EFFLUENT AND WASTE DISPOSAL

SEMIANNUAL REPORT

FOR

FIRST AND SECOND QUARTERS, 1994

VERMONT YANKEE NUCLEAR POWER STATION

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TABLE 1A

Vermont Yankee

Effluent and Waste Disposal Semiannual Report

First and Second Quarters, 1994

Gaseous Effluents - Summation of All Releases

		Unit	Quarter 1	Quarter 2	Est. Total Error, %
A.	Fission and Activation Gases				
1.	Total release	Ci	<7.77E+02	<7.86E+02	±1.00E+02
2.	Average release rate for period	uCi/sec	<9.88E+01	<1.00E+02	
3.	Percent of Tech. Spec. limit (1)	%			
B.	Iodines				
1.	Total Iodine-131	Ci	9.00E-04	8.91E-04	±5.00E+01
2.	Average release rate for period	uCi/sec	1.14E-04	1.13E-04	
3.	Percent of Tech. Spec. limit (1)	%			
C.	Particulates				
1.	Particulates with T-1/2 > 8 days	Ci	9.83E-04	4.94E-04	±5.00E+01
2.	Average release rate for period	uCi/sec	1.25E-04	6.28E-05	
3.	Percent of Tech. Spec. limit (1)	%			
4.	Gross alpha radioactivity	Ci	3.27E-06	1.98E-06	
D.	Tritium				
1.	Total release	Ci	5.54E+00	4.06E+00	±5.00E+01
2.	Average release rate for period	uCi/sec	7.05E-01	5.16E-01	
3.	Percent of Tech. Spec. limit (1)	%			

(1) Percent of Technical Specification limit will be provided in the Supplemental Effluent and Waste Disposal Report to be submitted per Technical Specification 6.7.C.1.

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TABLE 1B

Vermont Yankee

Effluent and Waste Disposal Semiannual Report

First and Second Quarters, 1994

Gaseous Effluents - Elevated Release

	Nuclides Released	Unit	Continuous Mode Quarter 1	Continuous Mode Quarter 2	Batch Mode(1) Quarter 1	Batch Mode(1) Quarter 2
1.	Fission Gases					
	Krypton-85	Ci	ND	ND		
	Krypton-85m	Ci	<2.99E+00	<3.06E+00		
	Krypton-87	Ci	<2.21E+01	<2.23E+01		
	Krypton-88	Ci	<1.08E+01	<1.12E+01		
	Xenon-133	Ci	<1.40E+00	<1.41E+00		
	Xenon-135	Ci	<1.91E+01	<1.94E+01		
	Xenon-135m	Ci	<1.47E+02	<1.44E+02		
	Xenon-138	Ci	<5.74E+02	<5.85E+02		
	Unidentified	Ci				
	Total for period	Ci	<7.77E+02	<7.86E+02		
2.	Iodines					
	Iodine-131	Ci	9.00E-04	8.91E-04		
	Iodine-133	Ci	4.33E-03	4.57E-03		
	Iodine-135	Ci	ND	ND		
	Total for period	Ci	5.23E-03	5.46E-03		
3.	Particulates					
	Strontium-89	Ci	3.94E-04	3.48E-04		
	Strontium-90	Ci	6.71E-06	1.08E-05		
	Cesium-134	Ci	ND	ND		
	Cesium-137	Ci	2.27E-06	1.21E-05		
	Barium-Lanthanum-140	Ci	5.60E-04	7.91E-05		
	Manganese-54	Ci	3.98E-06	ND		
	Chromium-51	Ci	ND	ND		
	Cobalt-58	Ci	ND	ND		
	Cobalt-60	Ci	1.61E-05	4.40E-05		
	Cerium-141	Ci	ND	ND		
	Zinc-65	Ci	ND	ND		
	Unidentified	Ci				
	Total for period	Ci	9.83E-04	4.94E-04		

(1) There were no batch mode gaseous releases for this reporting period.
ND Not detected at the plant stack.

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TABLE 1C

Vermont Yankee

Effluent and Waste Disposal Semiannual Report

First and Second Quarters, 1994

Gaseous Effluents - Ground Level Releases[1]

	Nuclides Released	Unit	Continuous Mode Quarter 1	Continuous Mode Quarter 2	Batch Mode Quarter 1	Batch Mode Quarter 2
1.	Fission Gases	Ci				
	Krypton-85	Ci				
	Krypton-85m	Ci				
	Krypton-87	Ci				
	Krypton-88	Ci				
	Xenon-133	Ci				
	Xenon-135	Ci				
	Xenon-135m	Ci				
	Xenon-138	Ci				
	Unidentified	Ci				
	Total for period	Ci				
2.	Iodines					
	Iodine-131	Ci				
	Iodine-133	Ci				
	Iodine-135	Ci				
	Total for period	Ci				
3.	Particulates					
	Strontium-89	Ci				
	Strontium-90	Ci				
	Cesium-134	Ci				
	Cesium-137	Ci				
	Barium-Lanthanum-140	Ci				
	Manganese-54	Ci				
	Chromium-51	Ci				
	Cobalt-58	Ci				
	Cobalt-60	Ci				
	Cerium-141	Ci				
	Zinc-65	Ci				
	Unidentified	Ci				
	Total for period	Ci				

(1) There were no ground level releases of gaseous effluents for this reporting period.

TABLE 1D

Vermont Yankee

Effluent and Waste Disposal Semiannual Report

First and Second Quarters, 1994

Gaseous Effluents - Nonroutine Releases[1]

	Nuclides Released	Unit	Continuous Mode Quarter 1	Continuous Mode Quarter 2	Batch Mode Quarter 1	Batch Mode Quarter 2
1.	Fission Gases	Ci				
	Krypton-85	Ci				
	Krypton-85m	Ci				
	Krypton-87	Ci				
	Krypton-88	Ci				
	Xenon-133	Ci				
	Xenon-135	Ci				
	Xenon-135m	Ci				
	Xenon-138	Ci				
	Unidentified	Ci				
	Total for period	Ci				
2.	Iodines					
	Iodine-131	Ci				
	Iodine-133	Ci				
	Iodine-135	Ci				
	Total for period	Ci				
3.	Particulates					
	Strontium-89	Ci				
	Strontium-90	Ci				
	Cesium-134	Ci				
	Cesium-137	Ci				
	Barium-Lanthanum-140	Ci				
	Manganese-54	Ci				
	Chromium-51	Ci				
	Cobalt-58	Ci				
	Cobalt-60	Ci				
	Cerium-141	Ci				
	Zinc-65	Ci				
	Unidentified	Ci				
	Total for period	Ci				

(1) There were no nonroutine gaseous releases for this reporting period.

TABLE 2A
Vermont Yankee
Effluent and Waste Disposal Semiannual Report
First and Second Quarters, 1994
Liquid Effluents - Summation of All Releases

There were no liquid releases during the first or second quarters of 1994.

TABLE 2B
Vermont Yankee
Effluent and Waste Disposal Semiannual Report
First and Second Quarters, 1994
Liquid Effluents - Nonroutine Releases

There were no nonroutine or accidental releases during the first or second quarters of 1994.

TABLE 3

Vermont Yankee

Effluent and Waste Disposal Semiannual Report

First and Second Quarters, 1994

Solid Waste and Irradiated Fuel Shipments

A. Solid Waste Shipped Off-Site for Burial or Disposal (Not Irradiated Fuel)

		Unit	6-Month Period	Est. Total Error, %
1.	Type of Waste			
	a. Spent resins, filter sludges, evaporator bottoms, etc.	m^3	4.35E+01	±7.50E+01
		Ci	1.12E+02	
	b. Dry compressible waste, contaminated equipment, etc.	m^3	1.84E+01	±7.50E+01
		Ci	4.52E+01	
	c. Irradiated components, control rods, etc.	m^3		
		Ci		
2.	Estimate of Major Nuclide Composition (By Type of Waste):			
	a. Zinc-65	%	2.75E+01	
	Cesium-137	%	2.16E+01	
	Cobalt-60	%	1.85E+01	
	Iron-55	%	7.59E+00	
	Cesium-134	%	7.45E+00	
	Manganese-54	%	7.39E+00	
	Nickel-63	%	4.13E+00	
	b. Iron-55	%	7.98E+01	
	Cobalt-60	%	9.04E+00	
	Manganese-54	%	7.27E+00	

3. Solid Waste Disposition:

Number of Shipments	Mode of Transportation	Destination
9 Resin Shipments	Truck	Barnwell, SC
48 Partial Shipments From Processor to Burial	Truck	Barnwell, SC

B. Irradiated Fuel Shipments (Disposition): None

C. Supplemental Information
1) Class of solid waste container shipped: 110A (Unstable), 7B
2) Types of containers used: 9 Type A, 108 Strong-Tight Container
3) Solidification agent or absorbent: None

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APPENDIX A

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT
Supplemental Information
First and Second Quarters, 1994

Facility: Vermont Yankee Nuclear Power Station

Licensee: Vermont Yankee Nuclear Power Corporation

1A. Technical Specification Limits - Dose and Dose Rate

Technical Specification and Category		Limit
a.	Noble Gases	
3.8.E.1	Total body dose rate	500 mrem/yr
3.8.E.1	Skin dose rate	3000 mrem/yr
3.8.F.1	Gamma air dose	5 mrad in a quarter
3.8.F.1	Gamma air dose	10 mrad in a year
3.8.F.1	Beta air dose	10 mrad in a quarter
3.8.F.1	Beta air dose	20 mrad in a year
b.	Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form With Half-Lives Greater Than 8 Days	
3.8.E.1	Organ dose rate	1500 mrem/yr
3.8.G.1	Organ dose	7.5 mrem in a quarter
3.8.G.1	Organ dose	15 mrem in a year
c.	Liquids	
3.8.B.1	Total body dose	1.5 mrem in a quarter
3.8.B.1	Total body dose	3 mrem in a year
3.8.B.1	Organ dose	5 mrem in a quarter
3.8.B.1	Organ dose	10 mrem in a year

2A. Technical Specification Limits - Concentration

Technical Specification and Category		Limit
a.	Noble Gases	No MPC limits (No ECL Limits)
b.	Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form With Half-Lives Greater Than 8 Days	No MPC limits (No ECL Limits)

c. Liquids

3.8.A.1 Total fraction of MPC (ECL) excluding noble gases (10CFR20, Appendix B, Table II, Column 2): ≤1.0

3.8.A.1 Total noble gas concentration: ≤2.00E-04 uCi/cc

3. Average Energy

Provided below are the average energy ($\bar{E}$) of the radionuclide mixture in releases of fission and activation gases, if applicable.

a. Average gamma energy: 1st Quarter 8.30E-01 MeV/dis
2nd Quarter 8.30E-01 MeV/dis

b. Average beta energy: Not Applicable

4. Measurements and Approximations of Total Radioactivity

Provided below are the methods used to measure or approximate the total radioactivity in effluents and the methods used to determine radionuclide composition.

a. Fission and Activation Gases

Continuous stack monitors monitor gross Noble Gas radioactivity released from the plant stack. Total Noble Gas release rates are calculated using this monitor. To determine the isotopic breakdown of the release, samples are taken of the Steam Jet Air Ejector, which is the source gas for the releases. These samples are analyzed by gamma spectroscopy to determine the isotopic composition. The isotopic composition is then proportioned to the gross releases determined from the stack monitor to quantify the individual isotopic releases. These are indicated in Table 1B and the totals of Table 1A.

The error involved in these steps may be approximately ±100 percent.

b. Iodines

Continuous isokinetic samples are drawn from the plant stack through a particulate filter and charcoal cartridge. The filters and cartridges are normally removed weekly and are analyzed for Iodine-131, 132, 133, 134, and 135. The error involved in these steps may be approximately ±50 percent.

c. Particulates

The particulate filters described in b. above are also counted for particulate radioactivity. The error involved in this sample is also approximately ±50 percent.

d. Liquid Effluents

Radioactive liquid effluents released from the facility are continuously monitored. Measurements are also made on a representative sample of each batch of radioactive liquid effluents released. For each batch, station records are retained of the total activity (mCi) released, concentration (uCi/ml) of gross radioactivity, volume (liters), and approximate total quantity of water (liters) used to dilute the liquid effluent prior to release to the Connecticut River.

Each batch of radioactive liquid effluent released is analyzed for gross gamma and gamma isotopic radioactivity. A monthly proportional composite sample, comprising an aliquot of each batch released during a month, is analyzed for tritium and gross alpha radioactivity. A quarterly proportional composite sample, comprising an aliquot of each batch released during a quarter, is analyzed for Sr-89, Sr-90, and Fe-55.

There were no liquid releases during the reporting period.

5. Batch Releases

a. Liquid

There were no routine liquid batch releases during the reporting period.

b. Gaseous

There were no routine gaseous batch releases during the reporting period.

6. Abnormal Releases

a. Liquid

There were no nonroutine liquid releases during the reporting period.

b. Gaseous

There were no nonroutine gaseous releases during the reporting period.

APPENDIX B

LIQUID HOLDUP TANKS

Requirement: Technical Specification 3.8.D.1 limits the quantity of radioactive material contained in any outside tank. With the quantity of radioactive material in any outside tank exceeding the limits of Technical Specification 3.8.D.1, a description of the events leading to this condition is required in the next Semiannual Effluent Release Report per Technical Specification 6.7.C.1.

Response: The limits of Technical Specification 3.8.D.1 were not exceeded during this reporting period.

APPENDIX C

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Requirement: Radioactive liquid effluent monitoring instrumentation channels are required to be operable in accordance with Technical Specification Table 3.9.1. If an inoperable radioactive liquid effluent monitoring instrument is not returned to operable status prior to a release pursuant to Note 4 of Table 3.9.1, an explanation in the next Semiannual Effluent Release Report of the reason(s) for delay in correcting the inoperability are required per Technical Specification 6.7.C.1.

Response: Since the requirements of Technical Specification Table 3.9.1 governing the operability of radioactive liquid effluent monitoring instrumentation were met for this reporting period, no response is required.

APPENDIX D

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Requirement: Radioactive gaseous effluent monitoring instrumentation channels are required to be operable in accordance with Technical Specification Table 3.9.2. If inoperable gaseous effluent monitoring instrumentation is not returned to operable status within 30 days pursuant to Note 5 of Table 3.9.2, an explanation in the next Semiannual Effluent Release Report of the reason(s) for the delay in correcting the inoperability is required per Technical Specification 6.7.C.1.

Response: Since the requirements of Technical Specification Table 3.9.2 governing the operability of radioactive gaseous effluent monitoring instrumentation were met for this reporting period, no response is required.

APPENDIX E

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Requirement: The Radiological Environmental Monitoring Program is conducted in accordance with Technical Specification 3.9.C. With milk samples no longer available from one or more of the sample locations required by Technical Specification Table 3.9.3, Technical Specification 6.7.C.1 requires the following to be included in the next Semiannual Effluent Release Report: (1) identify the cause(s) of the sample(s) no longer being available, (2) identify the new location(s) for obtaining available replacement samples, and (3) include revised ODCM figure(s) and table(s) reflecting the new location(s).

Response: No changes were needed in the milk sampling locations specified in Technical Specification Table 3.9.3 due to sample unavailability.

APPENDIX F

LAND USE CENSUS

Requirement: A land use census is conducted in accordance with Technical Specification 3.9.D. With a land use census identifying a location(s) which yields at least a 20 percent greater dose or dose commitment than the values currently being calculated in Technical Specification 4.8.G.1, Technical Specification 6.7.C.1 requires the identification of the new location(s) in the next Semiannual Effluent Release Report.

Response: The 1994 land use census was not performed during this reporting period. It will be completed during the next reporting period (second half of 1994).

APPENDIX G

PROCESS CONTROL PROGRAM

Requirement: Technical Specification 6.12.A.1 requires that licensee initiated changes to the Process Control Program (PCP) be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made.

Response: The licensee-initiated changes to the Process Control Program in this reporting period are:

1. The title, the Radiation Protection Supervisor, was changed to the Radiation Protection Manager.

2. In the Introduction Section, compliance with 10CFR20 was added.

3. In Section 2.0, the spent resin liners were replaced by casks for storing highly radioactive filters.

4. In Section 3.0, the Bird centrifuge for dewatering of depleted resins was replaced by the RDS-1000 Radioactive Waste Dewatering System. In addition, Class A resins that exceed 1.0 µCi/cc isotopes with greater than 5 year half-lives will be disposed of in an approved High Integrity Container (HIC). The RDS-1000 System has been tested by Chem-Nuclear for certification in meeting the Barnwell Site criteria and disposal requirement for freestanding liquid. This change of the dewatering system did not reduce the overall conformance of the dewatered spent resins/filter media waste product to existing criteria for solid waste shipments and disposal.

5. In Section 5.0, the use of a box compactor for compacting DAW was discontinued.

6. In Section 11.0, the Procedure OP-2527 Sampling and Analysis for Radioactive Classification, was added for determining 10CFR61 scaling factors.

7. The title for AP-0504 procedure was changed to "Shipment of Radioactive Materials."

8. In the list of procedures for the PCP implementation, AP-021, Maintenance Requests, was replaced by OP-2527, Sampling and Analysis for Radwaste Classification.

The revised Process Control Program is attached.

VERMONT YANKEE NUCLEAR POWER CORPORATION

PROCESS CONTROL PROGRAM

REV 3

1/15/94

Submitted _____
Radiation Protection Manager

Approved _____ 9404
PORC

Approved _____
Plant Manager

Approved _____
Manager of Operations

VERMONT YANKEE NUCLEAR POWER CORPORATION

PROCESS CONTROL PROGRAM

Introduction:

The Vermont Yankee Nuclear Power Corporation Process Control Program (PCP) describes the administrative and technical controls on the radioactive waste systems which provide assurance that Vermont Yankee meets federal shipping and burial site requirements.

The PCP complies with Technical Specification 6.12 by describing process parameters, controls, tests, sampling and analysis to ensure compliance with 10 CFR 20, 10 CFR 71, and 10 CFR 61 Energy, 49 CFR 172-173 Transportation, state, and burial site regulation requirements.

1.0 Solidification

Vermont Yankee Nuclear Power Corporation does not routinely solidify liquid waste. If the use of solidification to dispose of any liquid waste is required, it will be done by an outside vendor under the vendor's PCP. This PCP will be reviewed and approved by the Plant Health Physicist and the Radiation Protection Manager prior to implementation. This review is to identify that there is sufficient supporting documentation of the vendor's PCP to give assurance that the final product will meet all requirements for transport and burial, and that sufficient procedural controls exist to assure safe operations.

2.0 Cartridge Filter Elements

Low activity cartridge filter elements will be air dried and handled as dry active waste. Filters that are too radioactive to be disposed of in this manner will be placed in casks and shipped for disposal.

3.0 Resins

Normal operations produce radioactive waste in the form of depleted resins. These resins are processed in the burial container using a rapid dewatering system (RDS-1000) manufactured by Chem-Nuclear Systems, Inc. Procedures provide detailed instructions for remote vacuum dewatering and air drying of these resins.

The system has been tested, by Chem-Nuclear, for certification in meeting the Barnwell Site Criteria and disposal requirement for free standing liquid. These tests are described in Chem-Nuclear's Topical Report on the RDS-1000 Radioactive Waste Dewatering System. In addition, to comply with the statement, "Any liquids present in waste packages shall be non-corrosive with respect to the container", Vermont Yankee tested the pH of various resin mixtures used by the plant in solution with water. The range was found to be 4.2 - 8.4. A solution is not considered corrosive if the pH is greater than 4.0.

A resin sample is taken from each liner prior to shipment. The sample is counted to determine the activity and waste classification. Class A resins that exceed 1.0 μCi/cc of isotopes with greater than 5 year half-lives and all Class B and C resins will be disposed of in an approved High Integrity Container (HIC).

Vendor supplied or temporary methods of processing resins may be used in lieu of the above process provided that the vendor or temporary process meets the requirements of quality described above and does not conflict with accepted burial criteria or safety requirements. Such methods will be reviewed and approved by the Plant Health Physicist and the Radiation Protection Manager prior to implementation.

4.0 Filter Liners

During refueling outages and normal operation, liquid radwaste processing may require use of a decanting filter on the condensate phase separators. A floating suction is used to decant the water and resin into a filter liner. Filtered water is pumped from the liner. When use of the liner is completed, a vacuum pump is attached to dewater the resin in the liner. The liner is dewatered for a minimum of 48 hours and until no more water is viewed from the pump discharge. A resin sample is taken from the liner and counted to determine the activity and waste classification.

5.0 Dry Active Waste (DAW)

DAW is compacted, as practical, or shipped to a vendor that sorts the material for processing or recycling. All DAW is examined before being compacted or shipped. Any liquids or items found that would compromise the integrity of the package are removed and separated as specified by procedure. Containers used for DAW shipments meet the criteria of 49 CFR 173.425a. or b. "No leakage of radioactive material", as specified in 49 CFR 173.425.b.1 will be met provided that no radioactive materials in quantities equal to or exceeding those specified in 49 CFR 173.443 are detected on the external surfaces of the package at any time during shipment.

6.0 Chelating Agents

In order to comply with 10 CFR 20 Appendix F, chelating agents are controlled by the plant chemistry department using procedure AP 0620.

7.0 Explosive Waste

No waste capable of detonation or of explosive decomposition or reaction will be disposed as per 10 CFR 61.56(a)(4).

8.0 Toxic Waste

No waste capable of generating toxic gases, vapors, or fumes will be disposed as per 10 CFR 61.56(a)(5).

9.0 Pyrophoric Waste

No waste that is pyrophoric will be disposed as per 10 CFR 61.56(a)(6).

10.0 High Integrity Containers (HICs)

Vermont Yankee Nuclear Power Corporation has contracted with various suppliers of approved HICS. South Carolina has approved PCPs for HICs used by Vermont Yankee. Any HIC Vermont Yankee may choose to use at some future time, will meet all applicable requirements.

11.0 Waste class determination

Vermont Yankee periodically performs laboratory analysis on all waste streams to determine the activity of radionuclides listed in Tables 1 and 2 of 10 CFR 61. Correlation analysis show that the relative concentration of each radionuclide, with respect to the overall activity in a given Vermont Yankee waste stream, remains constant over time. A set of scaling factors is determined which allows the activity of 10 CFR 61 radionuclides to be estimated using the results of gamma spectrometric analysis or direct gamma dose rate measurements.

For resin wastes, analysis is performed on samples of each source of resin comprising the contents of a burial container. Scaling factors are applied to the activity of radionuclides identified by gamma spectrometry analysis to determine the activity of those radionuclides which are not detected in the gamma spectrum.

For DAW, dose rate-to-curie conversion calculations are performed to determine the total activity present in a container. Scaling factors are applied to the container's total curie content to determine the activity of individual radionuclides.

Specific procedures for determining 10 CFR 61 scaling factors are contained in OP 2527 Sampling and Analysis for Radwaste Classification. Once the activity of each radionuclide in a burial container is estimated, the waste classification is derived using methods required by 10 CFR 61. Specific procedures for waste class determination are contained in AP 0504, Shipment of Radioactive Material.

PROCEDURES WHICH IMPLEMENT THE PCP

1.	AP 0504	Shipment of Radioactive Materials
2.	OP 2511	Radwaste Cask, Drum, and Box Handling
3.	OP 2527	Sampling and Analysis for Radwaste Classification
4.	OP 2151	Liquid Radwaste
5.	OP 2153	Solid Radwaste
6.	AP 0620	Chemical Material Control

APPENDIX H

OFF-SITE DOSE CALCULATION MANUAL

Requirement: Technical Specification 6.13.A.1 requires that licensee initiated changes to the Off-Site Dose Calculation Manual (ODCM) be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective.

Response: The ODCM was amended to include new meteorological dispersion modeling that takes into account changes to both plant design and environmental air flow affects. The first change incorporates a new estimate of the plant stack exit flow rate (175,000 cfm) resulting from the Turbine Building roof ventilation reroute to the plant stack, with the addition of two new 25,000 cfm fans to the system. It was assumed that the normal flow profile would have one of the two new fans in service, adding 25,000 cfm to the previous nominal plant stack flow of 150,000 cfm. This increases the effective height of the plant stack, and therefore, provides additional credit for air dispersion of effluents. The second modification in the meteorological model is the inclusion of site specific recirculation correction factors. These factors provide adjustments to the straight-line steady-state dispersion models, and accounts for the potential return of effluents released from the plant to move back over the site area based on historical wind shift patterns. This adds additional conservatism to the existing dispersion estimates for both the plant stack and ground level (North Warehouse) release points. The recalculation of the dispersion estimates also utilized the most recent available five year site meteorological data, 1988 through 1992.

The effect of the revised dispersion modeling causes all the gaseous pathway dose conversion factors to be changed since they have factored into them X/Q values from the dispersion modeling. This is reflected in Sections 1, 3, 5, and example calculations in Appendix A. The same methodology for deriving dose factors that have been used and approved in the past have been reapplied with the addition of the new X/Q's.

A second issue addressed in the amendment is in response to an internal audit finding that indicated that a footnote should be added to Table 4.1 for TLD monitoring location DR-8, explaining how the measured values are averaged with other TLD data. This amendment provides an explanation of how the requirements in Technical Specification Table 3.9.3 for an inner ring direct radiation monitoring are satisfied with DR-8, but that its measured value is averaged as a site boundary TLD due to its

close proximity to the plant. No change in the existing program is made by this additional editorial clarification.

Section 6, including Figures 6.1 and 6.2, have also been amended to clarify that the system description and simplified one line drawing of effluent release pathways only address pathways that are, or likely to contain, radioactive materials discharged from the site. This section is not intended or required to reflect all types of site release, non-radioactive pathways. No change in the technical description or use of the radioactive system is affected by this editorial clarification.

The revised ODCM pages for the above revisions (Amendment No. 17) are attached.

VERMONT YANKEE NUCLEAR POWER STATION

OFF-SITE DOSE CALCULATION MANUAL

REVISION 17

Reviewed: R. McCullough 94-42 — 5/25/94
Plant Operations Review Committee — Date

Approved: B Wanczyk — 5/26/94
Plant Manager — Date

Approved: [signature] — 5/27/94
Vice President, Operations — Date

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LIST OF AFFECTED PAGES

Page	Revision	Date
i	17	05/31/94
ii	13	05/01/92
iii	17	05/31/94
iv	0	03/01/84
v-x	17	05/31/94
1.1-1.2	15	07/08/93
1.3-1.4	17	05/31/94
1.5-1.6	15	07/08/93
1.7-1.9	17	05/31/94
1.10	15	07/08/93
1.11	17	05/31/94
1.12	16	10/28/93
1.13-1.18	15	07/08/93
1.19-1.20	17	05/31/94
1.21	15	07/08/93
1.22	17	05/31/94
2.1-2.4	15	07/08/93
3.1	15	07/08/93
3.2-3.7	17	05/31/94
3.8-3.12	15	07/08/93
3.13-3.17	17	05/31/94
3.18	16	10/28/93
3.19	15	07/08/93
3.20-3.50	17	05/31/94
3.51	15	07/08/93
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DISCLAIMER OF RESPONSIBILITY

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ABSTRACT

The VYNPS ODCM (Vermont Yankee Nuclear Power Station Off-Site Dose Calculation Manual) contains the approved methods to estimate the maximum individual doses and radionuclide concentrations occurring at or beyond the boundaries of the plant due to normal plant operation. The effluent dose models are based on the U.S. NRC Regulatory Guide 1.109, Revision 1.

With initial approval by the U.S. Nuclear Regulatory Commission and the VYNPS Plant Management and approval of subsequent revisions by the Plant Management (as per the Technical Specifications) this ODCM is suitable to show compliance where referred to be the Plant Technical Specifications.

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1.0 INTRODUCTION

This ODCM (Off-Site Dose Calculation Manual) provides formal and approved methods for the calculation of off-site concentration, off-site doses, and effluent monitor setpoints in order to comply with the Vermont Yankee Technical Specifications 3.8/4.8 and 3.9/4.9, hereafter referred to as the Radiological Effluent Technical Specifications. The ODCM forms the basis for plant procedures and is designed for use by the procedure writer. In addition, the ODCM will be useful to the writer of periodic reports required by the NRC on the dose consequences of plant operation. The dose methods contained herein follow accepted NRC guidance for calculation of doses necessary to demonstrate compliance with the dose objectives of Appendix I to 10CFR50 (Regulatory Guide 1.109) unless otherwise noted in the text.

Demonstration of compliance with the dose limits of 40CFR190 (see Technical Specification 3.8.M) will be considered as demonstrating compliance with the 0.1 rem limit of 10CFR20.1301(a)(1) for members of the public in unrestricted areas (Reference 56 FR 23374, third column).

It shall be the responsibility of the Chemistry Manager and Radiation Protection Manager to ensure that the ODCM is used in the performance of the surveillance requirements of the appropriate portions of Technical Specifications. The administration of the program for the disposal of slightly contaminated septic waste, as described in Appendix B, is the responsibility of the Senior Environmental Program Manager.

All changes to the ODCM must be reviewed by PORC and approved by MOO, in accordance with Technical Specification 6.13, prior to implementation. All approved changes shall be submitted to the NRC for their information in the Semiannual Radioactive Effluent Report for the period in which the change(s) was made effective. The plant's Document Control Center (DCC) shall maintain the current version of the ODCM and issue under controlled distribution all approved changes to it.

1.1 Summary of Methods, Dose Factors, Limits, Constants, Variables and Definitions

This section summarizes the methods for the user. The first-time user should read Chapters 2 through 6. The concentration and setpoint methods are documented in Table 1.1-2 through Table 1.1-7, as well as the Method I Dose equations. Where more accurate dose calculations are needed use the Method II for the appropriate dose as described in Sections 3.2 through 3.9 and 3.11. The dose factors used in the equations are in Tables 1.1-10 through 1.1-12 and the Regulatory Limits are summarized in Table 1.1-1.

The variables and special definitions used in this ODCM are in Tables 1.1-8 and 1.1-9.

TABLE 1.1-1

Summary of Radiological Effluent Technical Specifications and Implementing Equations

	Technical Specification	Category	Method[1]	Limit
3.8.A.1	Liquid Effluent Concentration	Sum of the Fractions of Effluent Concentration Limits [Excluding Noble Gases]	Eq. 2-1	≤ 1.0
		Total Noble Gas Concentration	Eq. 2-2	≤ 2×10^{-4} µCi/cc
3.8.B.1	Liquid Effluent Dose	Total Body Dose	Eq. 3-1	≤ 1.5 mrem in a qtr.
				≤ 3.0 mrem in a yr.
		Organ Dose	Eq. 3-3	≤ 5 mrem in a qtr.
				≤ 10 mrem in a yr.
3.8.C.1	Liquid Radwaste Treatment Operability	Total Body Dose	Eq. 3-1	≤0.06 mrem in a mo.
		Organ Dose	Eq. 3-3	≤ 0.2 mrem in a mo.
3.8.E.1	Gaseous Effluents Dose Rate	Total Body Dose Rate from Noble Gases	Eq. 3-5 Eq. 3-39	≤ 500 mrem/yr.
		Skin Dose Rate from Noble Gases	Eq. 3-7 Eq. 3-38	≤ 3000 mrem/yr.
3.8.E.1	(Continued)	Organ Dose Rate from Iodines, Tritium and Particulates with $T_{1/2}$ > 8 Days	Eq. 3-16 Eq. 3-40	≤ 1500 mrem/yr.

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TABLE 1.1-1
(Continued)

Summary of Radiological Effluent Technical Specifications and Implementing Equations

	Technical Specification	Category	Method[1]	Limit
3.8.F.1	Gaseous Effluents Dose from Noble Gases	Gamma Air Dose from Noble Gases	Eq. 3-21 Eq. 3-41	≤ 5 mrad in a qtr. ≤ 10 mrad in a yr.
		Beta Air Dose from Noble Gases	Eq. 3-23 Eq. 3-43	≤ 10 mrad in a qtr. ≤ 20 mrad in a yr.
3.8.G.1	Gaseous Effluents Dose from Iodines, Tritium, and Particulates	Organ Dose from I-131, I-133, Tritium, and Particulates with $T_{1/2}$ > 8 Days	Eq. 3-25 Eq. 3-44	≤ 7.5 mrem in a qtr. ≤ 15 mrem in a yr.
3.8.I.1	Ventilation Exhaust Treatment	Organ Dose	Eq. 3-25	≤ 0.3 mrem in a mo.
3.8.M.1	Total Dose (from All Sources)	Total Body Dose	Footnote[2]	≤ 25 mrem in a yr.
		Organ Dose		≤ 25 mrem in a yr.
		Thyroid Dose		≤ 75 mrem in a yr.

TABLE 1.1-1
(Continued)

Summary of Radiological Effluent Technical Specifications and Implementing Equations

	Technical Specification	Category	Method[1]	Limit
3.9.A.1	Liquid Effluent Monitor Setpoint			
	Liquid Radwaste Discharge Monitor	Alarm Setpoint	Eq. 5-1	T.S. 3.8.A.1
3.9.B.1	Gaseous Effluent Monitor Setpoint			
	Plant Stack and AOG Offgas System Noble Gas Activity Monitors	Alarm/Trip Setpoint for Total Body Dose Rate	Eq. 5-9	T.S. 3.8.E.1a (Total Body)
		Alarm/Trip Setpoint for Skin Dose Rate	Eq. 5-10	T.S. 3.8.E.1a (Skin)
	SJAE Noble Gas Activity Monitors	Alarm Setpoint	Eq. 5-21	T.S. 3.8.K.1

(1) More accurate methods may be available (see subsequent chapters).

(2) Technical Specification 3.8.M.2 requires this evaluation only if twice the limit of Equations 3-1, 3-3, 3-21, 3-23, or 3-25 is reached. If this occurs a Method II calculation, using actual meteorology and identified pathways for a real individual, shall be made.

TABLE 1.1-2

Summary of Methods to Calculate
Unrestricted Area Liquid Concentrations

Equation Number	Category	Equation	Reference Section
2-1	Sum of the Fractions of Combined Effluent Concentrations in Liquids [Except Noble Gases]	$F_1^{ENG} = \sum_i \frac{C_i}{ECL_i} \leq 1$	2.1
2-2	Total Activity of Dissolved and Entrained Noble Gases from all Station Sources	$C_1^{NG} = \left(\frac{\mu Ci}{ml}\right) = \sum_i C_i^{NG}$ $\leq 2E\text{-}04$	2.1

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TABLE 1.1-3

Summary of Methods to Calculate
Off-Site Doses from Liquid Concentrations

Equation Number	Category	Equation	Reference Section
3-1	Total Body Dose	$D_{tb}\ (mrem) = \sum_i Q_i\ DFL_{itb}$	3.2.1
3-3	Maximum Organ Dose	$D_{mo}\ (mrem) = \sum_i Q_i\ DFL_{imo}$	3.3.1

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TABLE 1.1-4

Summary of Methods to Calculate Dose Rates

Equation Number	Category	Equation	Reference Section
3-5	Total Body Dose Rate from Noble Gases Released from Stack	$\dot{R}_{tbs}\left(\frac{mrem}{yr}\right)=0.61\sum_i \dot{Q}_i^{ST}\,DFB_i$	3.4.1
3-39	Total Body Dose Rate from Noble Gases Released from Ground	$\dot{R}_{tbg}\left(\frac{mrem}{yr}\right)=6.4\sum_i \dot{Q}_i^{GL}\,DFB_i$	3.4.1
3-7	Skin Dose Rate from Noble Gases Released from Stack	$\dot{R}_{skins}\left(\frac{mrem}{yr}\right)=\sum_i \dot{Q}_i^{ST}\,DF'_{is}$	3.5.1
3-38	Skin Dose Rate from Noble Gases Released from Ground	$\dot{R}_{sking}\left(\frac{mrem}{yr}\right)=\sum_i \dot{Q}_i^{GL}\,DF'_{ig}$	3.5.1
3-16	Critical Organ Dose Rate from Stack Release of I-131, I-133, Tritium, and Particulates with $T_{1/2}$ >8 Days	$\dot{R}_{cos}\left(\frac{mrem}{yr}\right)=\sum_i \dot{Q}_i^{STP}\,DFG'_{sico}$	3.6.1
3-40	Critical Organ Dose Rate from Ground Level Release of I-131, I-133, Tritium, and Particulates with $T_{1/2}$ >8 Days	$\dot{R}_{cog}\left(\frac{mrem}{yr}\right)=\sum_i \dot{Q}_i^{GLP}\,DFG'_{gico}$	3.6.1

TABLE 1.1-5

Summary of Methods to Calculate
Doses to Air from Noble Gases

Equation Number	Category	Equation	Reference Section	
3-21	Gamma Dose to Air from Noble Gases Released from Stack	$D^{\gamma}_{airs}\ (mrad) = 0.019 \sum_i Q_i^{ST}\ DF_i^{\gamma}$	3.7.1	I
3-41	Gamma Dose to Air from Noble Gases Released from Ground Level	$D^{\gamma}_{airg}\ (mrad) = 0.20 \sum_i Q_i^{GL}\ DF_i^{\gamma}$	3.7.1	I
3-23	Beta Dose to Air from Noble Gases Released from Stack	$D^{\beta}_{airs}\ (mrad) = 0.033 \sum_i Q_i^{ST}\ DF_i^{\beta}$	3.8.1	I
3-43	Beta Dose to Air from Noble Gases Released from Ground Level	$D^{\beta}_{airg}\ (mrad) = 1.12 \sum_i Q_i^{GL}\ DF_i^{\beta}$	3.8.1	I

TABLE 1.1-6

Summary of Methods to Calculate
Dose to an Individual from Tritium, Iodine, and Particulates in
Gas Releases and Direct Radiation

Equation Number	Category	Equation	Reference Section
3-25	Dose to Critical Organ from Stack Release of I-131, I-133, Tritium, and Particulates	$D_{cos}\ (mrem) = \sum_i Q_i^{STP}\ DFG_{sico}$	3.9.1
3-44	Dose to Critical Organ from Ground Level Release of I-131, I-133, Tritium, and Particulates	$D_{cog}(mrem) = \sum_i Q_i^{GLP}\ DFG_{gico}$	3.9.1
	Direct Dose		
3-27	Turbine Building	$D_d\ (mrem) = K_{N16}(L) \bullet E$	3.11.1
	North Warehouse		
3-29	Shielded End	$D_S = 0.25 \times \dot{R}_S$	3.11.2
3-30	Unshielded End	$D_U = 0.53 \times \dot{R}_U$	3.11.2
	LLW Storage Pad		
3-31	Direct Line (Module Short Side Out)	$D_{de} = 0.28 \times \dot{R}_d \times f_d$	3.11.3
3-32	Direct Line (Module Long Side Out)	$D_{dS} = 0.39 \times \dot{R}_d \times f_d$	3.11.3
3-33	Skyshine (Resin Liners)	$D_{SKR} = 0.016 \times \dot{R}_{SKR} \times f_{SK}$	3.11.3
3-34	Skyshine (DAW)	$D_{SKD} = 0.015 \times \dot{R}_{SKD} \times f_{SK}$	3.11.3
3-35	Resin Liner Transfer (Unshielded)	$D_{Tran} = 0.0025 \times \dot{R}_{Tran} \times T_{Tran}$	3.11.3
3-36	Intermodular Gap Dose	$D_{Gap} = 2.44E\text{-}2 \times W_{Gap} \times A_{RL} \times f_{Gap}$	3.11.3

TABLE 1.1-7

Summary of Methods for
Setpoint Determinations

Equation Number	Category	Equation	Reference Section
5-1	Liquid Effluents:		
	Liquid Radwaste Discharge Monitor (17/350)	R^{L}_{spt} (cps) = $\frac{DF}{DF_{min}} S_1 \sum_i C_{mi}$	5.1.1.1
	Gaseous Effluents:		
	Plant Stack (RR-108-1A, RR-108-1B) and AOG Offgas System (3127, 3128) Noble Gas Activity Monitors		
5-9	Total Body	R^{tb}_{spt} (cpm) = 818 $S_g \frac{1}{F} \frac{1}{DFB_c}$	5.2.1.1
5-10	Skin	R^{skin}_{spt} (cpm) = 3000 $S_g \frac{1}{F} \frac{1}{DF'_c}$	5.2.1.1
5-21	SJAE Noble Gas Activity Monitors (17/150A, 17/150B)	R^{SJAE}_{spt} (mR/hr) = 1.6E+05 $S_g \frac{1}{F}$	5.2.2.1

TABLE 1.1-8

Summary of Variables

Variable		Definition	Units
A_{RL}	=	Total gamma activity contained in a resin liner in storage directly in line with a gap between adjacent storage modules.	Ci
C_{1i}^{NG}	=	Concentration at point of discharge to an unrestricted area of dissolved and entrained noble gas "i" in liquid pathways from all station sources.	µCi/ml
C_1^{NG}	=	Total activity of all dissolved and entrained noble gases in liquid pathways from all station sources.	$\frac{\mu Ci}{ml}$
C_{di}	=	Concentration of radionuclide "i" at the point of liquid discharge to an unrestricted area.	$\frac{\mu Ci}{ml}$
C_i	=	Concentration of radionuclide "i".	$\frac{\mu Ci}{cc}$
C_{pi}	=	Concentration, exclusive of noble gases, of radionuclide "i" from tank "p" at point of discharge to an unrestricted area.	$\frac{\mu Ci}{ml}$
C_{mi}	=	Concentration of radionuclide "i" in mixture at the monitor.	$\frac{\mu Ci}{ml}$
D_{airs}^{β}	=	Beta dose to air from stack release.	mrad
D_{airg}^{β}	=	Beta dose to air from ground level release.	mrad
D_{airs}^{γ}	=	Gamma dose to air from stack release.	mrad
D_{airg}^{γ}	=	Gamma dose to air from ground level release.	mrad
D_{cos}	=	Dose to critical organ from stack release.	mrem
D_{cog}	=	Dose to the critical organ from ground level release.	mrem
D_d	=	Direct dose (Turbine Building).	mrem

TABLE 1.1-8
(Continued)

Summary of Variables

Variable		Definition	Units
$\dot{R}_d$	=	Dose rate at 3′ from unobstructed side of storage module facing site boundary.	$\frac{mrem}{hr}$
D_{dE}	=	Direct dose at site boundary per unobstructed storage module (short end).	$\frac{mrem}{yr\text{-}module}$
D_{dS}	=	Direct dose at site boundary per unobstructed storage module (long side).	$\frac{mrem}{yr\text{-}module}$
D^{γ}_{finite}	=	Gamma dose to air, corrected for finite cloud.	mrad
D_{Gap}	=	Intermodular gap dose projected to the maximum site boundary location from resin waste not directly shielded by DAW modules.	$\frac{mrem}{yr}$
D_{mo}	=	Dose the maximum organ.	mrem
D^{S}	=	Dose to skin from beta and gamma.	mrem
$\dot{R}_S$	=	Dose rate at 1 meter from source in shielded end of North Warehouse.	$\frac{mrem}{hr}$
D_S	=	Annual dose at site boundary from fixed sources in shielded end of North Warehouse.	$\frac{mrem}{yr}$
$\dot{R}_{SKD}$	=	Maximum dose rate at 3′ over top of DAW in a storage module.	$\frac{mrem}{hr}$
$\dot{R}_{SKR}$	=	Maximum dose rate at 3′ over top of each resin liner in a storage module.	$\frac{mrem}{hr}$
D_{SKD}	=	Skyshine dose at the site boundary from DAW in storage modules (unobstructed top surfaces).	$\frac{mrem}{yr\text{-}module}$
D_{SKR}	=	Skyshine dose at the site boundary from resin liners in storage modules (unobstructed top surfaces).	$\frac{mrem}{yr\text{-}liner}$

TABLE 1.1-8
(Continued)

Summary of Variables

Variable		Definition	Units
D_{tb}	=	Dose to the total body.	mrem
$K_{N/6}^{(L)}$	=	The direct dose conversion factor for N-16 scatter from the turbine hall to Location (L)	$\frac{mrem}{MW_e h}$
$\dot{R}_{Tran}$	=	Dose rate at contact from the unshielded top surface of resin liner.	$\frac{rad}{hr}$
D_{Tran}	=	Dose at the site boundary from unshielded movement of resin liner between transfer cask and storage module.	mrem
$\dot{R}_U$	=	Dose rate at 1 meter from source in unshielded end of North Warehouse.	$\frac{mrem}{hr}$
D_U	=	The annual dose at the site boundary from fixed sources in the unshielded end of North Warehouse.	$\frac{mrem}{yr}$
DF	=	Dilution factor.	ratio
DF_{min}	=	Minimum allowable dilution factor.	ratio
DF_c'	=	Composite skin dose factor.	$\frac{mrem\text{-}sec}{pCi\text{-}yr}$
DFB_i	=	Total body gamma dose factor for nuclide "i".	$\frac{mrem\text{-}m^3}{pCi\text{-}yr}$
DFB_c	=	Composite total body dose factor.	$\frac{mrem\text{-}m^3}{pCi\text{-}yr}$
DFL_{itb}	=	Site-specific, total body dose factor for a liquid release of nuclide "i".	$\frac{mrem}{Ci}$
DFL_{imo}	=	Site-specific, maximum organ dose factor for a liquid release of nuclide "i".	$\frac{mrem}{Ci}$

TABLE 1.1-8
(Continued)

Summary of Variables

Variable		Definition	Units
DFG_{sico}	=	Site-specific, critical organ dose factor for a stack gaseous release of nuclide "i".	$\frac{mrem}{Ci}$
DFG'_{sico}	=	Site-specific, critical organ dose rate factor for a stack gaseous release of nuclide "i".	$\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}$
DFG_{gico}	=	Site-specific, critical organ dose factor for a ground level gaseous release of nuclide "i".	$\frac{mrem}{Ci}$
DFG'_{gico}	=	Site-specific, critical organ dose rate factor for a ground level gaseous release of nuclide "i".	$\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}$
DFS_i	=	Beta skin dose factor for nuclide "i".	$\frac{mrem\text{-}m^3}{pCi\text{-}yr}$
DF'_{is}	=	Combined skin dose factor for nuclide "i" from a stack release.	$\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}$
DF'_{ig}	=	Combined skin dose factor for nuclide "i" from a ground level release.	$\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}$
DF_i^{γ}	=	Gamma air dose factor for nuclide "i".	$\frac{mrad\text{-}m^3}{pCi\text{-}yr}$
DF_i	=	Beta air dose factor for nuclide "i".	$\frac{mrad\text{-}m^3}{pCi\text{-}yr}$
$\dot{R}_{cos}$	=	Critical organ dose rate due to iodines and particulates released from stack.	$\frac{mrem}{yr}$
$\dot{R}_{cog}$	=	Critical organ dose rate due to iodines and particulates released from ground.	$\frac{mrem}{yr}$
$\dot{R}_{skins}$	=	Skin dose rate due to stack release of noble gases.	$\frac{mrem}{yr}$
$\dot{R}_{sking}$	=	Skin dose rate due to ground release of noble gases.	$\frac{mrem}{yr}$

TABLE 1.1-8
(Continued)

Summary of Variables

Variable		Definition	Units
$\dot{R}_{tbs}$	=	Total body dose rate due to noble gases from stack release.	$\frac{mrem}{yr}$
$\dot{R}_{tbg}$	=	Total body dose rate due to noble gases from ground level release.	$\frac{mrem}{yr}$
D/Q	=	Deposition factor for dry deposition of elemental radioiodines and other particulates.	$\frac{1}{m^2}$
E	=	Gross electric output over the period of interest.	MW_eh
f_d	=	Fraction of a year that a storage module is in use with an unobstructed side oriented toward west site boundary.	fraction
f_{Gap}	=	Fraction of a year that the intermodular gap is not shielded.	fraction
f_{SK}	=	Fraction of a year that a storage module is in use with an unobstructed top surface.	fraction
F_d	=	Flow rate out of discharge canal.	gpm
F_m	=	Flow rate past liquid radwaste monitor.	gpm
F	=	Flow rate past gaseous radwaste monitor.	$\frac{cc}{sec}$
F_1^{ENG}	=	Sum of the fractions of combined effluent concentrations in liquid pathways (excluding noble gases).	fraction
ECL_i	=	Annual average effluent concentration limit for radionuclide "i" (10CFR20.1001-20.2401, Appendix B, Table 2, Column 2)	$\frac{\mu Ci}{cc}$
Q_i	=	Release for radionuclide "i" from the point of interest.	curies

TABLE 1.1-8
(Continued)

Summary of Variables

Variable		Definition	Units
$\dot{Q}_i$	=	Release rate for radionuclide "i" at the point of interest.	$\frac{\mu Ci}{sec}$
$\dot{Q}_i^{ST}$	=	The noble gas radionuclide "i" release rate at the plant stack.	$\frac{\mu Ci}{sec}$
$\dot{Q}_i^{GL}$	=	The noble gas radionuclide "i" release rate from ground level.	$\frac{\mu Ci}{sec}$
$\dot{Q}_i^{SJAE}$	=	The noble gas radionuclide "i" release rate at the steam jet air ejector.	$\frac{\mu Ci}{sec}$
$\dot{Q}_i^{AOG}$	=	The noble gas radionuclide "i" release rate at the exhaust of the augmented Off-Gas System	$\frac{\mu Ci}{sec}$
$\dot{Q}_i^{STP}$	=	The iodine, tritium, and particulate radionuclide "i" release rate from the plant stack.	$\frac{\mu Ci}{sec}$
$\dot{Q}_i^{GLP}$	=	The iodine, tritium, and particulate radionuclide "i" release rate from ground level.	$\frac{\mu Ci}{sec}$
Q_i^{ST}	=	The release of noble gas radionuclide "i" from the plant stack.	curies
Q_i^{GL}	=	The release of noble gas radionuclide "i" from ground level.	curies
Q_i^{STP}	=	The release of iodine, tritium, and particulate radionuclide "i" from the plant stack.	curies
Q_i^{GLP}	=	The release of iodine, tritium, and particulate radionuclide "i" from ground level.	curies
R_{spt}^{L}	=	Liquid monitor response for the limiting concentration at the point of discharge.	cps

TABLE 1.1-8
(Continued)

Summary of Variables

Variable		Definition	Units
R_{spt}^{skin}	=	Response of the noble gas monitor at the limiting skin dose rate.	cpm
R_{spt}^{tb}	=	Response of the noble gas monitor to limiting t[illegible]body dose rate.	cpm
S_F	=	Sh[illegible]g factor.	Ratio
S_g	=	Detector counting efficiency from the most recent gas monitor calibration.	$\frac{cpm}{\mu Ci/cc}$ or $\frac{mR/hr}{\mu Ci/cc}$
S_{gi}	=	Detector count[illegible] efficiency for noble gas "i".	$\frac{cpm}{\mu Ci/cc}$ or $\frac{mR/hr}{\mu Ci/cc}$
S_l	=	Detector counting efficiency from the most recent liquid monitor calibration.	$\frac{cps}{\mu Ci/ml}$
S_{li}	=	Detector counting efficiency for radionuclide "i".	$\frac{cps}{\mu Ci/ml}$
T_{Tran}	=	Time that an [illegible]hielded resin liner is exposed in the storag[illegible]ad area.	hours
W_{Gap}	=	Intermodule gap width between adjacent DAW storage modules which shield resin liner storage modules from the west site boundary.	inches
X/Q_s	=	Annual or long-term average undepleted atmospheric dispersion factor for stack release.	$\frac{sec}{m^3}$
X/Q_g	=	Annual or long-term average undepleted atmospheric dispersion factor for ground level release.	$\frac{sec}{m^3}$
$[X/Q]_s^\gamma$	=	Effective annual or long-term average gamma atmospheric dispersion factor.	$\frac{sec}{m^3}$
$[X/Q]_g^\gamma$	=	Effective annual or long-term average gamma atmospheric dispersion factor for a ground level release.	$\frac{sec}{m^3}$

TABLE 1.1-10

Dose Factors Specific for Vermont Yankee
for Noble Gas Releases

Radionuclide	Gamma Total Body Dose Factor DFB_i $\left(\frac{mrem\text{-}m^3}{pCi\text{-}yr}\right)$	Beta Skin Dose Factor DFS_i $\left(\frac{mrem\text{-}m^3}{pCi\text{-}yr}\right)$	Combined Skin Dose Factor (Stack Release) DF'_{is} $\left(\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}\right)$	Beta Air Dose Factor DF_i^{β} $\left(\frac{mrad\text{-}m^3}{pCi\text{-}yr}\right)$	Gamma Air Dose Factor DF_i^{γ} $\left(\frac{mrad\text{-}m^3}{pCi\text{-}yr}\right)$
Ar-41	8.84E-03*	2.69E-03	9.12E-03	3.28E-03	9.30E-03
Kr-83m	7.56E-08	-----	1.31E-05	2.88E-04	1.93E-05
Kr-85m	1.17E-03	1.46E-03	2.35E-03	1.97E-03	1.23E-03
Kr-85	1.61E-05	1.34E-03	1.41E-03	1.95E-03	1.72E-05
Kr-87	5.92E-03	9.73E-03	1.43E-02	1.03E-02	6.17E-03
Kr-88	1.47E-02	2.37E-03	1.28E-02	2.93E-03	1.52E-02
Kr-89	1.66E-02	1.01E-02	2.23E-02	1.06E-02	1.73E-02
Kr-90	1.56E-02	7.29E-03	1.87E-02	7.83E-03	1.63E-02
Xe-131m	9.15E-05	4.76E-04	6.01E-04	1.11E-03	1.56E-04
Xe-133m	2.51E-04	9.94E-04	1.26E-03	1.48E-03	3.27E-04
Xe-133	2.94E-04	3.06E-04	5.58E-04	1.05E-03	3.53E-04
Xe-135m	3.12E-03	7.11E-04	3.02E-03	7.39E-04	3.36E-03
Xe-135	1.81E-03	1.86E-03	3.24E-03	2.46E-03	1.92E-03
Xe-137	1.42E-03	1.22E-02	1.37E-02	1.27E-02	1.51E-03
Xe-138	8.83E-03	4.13E-03	1.06E-02	4.75E-03	9.21E-03

*8.84E-03 = 8.84×10^{-3}

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TABLE 1.1-10A

Combined Skin Dose Factors Specific for Vermont Yankee Ground Level Noble Gas Releases

Radionuclide	DF'_{ig} $\left(\frac{\text{mrem-sec}}{\mu\text{Ci-yr}}\right)$
AR-41	1.61E-01
KR-83M	1.38E-04
KR-85M	6.02E-02
KR-85	4.73E-02
KR-87	3.86E-01
KR-88	1.92E-01
KR-89	4.79E-01
KR-90	3.73E-01
XE-131M	1.79E-02
XE-133M	3.72E-02
XE-133	1.33E-02
XE-135M	4.90E-02
XE-135	7.92E-02
XE-137	4.40E-01
XE-138	2.11E-01

TABLE 1.1-11

Dose Factors Specific for Vermont Yankee
for Liquid Releases

Radionuclide	Total Body Dose Factor $DFL_{itb}\left(\frac{mrem}{Ci}\right)$	Maximum Organ Dose Factor $DFL_{imo}\left(\frac{mrem}{Ci}\right)$
H-3	2.06E-04	2.06E-04
Na-24	3.38E-02	3.38E-02
Cr-51	3.10E-04	6.96E-02
Mn-54	2.08E-01	3.00E+00
Mn-56	8.53E-06	5.29E-03
Fe-55	4.18E-02	2.54E-01
Fe-59	2.49E-01	1.84E+00
Co-58	5.97E-02	4.34E-01
Co-60	2.13E-01	1.28E+00
Zn-65	8.06E+00	1.64E+01
Sr-89	2.55E-01	8.91E+00
Sr-90	4.23E+01	1.67E+02
Zr-95	4.21E-04	1.36E-01
Mo-99	4.79E-03	4.51E-02
Tc-99m	5.04E-06	2.33E-04
Ag-110m	6.90E-03	7.02E-01
Sb-124	8.44E-03	2.22E-01
Sb-125	7.52E-03	1.15E-01
I-131	2.57E-02	1.47E+01
I-132	3.10E-06	1.29E-04
I-133	3.31E-03	1.63E+00
I-135	3.16E-04	5.90E-02
Cs-134	1.28E+02	1.60E+02
Cs-137	7.58E+01	1.21E+02
Ba-140	4.08E-03	9.72E-02
Ce-141	2.31E-05	4.10E-02
W-187	1.18E-02	8.90E+00

TABLE 1.1-12

Dose and Dose Rate Factors Specific for Vermont Yankee for Iodines, Tritium, and Particulate Releases

Radionuclide	Stack Release: Critical Organ Dose Factor DFG_{sico} $\left(\frac{mrem}{Ci}\right)$	Stack Release: Critical Organ Dose Rate Factor DFG'_{sico} $\left(\frac{mrem\text{-}sec}{yr\text{-}\mu Ci}\right)$	Ground Level Release*: Critical Organ Dose Factor DFG_{gico} $\left(\frac{mrem}{Ci}\right)$	Ground Level Release*: Critical Organ Dose Rate Factor DFG'_{gico} $\left(\frac{mrem\text{-}sec}{yr\text{-}\mu Ci}\right)$
H-3	3.13E-04	9.87E-03	1.06E-02	3.34E-01
C-14	1.90E-01	5.99E+00	6.43E+00	2.03E+02
Cr-51	6.11E-03	2.11E-01	4.16E-02	1.43E+00
Mn-54	7.01E-01	2.77E+01	4.71E+00	1.84E+02
Fe-55	3.17E-01	1.00E+01	2.05E+00	6.47E+01
Fe-59	6.99E-01	2.32E+01	4.60E+00	1.52E+02
Co-58	3.62E-01	1.30E+01	2.39E+00	8.52E+01
Co-60	7.63E+00	3.41E+02	4.99E+01	2.16E+03
Zn-65	3.71E+00	1.20E+02	2.36E+01	7.63E+02
Sr-89	1.14E+01	3.60E+02	7.27E+01	2.29E+03
Sr-90	4.31E+02	1.36E+04	2.82E+03	8.89E+04
Zr-95	6.91E-01	2.28E+01	4.51E+00	1.49E+02
Sb-124	1.26E+00	4.23E+01	8.35E+00	2.79E+02
Sb-125	1.25E+00	4.89E+01	8.01E+00	3.13E+02
I-131	7.71E+01	2.43E+03	5.02E+02	1.58E+04
I-133	8.21E-01	2.59E+01	8.29E+00	2.61E+02
Cs-134	1.58E+01	5.27E+02	1.02E+02	3.37E+03
Cs-137	1.63E+01	5.55E+02	1.04E+02	3.53E+03
Ba-140	1.13E-01	3.66E+00	2.18E+00	6.94E+01
Ce-141	1.70E-01	5.42E+00	1.19E+00	3.78E+01
Ce-144	3.85E+00	1.22E+02	2.52E+01	7.98E+02

* The release point reference is the North Warehouse. These dose and dose rate factors are conservative for potential release applications associated with ground level effluents from other major facilities (i.e., Turbine Building, Reactor Building, AOG, and CAB).

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3.0 OFF-SITE DOSE CALCULATION METHODS

Chapter 3 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 Technical Specifications (hereafter called RETS). A simple, conservative method (called Method I) is listed in Tables 1.1-2 to 1.1-7 for each of the requirements of the RETS. Each of the Method I equations is presented, along with their bases in Sections 3.2 through 3.9 and Section 3.11. Appendix A provides example calculations for all Method I dose equations as guidance to their use. In addition, reference is provided to more sophisticated 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 5.

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).

3.1 Introductory Concepts

The Radiological Effluent Technical Specifications (RETS) 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 Technical Specification 3.6.E.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 RETS.

The dose D is the quantity calculated by the Chapter 3 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. 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.

$\dot{R}$ is the quantity calculated in the Chapter 3 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 3, and are summarized in Tables 1.1-1 to 1.1-7. Each method has two levels of complexity and conservative margin and are called Method I and Method II. Method I has the greatest margin and 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. Guidance is provided but the appropriate margin and depth of analysis are determined in each instance at the time of analysis under Method II.

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 Technical Specifications.

3.2 Method to Calculate the Total Body Dose from Liquid Releases

Technical Specification 3.8.B.1 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. Technical Specification 3.8.C.1 requires liquid radwaste treatment when the total body dose estimate exceeds 0.06 mrem in any month. Technical Specification 3.8.M.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 (Specification 3.8.C.1). To evaluate the total body dose, use Equation 3.1 to estimate the dose from the planned release and add this to the total body dose accumulated from prior releases during the month.

3.2.1 Method I

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

$$D_{tb} = \sum_i Q_i \, DFL_{itb} \qquad (3\text{-}1)$$

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

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 for radionuclide "i". (For strontiums and Fe-55, use the most recent measurement available.)

Equation 3-1 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.

3.2.2 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 Technical Specifications which limit off-site total body dose from liquids (3.8.B.1 and 3.8.C.1) have been met for releases over the appropriate periods. Technical Specification 3.8.B.1 is based on the ALARA design objectives in 10CFR50, Appendix I Subsection II A. Technical Specification 3.8.C.1 is an "appropriate fraction", determined by the NRC, of that design objective (hereafter called the Objective). Technical Specification 3.8.M.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 0.0346 which corresponds to a minimum regulated river flow of 1250 cfs at the Vernon Dam just below the plant discharge outfall.* Tables 3.2-1 and 3.2-2 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:

$$\Delta D_{tb} = Q_i \, DFL_{itb} \qquad (3\text{-}2)$$

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

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".

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* 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.

3.2.3 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. Analyses requiring Method II calculations should be referred to YNSD to be performed and documented.

TABLE 3.2-1

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	0.0	0.0	0.0	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	(L/D)				1440.0	24.0	2160.0	2160.0
QAW	Water Uptake Rate for Animal	(KG/D)						50.0	60.0
QF	Feed Uptake Rate for Animal							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						

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TABLE 3.2-2

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

3.3 Method to Calculate Maximum Organ Dose from Liquid Releases

Technical Specification 3.8.B.1 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. Technical Specification 3.8.C.1 requires liquid radwaste treatment when the maximum organ dose estimate exceeds 0.2 mrem in any month. Technical Specification 3.8.M.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 (Specification 3.8.C.1). To evaluate the maximum organ dose, use Equation 3-3 to estimate the dose from the planned release and add this to the maximum organ dose accumulated from prior releases during the month.

3.3.1 Method I

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

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

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

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 3-3 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.

3.3.2 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 3.2.2). 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:

$$\Delta D_{mo} = Q_i \, DFL_{imo} \qquad (3\text{-}4)$$

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

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.

3.3.3 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. Analyses requiring Method II calculations should be referred to YNSD to be performed and documented.

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

Technical Specification 3.8.E.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 Technical Specification 3.8.E.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 Technical Specification limit (i.e., use of actual meteorology over the period of interest) to determine if, in fact, Technical Specification 3.8.E.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 Technical Specification 3.8.E.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 Technical Specifications (3.8.E.1) are performed when the effluent monitor alarm setpoint is exceeded and the corrective action required by Specification 3.8.E.2 is unsuccessful, or as required by the notations to Technical Specification Table 3.9.2 when the stack noble gas monitor is inoperable.

3.4.1 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 3-5 (an example calculation is provided in Appendix A):

$$\dot{R}_{tbs} = 0.61 \sum_i \dot{Q}_i^{ST} \; DFB_i \qquad (3\text{-}5)$$

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

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 the measured radionuclide distribution in the off-gas at the Steam Jet Air Ejector (SJAE) during plant operation when the activity at the stack is below detectable levels, and the recorded total gas effluent count rate from the Stack Gas Monitor I or II. The release rate at the stack can also be stated as follows:

$$\dot{Q}_i^{ST} = \frac{\dot{Q}_i^{SJAE}}{\sum_i \dot{Q}_i^{SJAE}} \; M \; \frac{1}{Sg} \; F \qquad (3\text{-}28)$$

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

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:

$$\dot{R}_{tbg} = 6.4 \sum_i \dot{Q}_i^{GL} \; DFB_i \qquad (3\text{-}39)$$

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

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 $\bar{Q}_i^{SJAE}$ to the sum of all $\bar{Q}_i^{SJAE}$ in Equation 3-28 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 3-5. Alternately, a relative radionuclide "i" mix fraction (f_i) may be taken from Table 5.2-1 as a function of time after shutdown, and substituted in place of the ratio of $\bar{Q}_i^{SJAE}$ to the sum of all $\bar{Q}_i^{SJAE}$ in Equation (3-28) above to determine the relative fraction of each noble gas potentially available for release to the total (example calculations can be found in Appendix A). 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 3-5 and 3-39 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.

3.4.2 Basis for Method I

Method I may be used to show that the Technical Specification which limits total body dose rate from noble gases released to the atmosphere (Technical Specification 3.8.E.1) has been met for the peak noble gas release rate.

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

$$\dot{R}_{tbs} = 1E+06\ S_F\ [X/Q]_S^Y \sum_i \dot{Q}_i^{ST}\ DFB_i \qquad (3\text{-}6)$$

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

where:

S_F = Shielding factor = 1.0 for dose rate determination.

$[X/Q]_S^Y$ = 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" (μCi/sec).

DFB_i = Gamma total body dose factor, $\left(\frac{\text{mrem-m}^3}{\text{pCi-yr}}\right)$. See Table 1.1-10.

Equation 3-6 reduces to:

$$\dot{R}_{tbs} = 0.61 \sum_i \dot{Q}_i^{ST} \; DFB_i \qquad (3\text{-}5)$$

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

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

$$\dot{R}_{tbg} = 1E{+}06 * 6.42E{-}06 \sum_i \dot{Q}_i^{GL} \; DFB_i \qquad (3\text{-}39)$$

or

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

The selection of critical receptor, outlined in Section 3.10, 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 $\dot{R}_{tbs}$ and $\dot{R}_{tbg}$ 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.

3.4.3 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. Analyses requiring Method II calculations should be referred to YNSD to be performed and documented.

3.5 Method to Calculate the Skin Dose Rate from Noble Gases

Technical Specification 3.8.E.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 Technical Specification 3.8.E.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 Technical Specification limits (i.e., use of actual meteorology over the period of interest) to determine if, in fact, Technical Specification 3.8.E.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 Technical Specification 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 Technical Specifications (3.8.E.1) are performed when the effluent monitor alarm setpoint is exceeded and the corrective action required by Specification 3.8.E.2 is unsuccessful, or as required by the notations to Technical Specification Table 3.9.2 when the stack noble gas monitor is inoperable.

3.5.1 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 3-7 (an example calculation is provided in Appendix A):

$$\dot{R}_{skins} = \sum_i \dot{Q}_i^{ST} \quad DF'_{is} \qquad (3\text{-}7)$$

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

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 the measured radionuclide distribution in the off-gas at the Steam Jet Air Ejector (SJAE) during plant operation when the activity at the stack is below detectable levels, and the recorded total gas effluent count rate from the Stack Gas Monitor I or II. The release rate at the stack can also be stated as follows:

$$\dot{Q}_i^{ST} = \frac{\dot{Q}_i^{SJAE}}{\sum_i \dot{Q}_i^{SJAE}} \quad M \quad \frac{1}{Sg} \quad F \qquad (3\text{-}28)$$

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

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 3-38:

$$\dot{R}_{sking} = \sum_i \dot{Q}_i^{GL} DF'_{ig} \qquad (3\text{-}38)$$

where:

$\dot{Q}_i^{GL}$ = 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 $\dot{R}_{skins} + \dot{R}_{sking}$.

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 3-28 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 3-7. Alternately, a relative radionuclide "i" mix fraction (f_i) may be taken from Table 5.2-1 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 3-28 above to determine the relative fraction of each noble gas potentially available for release to the total (example calculations can be found in Appendix A). 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 3-7 and 3-38 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.

3.5.2 Basis For Metnod I

The methods to calculate skin dose rate parallel the total body dose rate methods in Sectiсn 3.4.3. Only the differences are presented here.

Method I may be used to show that the Technical Specification which limits skin dose rate from noble gases released to the atmosphere (Technical Specification 3.8.E.1) has been met for the peak noble gas release rate.

Method I was derived from Regulatory Guide 1.109 as follows:

$$D^S = 1.11 \; S_F \; D^{\gamma}_{air} + 3.17E{+}04 \sum_i Q_i \; [X/Q]_S \; DFS_i \qquad (3\text{-}8)$$

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

where:

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

$$D^{\gamma}_{air} = 3.17E{+}04 \sum_i Q_i \; [X/Q]_s \; DF^{\gamma}_i \qquad (3\text{-}9)$$

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

$$\text{now} \quad D^{\gamma}_{finite} = D^{\gamma}_{air} \; [X/Q]^{\gamma}_s / \; [X/Q]_s \qquad (3\text{-}10)$$

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

and $\quad Q_i = 31.54 \quad \dot{Q}_i^{ST}$ (3-11)

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

so $\quad \dot{R}_{skins} = 1.11 \quad S_F \quad 1E{+}06 \; [X/Q]_s^Y \sum_i \dot{Q}_i^{ST} \quad DF_i^Y$ (3-12)

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

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

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

substituting

$[X/Q]_s^Y$ = 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

$$\dot{R}_{skins} = 0.68 \sum_i \dot{Q}_i^{ST} \quad DF_i^Y + 1.04 \sum_i \dot{Q}_i^{ST} \quad DFS_i \quad (3\text{-}13)$$

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

$$= \sum_i \dot{Q}_i^{ST} [0.68\ DF_i^{\gamma} + 1.04\ DFS_i] \qquad (3\text{-}14)$$

define

$$DF'_{is} = 0.68\ DF_i^{\gamma} + 1.04\ DFS_i \qquad (3\text{-}15)$$

then

$$\dot{R}_{skins} = \sum_i \dot{Q}_i^{ST} \quad DF'_{is} \qquad (3\text{-}7)$$

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

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

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

then $$DF'_{ig} = 7.13\ DF_i^{\gamma} + 35.2\ DFS_i \qquad (3\text{-}37)$$

and $$\dot{R}_{sking} = \sum_i Q_i^{GL}\ DF'_{ig} \qquad (3\text{-}38)$$

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 3.10 is inherent in Method I, as it determined the maximum expected off-site atmospheric dispersion factors based on past long-term site-specific meteorology.

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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.

3.5.3 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. Analyses requiring Method II calculations should be referred to YNSD to be performed and documented.

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

Technical Specification 3.8.E.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 Technical Specification 3.8.E.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 Technical Specification limits (i.e., use of actual meteorology over the period of interest) to determine if, in fact, Technical Specification 3.8.E.1.b had actually been exceeded during the sampling period.

3.6.1 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 3-16 (an example calculation is provided in Appendix A):

$$\dot{R}_{COS} = \sum_i \dot{Q}_i^{STP} \; DFG'_{sico} \qquad (3\text{-}16)$$

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

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:

$$\dot{R}_{cog} = \sum_i \dot{Q}_i^{GLP} DFG'_{gico} \qquad (3\text{-}40)$$

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 3-16 and 3-40 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.

3.6.2 Basis for Method I

The methods to calculate critical organ dose rate parallel the total body dose rate methods in Section 3.4.3. Only the differences are presented here.

Method I may be used to show that the Technical Specification which limits 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 (Technical Specification 3.8.E.1.b) has been met for the peak I + P release rates.

The equation for $\dot{R}_{cos}$ and $\dot{R}_{cog}$ is derived by modifying Equation 3-25 from Section 3.9 as follows:

$$\dot{D}_{cos} = \sum_i Q_i \; DFG_{ico} \qquad (3\text{-}17)$$

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

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:

$$\dot{R}_{cos} = 31.54 \sum_i \dot{Q}_i^{STP} \; DFG_{sico} \qquad (3\text{-}18)$$

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

Equation 3.8 is written in the form:

$$\dot{R} = 31.54 \sum_i \dot{Q}_i^{STP} \; DFG_{sico} \qquad (3\text{-}19)$$

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

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 3.10 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.

3.6.3 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. Analyses requiring Method II calculations should be referred to YNSD to be performed and documented.

3.7 Method to Calculate the Gamma Air Dose from Noble Gases

Technical Specification 3.8.F.1 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 Technical Specification 3.8.F.1 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.

3.7.1 Method I

The gamma air dose from plant stack releases is:

$$D_{airs}^{\gamma} = 0.019 \sum_{i} Q_i^{ST} \quad DF_i^{\gamma} \qquad (3\text{-}21)$$

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

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:

$$D_{airg}^{\gamma} = 0.20 \sum_{i} Q_i^{GL} DF_i^{\gamma} \qquad (3\text{-}41)$$

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 $D^{\gamma}_{airs} + D^{\gamma}_{airg}$.

Equations 3-21 and 3-41 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.

3.7.2 Basis for Method I

Method I may be used to show that the Technical Specification which limits off-site gamma air dose from gaseous effluents (3.8.F.1) has been met for releases over appropriate periods. This Technical Specification 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 $D^{\gamma}_{finite} = D^{\gamma}[X/Q]^{\gamma}/[X/Q]$:

$$D^{\gamma}_{airs} = 3.17E{+}04\,[X/Q]^{\gamma}_{s} \sum_i Q_i^{ST}\, DF_i^{\gamma} \qquad (3\text{-}22)$$

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

where:

$[X/Q]_s^T$ = 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:

$$D_{airs}^T = 0.019 \sum_i Q_i^{ST} \quad DF_i^T \qquad (3\text{-}21)$$

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

For the ground level release:

$$D_{airg}^T = 3.17E{+}04\ [X/Q]_g^T \sum_i Q_i^{GL}\ DF_i^T \qquad (3\text{-}42)$$

where:

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

= 6.42E-06 sec/m^3

leading to:

$$D_{airg}^T = 0.20 \sum_i Q_i^{GL}\ DF_i^T \qquad (3\text{-}41)$$

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 $[X/Q]^r$ rather than use the maximum long-term average value obtained for the years 1981 to 1985.

3.7.3 Method II

If the Method I dose determination indicates that the Technical Specification 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. Analyses requiring Method II calculations should be referred to YNSD to be performed and documented.

3.8 Method to Calculate the Beta Air Dose from Noble Gases

Technical Specification 3.8.F.1 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 Technical Specification 3.8.F.1 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.

3.8.1 Method I

The beta air dose from plant vent stack releases is:

$$D^{\beta}_{airs} = 0.033 \sum_{i} Q_i^{ST} \, DF_i^{\beta} \qquad (3\text{-}23)$$

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

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:

$$D^{\beta}_{air_g} = 1.12 \sum_i Q_i^{GL} DF_i^{\beta} \qquad (3\text{-}43)$$

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 3-23 and 3-43 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.

3.8.2 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 3.7.3. Only the differences are presented here.

Method I may be used to show that the Technical Specification which limits off-site beta air dose from gaseous effluents (3.8.A.1) has been met for releases over appropriate periods. This Technical Specification 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:

$$D^{\beta}_{airs} = 3.17E{+}04 \quad X/Q_S \sum_i Q_i^{ST} \quad DF_i^{\beta} \qquad (3\text{-}24)$$

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

substituting

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

= 1.04E-06 sec/m^3

We have

$$D^{\beta}_{airs} = 0.033 \quad \sum_i Q_i^{ST} \quad DF_i^{\beta} \qquad (3\text{-}23)$$

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

For the ground level release:

$$D^{\beta}_{airg} = 3.17E{+}04 \; (X/Q)_g \sum_i Q_i^{GL} \; DF_i^{\beta} \qquad (3\text{-}44)$$

where:

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

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

leading to:

$$D^{\beta}_{airg} = 1.12 \sum_i Q_i^{GL} DF_i^{\beta} \qquad (3\text{-}43)$$

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.

3.8.3 Method II

If Method I cannot be applied, or if the Method I dose determination indicates that the Technical Specification 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. Analyses requiring Method II calculations should be referred to YNSD to be performed and documented.

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

Technical Specification 3.8.G.1 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 Technical Specification 3.8.G.1 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).

3.9.1 Method I

$$D_{cos} = \sum_i Q_i^{STP} \, DFG_{sico} \qquad (3\text{-}25)$$

(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:

$$D_{cog} = \sum_i Q_i^{GLP} \, DFG_{gico} \qquad (3\text{-}44)$$

(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 3-25 and 3-44 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.

3.9.2 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 Technical Specifications which limit off-site organ dose from gases (3.8.G.1) have 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 Technical Specifications 3.8.G.2 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 3.9-1 and 3.9-2 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 3.10. However, he is exposed, conservatively, to all pathways (see Section 3.10). 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:

$$\Delta D_{ico} = Q_i DFG_{ico} \qquad (3\text{-}26)$$

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 Technical Specification limits 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.

3.9.3 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. Analyses requiring Method II calculations should be referred to YNSD to be performed and documented.

TABLE 3.9-1

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.
	Soil Surface Density	(KG/M^2)	240.	240.	240.	240.	240.	240.	240.	240.
T	Transport Time to User	(HRS)			48.	48.	48.	48.	480.	480.
TB	Soil Exposure Time(1)	(HRS)	131400.	131400.	131400.	131400.	131400.	131400.	131400.	131400.
TF	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.

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TABLE 3.9-1
(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).

TABLE 3.9-2

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

3.10 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 3.10.1 details the selection of important off-site locations and receptors. Section 3.10.2 describes the atmospheric model used to convert meteorological data into atmospheric dispersion factors. Section 3.10.3 presents the maximum atmospheric dispersion factors calculated at each of the off-site receptor locations.

3.10.1 Receptor Locations

Distances to the site boundary from the two evaluated gaseous release pathways (the Stack and North Warehouse) are provided in Table 3.10-2. 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 3.10-1) 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.

3.10.2 Vermont Yankee Atmospheric Dispersion Model

The long-term average atmospheric dispersion factors are computed for routine releases using Yankee Atomic Electric Company's (YAEC) 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 3.10-3.

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 $[X/Q^{\gamma}]$.

3.10.3 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 3.10-1. Table 3.10-1 also indicates which atmospheric dispersion factors are used to calculate the various doses or dose rates of interest.

TABLE 3.10-1

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)	-

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TABLE 3.10-2

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 3.10-3

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

3.11 Method to Calculate Direct Dose From Plant Operation

Technical Specification 3.8.M.1 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).

3.11.1 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:

where: (3-27)

D_d = The dose contribution from N-16 decay at either the site boundary of maximum impact (west site boundary) or closest 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) 3.23E-06 for the maximum west site boundary; or
(2) 1.29E-06 for the closest residence ($mrem/MW_e h$).

3.11.2 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:

(3-28)

and $D_U = 0.53 \times \dot{R}_U$ for the unshielded end (3-29)

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

where:

$$D_S = 0.25 \times \dot{R}_S \quad \text{for the shielded end} \qquad (3\text{-}28)$$

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

$$\text{and} \qquad D_U = 0.53 \times \dot{R}_U \quad \text{for the unshielded end} \qquad (3\text{-}29)$$

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

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 $\left(\frac{\text{mrem/yr}}{\text{mrem/hr}}\right)$.

3.11.3 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)

$$D_{dE} = 0.28 \times \dot{R}_d \times f_d \quad (3\text{-}30)$$

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

or

$$D_{dS} = 0.39 \times \dot{R}_d \quad f_d \quad (3\text{-}31)$$

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

where:

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' 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' 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' from the long side of the rectangular storage module when that face is orientated toward the west boundary.

(b) Scatter From Skyshine

$$D_{SKR} = 0.016 \times \dot{R}_{SKR} \times f_{sk} \qquad (3\text{-}32)$$

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

and

$$D_{SKD} = 0.015 \times \dot{R}_{SKD} \times f_{SK} \qquad (3\text{-}33)$$

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

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' over the top of each liner in a storage module (mrem/hr).

$\dot{R}_{SKD}$ = For DAW, the maximum dose rate measured at 3' 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' 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' 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:

$$D_{trans} = 0.0025 \times \dot{R}_{tran} \times T_{trans} \qquad (3\text{-}34)$$

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

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 trans 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 (3-30 and 3-31) 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 (3-35) accounts for any physical spacing between the outside storage modules which have not been covered by additional external shielding.

$$D_{Gap} = 2.44E\text{-}2 \times W_{Gap} \times A_{RL} \times f_{Gap} \qquad (3\text{-}36)$$

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

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).

3.11.4 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.

3.12 Cumulative Doses

Cumulative Doses for a calendar quarter and a calendar year must be maintained to demonstrate a compliance with Technical Specifications 3.8.B.1, 3.8.F.1 and 3.8.G.1 (10CFR50, Appendix I dose objectives). In addition, if the requirements of Technical Specification 3.8.M.2 dictate, cumulative doses over a calendar year must be determined for Technical Specification 3.8.M.1 (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.

4.0 ENVIRONMENTAL MONITORING PROGRAM

The radiological environmental monitoring stations are listed in Table 4.1. The locations of the stations with respect to the Vermont Yankee plant are shown on the maps in Figures 4-1 to 4-6.

4.1 Intercomparison Program

All routine radiological analyses for environmental samples are performed at the Yankee Atomic Environmental Laboratory (YAEL). The YAEL participates in the U.S. Environmental Protection Agency's Environmental Radioactivity Laboratory Intercomparison Studies Program for all appropriate species and matrices offered by the agency.

4.2 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 18 years of operational data (as of 1990). 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, 11.3 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, 16.1 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, 3.4 km), which is along the western side of the valley in general proximity of historical dairy operations. (The sixth location is not a specific Technical Specification requirement.)

4.3 Distances and Directions to Monitoring Stations

It should be noted that the distances and directions for direct radiation monitoring locations in Table 4.1, as well as the sectors shown in Figures 4.5 and 4.6, 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. These distances and directions are also used in the semiannual reports.

Technical Specification 6.7 and Table 3.9.3, Footnote a, specify that in the Annual Radiological Environmental Surveillance 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 Surveillance Reports and ODCM for the sampling and TLD monitoring stations, respectively.

Table 4.1

Radiological Environmental Monitoring Stations[1]

Exposure Pathway and/or Sample		Sample Location and Designated Code[2]		Distance (km)[6]	Direction[6]
1. AIRBORNE (Radioiodine and Particulate)					
		AP/CF-11	River Station No. 3.3	1.9	SSE
		AP/CF-12	N. Hinsdale, NH	3.6	NNW
		AP/CF-13	Hinsdale Substation	3.1	E
		AP/CF-14	Northfield, MA	11.3	SSE
		AP/CF-15	Tyler Hill Road[4]	3.2	WNW
		AP/CF-21	Spofford Lake	16.1	NNE
2. WATERBORNE					
	a. Surface	WR-11	River Station No. 3.3	1.9	Downriver
		WR-21	Rt. 9 Bridge	12.8	Upriver
	b. Ground	WG-11	Plant Well	--	On-Site
		WG-12	Vernon Nursing Well	2.0	SSE
		WG-22	Skibniowsky Well	14.3	N
	c. Sediment From Shoreline	SE-11	Shoreline Downriver	0.8	SSE
		SE-12	North Storm Drain Outfall[3]	0.15	E
3. INGESTION					
	a. Milk	TM-11	Miller Farm	0.8	WNW
		TM-12	Dominick[5]	5.2	E
		TM-14	Brown Farm	2.1	S
		TM-16	Meadow Crest Farm	4.4	WNW/NW
		TM-18	Blodgett Farm[4]	3.4	SE
		TM-24	County Farm	22.5	N
	b. Mixed Grasses	TG-11	River Station No. 3.3	1.9	SSE
		TG-12	N. Hinsdale, NH	3.6	NNW
		TG-13	Hinsdale Substation	3.1	E
		TG-14	Northfield, MA	11.3	SSE
		TG-15	Tyler Hill Rd.[4]	3.2	WNW
		TG-21	Spofford Lake	16.1	NNE

Table 4.1
(Continued)

Radiological Environmental Monitoring Stations[1]

Exposure Pathway and/or Sample	Sample Location and Designated Code[2]		Distance (km)[6]	Direction[6]
c. Silage	TC-11	Miller Farm	0.8	WNW
	TC-12	Dominick[5]	5.2	E
	TC-14	Brown Farm	2.1	S
	TC-16	Meadow Crest Farm	4.4	WNW/NW
	TC-18	Blodgett Farm[4]	3.4	SE
	TC-24	County Farm	22.5	N
d. Fish	FH-11	Vernon Pond	(7)	(7)
	FH-21	Rt. 9 Bridge	12.8	Upriver
4. DIRECT RADIATION				
	DR-1	River Station No. 3.3	1.6	SSE
	DR-2	N. Hinsdale, NH	3.9	NNW
	DR-3	Hinsdale Substation	3.0	E
	DR-4	Northfield, MA	11.0	SSE
	DR-5	Spofford Lake	16.3	NNE
	DR-6	Vernon School	0.46	WSW
	DR-7	Site Boundary	0.27	W
	DR-8	Site Boundary(8)	0.25	SW
	DR-9	Inner Ring	2.1	N
	DR-10	Outer Ring	4.6	N
	DR-11	Inner Ring	2.0	NNE
	DR-12	Outer Ring	3.6	NNE
	DR-13	Inner Ring	1.4	NE
	DR-14	Outer Ring	4.3	NE
	DR-15	Inner Ring	1.4	ENE
	DR-16	Outer Ring	2.9	ENE
	DR-17	Inner Ring	1.2	E
	DR-18	Outer Ring	3.0	E
	DR-19	Inner Ring	3.5	ESE
	DR-20	Outer Ring	5.3	ESE
	DR-21	Inner Ring	1.8	SE
	DR-22	Outer Ring	3.2	SE
	DR-23	Inner Ring	1.8	SSE
	DR-24	Outer Ring	3.9	SSE
	DR-25	Inner Ring	2.0	S

Table 4.1
(Continued)

Radiological Environmental Monitoring Stations[1]

Exposure Pathway and/or Sample	Sample Location and Designated Code[2]		Distance (km)[6]	Direction[6]
	DR-26	Outer Ring	3.7	S
	DR-27	Inner Ring	1.0	SSW
	DR-28	Outer Ring	2.2	SSW
	DR-29	Inner Ring	0.7	WSW
	DR-30	Outer Ring	2.3	SW
	DR-31	Inner Ring	0.8	W
	DR-32	Outer Ring	5.0	WSW
	DR-33	Inner Ring	0.9	WNW
	DR-34	Outer Ring	4.9	W
	DR-35	Inner Ring	1.4	WNW
	DR-36	Outer Ring	4.7	WNW
	DR-37	Inner Ring	3.0	NW
	DR-38	Outer Ring	7.7	NW
	DR-39	Inner Ring	3.2	NNW
	DR-40	Outer Ring	5.8	NNW

(1) Sample locations are shown on Figures 4.1 to 4.6.

(2) Station Nos. 10 through 19 are indicator stations. Station Nos. 20 through 29 are control stations (for all but the direct radiation stations).

(3) To be sampled and analyzed semiannually.

(4) Non-Tech Spec station.

(5) Non-Tech Spec station. Sample will be collected as available.

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

(7) Fish samples are collected from anywhere in Vernon Pond, which is adjacent to the plant (see Figure 4-1).

(8) DR-8 satisfies Technical Specification Table 3.9.3 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.

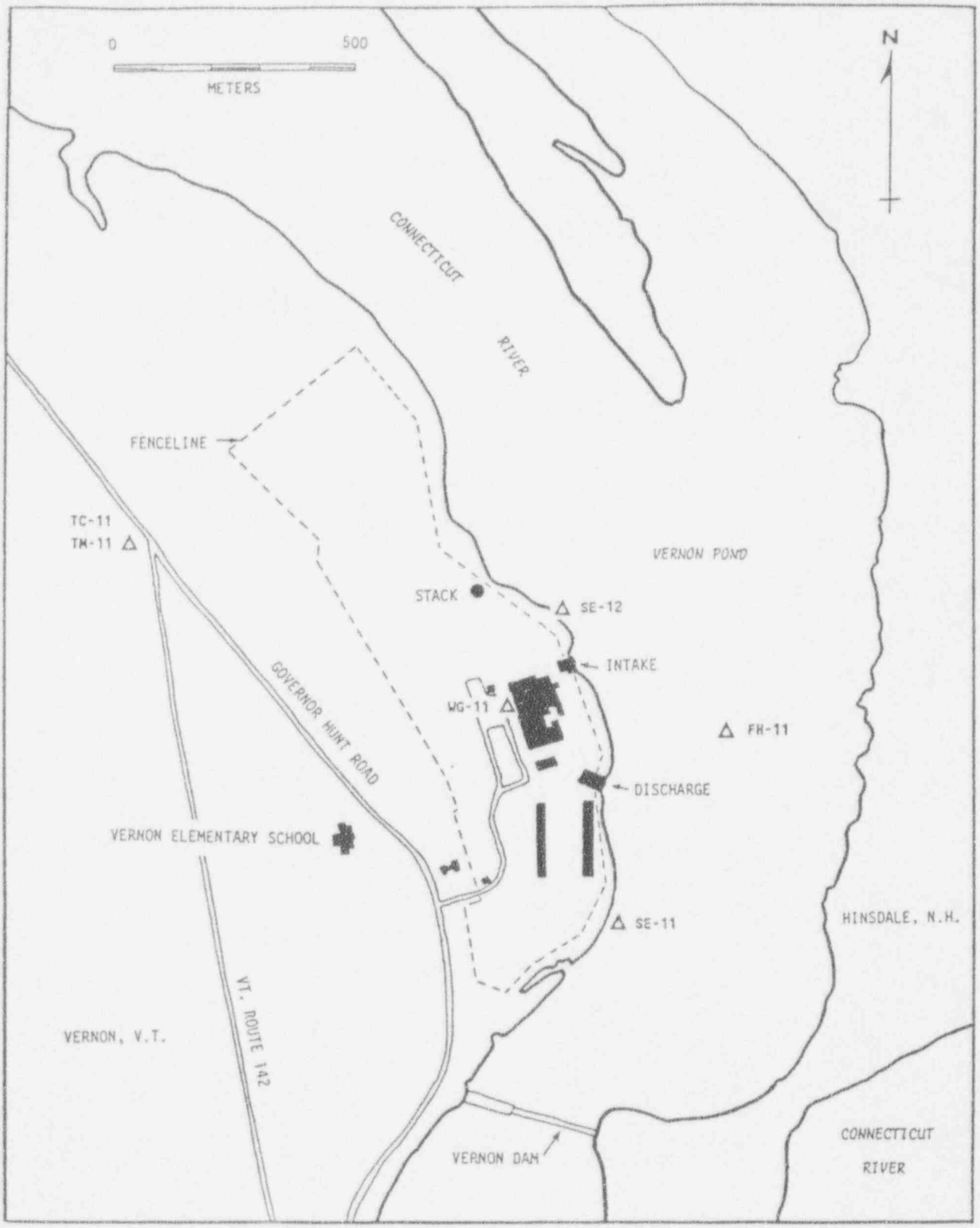

Figure 4-1 Environmental Sampling Locations in Close Proximity to Plant

Revision 10 Date 4/4/91

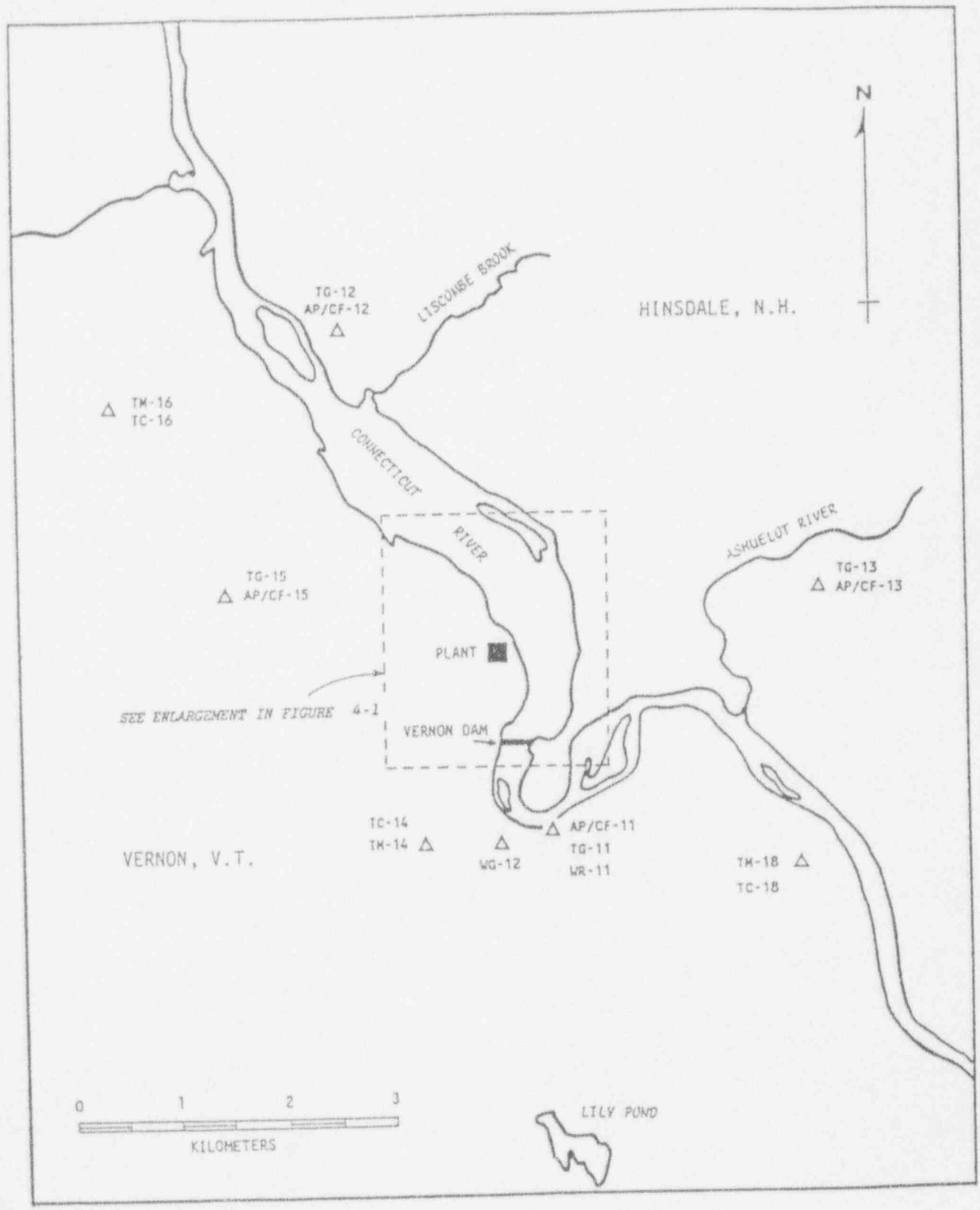

Figure 4-2 Environmental Sampling Locations Within 5 km of Plant

Revision 16 Date 10/28/93

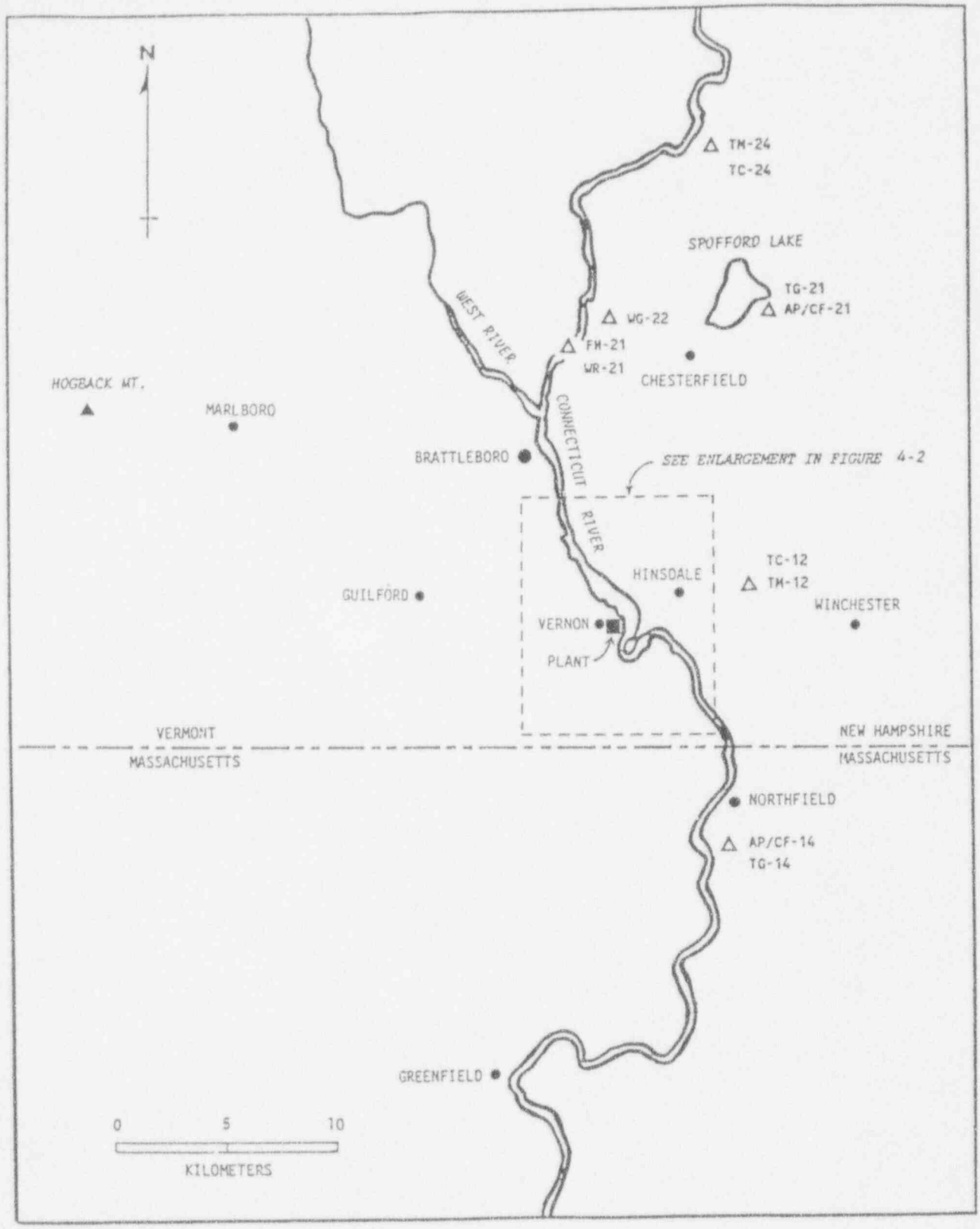

Figure 4-3 Environmental Sampling Locations Greater than 5 km from Plant

Revision 10 Date 4/4/91

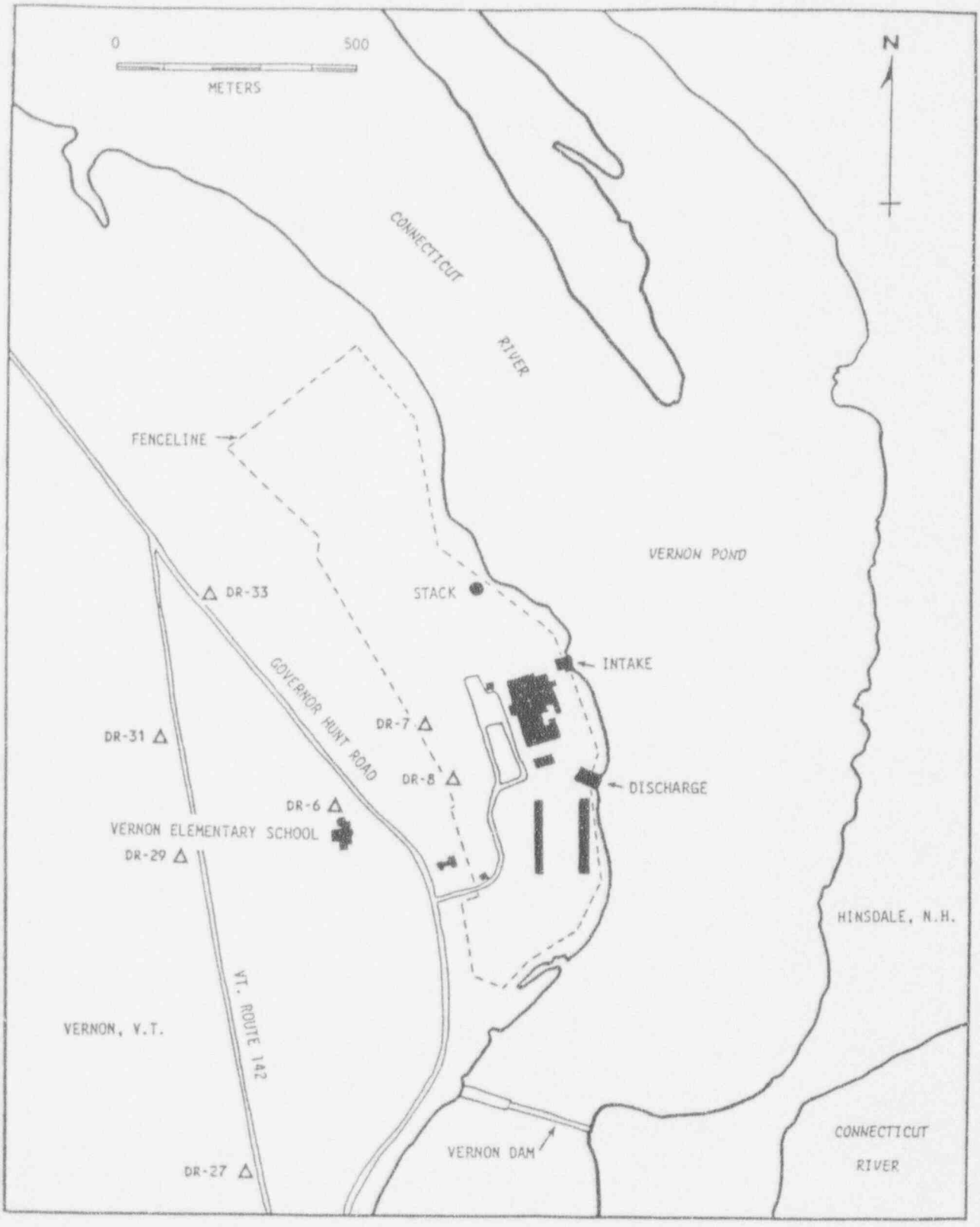

Figure 4-4 TLD Locations in Close Proximity to Plant

Revision 10 Date 4/4/91

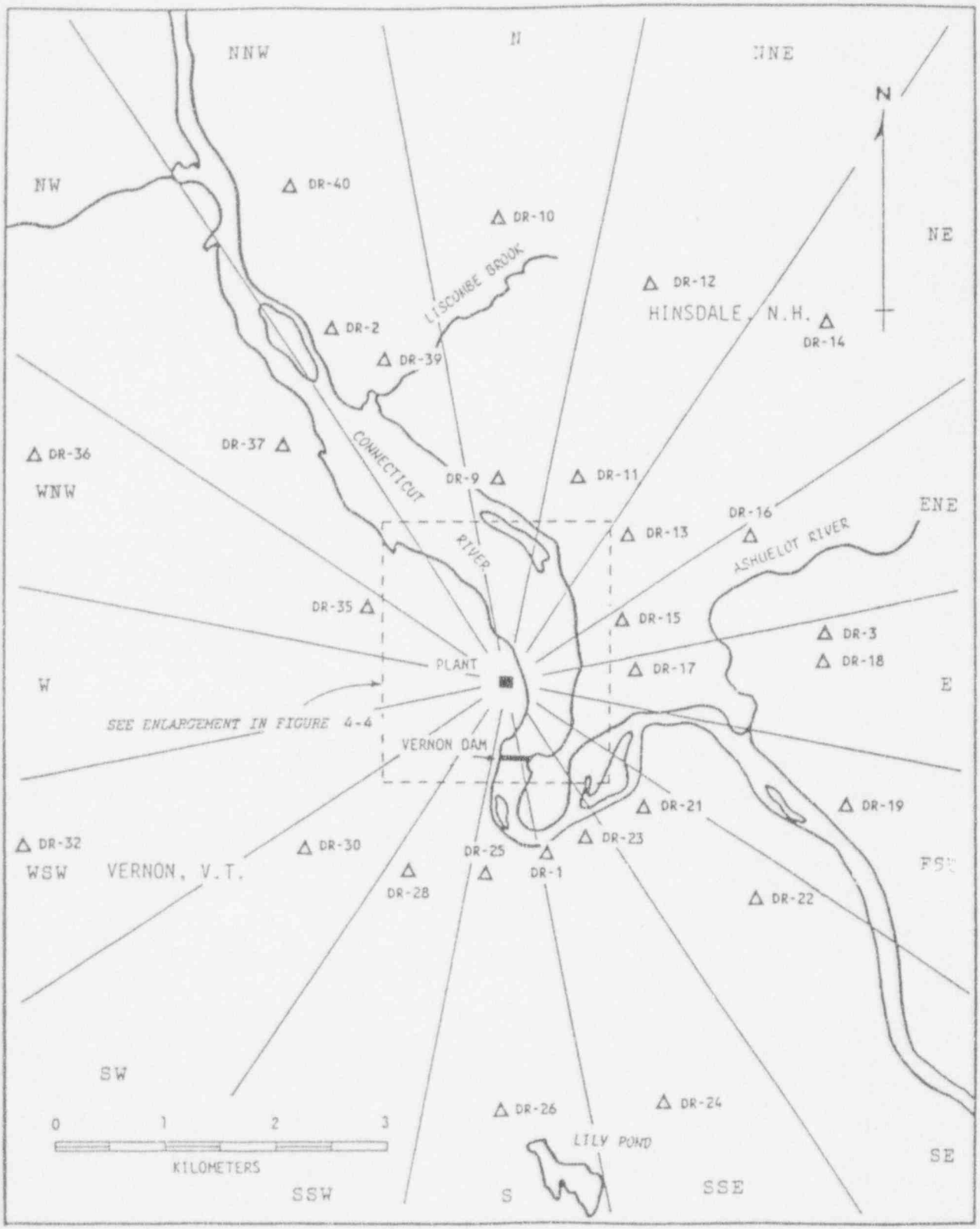

Figure 4-5 TLD Locations Within 5 km of Plant

Revision 10 Date 4/4/91

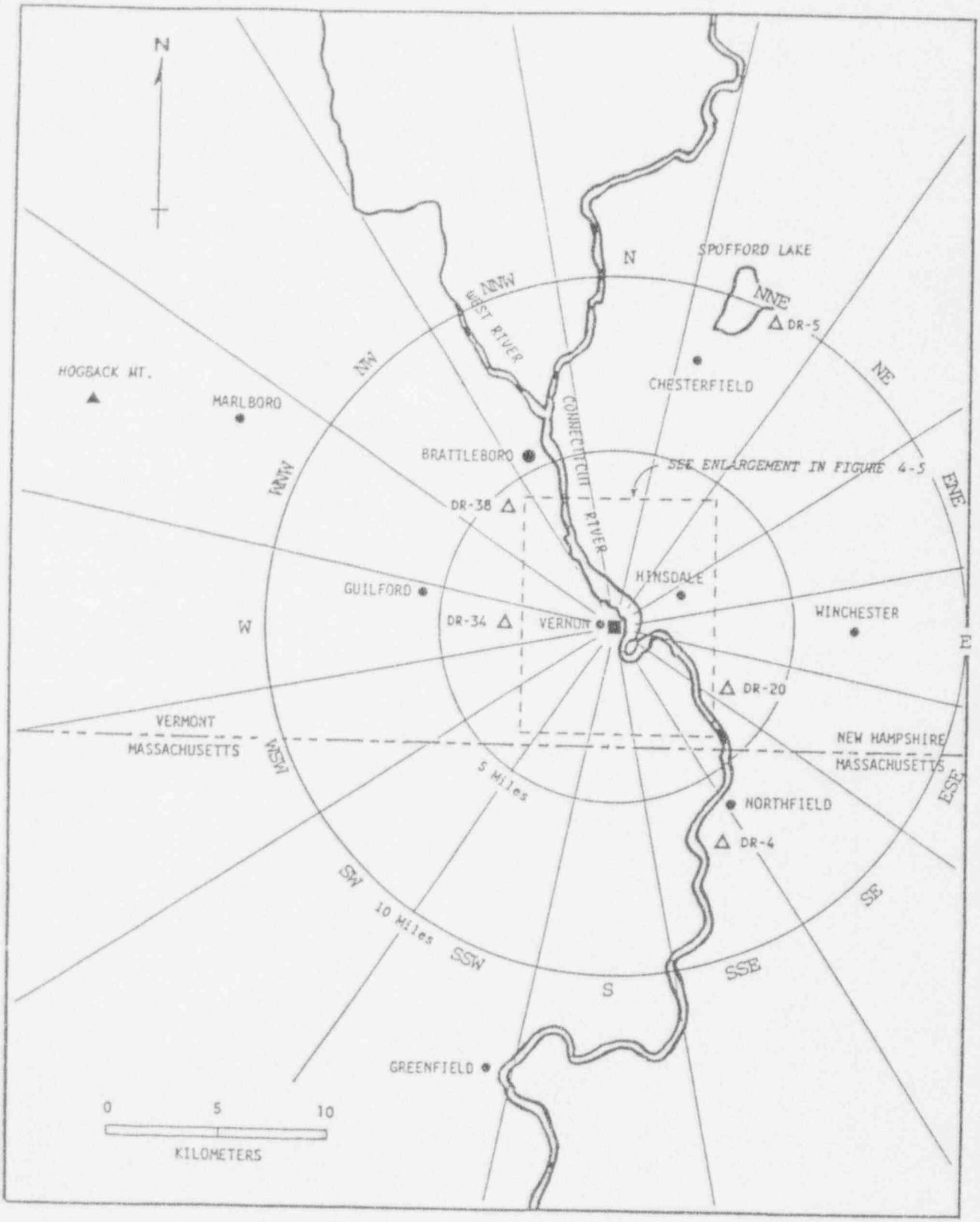

Figure 4-6 TLD Locations Greater than 5 km from Plant

Revision 10 Date 4/4/91

5.0 SETPOINT DETERMINATIONS

Chapter 5 contains the basis for plant procedures that the plant operator requires to meet the setpoint requirements of the Radioactive Effluent Monitoring Systems Technical Specifications. They are Specification 3.9.A.1 for liquids and Specification 3.9.B.1 for gases. Each outlines the instrumentation channels and the basis for each setpoint.

5.1 Liquid Effluent Instrumentation Setpoints

Technical Specification 3.9.A.1 requires that the radioactive liquid effluent instrumentation in Table 3.9.1 of the Technical Specifications have alarm setpoints in order to ensure that Specification 3.8.A.1 is not exceeded. Specification 3.8.A.1 limits the activity concentration at any time in liquid effluents to the appropriate effluent concentration values in Appendix B, Table 2, Column 2 of 10CFR20, and a total noble gas concentration limit of 2E-04 µCi/ml.

5.1.1 Liquid Radwaste Discharge Monitor (17/350)

The sample tank pathways shown on Figure 6-1 are monitored by the liquid radwaste discharge monitor (17/350). Periodic batch releases may be made from the waste sample tanks, detergent waste tank or floor drain sample tank.

5.1.1.1 Method to Determine the Setpoint of the Liquid Radwaste Discharge Monitor (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:

$$R_{setpoint} = \frac{DF}{DF_{min}} \quad S_1 \quad \sum_i \quad C_{mi} \qquad (5\text{-}1)$$

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

Where:

$DF = \frac{F_d}{F_m}$ = Dilution factor (as a conservative measure, a DF of at least 1000 is used) (dimensionless). (5-2)

F_m = Flow rate past monitor (gpm)

F_d = Flow rate out of discharge canal (gpm)

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

$$= \sum_i \frac{C_{mi}}{ECL_i} \qquad (5\text{-}3)$$

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_1	=	Detector counting efficiency from the most recent liquid radwaste discharge monitor calibration curve (cps/(μCi/ml))

5.1.1.2 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_1, 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

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

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

$$= 1.22E\text{-}04$$

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

$$DF_{min} = \sum_i \frac{C_{mi}}{ECL_i} \qquad (5\text{-}3)$$

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

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

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

$$= 107.2$$

The minimum dilution factor, DF_{min}, needed to discharge the mixture of radionuclides in this example is 108 (roundup). 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:

$$F_m = \frac{F_1}{DF} \qquad (5\text{-}4)$$

(gpm) (gpm)

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

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

$$R_{setpoint} = \frac{DF}{DF_{min}} \quad S_1 \quad \sum_i \quad C_{mi} \qquad (5\text{-}1)$$

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

$$= \frac{1.000}{108} \quad 4.9E{+}06 \quad 1.22E{-}04$$

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

$$= 5.535 \text{ cps}$$

In this example, the count rate alarm of the liquid radwaste discharge monitor should be set at 5.535 cps above background.

5.1.1.3 Basis for the Liquid Radwaste Discharge Monitor Setpoint

The liquid radwaste discharge monitor setpoint must ensure that Specification 3.8.A.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 as follows:

$$R = \sum_i \quad C_{mi} \quad S_{1i} \qquad (5\text{-}5)$$

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

Where:

R	=	Response of the monitor (cps)
S_{1i}	=	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 5-5 one may substitute S_1 for S_{1i}, where S_1 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.

$$R = S_1 \sum_i C_{mi} \qquad (5\text{-}6)$$

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

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:

$$\sum_i \frac{C_{di}}{ECL_i} \leq 1 \qquad (5\text{-}7)$$

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

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:

$$C_{di} = C_{mi} \frac{F_m}{F_d} \qquad (5\text{-}8)$$

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

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 5-8 for C_{di} in Equation 5-7 and solving for F_d/F_m yields the minimum dilution factor needed to comply with Equation 5-7:

$$DF_{min} \leq \frac{F_d}{F_m} \geq \sum_i \frac{C_{mi}}{ECL_i} \qquad (5\text{-}3)$$

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

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)

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:

$$\frac{F_d}{F_m} \geq DF_{min} \qquad (5\text{-}3)$$

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

Usually F_d/F_m is greater than DF_{min} (i.e., there is more dilution than necessary to comply with Equation 5-7). The response of the liquid radwaste discharge monitor at the setpoint is therefore:

$$R_{setpoint} = \frac{DF}{DF_{min}} S_1 \sum_i C_{mi} \qquad (5\text{-}1)$$

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

5.1.2 Service Water Discharge Monitor (17/351)

The service water pathway shown on Figure 6-1 is continuously monitored by the service water discharge monitor (17/351). The water in this line is not radioactive under normal operating conditions. The alarm setpoint on this monitor is set at a level which is three times the background of the instrument. The 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.

5.2 Gaseous Effluent Instrumentation Setpoints

Technical Specification 3.9.B.1 requires that the radioactive gaseous effluent instrumentation in Table 3.9.2 of the Technical Specifications have their alarm setpoints set to insure that Technical Specifications 3.8.E.1 and 3.8.K.1 are not exceeded. Technical Specification 3.8.K.1 limits the gross radioactivity release rate at the steam jet air ejector (SJAE) to 0.16 Ci/sec.

5.2.1 Plant Stack Noble Gas Activity Monitors (RR-108-1A and RR-108-1B) and Augmented Off-Gas System Noble Gas Activity Monitors (3127 and 3128)

The plant stack and AOG noble gas activity monitors are shown on Figure 6-2.

5.2.1.1 Method to Determine the Setpoint of the Plant Stack Noble Gas Activity Monitors (RR-108-1A and RR-108-1B) and the Augmented Off-Gas System Noble Gas Activity Monitors (3127 and 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:

$$R_{spt}^{tb} = 818 \; S_g \; \frac{1}{F} \; \frac{1}{DFB_c} \qquad (5\text{-}9)$$

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

and:

$$R_{spt}^{skin} = 3{,}000 \; S_g \; \frac{1}{F} \; \frac{1}{DF_c'} \qquad (5\text{-}10)$$

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

where:

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

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

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/m^3)

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 (cm^3/sec)

DFB_C = Composite total body dose factor ($mrem\text{-}m^3/pCi\text{-}yr$)

$$= \frac{\sum_i \dot{Q}_i DFB_i}{\sum_i \dot{Q}_i} \qquad (5\text{-}11)$$

$\dot{Q}_i$ = 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) ($mrem\text{-}m^3/pCi\text{-}yr$)

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)

$$= \frac{\sum_i \dot{Q}_i DF'_{is}}{\sum_i \dot{Q}_i} \qquad (5\text{-}12)$$

DF'_{is} = Combined skin dose factor (see Table 1.1-10) (mrem-sec/μCi-yr)

5.2.1.2 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 5-9 and 5-10 for determining setpoints.

The plant stack noble gas activity monitors, referred to as "Stack Gas I" (RR-108-1A) and "Stack Gas II" (RR-108-1B), consist of beta sensitive scintillation detectors, electronics, an analog 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 1E+08 cpm per 1 µCi/cc of noble gas. The nominal plant stack flow is 8.3E+07 cc/sec ((175,000 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	$\dot{Q}_i^{SJAE}$ $(\frac{\mu Ci}{sec})$	DFB_i $(\frac{mrem\text{-}m^3}{pCi\text{-}yr})$	DF'_{is} $(\frac{mrem\text{-}sec}{\mu Ci\text{-}yr})$
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+02	1.81E-03	3.24E-03
Xe-133	1.97E+01	2.94E-04	5.58E-04

$$DFB_c = \frac{\sum_i \dot{Q}_i^{SJAE} DFB_i}{\sum_i \dot{Q}_i^{SJAE}} \qquad (5\text{-}11)$$

$$\sum_i \dot{Q}_i^{SJAE} DFB_i = (1.03E+04)(8.83E-03) + (4.73E-02)(5.92E-03)$$
$$+ (2.57E+02)(1.47E-02) + (1.20E+02)(1.17E-03)$$
$$+ (3.70E+02)(1.81E-03) + (1.97E+01)(2.94E-04)$$
$$= 9.83E+01 \ (\mu Ci\text{-}mrem\text{-}m^3/sec\text{-}pCi\text{-}yr)$$

$$\sum_i \dot{Q}_i^{SJAE} = 1.03E+04 + 4.73E+02 + 2.57E+02$$
$$+ 1.20E+02 + 3.70E+02 + 1.97E+01$$
$$= 1.15E+04 \ \mu Ci/sec$$

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

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

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

$$= (818) \ (1E+08) \ \frac{1}{(8.3E+07)} \ \frac{1}{(8.52E-03)}$$

$$= 115{,}674 \text{ cpm}$$

Next:

$$DF'_c = \frac{\sum_i \dot{Q}_i^{SJAE} DF'_{is}}{\sum_i \dot{Q}_i^{SJAE}} \qquad (5\text{-}11)$$

$$\sum_i \dot{Q}_i^{SJAE} DF'_{is} = (1.03E+04)(1.06E-02) + (4.73E-02)(1.43E-02)$$
$$+ (2.57E+02)(1.28E-02) + (1.20E+02)(2.35E-03)$$
$$+ (3.70E+02)(3.24E-03) + (1.97E+01)(5.58E-04)$$
$$= 1.14E+02 \ (\mu Ci\text{-}mrem\text{-}sec/sec\text{-}\mu Ci\text{-}yr)$$

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

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

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

$$= (3,000) \ (1E+08) \ \frac{1}{(8.3E+07)} \ \frac{1}{(9.91E-03)}$$

$$= 364,728 \ cpm$$

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 115,674 cpm above background or at some conservative value below this (such as that which 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 Technical Specification whole body dose limit 500 rem/yr (115,674 cpm) is chosen, then the noble gas monitor alarms should be set at no more than 80,972 cpm above background (0.7 x 115,674 = 80,972).

5.2.1.3 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 Technical Specification 3.8.E.1.a is not exceeded. Sections 3.4 and 3.5 show that Equations 3-5 and 3-7 are acceptable methods for determining compliance with that Technical Specification. 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 5-9 and 5-10 begin with the general equation for the response R of a radiation monitor:

$$R = \sum_i S_{gi} \quad C_{mi} \tag{5-13}$$

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

where:

R = Response of the instrument (cpm)

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

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

The relative release rate of each noble gas, $\dot{Q}_i$ ($\mu Ci/sec$), 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 present 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 5.2-1 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.

$$C_{mi} = \dot{Q}_i \frac{1}{F} \quad (5\text{-}14)$$

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

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 5-14 into Equation 5-13 for C_{mi} yields:

$$R = \sum_i S_{gi} \dot{Q}_i \frac{1}{F} \quad (5\text{-}15)$$

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

The detector calibration procedure establishes a counting efficiency for a given mix of nuclides seen by the detector. Therefore, in Equation 5-15 one may substitute S_g for S_{gi}, where S_g 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.

$$R = S_g \frac{1}{F} \sum_i \dot{Q}_i \quad (5\text{-}16)$$

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

The total body dose rate due to noble gases is determined with Equation 3-5:

$$\dot{R}_{tbs} = 0.61 \sum_i \dot{Q}_i \, DFB_i \qquad (3\text{-}5)$$

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

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:

$$DFB_c \sum_i \dot{Q}_i = \sum_i \dot{Q}_i \, DFB_i \qquad (5\text{-}17)$$

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

Solving Equation 5-23 for DFB_c yields:

$$DFB_c = \frac{\sum_i \dot{Q}_i \, DFB_i}{\sum_i \dot{Q}_i} \qquad (5\text{-}11)$$

Technical Specification 3.8.E.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 $\dot{R}_{tb}$ equal to 500 mrem/yr and substituting DFB_c for DFB_i in Equation 3-5, one may solve for $\sum_i \dot{Q}_i$ at the limiting whole body noble gas dose rate:

$$\sum_i \dot{Q}_i = \frac{818}{\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem\text{-}\mu Ci\text{-}m^3}{yr\text{-}pCi\text{-}sec}\right)} \quad \frac{1}{\left(\frac{pCi\text{-}yr}{mrem\text{-}m^3}\right)} \qquad (5\text{-}18)$$

Substituting this result for $\sum \dot{Q}_i$ in Equation 5-16 yields R_{spt}^{tb}, the response of the monitor at the limiting noble gas total body dose rate:

$$R_{spt}^{tb} = \frac{818}{(cpm)\left(\frac{mrem\text{-}\mu Ci\text{-}m^3}{yr\text{-}pCi\text{-}sec}\right)} \quad \frac{S_g}{\left(\frac{cpm\text{-}cm^3}{\mu Ci}\right)} \quad \frac{1}{\left(\frac{sec}{cm^3}\right)} \quad \frac{1}{\left(\frac{pCi\text{-}yr}{mrem\text{-}m^3}\right)} \qquad (5\text{-}9)$$

The skin dose rate due to noble gases is determined with Equation 3-7:

$$\dot{R}_{spt}^{skin} = \sum_i \frac{\dot{Q}_i}{\left(\frac{mrem}{yr}\right)} \quad \frac{DF'_{is}}{\left(\frac{\mu Ci}{sec}\right)\left(\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}\right)} \qquad (3\text{-}7)$$

Where:

$\dot{R}_{spt}^{skin}$ = 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:

$$DF'_c \sum_i \dot{Q}_i = \sum_i \dot{Q}_i \, DF'_{is} \qquad (5\text{-}19)$$

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

Solving Equation 5-19 for DF' yields:

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

Technical Specification 3.8.E.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 3-7 one may solve for $\sum_i \dot{Q}_i$ at the limiting skin noble gas dose rate:

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

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

Substituting this result for $\sum_i \dot{Q}_i$ in Equation 5-16 yields R_{spt}^{skin}, the response of the monitor at the limiting noble gas skin dose rate:

$$R_{spt}^{skin} = 3.000 \quad S_g \quad \frac{1}{F} \quad \frac{1}{DF'_c} \qquad (5\text{-}10)$$

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

TABLE 5.2-1

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 ≤ t < 48 h	---	.003	.004	--	.001	.004	.022	.758	.010	.198	---
48 h ≤ t < 5 d	---	---	.005	---	---	.006	.024	.907	.001	.058	---
5 d ≤ t < 10 d	---	---	.007	---	---	.008	.016	.969	---	---	---
10 d ≤ t < 15 d	---	---	.014	---	---	.014	.006	.966	---	---	---
15 ≤ t < 20 d	---	---	.026	---	---	.022	.002	.950	---	---	---
20 ≤ t < 30 d	---	---	.048	---	---	.034	.001	.917	---	---	---
30 ≤ t < 40 d	---	---	.152	---	---	.070	---	.777	---	---	
40 ≤ t < 50 d	---	---	.378	---	---	.105	---	.517	---	---	---
50 ≤ t < 60 d	---	---	.652	---	---	.108	---	.240	---	---	---
60 ≤ t < 70 d	---	---	.835	---	---	.083	---	.082	---	---	---
t ≥ 70 d	---	---	.920	---	---	.055	---	.024	---	---	---

5.2.2 Steam Jet Air Ejector (SJAE) Noble Gas Activity Monitors (17/150A and 17/150B)

The SJAE noble gas activity monitors are shown in Figure 6-2.

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

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

$$R_{spt}^{SJAE} = 1.6E+05 \quad S_g \quad \frac{1}{F} \qquad (5\text{-}21)$$

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

Where:

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)

5.2.2.2 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 5-21 is straightforward. Simply taking Equation 5-16 and substituting the limiting release rate at the SJAE for $\dot{Q}$ yields Equation 5-21, the setpoint equation for the SJAE noble gas activity monitor.

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

Figure 6-1 shows the normal (design) radioactive liquid effluent streams, radiation monitors, and the appropriate Liquid Radwaste Treatment System. Figure 6-2 shows the normal (design) gaseous effluent systems, radiation monitors, and the appropriate Gaseous Radwaste Treatment System.

6.1 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 Technical Specification 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 Technical Specification 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.

6.2 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

EXAMPLE PROBLEM NO. 9
(Continued)

The dose factor (DFG_{sico}) for each radionuclide detected in the plant stack charcoal and particulate filter sample (plus tritium) is taken from Table 1.1-12 of the ODCM.

Therefore:

$$D_{cos} = (5.42E\text{-}04)(1.14E+01) + (1.10E\text{-}02)(4.31E+02) + (2.30E\text{-}01)(7.63E+00) + (1.15E\text{-}02)(1.63E+01) + (2.60E\text{-}02)(3.71E+00) + (4.30E\text{-}03)(7.71E+01) + (1.12E\text{-}04)(8.21E\text{-}01) + (0.15)(3.13E\text{-}04) =$$

Answer

D_{cos} = 7.12 mrem maximum organ dose for the month.

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/absorber 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/absorber subsystem may be bypassed if it becomes unavailable provided the releases are within Technical Specification limits. With the dryer/absorber 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.

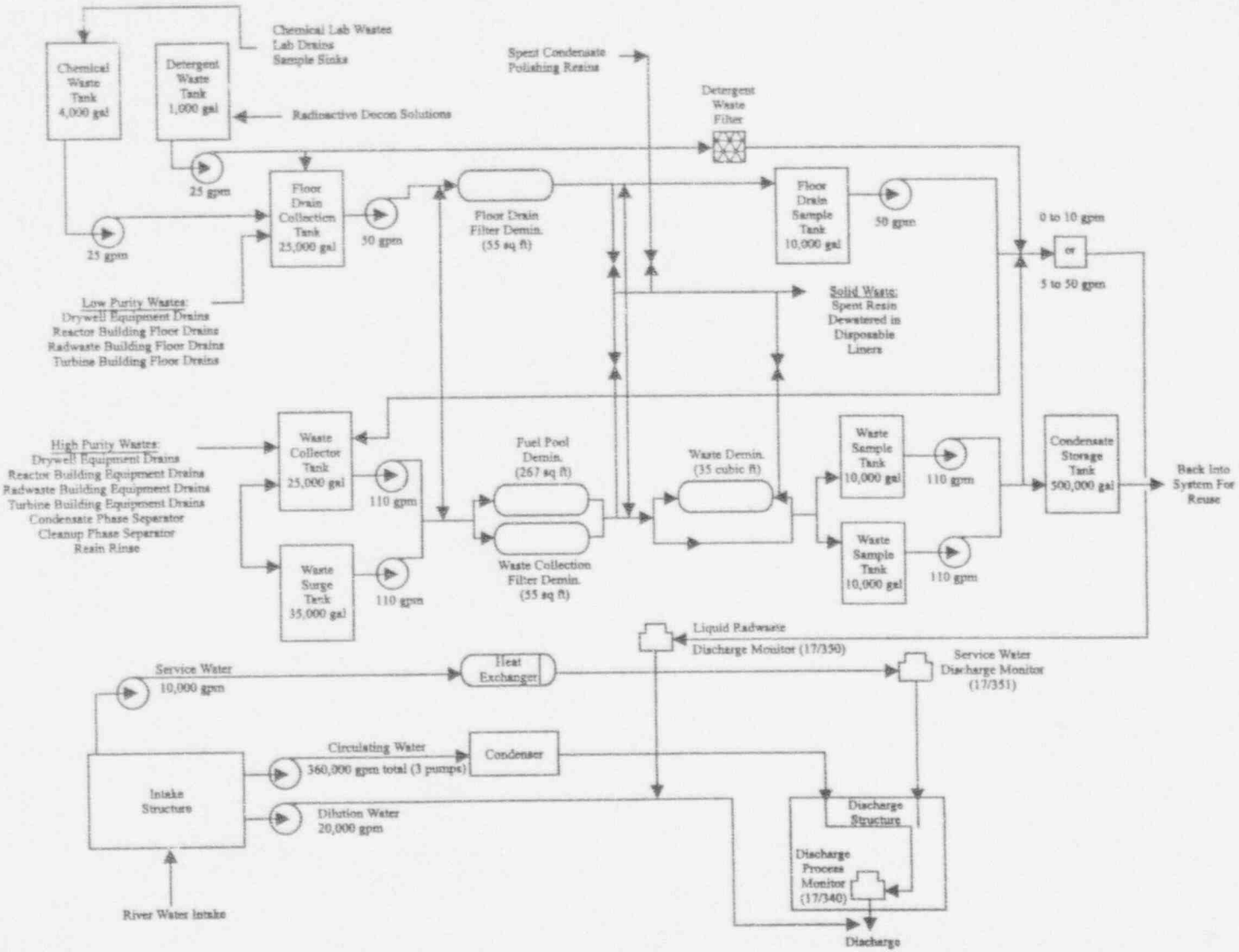

FIGURE 6-1: Radioactive Liquid Effluent Streams, Radiation Monitors, and Radwaste Treatment System at Vermont Yankee*

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

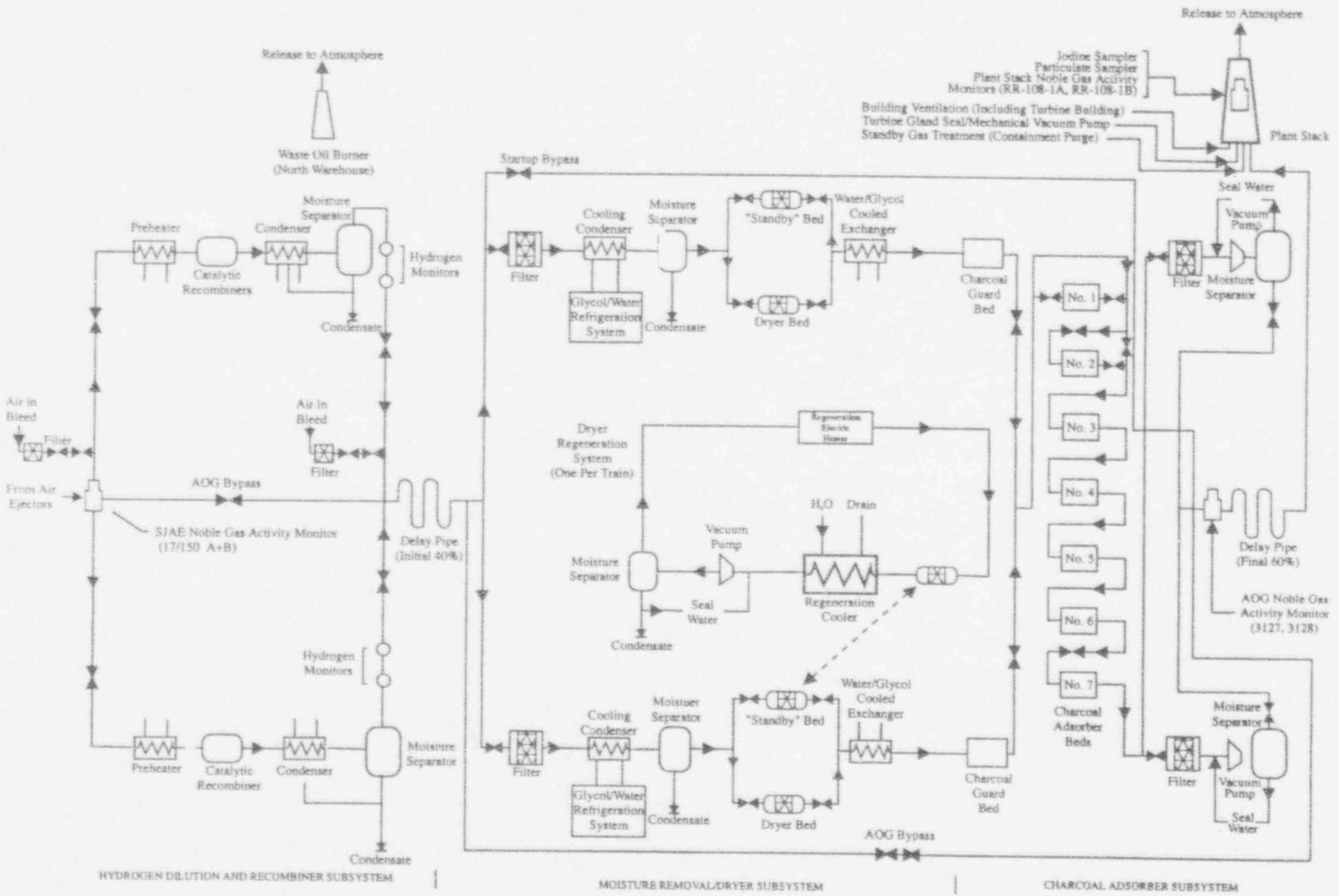

FIGURE 6-2: Radioactive Gaseous Effluent Streams, Radiation Monitors, and Radwaste Treatment System at Vermont Yankee*

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

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E. Slade, D. H., "Meteorology and Atomic Energy - 1968, USAEC, July 1968.

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APPENDIX A

METHOD I EXAMPLE CALCULATIONS

EXAMPLE CALCULATION NO. 1

Type

Total Body Dose From Liquid Effluents

References

a) ODCM Sections 3.2 and 3.3 (Method I).

b) Technical Specifications 3.8.B.1 and 3.8.C.1.

Problem

Calculate the off-site dose to the total body and maximum organ resulting from the batch release of radioactive liquid effluents.

Plant Data

a) Analysis from a representative grab sample of the liquid waste volume to be discharged indicates the following radionuclide activities were released to the Connecticut River:

i	Activity Q_i Released (Ci)	DFL_{itb}* (mrem/Ci)	DFL_{imo}** (mrem/Ci)
H-3	3.40E-01	2.60E-04	2.06E-04
Co-60	1.72E-05	2.13E-01	1.28E+00
Co-134	3.51E-05	1.28E+02	1.60E+02
I-131	7.20E-04	2.57E-02	1.47E+01

*Total body dose factor from Table 1.1-11.

**Maximum organ dose factor from Table 1.1-11.

Calculation

The total body dose is calculated from Equation (3-1):

$$D_{tb} = \sum_i \quad Q_i \quad DFL_{itb} \qquad (3\text{-}1)$$

(mrem) (Ci) (mrem/Ci)

Therefore:

$$D_{tb} = (3.40E\text{-}01)(2.06E\text{-}04) + (1.72E\text{-}05)(2.13E\text{-}01) + (3.51E\text{-}05)(1.28E+02) + (7.20E\text{-}04)(2.57E\text{-}02) =$$

Answer (1)

D_{tb} = 4.59E-03 mrem to the total body.

Next, the maximum organ dose is calculated from Equation (3-3):

$$D_{mo} = \sum_i \quad Q_i \quad DFL_{imo}$$

(mrem) (Ci) (mrem/Ci)

Therefore:

$$D_{mo} = (3.40E\text{-}01)(2.06E\text{-}04) + (1.72E\text{-}05)(1.28E+00) + (3.51E\text{-}05)(1.60E+02) + (7.20E\text{-}04)(1.47E+01) =$$

Answer (2)

D_{mo} = 1.63E-02 mrem to maximum organ.

EXAMPLE CALCULATION NO. 2

Type

Total Body Dose Rate From Noble Gases

References

a) ODCM Section 3.4 (Method I).

b) Technical Specification 3.8.E.1.a.

Problem

Calculate the off-site total body dose rate resulting from the release of noble gases from the plant stack during power operations.

Plant Data

a) Maximum plant stack gas monitor (I or II) Count rate during period of interest (M): 80,000 cpm

b) Stack flow rate during release (F): 8.26E+07 cc/sec

(175,000 cfm x 4.72E+02 $\frac{cc/sec}{cfm}$ =)

c) Plant stack monitor detector counting efficiency (Sg): 1E+08 cpm per µCi/cc

d) The last measured release rate mix of $\dot{Q}_i^{SJAE}$ noble gas from the SJAE and corresponding dose factor DFB_i from Table 1.1-10.

i	$\dot{Q}^{SJAE}$ (µCi/sec)	DFB_i (mrem-m^3/pCi-yr)
Xe-138	5.15E+03	8.83E-03
Kr-87	2.37E+02	5.92E-03
Kr-88	1.29E+02	1.47E-02
Xe-135	1.85E+02	1.81E-03

Calculation

The dose rate is calculated from Equations (3-5) and (3-28):

$$\dot{R}_{tbs} = 0.61 \sum_i \dot{Q}_i^{ST} \quad DFB_i \qquad (3\text{-}5)$$

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

and where the stack release rate is determined from:

$$\dot{Q}_i^{ST} = \frac{\dot{Q}_i^{SJAE}}{\sum_i \dot{Q}_i^{SJAE}} \quad M \quad \frac{1}{Sg} \quad F \qquad (3\text{-}28)$$

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

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

First, determine the sum ($\sum_i$) of all $\dot{Q}_i^{SJAE}$ and the fraction that each noble gas i represents in the total gas mix.

$$\sum_i \dot{Q}_i^{SJAE} = (5.15E+03) + (2.37E+02) + (1.29E+02) + (1.85E+02)$$

$$= 5.70E+03 \ \mu Ci/sec$$

and the relative fraction of each noble gas:

EXAMPLE CALCULATION NO. 2
(Continued)

i	$\dot{Q}_i^{SJAE}$ /5.70E+03		f_i Relative Fraction of Total
Xe-138	5.15E+03/5.70E+03	=	0.904
Kr-87	2.37E+02/5.70E+03	=	0.042
Kr-88	1.29E+02/5.70E+03	=	0.023
Xe-135	1.85E+02/5.70E+03	=	0.032

Next, the stack release rate of each noble gas i from Equation (3-28) can be substituted into Equation (3-5) to give the dose rate as:

$$\dot{R}_{tbs} = 0.61 \; M \; \frac{1}{Sg} \; F \; \sum_i \; f_i \; DFB_i$$

$$= 0.61 \quad 80,000 \quad 1/1E+08 \quad 8.26E+07 \; \sum_i \; f_i \; DFB_i$$

$$= 4.03E+04 \; [(0.904)(8.83E-03) + (0.042)(5.92E-03) +$$

$$(0.023)(1.47E-02) + (0.032)(1.81E-03)] =$$

Answer

$\dot{R}_{tbs}$ = 348 mrem/year noble gas total body dose rate.

EXAMPLE CALCULATION NO. 3

Type

Total Body Dose Rate From Noble Gases

References

a) ODCM Section 3.4 (Method I).

b) Technical Specification 3.8.E.1.a.

Problem

Calculate the off-site total body dose rate resulting from the release of noble gases from the plant stack recorded to have occurred 32 days after plant shutdown.

Plant Data

a) Maximum plant stack gas monitor (I or II) Count rate during period of interest (M): 80,000 cpm

b) Stack flow rate during release (F): 8.26E+07 cc/sec

(175,000 cfm x 4.72E+02 $\frac{cc/sec}{cfm}$ =)

c) Plant stack monitor detector counting efficiency (Sg): 1E+08 cpm per µCi/cc

d) The noble gas mix fractions $f_i(t)$ corresponding to 32 days taken from Table 5.2-1.

i	f_i (32 day)*	DFB_i**
Kr-85	0.152	1.61E-05
Xe-131m	0.070	9.15E-05
Xe-133	0.777	2.94E-04

---

* Fraction of nuclide in mix as function of time (see Table 5.2-1).

** Dose factors from Table 1.1-10.

Calculation

The dose rate is calculated from Equations (3-5) and (3-28):

$$\dot{R}_{tbs} = 0.61 \sum_i \dot{Q}_i^{ST} \quad DFB_i \qquad (3\text{-}5)$$

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

and where the stack release rate is determined from:

$$\dot{Q}_i^{ST} = \frac{\dot{Q}_i^{SJAE}}{\sum_i \dot{Q}_i^{SJAE}} \quad M \quad \frac{1}{Sg} \quad F \qquad (3\text{-}28)$$

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

However, for a time (t) after shutdown, the ratio of $\dot{Q}_i^{SJAE}$ to the sum release rate of all noble gases can be replaced in Equation (3-28) by the relative fraction [$f_i(t)$] of each noble gas available in the system; therefore, Equation (3-28) can be written:

$$\dot{Q}_i^{ST} = f_i(t) \quad M \quad \frac{1}{Sg} \quad F$$

Therefore, using the above data for a time period 32 days after shutdown, the dose rate equation can also be written as:

$$\dot{R}_{tbs} = 0.61 \quad M \quad \frac{1}{Sg} \quad F \sum_i f_i(t) \quad DFB_i$$

$$= 0.61 \quad 80{,}000 \quad 1/1E{+}08 \quad 8.26E{+}07 \; [(0.152)(1.61E{-}05) +$$

$$(0.070)(9.15E{-}05) + (0.777)(2.94E{-}04)] =$$

Answer

$\dot{R}_{tbs}$ = 9.6 mrem/year noble gas total body dose rates at 32 days after shutdown.

EXAMPLE CALCULATION NO. 4

Type

Skin Dose Rate From Noble Gases

References

a) ODCM Section 3.5 (Method I).

b) Technical Specification 3.8.E.1.a.

Problem

Calculate the off-site skin dose rate resulting from the release of noble gases from the plant stack during power operations.

Plant Data

a) Maximum plant stack gas monitor (I or II) Count rate during period of interest (M): 80,000 cpm

b) Stack flow rate during release (F): 8.26E+07 cc/sec

(175,000 cfm x 4.72E+02 $\frac{cc/sec}{cfm}$ =)

c) Plant stack monitor detector counting efficiency (Sg): 1E+08 cpm per µCi/cc

d) The last measured release rate mix of noble gas from the SJAE ($\dot{Q}_i^{SJAE}$), and corresponding dose factor DF_i' from Table 1.1-10.

i	$\dot{Q}_i^{SJAE}$ (µCi/sec)	DF'_{is} (mrem-sec/µCi-yr)
Xe-138	5.15E+03	1.06E-02
Kr-87	2.37E+02	1.43E-02
Kr-88	1.29E+02	1.28E-02
Xe-135	1.85E+02	3.24E-03

Calculation

The skin dose rate is calculated from Equations (3-7) and (3-28):

$$\dot{R}_{skins} = \sum_i \dot{Q}_i^{ST} \quad DF'_{is} \qquad (3\text{-}7)$$

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

and where the stack release rate is determined from:

$$\dot{Q}_i^{ST} = \frac{\dot{Q}_i^{SJAE}}{\sum_i \dot{Q}_i^{SJAE}} \quad M \quad \frac{1}{Sg} \quad F \qquad (3\text{-}28)$$

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

First, determine the sum ($\sum_i$) of all $\dot{Q}_i^{SJAE}$ and the fraction that each noble gas i represents in the total gas mix.

$$\sum_i \dot{Q}_i^{SJAE} = (5.15E{+}03) + (2.37E{+}02) + (1.29E{+}02) + (1.85E{+}02)$$

$$= 5.70E{+}03\ \mu Ci/sec$$

and the relative fraction of each noble gas:

i	$\dot{Q}_i^{SJAE}/5.70E{+}03$		f_i Relative Fraction of Total
Xe-138	5.15E+03/5.70E+03	=	0.904
Kr-87	2.37E+02/5.70E+03	=	0.042
Kr-88	1.29E+02/5.70E+03	=	0.023
Xe-135	1.85E+02/5.70E+03	=	0.032

Next, the stack release rate of each noble gas i from Equation (3-28) can be substituted into Equation (3-5) to give the skin dose rate as:

$$\dot{R}_{skins} = M \frac{1}{Sg} F \sum_i f_i DF'_{is}$$

$$= 80,000 \quad 1/1E+08 \quad 8.26E+07 \,[(0.904)(1.06E-02) +$$

$$(0.042)(1.43E-02) + (0.023)(1.28E-02) + (0.032)(3.24E-03)] =$$

$$= 6.61E+04 \,(9.58E-03 + 6.01E-04 + 2.94E-04 + 1.04E-04)$$

$$= 6.61E+04 \,(1.06E-02)$$

Answer

$\dot{R}_{skins}$ = 699 mrem/year noble gas skin dose rate.

EXAMPLE CALCULATION NO. 5

Type

Skin Dose Rate From Noble Gases

References

a) ODCM Section 3.5 (Method I).

b) Technical Specification 3.8.E.1.a.

Problem

Calculate the off-site skin dose rate resulting from the release of noble gases from the plant stack six days after plant shutdown.

Plant Data

a) Maximum plant stack gas monitor (I or II) Count rate during period of interest (M): 120,000 cpm

b) Stack flow rate during release (F): 8.26E+07 cc/sec

(175,000 cfm x 4.72E+02 $\frac{cc/sec}{cfm}$ =)

c) Plant stack monitor detector counting efficiency (Sg): 1[illegible]08 cpm per µCi/cc

d) Since the plant is shut down for more than five days, Xe-133 may be used as the referenced radionuclide in place of the ratio of $\dot{Q}_i^{SJAE}$ to the sum of all $\dot{Q}_i^{SJAE}$ in Equation (3-28).

i	f_i (t >5 days)	DF'_{is} $\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}$
Xe-133	1.	5.58E-04

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Calculation

The skin dose rate is calculated from Equations (3-7) and (3-28):

$$\dot{R}_{skins} = \sum_i \dot{Q}_i^{ST} \quad DF'_{is} \qquad (3\text{-}7)$$

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

and, the stack release rate is determined from:

$$\dot{Q}_i^{ST} = \frac{\dot{Q}_i^{SJAE}}{\sum_i \dot{Q}_i^{SJAE}} \quad M \quad \frac{1}{Sg} \quad F \qquad (3\text{-}28)$$

However, for times greater than five days after shutdown, Xe-133 may be used as the referenced radionuclide alone. Therefore, in Equation (3-28) the ratio of $\dot{Q}_i^{SJAE}$ to the sum of all $\dot{Q}_i^{SJAE}$ can be replaced by a value of 1 which indicates that all tne contribution to the release is from Xe-133.

Therefore:

$$\dot{Q}_{Xe\text{-}133}^{ST} = 1.0 \times 120{,}000 \times 1/1E{+}08 \times 8.26E{+}07$$

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

$$\dot{Q}_{Xe\text{-}133}^{ST} = 99{,}120\ \mu Ci/sec$$

Therefore, replacing this value of $\dot{Q}_i^{ST}$ into Equation (3-7) we find the skin dose rate as:

$$\dot{R}_{skins} = 99{,}120 \times 5.58E{-}04$$

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

EXAMPLE CALCULATION NO. 5
(Continued)

Answer

R_{skins} = 55.3 mrem/year.

EXAMPLE PROBLEM NO. 6

Type

Critical Organ Dose Rate From Iodine, Tritium, and Particulates

References

a) ODCM Section 3.6 (Method I).

b) Technical Specification 3.8.E.1.b.

Problem

Calculate the critical organ dose rate due to measured effluent data taken from the plant stack for a seven-day sample collection period.

Plant Data

a) Stack particulate analysis for the seven-day period of interest.

i	Activity $\dot{Q}_i^{STP}$ (μCi-sec)	DFG'_{sico} [1] ($\frac{mrem\text{-}sec}{yr\text{-}\mu Ci}$)
Sr-89*	1.42E-04*	3.60E+02
Sr-90*	3.50E-03*	1.36E+04
Co-60	4.89E-02	3.41E+02
Cs-137	3.90E-03	5.55E+02
Zn-65	1.01E-02	1.20E+02
Na-24**	2.76E-03**	--
Mn-54+	<2.87E-06+	2.77E+01

Notes

(1) DFG'_{sico} dose rate factor for each radionuclide is taken from Table 1.1-12.

* For Sr-89/90, use the most recent available measurement from quarters composite analysis.

** Na-24 has a half life of less than 8-1/2 days, and therefore is not included in the dose analysis per requirements of Technical Specification 3.8.E.1.b even though it was detected.

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+ Mn-54 is not included in the dose analysis since it was not detected as being present after counting to at least the LLD.

b) Stack iodine (charcoal and particulate activities combined for the seven-day period of interest):

i	Activity $\hat{Q}_i^{STP}$ (μCi-sec)	DFG'_{sico} (1) ($\frac{mrem\text{-}sec}{yr\text{-}\mu Ci}$)
I-131	1.16E-03	2.43E+03
I-133*	<6.35E-05*	2.59E+01
I-135**	7.21E-03**	--
and		
H-3+	3.17E-02	9.87E-03

Notes

* I-133 is not included in the dose analysis for this case since it was not detected is being present in the stack analysis.

** I-135 is not included in the dose analysis because it has a half life less than 8-1/2 for particulates, and is not included as a required iodine in Technical Specification 3.8.E.1.b.

+ Tritium value based as latest available stack grab sample.

Calculation

The dose rate is calculated from Equation (3-16):

$$\hat{R}_{cos} = \sum_i \hat{Q}_i^{STP} \; DFG'_{sico} \qquad (3\text{-}16)$$

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

The dose rate factors (DFG'_{sico}) for each of the radionuclides detected in the plant stack charcoal and particulate filter sample (plus tritium) is taken from Table 1.1-12 of the ODCM.

EXAMPLE PROBLEM NO. 6
(Continued)

Therefore:

$$\dot{R}_{cos} = (1.42E\text{-}04)(3.60E\text{+}02) + (3.50E\text{-}03)(1.36E\text{+}04) + (4.89E\text{-}02)(3.41E\text{+}02) + (3.90E\text{-}03)(5.55E\text{+}02) + (1.01E\text{-}02)(1.20E\text{+}02) + (1.16E\text{-}03)(2.43E\text{+}03) + (3.17E\text{-}02)(9.87E\text{-}03) =$$

Answer

$\dot{R}_{cos}$ = 70.5 mrem/year critical organ dose rate from iodine, tritium, and particulate.

EXAMPLE PROBLEM NO. 7

Type

Gamma Air Dose From Noble Gases released from stack

References

a) ODCM Section 3.7 (Method I).

b) Technical Specification 3.8.F.1.

Problem

Calculate the maximum gamma air dose resulting from noble gases released from the plant stack over a calendar month.

Plant Data

Based on the daily off-gas analysis, the total activity released during the month of interest is:

i	Activity Q_i^{ST} (Ci)	$DF_i^{\gamma *}$ (mrad-m^3 pCi-yr)
Kr-88	3.55E-01	1.52E-02
Kr-85m	4.71E+00	1.23E-03
Xe-138	2.75E+00	9.21E-03
Xe-135	3.51E+01	1.92E-03
Xe-133	9.42E+01	3.53E-04

*Gamma air dose factors taken from Table 1.1-10.

EXAMPLE PROBLEM NO. 7
(Continued)

Calculation

The maximum gamma air dose off-site is calculated from Equation (3-21):

$$D^{\gamma}_{airs} = 0.019 \sum_{i} Q_i^{ST} \; DF_i^{\gamma} \qquad (3\text{-}21)$$

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

Therefore:

$$D^{\gamma}_{airs} = 0.019 \; [(3.55E\text{-}01)(1.52E\text{-}02) + (4.71E+00)(1.23E\text{-}03) + (2.75E+00)(9.21E\text{-}03) + (3.51E+01)(1.92E\text{-}03) + (9.42E+01)(3.52E\text{-}04)]$$

$$= 0.019 \; (5.40E\text{-}03 + 5.79E\text{-}03 + 2.53E\text{-}02 + 6.74E\text{-}02 + 3.31E\text{-}02)$$

Answer

D^{γ}_{airs} = 2.60E-03 mrad gamma air dose during the month.

EXAMPLE CALCULATION NO. 8

Type

Beta Air Dose From Noble Gases released from stack

References

a) ODCM Section 3.8 (Method I).

b) Technical Specification 3.8.F.1.

Problem

Calculate the maximum beta air dose resulting from the same noble gas releases given in Example Calculation No. 7.

Plant Data

From Example No. 7, the total activity determined to be released during the month is:

i	Activity Q_i^{ST} (Ci)	$DF_i^{\beta *}$ $\frac{(mrad\text{-}m^3}{pCi\text{-}yr)}$
Kr-88	3.55E-01	2.93E-03
Kr-85m	4.71E+00	1.97E-03
Xe-138	2.75E+00	4.75E-03
Xe-135	3.51E+01	2.46E-03
Xe-133	9.42E+01	1.05E-03

*Beta air dose factors taken from Table 1.1-10.

Calculation

The maximum beta air dose off-site is calculated from Equation (3-23):

$$D^{\beta}_{airs} = 0.033 \sum_{i} Q_i^{ST} \; DF_i^{\beta} \qquad (3\text{-}23)$$

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

Therefore:

$$D^{\beta}_{airs} = 0.033\,[(3.55E\text{-}01)(2.93E\text{-}03) + (4.71E+00)(1.97E\text{-}03) + (2.75E+00)(4.75E\text{-}03) + (3.51E+01)(2.46E\text{-}03) + (9.42E+01)(1.05E\text{-}03)]$$

$$= 0.033\,(1.04E\text{-}03 + 9.28E\text{-}03 + 1.31E\text{-}02 + 8.63E\text{-}02 + 9.89E\text{-}02)$$

Answer

D^{β}_{airs} = 6.88E-03 mrad beta air dose during the month.

EXAMPLE PROBLEM NO. 9

Type

Critical Organ Dose From Iodine, Tritium, and Particulates

References

a) ODCM Section 3.9 (Method I).

b) Technical Specification 3.8.G.1.

Problem

Calculate the critical organ dose due to the total activity recorded as being released from the plant stack during a calendar month.

Plant Data

a) From the combined stack analyses during the month, the following activity released is:

i	Q_i^{STP} (Ci)	DFG_{sico} (1) ($\frac{mrem}{Ci}$)
Sr-89*	5.42E-04*	1.14E+01
Sr-90*	1.10E-02*	4.31E+02
Co-60	2.30E-01	7.63E+00
Cs-137	1.15E-02	1.63E+01
Zn-65	2.60E-02	3.71E+00
Na-24**	7.11E-03**	--
Mn-54	<2.76E+06+	7.01E-01

Notes for Plant Data a) Above

(1) Critical organ dose factor taken from Table 1.1-12.

* For Sr-89/90, use the most recent available measurement from the quarterly composite analysis.

** Na-24 has a half life of less than 8-1/2 days, and therefore is not included in accordance with Technical Specification 3.8.G.1.

+ Mn-54 is not included in the dose analysis since it was not detected as being present after counting to at least the LLD.

b) Total iodine release for the month based on the combined charcoal and particulate filter samples taken during the month:

i	Q_i^{STP} (Ci)	DFG_{sico} ($\frac{mrem}{Ci}$)
I-131	4.30E-03	7.71E+01
I-133*	1.12E-04*	8.21E-01
I-135**	2.01E-02**	
and		
H-3+	0.15	3.13E-04

Calculation

The dose is calculated from Equation (3-25):

$$D_{cos} = \sum_i Q_i^{STP} \; DFG_{sico} \qquad (3\text{-}25)$$

(mrem) (Ci) (mrem/Ci)

Notes for Plant Data b) Above

* In this case, I-133 was found in one of the weekly stack samples to be present, and therefore based on that value is included in the dose analysis.

** I-135 is not included in the dose analysis because it has a half life less than 8-1/2 days for particulates and is not included as a required iodine in Technical Specification 3.8.8.1.

+ Tritium value based on the monthly stack grab sample.

EXAMPLE CALCULATION NO. 10

Type

Releases Limited to 20 times the effluent concentration limits of 10CFR20.1001-20.2401, Table 2, Column 1

References

a) 10CFR50.72

b) 10CFR50.73

Problem

Find the minimum stack gas monitor response which would require an assessment to determine if a four-hour notification to NRC is required per 10CFR50.72.

Assumptions

a)	Maximum expected stack flow rate (F) (175,000 cfm x 4.72E+02 $\frac{cc/sec}{cfm}$ ≈)	8.26E+07 cc/sec
b)	Plant stack monitor detector counting efficiency (Sg)	1E+08 cpm per µCi/cc
c)	Maximum off-site ground level dispersion parameter (X/Q undepleted) (from ODCM Table 3.10-1)	1.04-06 sec/m^3
d)	Most restrictive ECL value for noble gases (10CFR20, Appendix B, Table 2, Column 1) for Kr-88)	9E-09 µCi/cc

Calculation (Part I)

The setpoint R_{spt} for the stack monitor which would correspond to an instantaneous off-site air concentration of 20 x ELC for the most restrictive noble gas can be calculated by:

$$R_{spt} = 20\ ECL \quad Sg \quad \frac{1}{X/Q} \quad 1E+06 \quad \frac{1}{F}$$

$$(cpm)\ \left(\frac{\mu Ci}{cc}\right)\ \left(\frac{cpm}{\mu Ci/cc}\right)\ \left(\frac{m^3}{sec}\right)\ \left(\frac{cc}{m^3}\right)\ \left(\frac{sec}{cc}\right)$$

$$= 20 \times (9E-09)(1E+08)(1/1.04E-06)(1E+06)(1/8.26E+07)$$

R_{spt} = 209,536 cpm setpoint alarm value

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APPENDIX I

RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS

Requirement: Technical Specification 6.14.A requires that licensee initiated major changes to the radioactive waste systems (liquid, gaseous, and solid) be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Plant Operation Review Committee.

Response: There were no licensee initiated major changes to the radioactive waste systems (liquid, gaseous, and solid) during this reporting period.

APPENDIX J

ON-SITE DISPOSAL OF SEPTIC WASTE

Requirement: Off-Site Dose Calculational Manual, Appendix B requires that the dose impact due to on-site disposal of septic waste during the reporting year and from previous years be reported to the Commission in the Semiannual Radioactive Effluent Report filed after January 1, if disposals occur during the reporting year.

Response: No response is required for this reporting period.