HEPA) after-filters are used to remove charcoal fines prior to the vacuum pumps.

A water-sealed vacuum pump boosts the gas stream pressure to slightly over-atmospheric pressure before it is vented through the stack. To assure maintaining constant operating pressures in the system, a modulating bypass valve will recirculate process gas around the pump as required. During periods of high flow rates, both pumps can be operated in parallel.

Discharge of the vacuum pump then passes through the remaining 60% of the delay pipe prior to being vented through the station stack.

The gland seal off-gas subsystem collects gases from the gland seal condenser and the mechanical vacuum pump and passes them through a charcoal filter (if required) and then through holdup piping prior to release to the stack. The gases from the gland seal condenser system are discharged to the atmosphere via the ventilation stack after passing through the filter for iodine removal (if required) and then through the same 1-3/4 minute holdup piping that is used for the startup vacuum pump system. One automatic valve on the discharge side of each steam packing exhauster closes upon the receipt of high level radiation signal from the main steam line radiation monitoring subsystem to prevent the release of excessive radioactive material to the atmosphere. The exhausters are shut down at the same time the valves close. In addition, the mechanical vacuum pump is automatically isolated and stopped by a main steam line high radiation signal. The filter assembly is located in the air ejector room.

The release of significant quantities of gaseous and particulate radioactive material is prevented by the combination of the design of the air ejector AOG system and automatic isolation of the system from the stack. Gas flow from the main condenser stops when the air

ejectors are automatically isolated from the main condenser by either a high radiation signal in the main steam line or by high temperature and/or pressure signals from the AOG System. The gland seal off-gas system is automatically isolated and stopped by a main steam line high radiation signal. In addition, monitoring the stack release provides a backup warning of abnormal conditions.

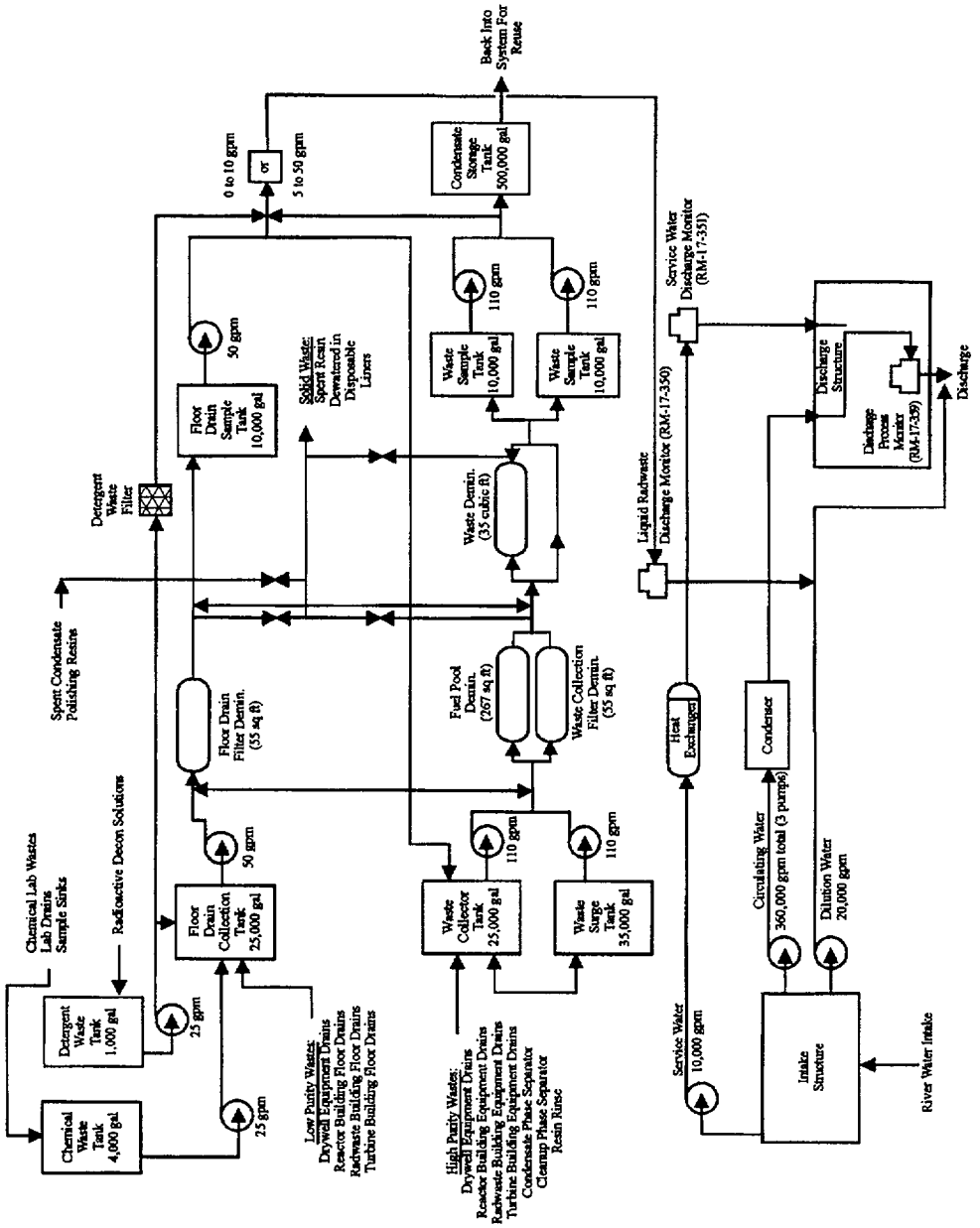

FIGURE 9-1: Radioactive Liquid Effluent Streams, Radiation Monitors, and Radwaste Treatment System at Vermont Yankee*

*Normal (design) radioactive process streams only are shown.

VY ZIPDISK\YOCOMNEW\FIGS

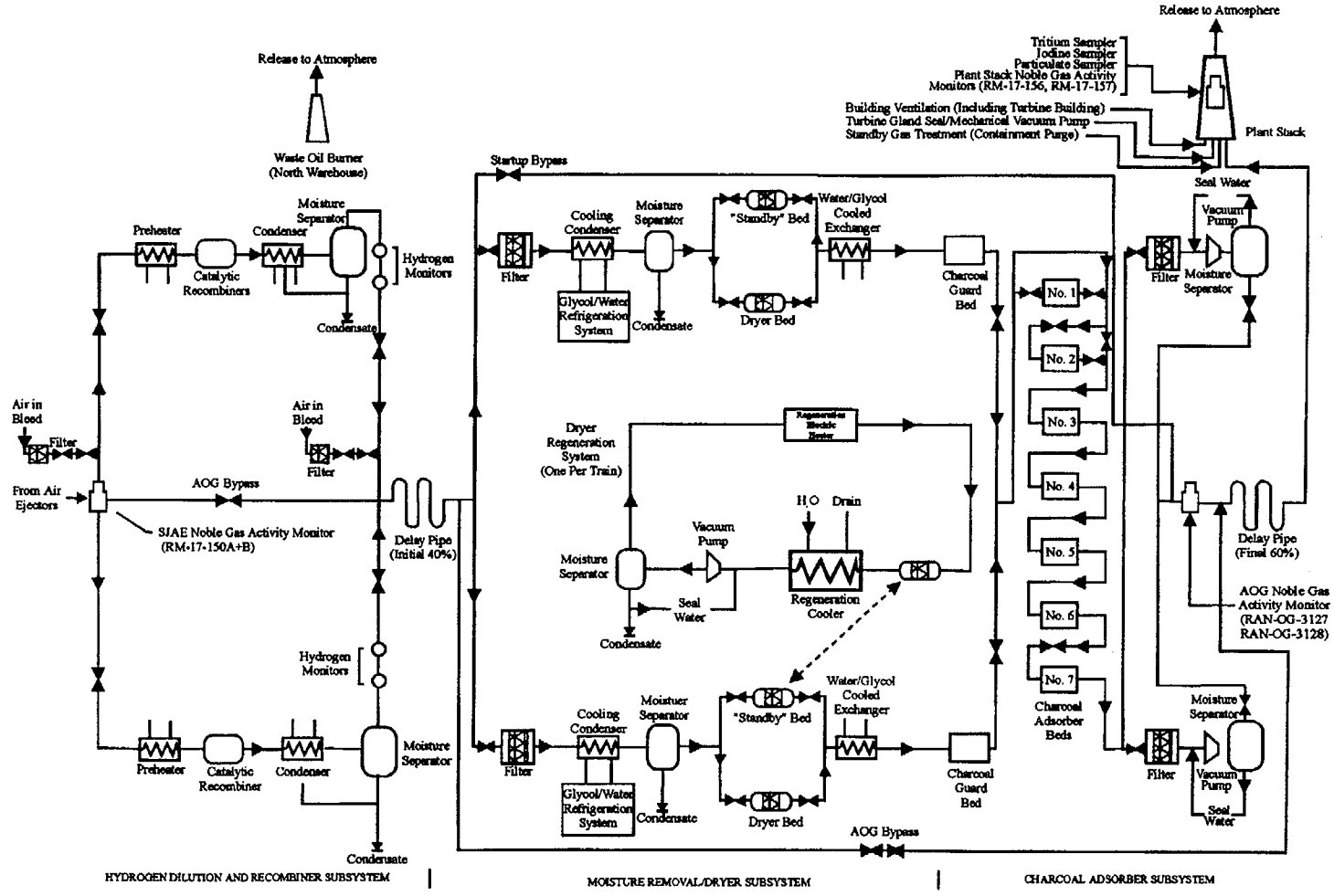

FIGURE 9-2: Radioactive Gaseous Effluent Streams, Radiation Monitors, and Radwaste Treatment System at Vermont Yankee*

*Normal (design) radioactive process streams only are shown.

VY ZIPDISK\VYODCM\NEWFIGS

10.0 UNIQUE REPORTING REQUIREMENTS

10.1 Annual Radioactive Effluent Release Report

In accordance with 10CFR 50.36a, the Radioactive Effluent Release Report covering the operation of the unit shall be submitted by May 15 of each year.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes the format for Table 3 in Appendix B of Regulatory Guide 1.21 shall be supplemented with three additional categories: class of solid wastes (as defined by 10CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity), and solidification agent or absorbent, if any.

In addition, the Radioactive Effluent Release Report shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. The Radioactive Effluent Release Report shall also include an assessment of the radiation doses from radioactive effluents to member(s) of the public due to any allowed recreational activities inside the site boundary during the previous calendar year. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports. For any batch or discrete gas volume releases, the meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. For radioactive materials released in continuous effluent streams, quarterly average meteorological conditions concurrent with the quarterly release period shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the Off-Site Dose Calculation Manual (ODCM).

With the limits of Control 3.4.1 being exceeded during the calendar year, the Radioactive Effluent Release Report shall also include an assessment of radiation doses to the

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

likely most exposed real member(s) of the public from reactor releases (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40CFR190, Environmental Radiation Protection Standards for Nuclear Power Operation.

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

With the quantity of radioactive material in any outside tank exceeding the limit of Technical Specification 3.8.D.1, describe the events leading to this condition in the next Radioactive Effluent Release Report.

If inoperable radioactive liquid effluent monitoring instrumentation is not returned to operable status prior to the next release pursuant to Note 4 of Control Table 3.1.1, explain in the next Radioactive Effluent Report the reason(s) for delay in correcting the inoperability.

If inoperable gaseous effluent monitoring instrumentation is not returned to operable status within 30 days pursuant to Note 5 of Control Table 3.1.2, explain in the next Radioactive Effluent Release Report the reason(s) for delay in correcting the inoperability.

With milk samples no longer available from one or more of the sample locations required by Control Table 3.5.1, identify the cause(s) of the sample(s) no longer being available, identify the new location(s) for obtaining available replacement samples, and include revised ODCM figure(s) and table(s) reflecting the new location(s) in the next Radioactive Effluent Release Report.

With a land use census identifying one or more locations which yield at least a 20 percent greater dose or dose commitment than the values currently being calculated in Control 4.3.3, identify the new location(s) in the next Radioactive Effluent Release Report.

Changes made during the reporting period to the Process Control Program (PCP) and to the Off-Site Dose Calculation Manual (ODCM), shall be identified in the next Radioactive Effluent Release Report.

10.2 Environmental Radiological Monitoring

The Annual Radiological Environmental Operating Report covering the operation of the unit during previous calendar year shall be submitted by May 15th of each year.

The report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period. The material provided shall be consistent with the objectives outlined in the ODCM and in 10CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of all radiological environmental samples taken during the report period pursuant to Table 7-1 and Figures 7-1 through 7-6. 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 possible in a supplementary report.

With the level of radioactivity in an environmental sampling media at one or more of the locations specified in Control Table 3.5.1 exceeding the reporting levels of Control Table 3.5.2, the condition shall be described in the next Annual Radiological Environmental Operating Report only if the measured level of radioactivity was not the result of plant effluents. With the radiological environmental monitoring program not being conducted as specified in Control Table 3.5.1, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence shall be included in the next Annual Radiological Environmental Operating Report.

The Annual Radiological Environmental Operating Report shall also include the results of the land use census required by Control 3.5.2. A summary description of the radiological environmental monitoring program including a map of all sampling locations keyed to a table giving distances and directions from the reactor shall be in the reports. If new environmental sampling locations are identified in accordance with Control 3.5.2, the new locations shall be identified in the next Annual Radiological Environmental Operating Report.

The reports shall also include a discussion of all analyses in which the LLD required by Control Table 4.5.1 was not achievable.

The results of license participation in the intercomparison program required by Control 3.5.3 shall be included in the reports. With analyses not being performed as required by Control 3.5.3, the corrective actions taken to prevent a recurrence shall be reported to the Commission in the next Annual Radiological Environmental Operating Report.

10.3 Special Reports

Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

10.3.1 Liquid Effluents (Controls 3.2.2 and 3.2.3)

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the limits of Control 3.2.2, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to assure that subsequent releases will be in compliance with the limits of Control 3.2.2.

With liquid radwaste being discharged without processing through appropriate treatment systems and estimated doses in excess of Control 3.2.3, prepare and submit to the Commission within 30 days a special report which includes the following information:

(1) explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reasons for the inoperability;

(2) action(s) taken to restore the inoperable equipment to operable status; and

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

10.3.2 Gaseous Effluents (Controls 3.3.2, 3.3.3, 3.3.4 and 3.3.5)

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the limits of Control 3.3.2, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and the corrective action(s) taken to assure that subsequent releases will be in compliance with the limits of Control 3.3.2. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and/or radionuclides in particulate form exceeding any of the limits of Control 3.3.3, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and the corrective action(s) taken to assure that subsequent releases will be in compliance with the limits of Control 3.3.3.

With gaseous radwaste being discharged without processing through appropriate treatment systems as defined in Control 3.3.4 for more than seven (7) consecutive days, or in excess of the limits of Control 3.3.5, prepare and submit to the Commission within 30 days a special report which includes the following information:

(1) explanation of why gaseous radwaste was being discharged without treatment (Control 3.3.4), or with resultant doses in excess of Control 3.3.5, identification of any inoperable equipment or subsystems, and the reasons for the inoperability;

(2) action(s) taken to restore the inoperable equipment to operable status; and

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

10.3.3 Total Dose (Control 3.4.1)

With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding the limits of Control 3.4.1, prepare and submit to the Commission within 30 days a special report which defines the corrective action(s) to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Control 3.4.1 and includes the schedule for achieving conformance with these limits. This special report, required by 10CFR Part 20.2203(a)(4), shall include an analysis that estimates the radiation exposure (dose) to a member of the public from station 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 doses exceed any of the limits of Control 3.4.1, and if the release condition resulting in violation of 40CFR Part 190 has not already been corrected, the special report shall include a request for a variance in accordance with the provisions of 40CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

10.3.4 Radiological Environmental Monitoring (Control 3.5.1)

With the level of radioactivity as the result of plant effluents in an environmental sampling media at one or more of the locations specified in Control Table 3.5.1 exceeding the reporting levels of Control Table 3.5.2, prepare and submit to the Commission within 30 days from the receipt of the Laboratory Analyses a special report

which includes an evaluation of any release conditions, environmental factors or other factors which caused the limits of Control Table 3.5.2 to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents, however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

10.3.5 Land Use Census (Control 3.5.2)

With a land use census not being conducted as required by Control 3.5.2, prepare and submit to the Commission within 30 days a special report which identifies the reasons why the survey was not conducted, and what steps are being taken to correct the situation.

10.4 Major Changes to Radioactive Liquid, Gaseous, and Solid Waste Treatment Systems**

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

A. Shall be reported to the commision in the Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

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

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

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

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

5. An evaluation of the change, which shows the expected maximum exposures to member(s) of the public at the site boundary and to the general population that differ from

**Licensee may choose to submit the information called for in this reporting requirement as part of the annual FSAR update.

those previously estimated in the license application and amendments thereto;

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

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

8. Documentation of the fact that the change was reviewed and found acceptable by PORC.

B. Shall become effective upon review and acceptance by PORC and approval by the Plant Manager.