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10 CFR 50.36a

BVY 22-014

April 28, 2022

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: 2021 Radiological Effluent Release Report
Vermont Yankee Nuclear Power Station
Docket No. 50-271
License No. DPR-28

Dear Sir or Madam:

In accordance with 10 CFR 50.36a and the Vermont Yankee (VY) Off-site Dose Calculation Manual, please find in Enclosure 1 a copy of the Annual Radiological Effluent Release Report for 2021.

In addition, the VY Off-site Dose Calculation Manual (ODCM) requires that changes made during the reporting period to the ODCM be identified in the next Radiological Effluent Release Report. Accordingly, Enclosure 2 provides Revision 41 of the ODCM that occurred during 2021. The changes associated with Revision 41 to the ODCM are discussed within the ODCM.

The VY Renewed Facility Operating License Condition 3.E.10 requires that similar information to that contained within the subject report be provided to the Massachusetts Metropolitan District Commission (MDC). However, since the MDC is currently part of the Massachusetts Department of Conservation and Recreation (DCR), this report is being provided to the DCR to satisfy License Condition 3.E.10.

This letter contains no new regulatory commitments.

Should you have any questions concerning this letter, or require additional information, please contact Mr. Thomas B. Silko at (802) 451-5354, Ext 2506.

Sincerely,

CRD/tbs

Enclosure 1: Annual Radiological Effluent Release Report for 2020.
Enclosure 2: Off-site Dose Calculation Manual (ODCM), Revision 41

cc: Mr. David C. Lew
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Enclosure 1

Vermont Yankee Nuclear Power Station

Annual Radiological Effluent Release Report for 2021
(52 pages excluding this cover sheet)

Vermont Yankee Nuclear Power Station

Annual Radiological Effluent Release Report

January - December 2021



NorthStar – Vermont Yankee.
Vermont Yankee Nuclear Power Station
320 Governor Hunt Road
Vernon, Vermont 05354

Docket No. 50-271

License No. DPR-28

**VERMONT YANKEE
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
FOR 2021**

Reviewed by:  / 4/27/2022
Scott Dorval, RP Manager ISFSI Date

Approved for Distribution:  / 4/27/2022
Mike Fletcher, RP Manager Plant Date

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1.0 INTRODUCTION

Tables 1 through 3 list the recorded radioactive liquid and gaseous effluents and solid waste shipments for 2021, with data summarized on a quarterly basis for both liquids and gases. Table 4A summarizes the estimated radiological dose commitments from all radioactive liquid and gaseous effluents released during 2021, to the maximally exposed individual member of the public, in response to the ALARA objectives of 10 CFR Part 50, Appendix I. Table 4B presents the estimate of direct dose from fixed station sources along the limiting west site boundary line. Tables 5A through 6G present the cumulative joint frequency distributions of wind speed, wind direction, and atmospheric stability for the 5-year period, 2012 through 2016. Radioactive effluents reported in Tables 1 and 2 were used to determine the dose to the maximum exposed individual member of the public for 2021.

Dose commitments resulting from the release of radioactive materials in liquids and gases during the reporting period were estimated in accordance with the plant's Off-Site Dose Calculation Manual (ODCM), Section 10.1 (Reference 1). These dose estimates were made using a "Method I" analysis as described in the ODCM, and as reported in Tables 4A and 4B of this report. A conservative "Method I" analysis incorporates the methodology of Regulatory Guide 1.109 (Reference 2) and uses nuclide specific dose factors. Dose factors are the dose per Ci released for the age group and organ receiving the highest dose. This method is conservative since the age and organ receiving the highest dose differs from one nuclide to another.

As required by ODCM Section 10.1, this report shall also include an assessment of the radiation doses from radioactive effluents to member(s) of the public due to allowed recreational activities inside the site boundary during the year. As discussed in Section 3.6, there were no such recreational activities permitted and, therefore, there is no associated dose assessment.

An assessment of radiation doses (including direct radiation) to the likely most exposed real member(s) of the public for the calendar year for the purposes of demonstrating conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," is also required to be included in this report if the conditions indicated in ODCM 3/4.4, "Total Dose," have been exceeded during the year. Since the conditions indicated in the action statement under ODCM 3/4.4 were not entered into during the year, no additional radiation dose assessment is required. However, Table 4B does provide the combination of off-site doses and dose commitments from plant effluents and direct radiation sources for the limiting member of the public as a demonstration of compliance with the dose standards of 40 CFR Part 190.

All calculated dose estimates for members of the public at the site boundary or beyond for the 2021 annual reporting period are below the dose criteria of 10 CFR Part 50, Appendix I, and 40 CFR Part 190.

Appendices B through J indicate the status of reportable items per the requirements of ODCM Section 10.1.

2.0 METEOROLOGICAL DATA

The scope of possible accidents is significantly reduced with all spent fuel in dry cask storage. The potential for any off-site consequences from radiological accidents are also substantially reduced. No reasonably conceivable beyond design basis event can result in a radiological release that exceeds Environmental Protection Agency (EPA) Protective Actions Guidelines. During the final decommissioning and dismantling phases, administrative controls over radiological source accumulation will preempt any significant radiological release to the environment. As a result, there is no need for continued on-site meteorological data accumulation or real time dose assessment. Conservative atmospheric dispersion factors have been developed from the 2012 through 2016 joint frequency data to provide both short term dose assessment and for assessment in the annual average dose from facility routine releases. There are no planned or existing Emergency Action Levels in the defueled state that could result in the need for real time accident release assessment with other than previously determined conservative atmospheric dispersion factors.

Actual measured meteorological data for the five-year period, 2012 through 2016, were analyzed to determine all the values and locations of the maximum off-site long-term average atmospheric dispersion and deposition factors. The highest offsite dispersion and deposition factors at any location beyond the site boundary, regardless of whether or not it was an actual location of a residence or food production, was used to calculate "Method I" dose factors for each nuclide. Each dose and dose rate calculation presented in the current Revision 41 of the ODCM incorporate the maximum applicable off-site long-term average atmospheric dispersion and deposition factors, and maximum organ dose to any age group from each nuclide.

Updated five-year average dispersion factors and deposition factors developed from 2012 through 2016 on-site meteorological hourly data and the nuclide specific dose factors can be used to assess either routine releases or estimate conservative off-site consequences from any on-site radiological event. As such, there is no need to continue to collect on-site meteorological data for either accessing routine releases or potential emergency events. Collection of data from on-site meteorological tower was terminated in November 2018.

The five-year aggregate joint frequency distribution tables for ground level releases are presented in Tables 5A through 5G. The five-year aggregate joint frequency distribution tables for elevated releases are presented in Tables 6A through 6G.

3.0 DOSE ASSESSMENT

3.1 Doses from Liquid Effluents

ODCM 3/4.2.2 limits total body doses (1.5 mrem per quarter, and 3 mrem per year) and organ doses (5 mrem per quarter, and 10 mrem per year) from liquid effluents to a member of the public to those specified in 10 CFR Part 50, Appendix I. By implementing the requirements of 10 CFR Part 50, Appendix I, ODCM 3/4.2.2 assures that the release of radioactive material in liquid effluents will be kept "as low as is reasonably achievable."

There were no continuous or batch routine liquid radioactive waste discharges during 2021.

Dose estimates of tritium-contaminated groundwater released from the site are based on Protected Area Boundary monitoring well data collected throughout 2021 and hydrological modeling of groundwater movement. The groundwater discharge rates from the developed portion of the site to the river are estimated using a streamtube approach based on Darcy's Law. Using a conservative estimate of groundwater flow through the affected area toward the river on a quarterly basis, an estimate of the total potential tritium released from the site during each quarter of 2021 was generated and reported in Table 2A. The quantity of tritium released from the site by groundwater in each quarter was then converted to dose by using the calculated dose conversion factors presented in ODCM Table 1.1.11. The resulting quarterly doses are presented in Table 4A.

ODCM Control 3.2.1 states, in part, that groundwater flowing to the Connecticut River from the site in radioactive concentrations above background (Unrestricted Areas for liquids is at the point of discharge from the plant discharge in Connecticut River) shall be limited to 10 times the concentrations specified in Appendix B to 10CFR Part 20.1001 – 20.2402, Table 2, Column 2. The tritium concentrations at the point of discharge from the plant ranged from 0.07% to 0.08% of the Table 2, Column 2 values.

3.2 Doses from Noble Gases

The dose limits specified in ODCM 3/4.3.2 have been deleted from Revision 40 of the ODCM. Noble gases were not produced or detected in releases from the plant stack or Reactor Building Exhaust in 2021.

3.3 Dose from Radionuclides in Particulate Form and Tritium

ODCM 3/4.3.3 limits the organ dose to a member of the public from tritium and radionuclides in particulate form in gaseous effluents released from the site to areas at and beyond the site boundary to those specified in 10 CFR Part 50, Appendix I (7.5 mrem per quarter and 15 mrem per year). By implementing the requirements of 10 CFR Part 50, Appendix I, ODCM 3/4.3.3 assures that the releases of any tritium and particulates in gaseous effluents will be kept "as low as is reasonably achievable."

There were no non-routine gaseous releases or batch releases in 2021. There was no radioactively contaminated used oil burned in 2021.

Continuous sampling of the plant stack or reactor building exhaust for tritium, per ODCM Table 4.3.1, was performed by using silica gel cartridges in 2021. The cartridges were analyzed monthly. Based upon the stack/reactor building exhaust flow rate and sample flow rates, the average release rate in $\mu\text{Ci/sec}$ and total release in Ci for each quarter was calculated. The quantity of tritium released from the site by the plant stack or reactor building exhaust in each quarter was then converted to dose by using the calculated dose conversion factors presented in ODCM Table 1.1.12. The resulting quarterly doses from H-3 were combined with the particulate releases and are presented in Table 4A.

Continuous sampling of the plant stack or reactor building exhaust for particulates, per ODCM Table 4.3.1, was performed in 2021. These samples are analyzed weekly for principle gamma emitters. Cobalt-60 was identified on the reactor building ventilation exhaust particulate filters in the fourth quarter in 2021. Cesium-137 was also identified on the reactor building ventilation exhaust particulate filters in the fourth quarter in 2021. The quantity of Cobalt 60 and Cs-137 did not exceed any dose limits. The doses from Co-60 and Cs-137 were combined with H-3 and are presented in Table 4A.

3.4 Whole Body Doses in Unrestricted Areas from Direct Radiation

As opposed to prior years before the permanent shut down when the majority of the dose in the unrestricted area consisted of direct and skyshine radiation from N-16 decay in the Turbine Building steam cycle during power operations, there was no such source during 2021 due to the elimination of its production and its short half-life.

The other fixed sources of direct and scatter radiation to the site boundary are the Independent Spent Fuel Storage Installation (ISFSI) and old turbine rotors and casings in the Turbine Storage Facility. The annual direct radiation dose at the site boundary is driven by the cask loading at the ISFSI. All spent fuel has been transferred to the two ISFSI pads since August 2018. The method to assess the direct radiation dose in unrestricted areas has been agreed upon with the State of Vermont. Site boundary TLDs are changed out monthly. The net dose at the location of TLD DR-53A has been chosen to assess direct radiation dose to unrestricted areas.

Table 4B lists the combination of the direct radiation dose at the limiting site boundary location and the maximum offsite dose from gaseous and liquid effluents for the purpose of demonstrating compliance with the dose standards contained in 40 CFR Part 190. For 2021, this annual dose was below the 25 mrem total body and organ limit, as well as the 75 mrem thyroid limit, of 40 CFR Part 190.

3.5 Doses from On-Site Disposal of Septic Waste, Cooling Tower Silt and Soil

ODCM Appendices B, F, and I require that all septic waste, cooling tower silt, and sand/soil applied within the approved designated disposal areas be controlled to ensure the dose to a maximally exposed individual during the period of Vermont Yankee site control is limited to less than 1 mrem/year to the whole body and any organ. After the period associated with Vermont Yankee operational control, the dose to the inadvertent intruder is to be limited to 5 mrem/year. The projected dose from on-site disposals of septic waste, cooling tower silt, and sand/soil mixes is given in Appendix J of this report. During 2021 there was no septic sludge spread on the southern on-site disposal field.

3.6 On-Site Recreational Activities

During 2021, no access to the on-site boat launching ramp located north of the intake structure was permitted for employees, their families, and guests. As such, there was no associated dose impact to members of the public due to any recreational activities on-site.

REFERENCES

1. Off-Site Dose Calculation Manual (ODCM), Revision 41, Entergy Nuclear Vermont Yankee, LLC, dated July 1, 2021.
2. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," U. S. Nuclear Regulatory Commission, Office of Standards Development, Revision 1, October 1977.
3. Safety Guide 1.23, "Onsite Meteorological Programs," U.S. Atomic Energy Commission, February 17, 1972.
4. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, Office of Standards Development, March 1976.
5. Meteorology and Atomic Energy, 1968, Section 5-3.2.2, "Cloud Depletion," page 204, U. S. Atomic Energy Commission, July 1968.
6. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Revision 2, June 2009.

Table 1A - Gaseous Effluents -Summation of All Releases

	Units	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Est. Total Error, %
A. Fission and Activation Gases						
1. Total release	Ci	ND	ND	ND	ND	
2. Average release rate for period	μCi/sec	ND	ND	ND	ND	
3. Percent of ODCM limit (1)	%	NA	NA	NA	NA	
B. Iodines						
1. Total Iodine	Ci	ND	ND	ND	ND	
2. Average release rate for period	μCi/sec	ND	ND	ND	ND	
3. Percent of ODCM limit (1)	%	NA	NA	NA	NA	
C. Particulates						
1. Particulates with T-1/2>8 days	Ci	ND	ND	ND	2.18E-06	1.80E+01
2. Average release rate for period	μCi/sec	ND	ND	ND	2.80E-07	
3. Percent of ODCM limit (2)	%	(2)	(2)	(2)	(2)	
4. Gross alpha radioactivity	Ci	ND	ND	ND	ND	
D. Tritium						
1. Total release	Ci	2.85E-02	2.11E-01	2.47E-01	2.42E-01	1.80E+01
2. Average release rate for period	μCi/sec	3.67E-03	2.70E-02	3.18E-02	3.11E-01	
3. Percent of ODCM limit (2)	%	1.33E-06	1.93E-04	2.27E-04	2.19E-04	
E. Carbon-14						
1. Total release	Ci	ND	ND	ND	ND	
2. Percent of ODCM limit (1)	%	NA	NA	NA	NA	

ND = Not Detected NA = Not Applicable

- (1) There is no ODCM Control for Iodines, C-14 and Fission and Activation Gases. All spent fuel is in casks on the ISFSI pads.
- (2) ODCM Control 3.3.3 for dose from Tritium and radionuclides in particulate form. The values in row D.3 are based upon the total particulate and Tritium activity in each quarter.

Table 1B - Gaseous Effluents -Elevated Releases

		Continuous Mode			
		Quarter			
Nuclides Released	Units	1	2	3	4
1. Fission Gases					
Krypton-85	Ci	ND	ND	ND	ND
Unidentified	Ci	ND	ND	ND	ND
Total for Period	Ci	ND	ND	ND	ND
2. Iodines	Ci	ND	ND	ND	ND
3. Particulates					
Strontium-90	Ci	ND	ND	ND	ND
Cesium-134	Ci	ND	ND	ND	ND
Cesium-137	Ci	ND	ND	ND	ND
Manganese-54	Ci	ND	ND	ND	ND
Cobalt-57	Ci	ND	ND	ND	ND
Cobalt-60	Ci	ND	ND	ND	ND
Zinc-65	Ci	ND	ND	ND	ND
Total for Period	Ci	ND	ND	ND	ND

ND Not Detected at the plant stack (The Plant Stack was no longer in service as of the last week in the first quarter of 2021.)

Table 1B - Gaseous Effluents -Elevated Releases
(Continued)

		Batch Mode			
		Quarter			
Nuclides Released	Units	1	2	3	4
1. Fission Gases					
Krypton-85	Ci				
Unidentified	Ci				
Total for Period	Ci	(1)	(1)	(1)	(1)
2. Iodines	Ci				
3. Particulates					
Strontium-90	Ci				
Cesium-134	Ci				
Cesium-137	Ci				
Manganese-54	Ci				
Cobalt-57	Ci				
Cobalt-60	Ci				
Zinc-65	Ci				
Total for Period	Ci	(1)	(1)	(1)	(1)

- (1) There were no batch mode gaseous releases for this reporting period. (The Plant Stack was no longer in service as of the last week in the first quarter of 2021.)

Table 1C – Gaseous Effluents Ground Level Releases

		Continuous Mode			
		Quarter			
Nuclides Released	Units	1	2	3	4
1. Fission Gases					
Krypton-85	Ci				
Unidentified	Ci				
Total for Period	Ci	(1)	(1)	(1)	(1)
2. Iodines	Ci				
3. Particulates					
Strontium-90	Ci	(1)			
Cesium-134	Ci	(1)			
Cesium-137	Ci	(1)			6.93E-07
Manganese-54	Ci	(1)			
Cobalt-57	Ci	(1)			
Cobalt-60	Ci	(1)			1.49E-06
Zinc-65	Ci	(1)			
Total for Period	Ci				2.18E-06

(1) There were no routine ground level gaseous releases for this reporting period. (the only release path that existed after the first quarter was a ground release path via the Reactor Building Exhaust pathway.)

Table 1C – Gaseous Effluents Ground Level Releases
(Continued)

		Batch Mode			
		Quarter			
Nuclides Released	Units	1	2	3	4
1. Fission Gases					
Krypton-85	Ci				
Unidentified	Ci				
Total for Period	Ci	(1)	(1)	(1)	(1)
2. Iodines	Ci				
3. Particulates					
Strontium-90	Ci				
Cesium-134	Ci				
Cesium-137	Ci				
Manganese-54	Ci				
Cobalt-57	Ci				
Cobalt-60	Ci				
Zinc-65	Ci				
Total for Period	Ci	(1)	(1)	(1)	(1)

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

Table 1D - Gaseous Effluents –Non-routine Releases

Nuclides Released	Units	Quarter			
		1	2	3	4
1. Fission Gases					
Krypton-85	Ci				
Unidentified	Ci				
Total for Period	Ci	(1)	(1)	(1)	(1)
2. Iodines	Ci				
3. Particulates					
Strontium-90	Ci				
Cesium-134	Ci				
Cesium-137	Ci				
Manganese-54	Ci				
Cobalt-57	Ci				
Cobalt-60	Ci				
Zinc-65	Ci				
Total for Period	Ci	(1)	(1)	(1)	(1)

(1) There were no non-routine gaseous releases for this reporting period.

Table 2A - Liquid Effluents - Summation of All Releases

There were continuous non-routine (groundwater flow to the Connecticut River) liquid releases during this reporting period. The data in this table is based upon monitoring well data collected throughout 2021 and hydrological modeling of groundwater movement.

Nuclides Released	Units	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Est. Total Error, %
A. Fission and Activation Products						
Total Release (not including tritium, gases, alpha)	Ci	ND	ND	ND	ND	
Average diluted concentration during period	μCi/ml	ND	ND	ND	ND	
Percent of applicable limit (2)	%	NA	NA	NA	NA	
B. Tritium						
Total Release	Ci	2.72E-03	2.55E-03	2.66E-03	2.93E-03	1.8E+01
Average diluted concentration during period	μCi/ml	7.41E-07	6.93E-07	7.23E-07	7.73E-07	
Percent of applicable limit (2)	%	3.74E-05	3.50E-05	3.65E-05	4.02E-05	
C. Dissolved and Entrained Gases						
Total Release	Ci	ND	ND	ND	ND	
Average diluted concentration during period	μCi/ml	ND	ND	ND	ND	
Percent of applicable limit (1)	%	NA	NA	NA	NA	
D. Gross Alpha Radioactivity						
Total release	Ci	ND	ND	ND	ND	
E. Volume of Waste Released (prior to dilution)	Liters	(3)	(3)	(3)	(3)	NA
F. Volume of Dilution Water Used During Period	Liters	3.67E+06	3.68E+06	3.68E+06	3.79E+06	(4)

ND = Not Detected NA = Not Applicable

- (1) All spent fuel is in casks on the ISFSI pads. Release of Dissolved and Entrained Gases is not possible.
- (2) The percent of limit is based on the ODCM Control 3.2.2 limiting dose (1.5 mrem/quarter to the total body) from liquid effluents and is related to the abnormal leakage of tritiated plant water into the underground environment.
- (3) Leakage of contaminated plant water to subsurface areas was stopped in February 2010. The release of contaminated groundwater to the Connecticut River is based on site boundary monitoring well data collected during 2021 and reported in Section B.
- (4) Dilution due to groundwater flow through the affected subsurface plume area toward the Connecticut River was estimated to be 7.38 gpm (times the number of days per quarter) during 2021. An estimated total error is not applicable.

Table 2B - Liquid Effluents - Routine Releases

		Continuous Mode			
		Quarter			
Nuclides Released	Units	1	2	3	4
Strontium-89	Ci				
Strontium-90	Ci				
Cesium-134	Ci				
Cesium-137	Ci				
Iodine-131	Ci				
Cobalt-58	Ci				
Cobalt-60	Ci				
Iron-59	Ci				
Zinc-65	Ci				
Manganese-54	Ci				
Chromium-51	Ci				
Zirconium-Niobium-95	Ci				
Molybdenum-99	Ci				
Technetium-99m	Ci				
Barium-Lanthanum-140	Ci				
Cerium-141	Ci				
Others-	Ci				
	Ci				
Unidentified	Ci				
Total for period (above)	Ci	(1)	(1)	(1)	(1)
Kr-85	Ci				
Xe-133	Ci				

(1) There were no continuous routine releases in this reporting period, only continuous non-routine liquid releases. See Table 2A.

Table 2B - Liquid Effluents - Routine Releases
(Continued)

		Batch Mode			
		Quarter			
Nuclides Released	Units	1	2	3	4
Strontium-89	Ci				
Strontium-90	Ci				
Cesium-134	Ci				
Cesium-137	Ci				
Iodine-131	Ci				
Cobalt-58	Ci				
Cobalt-60	Ci				
Iron-59	Ci				
Zinc-65	Ci				
Manganese-54	Ci				
Chromium-51	Ci				
Zirconium-Niobium-95	Ci				
Molybdenum-99	Ci				
Technetium-99m	Ci				
Barium-Lanthanum-140	Ci				
Cerium-141	Ci				
Others-	Ci				
	Ci				
Unidentified	Ci				
Total for period (above)	Ci	(1)	(1)	(1)	(1)
Kr-85	Ci				
Xe-133	Ci				

(1) There were no batch routine releases in this reporting period, only continuous non-routine liquid releases. See Table 2A.

Table 3 – Solid Waste and Irradiated Fuel Shipments

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

1. Type of Waste

Shipped from VY for Burial	Unit	Quarters 1 & 2	Est. Total Error %
a. Spent resins, filter sludges, etc.	m ³	8.01E+01	±25%
	Ci	9.06E+01	±25%
b. Dry Compressible Waste, equipment, etc.	m ³	2.99E+03	±25%
	Ci	9.24E-01	±25%
c. Irradiated components, control rods, etc.	m ³	1.07E+02	±25%
	Ci	4.24E+03	±25%
d. Other	m ³	1.32E+03	±25%
	Ci	3.95E-01	±25%

Shipped from Processor(s) for Burial	Unit	Quarters 1 & 2	Est. Total Error %
a. Spent resins, filter sludges, etc.	m ³	0	N/A
	Ci	0	N/A
b. Dry Compressible Waste, equipment, etc.	m ³	0	N/A
	Ci	0	N/A
c. Irradiated components, control rods, etc.	m ³	0	N/A
	Ci	0	N/A
d. Other	m ³	0	N/A
	Ci	0	N/A

2. Estimate of Major Nuclide Composition (By Type of Waste)

Spent resins, filter sludges		Dry Compressible Waste, equipment, etc.		Irradiated components, control rods, etc.		Other Waste	
Nuclide	Percent(1)	Nuclide	Percent(1)	Nuclide	Percent(1)	Nuclide	Percent(1)
Fe-55	43.76	H-3	0.29	Fe-55	39.53	H-3	90.96
Co-60	40.50	Mn-54	0.10	Co-60	47.72	Fe-55	0.18
Ni-63	15.58	Fe-55	26.55	Ni-63	12.65	Co-60	4.38
Cs-137	0.03	Co-60	64.89			Ni-63	0.24
		Ni-63	2.36			Cs-137	4.24
		Cs-137	5.54				

(1) Includes only those nuclides that are greater than 0.1% of the total activity

3. Disposition of Solid Waste Shipments (1st & 2nd Quarters)

No. of Shipments	From VY	From Processor	Mode	To Processor	To Burial
56	56	0	Rail	0	56
13	13	0	Truck	0	13

B. Irradiated Fuel Shipments (Disposition): None

C. Additional Data (1st & 2nd Quarters)

Supplemental Information	VY to Processor	VY to Burial	Processors to Burial
Class of Solid Waste Shipped	N/A	A/B/C	N/A
Type of Containers Used	N/A	GDC/IP-1/IP-2/Type B	N/A
Solidification Agent or Absorbent Used	N/A	None	N/A

GDC = General Design Container

Table 3 – Solid Waste and Irradiated Fuel Shipments (Continued)

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

1. Type of Waste

Shipped from VY for Burial	Unit	Quarters 3 & 4	Est. Total Error %
a.Spent resins, filter sludges, etc.	m ³	6.59E+01	±25%
	Ci	1.20E+03	±25%
b.Dry Compressible Waste, equipment, etc.	m ³	2.35E+03	±25%
	Ci	4.78E+00	±25%
c. Irradiated components, control rods, etc.	m ³	2.96E+01	±25%
	Ci	2.17E+04	±25%
d.Other	m ³	4.40E+03	±25%
	Ci	4.45E+01	±25%

Shipped from Processor(s) for Burial	Unit	Quarters 3 & 4	Est. Total Error %
a.Spent resins, filter sludges, etc.	m ³	0	N/A
	Ci	0	N/A
b.Dry Compressible Waste, equipment, etc.	m ³	0	N/A
	Ci	0	N/A
c. Irradiated components, control rods, etc.	m ³	0	N/A
	Ci	0	N/A
d.Other	m ³	0	N/A
	Ci	0	N/A

A. Estimate of Major Nuclide Composition (By Type of Waste)

Spent resins, filter sludges		Dry Compressible Waste, equipment, etc.		Irradiated components, control rods, etc.		Other Waste	
Nuclide	Percent(1)	Nuclide	Percent(1)	Nuclide	Percent(1)	Nuclide	Percent(1)
Fe-55	43.16	H-3	0.06	Fe-55	45.10	H-3	2.13
Co-60	43.78	Fe-55	6.29	Co-60	41.51	Fe-55	43.35
Ni-63	12.88	Co-60	44.6	Ni-59	0.13	Co-60	39.48
Cs-137	0.06	Ni-63	43.18	Ni-63	13.27	Ni-59	0.12
		Zn-65	0.15			Ni-63	14.76
		Cs-137	5.29			Cs-137	0.16

(2) Includes only those nuclides that are greater than 0.1% of the total activity

B. Disposition of Solid Waste Shipments (3rd & 4th Quarters)

No. of Shipments	From VY	From Processor	Mode	To Processor	To Burial
85	85	N/A	Rail	N/A	85
16	16	N/A	Truck	N/A	16

A. Irradiated Fuel Shipments (Disposition): None

B. Additional Data (1st & 2nd Quarters)

Supplemental Information	VY to Processor	VY to Burial	Processors to Burial
Class of Solid Waste Shipped	N/A	A/B/C	N/A
Type of Containers Used	N/A	GDC/IP-1/IP-2/Type B	N/A
Solidification Agent or Absorbent Used	N/A	None	N/A

GDC = General Design Container

Table 4A – Off-Site Doses from Liquid and Gaseous Releases
(10CFR50, Appendix I)

Source	Dose (mrem)				
	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Year
Liquid Effluents					
Total Body Dose	5.60E-07	5.26E-07	5.84E-07	6.04E-07	2.24E-06
Percent of Limit (1.5 mrem)	3.73E-05	3.51E-05	3.89E-05	4.05E-05	1.49E-04
Footnotes	(a)	(a)	(a)	(a)	
Organ Dose	5.60E-07	5.26E-07	5.84E-07	6.04E-07	2.24E-06
Percent of Limit (5 mrem)	1.12E-05	1.05E-05	1.17E-05	1.21E-05	4.48E-05
Footnotes	(a)	(a)	(a)	(a)	
Airborne Effluents					
Iodines, H-3, C-14 and Particulates	9.96E-06	1.45E-03	1.70E-03	1.64E-03	4.80E-03
Percent of Limit (7.5 mrem)	1.33E-04	1.93E-02	2.27E-02	2.19E-02	6.40E-02
Footnotes	(b)	(b)	(b)	(b)	
Noble Gases					
Beta Air (mrad)	--	--	--	--	--
Footnotes	(c)	(c)	(c)	(c)	
Gamma Air (mrad)	--	--	--	--	--
Footnotes	(c)	(c)	(c)	(c)	

- (a) The critical age group/organ for the Maximum Exposed Individual (MEI) is the Adult/Total Body and all organs (except Bone) from the release of H-3 to groundwater.
- (b) These doses are based upon the maximum offsite X/Qs and D/Qs. The doses are the maximum organ dose when the Vegetable, Meat, Cow Milk, and Inhalation pathways are summed.
- (c) There were no noble gas releases in this quarter.

Table 4B – Annual Off-Site Doses from Liquid, Gaseous and Direct Radiation
(40CFR190)

Pathway	Total Body (mrem)	Maximum Organ (mrem)	Thyroid (mrem)
Direct External (a) (b)	7.338	7.338	7.338
Liquids (c)	2.24E-06	2.24E-06	2.24E-06
Gases (c)	4.80E-05	4.80E-05	4.80E-05
Annual Total (d)	7.338	7.338	7.338

The location of the projected maximum individual doses from combined direct radiation plus liquid and gaseous effluents correspond to residences at the southwest boundary relative to the Turbine Hall.

- (a) No residential shielding credit or occupancy time fraction (i.e., occupancy is assumed to be 100%) is used. Expected direct external radiation doses would be reduced by approximately 54% with a realistic residential shielding credit and occupancy time (i.e., by using a 0.7 shielding factor from Regulatory Guide 1.109 (Reference 2) and an annual occupancy time of 6760 hours).
- (b) The direct dose reported here was calculated by summing the net monthly doses at TLD location DR-53A and represents the dose to the former nearest residence, which was located in the South sector at 385 meters from the stack prior to the vacancy of this residence in 2008 and the purchase of land by Vermont Yankee. Doses are driven by the casks on the ISFSI.
- (c) Maximum dose to any organ over all age groups for each release.
- (d) Annual dose limits contained in 40 CFR Part 190 are 25 mrem to the total body and any organ, and 75 mrem to the thyroid for any real member of the public.

Table 4C – Receptor Locations

Sector	Site Boundary ⁽¹⁾ (meters)	Nearest Resident ⁽²⁾ (meters)
N	400	1400
NNE	350	1384
NE	350	1255
ENE	400	966
E	500	933
ESE	700	1915
SE	750	1963
SSE	850	2044
S	385	644
SSW	300	451
SW	250	418
WSW	250	451
W	300	628
WNW	400	1062
NW	550	2253
NNW	550	1738

- (1) Site boundary locations are taken from the Stack Releases column in Table 6.10.2 of the ODCM.
- (2) The location(s) given are based on information from the Vermont Yankee 2016 Land Use Census and Table 7.1 of the ODCM and are relative to the plant stack. Gardens are assumed to be present at all resident locations.

Table 4D - Usage Factors for Environmental Pathways

Age Group	Fish (kg/yr)	Potable Water (l/yr)	Veg. (kg/yr)	Leafy Veg. (kg/yr)	Milk (l/yr)	Meat (kg/yr)	Inhalation (m3/yr)
Adult	21	730	520	64	310	110	8,000
Teen	16	510	630	42	400	65	8,000
Child	6.9	510	520	26	330	41	3,700
Infant	0	330	0	0	330	0	1,400

Data from Regulatory Guide 1.109, Table E-5 (Reference 2).

Table 4E - Environmental Parameters for Gaseous Effluents

		Vegetables		Cow Milk		Goat Milk		Meat	
Variable		Stored	Leafy	Pasture	Stored	Pasture	Stored	Pasture	Stored
YV	Agricultural Productivity (kg/m ²)	2	2	0.70	2	0.70	2	0.70	2
P	Soil Surface Density (kg/m ²)	240	240	240	240	240	240	240	240
T	Transport Time to User (hrs)	--	--	48	48	48	48	480	480
TB	Soil Exposure Time ^(a) (hrs)	131,400	131,400	131,400	131,400	131,400	131,400	131,400	131,400
TE	Crop Exposure Time to Plume (hrs)	1,440	1,440	720	1,440	720	1,440	720	1,440
TH	Holdup After Harvest (hrs)	1,440	24	0	2,160	0	2,160	0	2,160
QF	Animals Daily Feed (kg/day)	--	--	50	50	6	6	50	50
FP	Fraction of Year on Pasture	--	--	(b)	--	(b)	--	(b)	--
FS	Fraction Pasture Feed When on Pasture ^(c)	--	--	1	--	1	--	1	--

Note: Footnotes on following page.

Table 4E - Environmental Parameters for Gaseous Effluents
(Continued)

		Vegetables		Cow Milk		Goat Milk		Meat	
Variable		Stored	Leafy	Pasture	Stored	Pasture	Stored	Pasture	Stored
FG	Fraction of Stored Vegetables Grown in Garden	0.76	--	--	--	--	--	--	--
FL	Fraction of Leafy Vegetables Grown in Garden	--	1.0	--	--	--	--	--	--
FI	Fraction Elemental Iodine = 0.5	--	--	--	--	--	--	--	--
H	Absolute Humidity = 5.6 ^(d)	--	--	--	--	--	--	--	--

* From VY ODCM, Table 6.9.1 (Reference 1).

- (a) 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 8,760 hours (one year) for all pathways.
- (b) 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 (non-growing 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.
- (c) 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.
- (d) For all Method II analyses, an absolute humidity value equal to 5.6 (gm/m³) shall be used to reflect conditions in the Northeast (Reference: Health Physics Journal, Volume 39 (August), 1980; Pages 318-320, Pergammon Press).

Table 4F - Environmental Parameters for Liquid Releases (Tritium) Via Groundwater

Variable Name (Units)	Potable Water	Aquatic Food	Stored Veg.	Leafy Veg.	Meat	Cow Milk
Mixing Ratio	5.94E-06	1.27E-03	5.94E-06	5.94E-06	5.94E-06	5.94E-06
Transit Time (hrs)*	12	24	0	0	0	0
Water Uptake** (animal) (L/day)	--	--	--	--	50.0	60.0
Feed Uptake** (animal) (kg/day)	--	--	--	--	50.0	50.0

* Values are from Regulatory Guide 1.109, Table E-15 (Reference 2)

** Values are from Regulatory Guide 1.109, Table E-3 (Reference 2)

Tables 5A-5G - Frequency Distribution Tables for Ground Level Releases

Table 5A

Five Year Aggregate
PASQUILL A

35-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	15	11	30	32	7	2	0	97
NNE	1	2	9	3	0	0	0	15
NE	0	5	10	0	0	0	0	15
ENE	0	5	11	1	0	0	0	17
E	0	12	41	11	0	0	0	64
ESE	1	7	36	32	1	0	0	77
SE	0	3	12	16	0	0	0	31
SSE	1	0	7	23	4	0	0	35
S	0	1	2	9	2	0	0	14
SSW	0	0	0	4	0	0	0	4
SW	0	1	2	4	0	0	0	7
WSW	0	4	2	3	0	0	0	9
W	0	1	3	2	5	0	0	11
WNW	0	0	0	1	6	1	1	9
NW	0	0	11	11	3	1	0	26
NNW	1	10	35	60	19	5	0	130
TOTALS	19	62	211	212	47	9	1	561

Table 5B

Five Year Aggregate
PASQUILL B

35-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	2	10	65	36	5	1	0	119
NNE	0	12	15	9	0	0	0	36
NE	0	7	23	0	0	0	0	30
ENE	0	7	22	0	0	0	0	29
E	1	9	56	7	0	0	0	73
ESE	1	5	65	28	1	0	0	100
SE	0	2	39	22	0	0	0	63
SSE	0	4	31	65	8	1	0	109
S	0	3	15	29	11	1	0	59
SSW	0	0	6	4	0	0	0	10
SW	0	0	1	3	0	0	0	4
WSW	0	0	2	3	0	0	0	5
W	0	0	5	15	14	3	0	37
WNW	0	3	6	12	10	0	0	31
NW	0	2	17	18	14	7	0	58
NNW	1	5	64	72	45	6	0	193
TOTALS	5	69	432	323	108	19	0	956

Table 5C

Five Year Aggregate
PASQUILL C

35-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	2	19	103	68	6	0	0	198
NNE	1	31	31	13	0	0	0	76
NE	0	19	23	2	0	0	0	44
ENE	0	22	56	2	0	0	0	80
E	0	28	111	2	0	0	0	141
ESE	6	25	128	25	1	0	0	185
SE	0	14	94	21	0	0	0	129
SSE	0	6	74	99	11	0	0	190
S	0	2	30	63	8	0	0	103
SSW	0	5	12	9	1	0	0	27
SW	0	5	8	7	0	0	0	20
WSW	0	3	8	10	0	0	0	21
W	1	1	5	37	22	1	0	67
WNW	0	4	14	38	33	2	1	92
NW	1	11	63	53	44	4	0	176
NNW	0	9	123	109	49	6	1	297
TOTALS	11	204	883	558	175	13	2	1846

Table 5D

Five Year Aggregate
PASQUILL D

35-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	33	418	666	514	107	3	0	1741
NNE	21	244	203	58	2	0	0	528
NE	21	232	99	14	1	0	0	367
ENE	31	243	111	2	0	0	0	387
E	51	288	240	24	0	0	0	603
ESE	96	303	474	89	2	0	0	964
SE	102	349	901	101	7	0	0	1460
SSE	79	358	1167	583	48	3	0	2238
S	31	282	493	435	96	3	0	1340
SSW	5	206	177	92	6	0	0	486
SW	27	167	144	94	2	0	0	434
WSW	7	133	205	125	6	0	0	476
W	6	183	412	788	277	13	0	1679
WNW	3	235	370	908	471	42	1	2030
NW	10	400	895	713	359	39	2	2418
NNW	33	521	1440	1181	385	52	1	3613
TOTALS	556	4562	7997	5721	1769	155	4	20764

Table 5E

Five Year Aggregate
PASQUILL E

35-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	51	197	161	43	15	0	0	467
NNE	19	111	23	4	0	0	0	157
NE	14	89	9	1	1	0	0	114
ENE	14	71	7	0	0	0	0	92
E	13	91	36	1	0	0	0	141
ESE	25	156	74	7	0	0	0	262
SE	71	216	277	15	0	0	0	579
SSE	77	346	428	100	8	1	0	960
S	46	529	278	125	32	3	0	1013
SSW	34	676	101	17	2	0	0	830
SW	38	912	98	12	1	0	0	1061
WSW	19	827	214	13	0	1	0	1074
W	19	797	342	150	22	2	0	1332
WNW	21	640	396	144	36	0	0	1237
NW	21	601	577	148	29	1	0	1377
NNW	27	407	711	239	49	3	1	1437
TOTALS	509	6666	3732	1019	195	11	1	12133

Table 5F

Five Year Aggregate
PASQUILL F

35-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	8	78	12	0	0	0	0	98
NNE	7	31	3	0	0	0	0	41
NE	5	25	0	0	0	0	0	30
ENE	5	27	1	0	0	0	0	33
E	6	27	5	0	0	0	0	38
ESE	14	26	7	0	0	0	0	47
SE	17	75	13	0	0	0	0	105
SSE	19	136	39	1	1	0	0	196
S	30	344	59	1	0	0	0	434
SSW	50	501	56	0	0	0	0	607
SW	36	1061	68	0	0	0	0	1165
WSW	29	990	47	1	0	0	0	1067
W	11	677	78	1	0	0	0	767
WNW	12	349	69	2	0	0	0	432
NW	13	198	84	2	0	1	0	298
NNW	23	94	70	6	0	0	0	193
TOTALS	285	4639	611	14	1	1	0	5551

Table 5G

Five Year Aggregate
PASQUILL G

35-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	7	36	4	0	0	1	0	48
NNE	11	18	1	0	0	0	0	30
NE	4	16	0	0	0	0	0	20
ENE	1	10	0	0	0	0	0	11
E	6	22	0	0	0	0	0	28
ESE	10	22	5	0	0	0	0	37
SE	10	27	2	0	0	0	0	39
SSE	10	52	6	0	0	0	0	68
S	22	96	12	0	0	0	0	130
SSW	21	133	15	0	0	0	0	169
SW	7	198	28	0	0	0	0	233
WSW	8	205	8	0	0	0	0	221
W	4	140	17	1	0	0	0	162
WNW	7	90	15	0	0	0	0	112
NW	6	57	26	2	0	0	0	91
NNW	5	38	25	3	0	0	0	71
TOTALS	139	1160	164	6	0	1	0	1470

Tables 6A-6G - Frequency Distribution Tables for Elevated Releases

Table 6A

Five Year Aggregate
PASQUILL A

297-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	11	1	1	6	0	0	0	19
NNE	1	5	1	1	0	0	0	8
NE	0	1	1	1	0	0	0	3
ENE	0	1	2	0	0	0	0	3
E	0	1	0	0	0	0	0	1
ESE	0	3	2	3	1	0	0	9
SE	0	2	5	1	0	0	0	8
SSE	0	4	4	1	2	0	0	11
S	0	1	2	0	0	0	0	3
SSW	0	0	0	0	0	0	0	0
SW	0	0	0	0	0	0	0	0
WSW	0	0	0	1	0	0	0	1
W	0	1	0	0	2	0	0	3
WNW	0	0	2	0	0	0	0	2
NW	0	1	0	3	0	0	0	4
NNW	1	2	3	3	3	0	0	12
TOTALS	13	23	23	20	8	0	0	87

Table 6B

Five Year Aggregate
PASQUILL B

297-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	0	1	8	14	2	0	0	25
NNE	0	0	2	5	2	0	0	9
NE	0	1	1	6	0	0	0	8
ENE	0	0	2	3	0	0	0	5
E	0	1	1	3	1	0	0	6
ESE	1	4	13	26	4	0	0	48
SE	0	3	6	5	3	0	0	17
SSE	0	1	1	3	2	0	0	7
S	0	0	0	0	5	0	0	5
SSW	0	0	0	0	0	0	0	0
SW	0	0	0	0	0	0	0	0
WSW	0	0	0	1	1	0	0	2
W	0	0	0	3	1	1	0	5
WNW	0	0	3	0	2	2	3	10
NW	0	1	1	2	2	0	0	6
NNW	0	3	9	24	18	9	4	67
TOTALS	1	15	47	95	43	12	7	220

Table 6C

Five Year Aggregate
PASQUILL C

297-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	0	6	25	36	26	1	0	94
NNE	0	2	11	16	4	0	0	33
NE	0	2	7	7	2	0	0	18
ENE	0	3	14	8	0	0	0	25
E	0	4	16	9	1	0	0	30
ESE	0	7	51	47	3	0	0	108
SE	0	9	21	31	7	1	0	69
SSE	0	4	2	38	13	0	0	57
S	1	1	3	9	14	7	2	37
SSW	0	0	0	0	0	1	0	1
SW	0	0	1	4	0	0	0	5
WSW	0	0	1	2	3	0	0	6
W	0	0	0	5	3	9	1	18
WNW	0	0	2	7	7	8	0	24
NW	1	1	10	11	15	8	2	48
NNW	0	3	31	71	62	47	11	225
TOTALS	2	42	195	301	160	82	16	798

Table 6D

Five Year Aggregate
PASQUILL D

297-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	18	269	442	565	426	131	10	1861
NNE	3	182	145	127	55	4	1	517
NE	2	191	125	62	21	1	2	404
ENE	2	186	131	32	5	0	0	356
E	8	238	251	66	18	3	0	584
ESE	7	271	505	192	47	3	0	1025
SE	17	355	878	458	57	10	1	1776
SSE	10	243	811	1092	206	24	3	2389
S	13	102	395	1047	628	136	18	2339
SSW	5	72	124	220	71	8	1	501
SW	7	43	93	171	57	1	0	372
WSW	11	50	73	216	92	7	2	451
W	16	53	101	623	695	175	12	1675
WNW	3	82	138	689	901	285	28	2126
NW	2	173	344	430	564	229	52	1794
NNW	54	358	1172	1527	1240	512	153	5016
TOTALS	178	2868	5728	7517	5083	1529	283	23186

Table 6E

Five Year Aggregate
PASQUILL E

297-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	27	524	467	222	46	7	0	1293
NNE	8	354	66	27	5	0	0	460
NE	9	294	41	2	0	0	1	347
ENE	5	282	33	1	0	2	0	323
E	9	397	70	8	0	0	0	484
ESE	15	457	226	27	3	0	0	728
SE	25	474	747	171	6	1	0	1424
SSE	10	277	718	372	30	4	2	1413
S	6	160	281	306	144	23	3	923
SSW	2	90	118	114	40	8	0	372
SW	3	61	87	63	16	0	0	230
WSW	2	61	111	109	20	2	0	305
W	4	57	132	299	186	9	1	688
WNW	4	97	146	284	108	12	1	652
NW	1	206	361	258	83	13	1	923
NNW	21	441	1536	944	305	58	10	3315
TOTALS	151	4232	5140	3207	992	139	19	13880

Table 6F

Five Year Aggregate
PASQUILL F

297-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	9	217	176	19	6	0	0	427
NNE	2	161	24	3	0	0	0	190
NE	2	130	6	0	0	0	0	138
ENE	3	115	16	2	0	0	0	136
E	7	165	42	0	0	0	0	214
ESE	5	140	125	14	0	0	0	284
SE	8	188	343	81	0	0	0	620
SSE	5	149	296	79	0	1	0	530
S	2	76	172	51	8	0	0	309
SSW	3	63	60	29	4	0	0	159
SW	1	52	73	19	2	0	0	147
WSW	0	40	67	20	1	0	0	128
W	1	44	81	70	7	0	0	203
WNW	2	66	91	64	3	0	0	226
NW	2	115	154	40	4	0	0	315
NNW	3	167	429	184	9	0	0	792
TOTALS	55	1888	2155	675	44	1	0	4818

Table 6G

Five Year Aggregate
PASQUILL G

297-foot

	Class 1 Calms	Class 2	Class 3	Class 4	Class 5	Class 6	Class 7	
Wind mph	0.95	3	7	12	18	24	>24	TOTAL
N	0	18	19	3	1	0	0	41
NNE	0	6	2	0	0	0	0	8
NE	0	10	3	0	0	0	0	13
ENE	0	9	2	0	0	0	0	11
E	1	9	3	0	0	0	0	13
ESE	0	12	7	2	0	0	0	21
SE	0	19	48	19	0	0	0	86
SSE	0	19	50	18	0	0	0	87
S	1	14	39	14	4	1	0	73
SSW	0	8	19	12	2	0	0	41
SW	0	11	22	17	0	0	0	50
WSW	0	15	29	15	0	0	0	59
W	0	8	32	26	2	0	0	68
WNW	0	11	24	25	3	0	0	63
NW	1	20	23	20	0	0	0	64
NNW	0	21	37	19	0	0	0	77
TOTALS	3	210	359	190	12	1	0	775

APPENDIX A – SUPPLEMENTAL INFORMATION

1. 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.

Tritium

ODCM Table 4.3.1 requires that a continuous sample of stack effluents be collected for H-3 and analyzed monthly. The error involved in this sample is approximately ± 18 percent. This sampling was initiated at the reactor building exhaust in the second quarter of 2021 when the stack was abandoned prior to demolition.

Particulates

ODCM Table 4.3.1 requires that a continuous sample of stack effluents be collected and analyzed weekly for principal gamma emitters. The error involved in this sample is approximately ± 18 percent. This sampling was initiated at the reactor building exhaust in the second quarter of 2021 when the stack was abandoned prior to demolition.

2. BATCH RELEASES

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

3. ABNORMAL RELEASES

a. Liquid

VY had installed 32 groundwater wells to monitor the 2010 leak event or to monitor additional at-risk structures, systems or components (SSCs) that could cause a release of licensed material to the groundwater. One well (GZ-08) has been dry since installation and no samples were collected from it in 2021. A second well, GZ-24 was compromised by excavation activity in 2014 and is no longer able to be sampled. Due to the current condition of the facility a groundwater expert panel was convened to evaluate the status of the groundwater sampling wells. Based upon the plant configuration (removal of piping/components) and status of the plume it was prudent to reduce the number of active wells in the program. A summary of the sample results for the remaining wells are presented in Table A-1. There are only two (2) wells that had detectable activity ($>MDC$) in 2021 and all of these sample results are well below the EPA limit of 20,000 pCi/L for drinking water. None of the wells in this program supply drinking water, and no drinking water wells on site or adjacent to VY have shown tritium at detectable levels in regular surveillance samples.

b. Gaseous

There were no non-routine gaseous releases (measured) during the reporting period.

Table A-1: VY Groundwater Tritium Summary - 2021

Groundwater well Sampled	Number of analyses performed	Mean Concentration ¹	Concentration Range ¹	
			Min	Max
GZ-01	1	< 601	< 601	< 601
GZ-02 X	1	< 1840	< 1840	< 1840
GZ-03	1	< 662	< 662	< 662
GZ-04	2	< 602	< 602	< 602
GZ-05	1	< 602	< 602	< 602
GZ-06 X	1	< 665	< 665	< 665
GZ-07 X	0	**	**	**
GZ-09 X	1	< 594	< 594	< 594
GZ-10 X	1	< 593	< 593	< 593
GZ-11 X	1	< 592	< 592	< 592
GZ-12 X	1	< 592	< 592	< 592
GZ-12D X	2	< 640	< 609	< 670
GZ-13	1	< 571	< 571	< 571
GZ-13D	2	< 583	< 574	< 592
GZ-14	1	< 596	< 596	< 596
GZ-14D	4	980	< 667	1430
GZ-15 X	3	< 651	< 608	< 677
GZ-16 X	0	**	**	**
GZ-17 X	0	**	**	**
GZ-18	1	< 570	< 570	< 570
GZ-18D	1	< 568	< 568	< 568
GZ-19	1	< 571	< 571	< 571
GZ-19D	1	< 573	< 573	< 573
GZ-20 X	0	**	**	**
GZ-21 X	1	< 594	< 594	< 594
GZ-22D	12	1166	< 670	1770
GZ-23	9	< 612	< 514	< 691
GZ-25	1	< 597	< 597	< 597
GZ-26	1	< 599	< 599	< 599
GZ-27	1	< 648	< 648	< 648

Notes:

1. All concentrations are in units of pCi/L
2. Required LLD for tritium = 2,000 pCi/L
3. "<" denotes minimum detectable value for the analytical period
4. **Bold** values denote positive results (greater than minimum detectable values)
5. **"X"** denotes a well that was abandoned as of August 2021.

APPENDIX B – LIQUID HOLDUP TANKS

<u>Requirement</u>	With the quantity of radioactive material in any outside tank exceeding the limit of Section 4 of the Decommissioning Safety Analysis Report, describe the events leading to this condition in the next Radioactive Effluent Release Report.
<u>Response</u>	The limits for any outside tank were not exceeded during this reporting period.

APPENDIX C - RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The requirements for radioactive liquid effluent monitoring instrumentation channels in ODCM Table 3.1.1 were removed in ODCM Revision 40.

APPENDIX D - RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

<u>Requirement</u>	Radioactive gaseous effluent monitoring instrumentation channels are required to be functional in accordance with ODCM Table 3.1.2. If a non-functional, gaseous effluent monitoring instrumentation is not returned to functional status within 30 days pursuant to Note 5 of Table 3.1.2, an explanation in the next annual Radioactive Effluent Release Report of the reason(s) for the delay in correcting the inoperability is required per ODCM Section 10.1. The plant stack was removed from service in March 2021. The release path was relocated to reactor building exhaust path in the overhead of the 280 foot level.
<u>Response</u>	There were no issues related to the radioactive gaseous effluent monitoring instrumentation for this reporting period.

APPENDIX E – RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

This Appendix is no longer required by the ODCM Revision 40. This Appendix is retained as a placeholder to allow this report to be consistent with previous reports.

APPENDIX F – LAND USE CENSUS

<u>Requirement</u>	A land use census is conducted in accordance with ODCM Control 3/4.5.2. With a land use census identifying a location(s) that yields at least a 20 percent greater dose or dose commitment than the values currently being calculated pursuant to ODCM Control 4.3.3, the new location(s) must be identified in the next Annual Radioactive Effluent Release Report.
<u>Response</u>	The Land Use Census was completed during the third quarter of 2021. No locations were identified which yielded a 20 percent greater dose or dose commitment than the values currently being calculated pursuant to ODCM Control 4.3.3. See Table 4C for a listing of nearest residents in the site area as determined in the 2021 Land Use Census.

APPENDIX G – PROCESS CONTROL PROGRAM

<u>Requirement</u>	ODCM Section 10.1 requires that licensee-initiated changes to the Process Control Program (PCP) be submitted to the Commission in the annual Radioactive Effluent Release Report for the period in which the change(s) was made.
<u>Response</u>	There were no changes made to the Process Control Program during this reporting period.

APPENDIX H – OFF-SITE DOSE CALCULATION MANUAL

<u>Requirement</u>	ODCM Section 10.1 requires that licensee-initiated changes to the Off-Site Dose Calculation Manual (ODCM) be submitted to the Commission in the annual Radioactive Effluent Release Report for the period in which the change(s).
<u>Response</u>	There was one revision (Rev. 41) to the Offsite Dose Calculation Manual (ODCM) during 2021. The ODCM is included as Enclosure 2 to the letter forwarding this report. A description of the changes associated with Revision 41 to the ODCM are included within the Revision Summary portion of the ODCM.

APPENDIX I – RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS

<u>Requirement</u>	ODCM Section 10.5 requires that licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid) be reported to the Commission in the annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Independent Safety Review Committee.
<u>Response</u>	There were no licensee-initiated major changes to the radioactive waste systems during this reporting period.

APPENDIX J – ON-SITE DISPOSAL OF SEPTIC/SILT/SOIL WASTE

<u>Requirement</u>	Off-Site Dose Calculation Manual, Appendices B, F and I require that the dose impact due to on-site disposal of septic waste, cooling tower silt, and sand/soil type materials during the reporting year and from previous years be reported to the Nuclear Regulatory Commission in the Annual Radioactive Effluent Release Report if disposals occur during the reporting year. Vermont Yankee will report in the Annual Radioactive Effluent Release Report a list of the radionuclides present and the total radioactivity associated with the disposal activities on the Vermont Yankee site.
<u>Response</u>	There was no on-site disposal spreading during 2021.

Enclosure 2

Vermont Yankee Nuclear Power Station

Off-site Dose Calculation Manual (ODCM), Revision 41
(338 pages excluding this cover sheet)

VERMONT YANKEE NUCLEAR POWER STATION

OFF-SITE DOSE CALCULATION MANUAL

REVISION 41

Effective Date: 7/1/2021

Originator:	Michael Pletcher		<u>7/6/25/21</u>
	Print/Sign		Date
Reviewed:	John Gevster		<u>7/6/30/21</u>
	Print/Sign (Independent Safety Review)		Date
Approved:	Corey Daniels		<u>6/30/121</u>
	Print/Sign (ISFSI Senior Manager)		Date

REVISION SUMMARY

<u>DATE</u>	<u>REVISION</u>	<u>DESCRIPTION</u>
9/23/10	33	<p>Three sections of the ODCM were modified with minor changes to incorporate recommendations made in the 2009 RETS/REMP QA audit:</p> <ul style="list-style-type: none"> - A typographical error in ODCM Table 6.10.1 was corrected. The table listed a distance of 26,500 meters for the highest undepleted X/Q for skin dose calculations. The correct value is 2,650 meters. This table is descriptive in nature and is not used for calculation of doses. - The error in distance measurement in the first paragraph on page 2 of 12 of Section 7 of the ODCM was corrected. The distance of 0.5 miles was revised to 0.75 miles. This value is descriptive in nature and is not used for calculation of doses. - ODCM Table 3.5.1 provided distances in miles whereas other sections of this table contained distances in kilometers. Additionally, there was no explanation as to why some “inner” ring TLDs are located further from the plant than some “outer” ring TLDs. This table was changed to include both kilometers and miles where distances are required. Additionally, the table footnotes were revised to explain the method used to determine proper location for the inner and outer ring TLDs in each of the 16 compass sectors. - Footnote “e” to ODCM Table 4.5.1 did not fully explain how the determination of Barium/Lanthanum 140 activity is determined using daughter ingrowth. Additional information was provided in footnote “e” to provide the reader with a better explanation. - Table 7.1 was revised to clearly indicate the TLDs designated as “control” location TLDs.

REVISION SUMMARY (Continued)

7/08/2011	34	<p>Four main sections of the ODCM were modified with significant changes to incorporate the contaminated groundwater discharge pathway to the Connecticut River:</p> <ul style="list-style-type: none"> ○ Section 3 / 4 was revised to include the subsurface groundwater pathway in the Liquids Discharge description. Groundwater monitoring wells used to determine the extent of these releases are listed. The Southwest Well was added as Ground (Potable Drinking) Water sample location in the REMP description of Section 3 / 4. ○ Section 5 was revised to include a description of the determination of plant generated radionuclide concentrations in groundwater discharges. ○ Section 6 was revised to include methods for calculating radiation dose from plant generated radionuclides in groundwater discharges. ○ Section 9 was revised to include the method for determination of groundwater flows in the 17 identified streamtubes flowing from the plant site to the Connecticut River. <p>In addition to revisions of four main sections of the ODCM, the Table of Contents, Definitions and References Sections of the ODCM were revised to reflect the additional subsections, figures, tables, definitions and references in the ODCM.</p>
10/09/14	35	<ul style="list-style-type: none"> ○ Section 3/4 was revised to delete a reference requirement to Note 5 in Section 2 of Table 3.1.2 (Gaseous Effluent Monitoring Instrumentation). It was determined that Note 5 was in conflict with Note 2 for Section 2 and therefore should be removed as a requirement for Section 2 (CR-VTY-2013-04078 CA-0002). ○ Also, in Sections 2, 3/4, 6, 8 and 10, the word “operable” was found to be inappropriate and should be replaced by the word “functional”. Additionally, the word “inoperable” by the word “non-functional” and the word “operability” was replaced by “functionality” (EN-OP-104 and NRC Inspection Guide 9900) (CR-VTY-2013-04078 CA-0002). ○ In the tables of Section 7 of the Offsite Dose Calculation Manual, it was determined that the out-of-business dairy farms which had provided milk for the REMP but were no longer functional, should be eliminated from the description (WT-WTVTY-2011-00116). <p>Section 9 of the Offsite Dose Calculation Manual was revised to eliminate the references to the Off Gas “30 minute” delay line. This was previously evaluated under CR-VTY-2010-1676 and it was determined that the reference to the “30 minute” should be removed. (WT-WTVTY-2009-00009 CA-0010)</p>

02/03/16	36	<ul style="list-style-type: none"> ○ Section 3/4.2.1 was changed to add intercepted groundwater releases to Surveillance 4.2.1.a to clarify that controls on liquid releases also apply to intercepted groundwater being released. ○ Section 3/4.2.1, Table 4.2.1 was changed to add intercepted groundwater requirements to Table 4.2.1. This establishes sampling and analysis protocols that are as rigorous as liquid waste releases and consistent with subsurface groundwater analysis. These protocols provide data sufficient to support dose estimates and assignments and allows comparison between interception wells and other subsurface groundwater. ○ Section 3/4.2.1, Table 4.2.1 Notation (e) was revised to provide the same analytical specifications for intercepted groundwater and subsurface groundwater, enabling comparison of results for all groundwater samples. ○ Section 3/4.2.1, <u>Surveillance 4.2.1.a</u> was revised to ensure that the ODCM limit of 10 times 10CFR20 Appendix B applies to all water released to the Connecticut River. ○ Section 3/4.6 was revised to ensure that liquid effluent instrumentation is not required for intercepted groundwater releases. ○ Section 5.1 was revised to include groundwater intercept release tank(s) as a potential release point and to clarify that intercepted groundwater releases are undiluted. ○ Section 5.2.1 was revised to include intercepted groundwater tanks to the list of tanks from which batch releases are made. Additionally, this section recognizes alternate sample point(s) other than the radwaste sample sink are acceptable for collecting samples for analysis prior to release. ○ Section 5.2.4 is added to describe the new release pathway for intercepted groundwater and to describe it's use in a manner consistent with liquid waste treatment systems and subsurface groundwater. ○ Section 6.2 is revised to add dose calculation from intercepted groundwater and ensure it is performed the same as subsurface groundwater. This clarifies that total body dose estimates are only required prior to liquid waste releases. ○ Table 6.2.1 is revised to add a footnote clarifying that the mixing ratio is based on a 20,000 gpm flow. ○ Section 6.2.1, Equation 6-1 is revised to add a flow correction term. This allows dose factors DFL to be adjusted for release flow rates if the flow rate is different than the rate used for Table 1.1.11. ○ Section 6.2.2, Equation 6-2 is revised to add a flow correction term. This allows dose factors DFL to be adjusted for release flow rates if the flow rate is different than the rate used for Table 1.1.11. ○ Section 6.3.1, Equation 6-3 is revised to add a flow correction term. This allows dose factors DFL to be adjusted for release
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		<p>flow rates if the flow rate is different than the rate used for Table 1.1.11.</p> <ul style="list-style-type: none"> ○ Section 6.3.2, Equation 6-4 is revised to add a flow correction term. This allows dose factors DFL to be adjusted for release flow rates if the flow rate is different than the rate used for Table 1.1.11. ○ Section 7.2 and Table 7.1 were revised to delete non-required control stations from the REMP Sample Station listing for airborne, waterborne, mixed grasses, milk and silage.
12/01/16	37	<ul style="list-style-type: none"> ○ Delete Steam Jet Air Ejector (SJAE) Monitoring from Table 1.1.1 ○ Delete “iodines” from Table 1.1.1 ○ Delete Thyroid dose from Table 1.1.1 ○ Delete “iodines” from Table 1.1.4 ○ Delete iodine and N-16 terms from Table 1.1.6 ○ Delete direct radiation calculations from Table 1.1.6 and replace with TLD readings from site boundary ○ Delete reference to North Warehouse from Table 1.1.6 ○ Delete SJAE requirements from Table 1.1.7 ○ Delete Total body calculation for noble gas from Table 1.1.7 ○ Remove I-131, I-133 and SJAE from Table 1.1.8 ○ Delete all nuclides other than Kr-85 from Table 1.1.10 ○ Delete all nuclides other than Kr-85 from Table 1.1.10A ○ Delete nuclides with short half-life from Table 1.1.11 ○ Delete nuclides with short half-life from Table 1.1.12 ○ Delete Gaseous Radwaste Treatment system, Hot Standby and Refueling Outage definitions from Table 2.1.1 ○ Delete variables from Table 2.1.2 that are no longer applicable ○ Delete iodines from Table 2.1.2 ○ Delete SJAE and AOG terms from Table 2.1.2 ○ Delete SJAE and AOG requirements from Table 3.1.2 ○ Delete Notes 2,3,6,8 & 9 from Table 3.1.2 ○ Delete references to Tech Spec 3.8.K and 3.8.J ○ Delete SJAE and AOG requirements from Table 4.1.2 ○ Delete SJAE and AOG monitor Notes from Table 4.1.2 ○ Delete I-131 analysis and LLD from Table 4.2.1 ○ Delete I-131, Ba-140 & La-140 from Table 4.2.1 Note (e) ○ Delete iodine requirements from 3/4.3.1 ○ Delete SJAE, I-131 and short-lived noble gas requirements from Table 4.3.1 ○ Delete start-up, shutdown, power change requirements, SJAE noble gas sample and short-lived gamma emitter list from Table 4.3.1 Notation ○ Delete iodines from 3/4.3.3 ○ Delete 3/4.3.4 and Revise Figure 9.2 to reflect change ○ Delete AOG requirements from 3/4.3.5 ○ Delete 3/4.3.6

		<ul style="list-style-type: none"> ○ Delete 3/4.3.7 ○ Reduce the number of REMP air sample stations due to reduced risk of release and remove iodine canisters ○ Reduce REMP TLDs to site boundary and offsite controls ○ Delete Table 3.5.1 4.a. Milk sampling ○ Delete Table 3.5.1 Notation (g) ○ Delete I-131, Ba-140, La-140 and Notation (d) from Table 3.5.2 ○ Delete I-131, Ba-140, La-140, milk and Notes (e) and (g) from Table 4.5.1 ○ Delete requirement to identify location of nearest milk animal in each meteorological sector from the land use census. ○ Delete infant thyroid dose from 3/4.6 Basis 3.3.1 ○ Delete iodine dose from 3/4.6 Basis 3.3.3 ○ Delete 3/4.6 Basis 3.3.4 ○ Delete AOG from 3/4.6 Basis 3.3.5 ○ Delete 3/4.6 Basis 3.3.6 ○ Delete 3/4.6 Basis 3.3.7 ○ Delete reference to multiple reactor site from 3/4.6 Basis 3.4.1 ○ Revise wording in 3/4.6 Basis 3.5.1 to indicate the cessation of milk sampling. ○ Revise wording in 3/4.6 Basis 3.5.2 to indicate the cessation of milk sampling. ○ Revise wording of 5.2.1 Sample Tank Pathway to reflect current operational status. ○ Delete Section 5.2.3, Circulating Water Pathway ○ Revise Section 6.1 to delete discussion of burning waste oil ○ Section 6.2.2, reformat to align terms correctly (editorial) ○ Delete thyroid dose from Section 6.3 ○ Re-format terms under Equation 6-3 (editorial) ○ Revise 6.4.1 Method I to remove SJAE calculations and calculate based on stack gas grab samples. Due to decay, remove summation and calculate based solely on Kr-85 for stack and ground calculations ○ Revise 6.4.2 Method I equations to be specific for Kr-85 and remove reference to SJAE and AOG ○ Delete references to AOG and SJAE and make Section 6.5.1 Method I specific to Kr-85 ○ Revise 6.5.2 Method I to make Kr-85 specific (delete summation in Eq 6-8, 6-9, 6-12, 6-13, 6-7, 6-37, 6-38 and make all D and Q terms specific for Kr-85 ○ Section 6.5.2 Basis for Method I delete references to the North Warehouse as a designated ground level release point and the waste oil burner. Designate the Reactor Building as a ground level release point. ○ Delete iodine from Section 6.6 ○ Delete references to the North Warehouse waste oil burner and designate the Reactor Building as a ground level release point in Section 6.6
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		<ul style="list-style-type: none"> ○ Delete Section 6.7; beta air dose is bounding based on Kr-85 ○ Revise Section 6.8 due to decay, remove summation and calculate based solely on Kr-85 for stack and ground calculations; make all D and Q terms specific for Kr-85 ○ Section 6.8.2 Basis for Method I delete references to the North Warehouse as a designated ground level release point and the waste oil burner. Designate the Reactor Building as a ground level release point. ○ Delete iodine from Section 6.9 ○ Delete milk pathway, iodines and noble gas gamma air dose from Section 6.10 ○ Revise the stack flow rate in Section 6.10 ○ Table 6.10.1 was revised to include new atmospheric dispersion factors for the Reactor Building ○ Revise Table 6.10.2 to show distances to site boundary from the Reactor Building ○ Delete Section 6.11.1 due to no dose from N-16 ○ Delete Section 6.11.2 ○ Delete Section 6.11.3 due to the Low Level Waste Pad being demolished ○ Delete calculation of ISFSI direct dose and measure dose using REMP TLDs ○ Revise Section 6.11.4 to replace calculations of direct dose with REMP TLDs ○ Delete Section 6.11.5 ○ Revise Section 7.2 Airborne Pathway to delete sampling in nearby communities ○ Revise Section 7.3 to replace Turbine Building with Reactor Building for distances to sample stations ○ Delete reference to Tech Spec 3.8.K.1 and SJAЕ in Section 8.2 ○ Delete AOG from Section 8.2.1 ○ Delete AOG from Section 8.2.1.1 ○ Delete total body setpoint equation; beta skin dose is limiting for noble gas ○ Revise Section 8.2.1.2 to add Kr-85 values for Sg, revise the stack flow and make the example specific for Kr-85 and make set point based on Kr-85 skin dose ○ Revise Section 8.2.1.3 to remove reference to AOG and SJAЕ, make specific to Kr-85 and remove basis for total body set point ○ Delete Section 8.2.2, SJAЕ Noble Gas Activity Monitors ○ Revise Section 9.2 to delete references to equipment no longer in service (AOG, recombiner system, charcoal absorber system, gland seal system) ○ Delete met tower reporting requirement from Section 10.1 ○ Revise Section 10.2 to delete milk sample reference. ○ Revise Section 10.4.2 to delete iodine from gaseous effluents
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02/06/17	38	<ul style="list-style-type: none"> ○ Table 4.2.1: change “Perimeter” wells to “Monitoring” wells in the table, consistent with the function of the well as defined in site procedures. ○ Table 4.2.1 Notation g.: change “Perimeter” wells to “Monitoring” wells, consistent with the function of the well as defined in site procedures. ○ Table 4.2.1, Add Notation i. to provide basis for sampling intervals in Table 4.2.1. ○ Table 4.2.1, Add Notation j. to provide basis for sampling intervals in Table 4.2.1.
03/05/18	39	<ul style="list-style-type: none"> ○ Delete Liquid Radwaste Monitor from Table 1.1.1 ○ Delete Liquid Radwaste Monitor and set points from Table 1.1.7 ○ Revise Definition 7 in Table 2.1.1 (Liquid Waste Discharge) ○ Revise Definition 8 in Table 2.1.1 (ODCM) ○ Revise definitions of dilution flow and discharge flow rates and delete set point and efficiency terms in Table 2.1.2 ○ Delete Liquid Radwaste Monitor and Flow Rate Measurement Devices from Table 3.1.1 ○ Delete Note 1 and Note 3 in Table 3.1.1 ○ Delete Liquid Radwaste Monitor from Table 4.1.1 ○ Revise Control 3/4.2.1 to allow intercepted groundwater release through liquid waste effluent line ○ Revise Table 4.2.1 to institute requirements for release without liquid radwaste monitor deleted from Table 3.1.1 Note 1. ○ Add Note (k) to Table 4.2.1 implement Table 3.1.1 requirements for release without liquid radwaste monitor ○ Revise 3/4.6 Bases 3.1.1 to pertain to flow rate meter and delete language on release frequency ○ Revise 3/4.6 Bases 3.2.1 to allow intercepted groundwater release through liquid waste effluent line ○ Revise Cpi term in 5.1 to remove names of SSCs to reflect SAFSTOR configuration ○ Revise 5.2.1 Sample Tank Pathway to reflect SAFSTOR configuration ○ Revise 5.2.4 Intercepted Groundwater Pathway to allow intercepted groundwater release through liquid waste effluent line ○ Delete 8.1.1 Liquid Radwaste Discharge Monitor (RM-17-350) ○ Revise Section 9.1 to reflect SAFSTOR configuration ○ Revise Figure 9-1 to reflect SAFSTOR configuration

10/23/18	40	<ul style="list-style-type: none"> ○ Add definitions for X/Q and D/Q ○ Changed Reactor Building release to Ground Release throughout and eliminated elevated throughout. Put notes under Table 1.1.12 and Table 6.10.1 that Reactor Building releases are considered ground releases. Changed ground level vents to ground level releases throughout. ○ Remove North Warehouse throughout document. ○ Delete references to noble gas controls and calculations, and ventilation exhaust treatment system for AOG and waste building from Table 1.1.1 Summary of Radiological Effluent Controls and Implementing Equations and equations 6-21 and 6-23 in Note 2 ○ Delete noble gas requirements from Table 1.1.2 Summary of Methods to Calculate Unrestricted Area Liquid Concentrations ○ Delete the following direct noble gas dose calc terms from TABLE 1.1.4 Summary of Methods to Calculate Dose Rates <ul style="list-style-type: none"> Eq. No. Title ○ 6-5 Total Body Dose Rate from Noble Gases Released from Stack ○ 6-39 Total Body Dose Rate from Noble Gases Released from Ground ○ 6-7 Skin Dose Rate from Noble Gases Released from Stack ○ 6-38 Skin Dose Rate from Noble Gases Released from Ground ○ 6-21 Gamma Dose to Air from Noble Gases Released from Stack ○ 6-41 Gamma Dose to Air from Noble Gases Released from Ground Level ○ 6-23 Beta Dose to Air from Noble Gases Released from Stack ○ 6-43 Beta Dose to Air from Noble Gases Released from Ground Level ○ 6-27b Direct Dose ○ 6-27c Direct Dose ○ 6-27d Direct Dose ○ Gaseous Effluents: ○ Plant Stack (RR-108-1A, RR-108-1B) Noble Gas Activity Monitors ○ 8-10 Skin ○ Delete TABLE 1.1.5 Summary of Methods to Calculate Doses to Air from Noble Gases entirely ○ Delete Table 1.1.6 reference to direct dose from main steam line radiation monitor associated with N-16 ○ Delete Table 1.1.7 Summary of Methods for Setpoint Determinations, stack monitoring consists of grab and composite sampling ○ Delete items from Table 1.1.8 that no longer apply associated
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		<p>with noble gases and primary gas treatment systems</p> <ul style="list-style-type: none"> ○ Delete TABLE 1.1.10 Dose Factors Specific for Vermont Yankee for Noble Gas Releases ○ Delete TABLE 1.1.10A Combined Skin Dose Factors Specific for Vermont Yankee Ground Level Noble Gas Releases. ○ Modify Table 1.1.12 values incorporating new meteorology and dispersion factors. ○ Delete the Noble Gas effluent release and dose calculation terms from TABLE 2.1.2 Summary of Variables, added new terms as applicable to coincide with methodology changes. ○ Modify 3/4.1.2 and Table 2.1.2 to recognize the removal of the gaseous monitor channel for noble gases ○ Delete Notes 7 and 10 from TABLE 3.1.2 NOTATION along with removal of noble gas requirements ○ Delete iodine sampler cartridge and Noble Gas Activity Monitor (RM 17 156, RM 17 157) from Table 3.1.2. ○ Delete 3/4.1.1 Radioactive Liquid Effluent Instrumentation since Service Water Discharge Monitor (RM 17 351) is removed from service. ○ Delete Table 3.1.1 Service Water Discharge Monitor (RM 17 351) will be removed from Service. ○ Delete Table 4.1.1 and Notations Service Water Discharge Monitor (RM 17 351) will be removed from Service. ○ Delete noble gas activity monitor from Table 4.1.2 ○ Add Service Water weekly grab sample and quarterly composite to Table 4.2.1 Radioactive Liquid Sampling and Analysis Program ○ Delete 3/4.6 EFFLUENT AND ENVIRONMENTAL CONTROL BASES INSTRUMENTATION Liquid Effluent Instrumentation (3.1.1) ○ Revise 5.2.2 Service Water Pathway to eliminate continuous monitoring and require routine sampling. ○ Delete Section 8.1.2 Service Water Discharge Monitor (RM 17 351) ○ Delete reference to noble gases in 3/4.2.1 ○ Delete reference to dissolved noble gases in Table 4.2.1 ○ Add weekly storm drain grab sample to Table 4.2.1. ○ Change MDC to LLD in Table 4.2.1 Note j. ○ Delete the Noble Gas requirements from Section 3/4.3.1 Gaseous Effluents Dose Rate ○ Delete Section 3/4.3 RADIOACTIVE GASEOUS EFFLUENTS for noble gases ○ Delete 3/4. Ventilation exhaust treatment system is no longer needed to maintain doses below 0.3 mrem to any organ over one month during decommissioning. ○ Delete monthly stack grab sample. Change to continuous tritium sample analyzed monthly. ○ Delete Control 4.3.2 in 3/4.4.1 noble gas controls no longer applicable.
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		<ul style="list-style-type: none"> ○ Modified wording explaining the REMP direct radiation monitoring sites to be consistent with the 10 chosen locations. ○ Modify note b in Table 3.5.1 to coincide with number of direct radiation monitoring stations. ○ Change from weekly to Monthly the frequency for change out of particulate sampling (weekly sampling was originally based on radioiodine sampling requirements). ○ Remove statements from 3/4.6 basis (for 3.2.1 and 3.3.2) related to noble gas dose and dose rate limits, no longer applicable. ○ Delete reference to ventilation exhaust treatment system, systems supported by this system are no longer in service or needed per NUREG-0473 ○ Delete Table 4.5.1 Note (b). No longer used in the table. ○ Typo, repeat paragraph in 3/4.6 after discussion of Interlaboratory Comparison Program basis statement. ○ Delete the noble gas terms from Section 5.0 METHOD TO CALCULATE OFF SITE LIQUID CONCENTRATIONS ○ Delete References to noble gases in Section 6.1 on introductory concepts to dose calculations ○ Delete release release in fourth paragraph of 6.1 Introductory Concepts ○ Delete sections 6.4, 6.5 and 6.8 related to methods associated with noble gas dose and dose rate calculations ○ In Section 6.6.2, remove statement referring to Section 6.4.3 since that section has been deleted. ○ Modify Section 6.6 for tritium and particulates to present calculational methods that are comparable and consistent with NUREG-0133 "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants ○ Changed reference for equation 6-19 from 6.8 to 6-44 which is the ground release equation. ○ Revise equation 6-19 to use ground release terms R_{cog}, Q_{iGLP} and DFG_{gic} ○ Delete references to noble gas doses, air, and skin doses in Section 6.10 and 6.10.1 ○ In section 6.10.2 change the computer code used to calculate dispersion factors to the NRC sponsored code XOQDOQ ○ In Section 6.10.2 deleted discussion on effective stack height being above 100 m, XOQDOQ program code was run with a stack height less than 100 meters. Also changed basis for depleted X/Q and calculated D/Q from Regulatory Guide 1.111 to NRC approved software XOQDOQ. ○ Changed date range for meteorological data to 2012 to 2016. ○ In Section 6.10 add in ingestion of milk since this is still an exposure pathway that will contribute to dose from particulate radionuclides. ○ Table 6.10.1 was modified to remove atmospheric dispersion data applicable to noble gases and incorporate the new factors
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		<p>determined from the XOQDOQ program code with the updated meteorological data.</p> <ul style="list-style-type: none"> ○ Revised 6.11.4 and reference H to reflect 58 Spent Fuel Dry Storage Casks and the 10 CFR 72.104 study of direct radiation dose rates. ○ Renumbered section after 6.11.4. 6.11.4 was used twice. ○ Deleted control 3.3.2 related to noble gases from statement in Section 6.12. ○ Added in information on the change in meteorological data wind frequencies as compared to original plant data in Sector 7.1. ○ In Table 7.1 added in four additional REMP direct radiation monitoring sites to account for the committed number of 10 in Table 3.5.1. These added sites include those associated with the ISFSI siting. ○ Deleted silage as an exception in note 2 of Table 7.1 ○ Delete radioiodine sampling from Table 7.1 Radiological Environmental Monitoring Stations ○ Updated Figure 7.4 to include all REMP direct radiation monitoring sites. ○ Delete section 8.1 Liquid Effluent Instrumentation Setpoints because there is no liquid effluent instrumentation and there are no liquid effluent releases occurring. All water is being shipped off site for disposal. ○ Delete Section 8.2.1 on setpoint determination for the noble gas monitor, no longer required. ○ Modified Figure 9.2 to show current configuration of gaseous effluent pathway and monitoring. ○ Delete Section 10.1 Annual Meteorological Report Requirements ○ Changed outside tank limit location in 6th paragraph from Techs Specs to DSAR section 4. ○ Fixed typo in Section 10.3 where radioactive was misspelled. ○ Modified Section 10.4.2 to remove reference to reporting for noble gases and the abandonment of the primary vet gas and AOG treatment systems. ○ In Reference Section item B Change computer code for calculating dispersion coefficients to the NRC program XOQDOQ which formed the basis for the AEOLUS-2 computer code
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7/1/21	41	<ul style="list-style-type: none"> ○ Deleted proprietary statement on the cover page. ○ Modified references to the Technical Specifications throughout the ODCM. The reference information is now contained in the Vermont Yankee Nuclear Power Station Quality Assurance Program Manual (QAPM). ○ Table 1.1.1, Note 2. Removed reference to Equation 6-21 and 6-23 as the equations related to noble gas releases which had been removed in the previous revision to the ODCM. ○ Deleted 3.3.2.a and 3.3.2.b from 3/4.4.1 action statement, the information was deleted in Rev. 40 of the ODCM. ○ Table 3.5.1, Removed the composite sampling requirement for the downstream river sample. Vermont Yankee is no longer discharging to the river continuously. This sample requirement has been changed to a monthly grab sample. ○ Table 3.5.1, Notation – Note f. was deleted, it was a reference to composite sampling. ○ Table 4.2.1, Deleted Service Water sampling, service water system has been abandoned. ○ 5.2.2, Service Water Pathway, section deleted because service water system has been abandoned. ○ 6.10.1, Added back in the remaining Atmospheric Dispersion Factor that had been inadvertently left out of the previous version of the ODCM. ○ 6.11.4, Modified the location description of the ISFSI Pad based upon current site configuration. ○ 9.3.1, typo correction – steam should be stream. ○ 10.1, typo correction – correct spelling of “functional”. ○ Revised the entire document such that it is Revision (Rev) 41. ○ Corrected minor typographical errors such as references to formally deleted sections, tables or figures. ○ Deleted definition #16 from Table 2.1.1 “Ventilation Exhaust Treatment System.” This is a legacy removal as the use of this definition and the corresponding equations were removed during Revision 40.
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ABSTRACT

The VYNPS ODCM (Vermont Yankee Nuclear Power Station Off-Site Dose Calculation Manual) contains the effluent and environmental control limits, and 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 manager responsible for overall operational activities (as per the VY Quality Assurance Program Manual) the methods contained in the ODCM are suitable to demonstrate compliance with effluent controls.

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1.0 INTRODUCTION

The 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 Control Limits which implement the program requirements of Quality Assurance Program Manual (QAPM), Appendix D, Section 2.5.B. 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 10 CFR 50 (Regulatory Guide 1.109) unless otherwise noted in the text.

Demonstration of compliance with the dose limits of 40 CFR 190 (see Control 3.4.1) will be considered as demonstrating compliance with the 0.1 rem limit of 10 CFR 20.1301(a)(1) for members of the public in unrestricted areas (Reference 56 FR 23374, third column.)

It shall be the responsibility of the RP/Chemistry Manager to ensure that the ODCM is used in the performance of the surveillance requirements of the appropriate portions of ODCM Controls. The administration of the program for the onsite disposal of slightly contaminated waste, as described in Appendices, is also the responsibility of the RP/Chemistry Manager.

All changes to the ODCM must be reviewed by the Independent Safety Review and approved by the manager responsible for overall operational activities, in accordance with QAPM Appendix D, prior to implementation. All approved changes shall be submitted to the NRC for their information in the Radioactive Effluent Release 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.

1.1 Summary of Methods, Dose Factors, Limits, Constants, and Radiological Effluent Control Cross-References

This section summarizes the dose calculation methods. The concentration and setpoint methods are also summarized in Table 1.1.2 through Table 1.1.4 and Table 1.1.6, 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 6.2, 6.3, 6.6, 6.9 and 6.11. The dose factors used in the equations are in Tables 1.1.11 and 1.1.12 and the Regulatory Limits are summarized in Table 1.1.1.

A cross-reference of old Technical Specification sections to the new ODCM sections containing the equivalent Controls is presented in Table 1.1.8.

Special definitions and equation variables used in the ODCM are in Tables 2.1.1 and 2.1.2.

TABLE 1.1.1
Summary of Radiological Effluent Controls
and Implementing Equations

Control	Category	Method ⁽¹⁾	Limit
3.2.1	Liquid Effluent Concentration	Sum of the Fractions of Effluent Concentration Limits [Excluding Noble Gases]	Eq. 5-1 ≤10
3.2.2	Liquid Effluent Dose	Total Body Dose Organ Dose	Eq. 6-1 Eq. 6-3 ≤1.5 mrem in a qtr. ≤3.0 mrem in a yr. ≤5 mrem in a qtr. ≤10 mrem in a yr.
3.2.3	Liquid Radwaste Treatment Operability	Total Body Dose Organ Dose	Eq. 6-1 Eq. 6-3 ≤0.06 mrem in a mo. ≤0.2 mrem in a mo.
3.3.1	Gaseous Effluents Dose Rate	Organ Dose Rate from Tritium and Particulates with T _{1/2} >8 Days	Eq. 6-16 Eq. 6-40 ≤1500 mrem/yr.
3.3.3	Gaseous Effluents Dose from Tritium and Particulates	Organ Dose from Tritium and Particulates with T _{1/2} >8 Days	Eq. 6-25 Eq. 6-44 ≤ 7.5 mrem in a qtr. ≤15 mrem in a yr.
3.4.1	Total Dose (from All Sources)	Total Body Dose	Footnote (2) ≤25 mrem in a yr.
		Organ Dose	No Thyroid ≤25 mrem in a yr.

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

(2) Effluent Control 3.4.1 requires this evaluation only if twice the limit of Equations 6-1, 6-3, or 6-25 is reached. If this occurs a Method II calculation shall be made considering available information for pathways of exposure to real individuals from liquid, gaseous, and direct radiation sources.

TABLE 1.1.2

Summary of Methods to Calculate
Unrestricted Area Liquid Concentrations

Equation Number	Category	Equation	Reference Section
5-1	Sum of the Fractions of Combined Effluent Concentrations in Liquids [Except Noble Gases]	$F_1^{ENG} = \sum_i \frac{C_{pi}}{ECL_i} \leq 10$	5.1

TABLE 1.1.3

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

Equation Number	Category	Equation	Reference Section
6-1	Total Body Dose	$D_{tb}(\text{mrem}) = \sum_i Q_i \text{DFL}_{itb}$	6.2.1
6-3	Maximum Organ Dose	$D_{mo}(\text{mrem}) = \sum_i Q_i \text{DFL}_{imo}$	6.3.1

TABLE 1.1.4

Summary of Methods to Calculate
Dose Rates

Equation Number	Category	Equation	Reference Section
6-16	Critical Organ Dose Rate from Stack Release of Tritium and Particulates with $T_{1/2} > 8$ Days	$\dot{R}_{\text{cos}} \left(\frac{\text{mrem}}{\text{yr}} \right) = \sum_i \dot{Q}_i^{\text{STP}} \text{DFG}'_{\text{sico}}$	6.6.1
6-40	Critical Organ Dose Rate from Ground Level Release of Tritium and Particulates with $T_{1/2} > 8$ Days	$\dot{R}_{\text{cog}} \left(\frac{\text{mrem}}{\text{yr}} \right) = \sum_i \dot{Q}_i^{\text{GLP}} \text{DFG}'_{\text{gico}}$	6.6.1

TABLE 1.1.5

Summary of Methods to Calculate
Doses to Air from Noble Gases

Deleted

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
6-25	Dose to Critical Organ from Stack Release of Tritium and Particulates	$D_{\text{cos}}(\text{mrem}) = \sum_i Q_i^{\text{STP}} \text{DFG}_{\text{sico}}$	6.9.1
6-44	Dose to Critical Organ from Ground Level Release of Tritium and Particulates	$D_{\text{cog}}(\text{mrem}) = \sum_i Q_i^{\text{GLP}} \text{DFG}_{\text{gico}}$	6.9.1
	Deleted		

TABLE 1.1.7

Summary of Methods for
Setpoint Determinations

Deleted

TABLE 1.1.8Effluent and Environmental Controls Cross-Reference

Control Topic	Original Technical Specification Section	Revised ODCM Control Section
INSTRUMENTATION		
Radioactive Liquid Effluent Instrumentation	3/4.9.A	3/4.1.1
Effluent instrumentation list	Table 3.9.1	Table 3.1.1
Instrument surveillance requirements	Table 4.9.1	Table 4.1.1
Radioactive Gaseous Effluent Instrumentation	3/4.9.B	3/4.1.2
Effluent instrumentation list	Table 3.9.2	Table 3.1.2
Instrumentation requirements	Table 4.9.2	Table 4.1.2
RADIOACTIVE LIQUID EFFLUENTS		
Concentration	3/4.8.A	3/4.2.1
Liquid waste sampling & analysis program	Table 4.8.1	Table 4.2.1
Dose – Liquids	3/4.8.B	3/4.2.2
Liquid Radwaste Treatment	3/4.8.C	3/4.2.3
RADIOACTIVE GASEOUS EFFLUENTS		
Dose Rate	3/4.8.E	3/4.3.1
Gaseous waste sampling & analysis program	Table 4.8.2	Table 4.3.1
Dose from Tritium and Radionuclides in Particulate Form	3/4.8.G	3/4.3.3
TOTAL DOSE		
Total Dose	3/4.8.M	3/4.4.1

TABLE 1.1.8 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING		
Radiological Environmental Monitoring Program	3/4.9.C	3/4.5.1
Listing of required monitoring criteria	Table 3.9.3	Table 3.5.1
Reporting levels for radioactivity in samples	Table 3.9.4	Table 3.5.2
Detector capability for environmental analysis	Table 4.9.3	Table 4.5.1
Land Use Census	3/4.9.D	3/4.5.2
Intercomparison Program	3/4.9.E	3/4.5.3
EFFLUENT CONTROL BASES	Bases: 3.8 & 3.9	3/4.6
UNIQUE REPORTING REQUIREMENTS		
Annual Radioactive Effluent Release Report	6.7.C.1	10.1
Environmental Radiological Monitoring	6.7.C.3	10.2
Special Reports	6.7.C.2	10.3
Major Changes to Radioactive Liquid, Gaseous, and Solid Waste Treatment Systems	6.14	10.4

TABLE 1.1.9

(Deleted)

TABLE 1.1.10

Dose Factors Specific for Vermont Yankee
for Noble Gas Releases

Deleted

TABLE 1.1.10A

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

Deleted

TABLE 1.1.11

Dose Factors Specific for Vermont Yankee
for Liquid Releases

Radionuclide	Total Body Dose Factor $DF_{litb} \left(\frac{\text{mrem}}{\text{Ci}} \right)$	Maximum Organ Dose Factor $DF_{limo} \left(\frac{\text{mrem}}{\text{Ci}} \right)$
H-3	2.06E-04	2.06E-04
Mn-54	2.08E-01	3.00E+00
Fe-55	4.18E-02	2.54E-01
Co-60	2.13E-01	1.28E+00
Zn-65	8.06E+00	1.64E+01
Sr-90	4.23E+01	1.67E+02
Zr-95	4.21E-04	1.36E-01
Ag-110m	6.90E-03	7.02E-01
Sb-125	7.52E-03	1.15E-01
Cs-134	1.28E+02	1.60E+02
Cs-137	7.58E+01	1.21E+02

TABLE 1.1.12
Dose and Dose Rate Factors Specific for Vermont Yankee
for Tritium and Particulate Releases

Nuclide	Stack		Ground ⁽¹⁾	
	Dose mrem/Ci	Dose Rate mrem/yr per μ Ci/sec	Dose mrem/Ci	Dose Rate mrem/yr per μ Ci/sec
H-3	1.11E-04	3.52E-03	6.90E-03	2.18E-01
C-14	1.18E-01	3.71E+00	7.29E+00	2.30E+02
Mn-54	8.10E-02	2.55E+00	3.25E+00	1.03E+02
Fe-55	8.44E-02	2.66E+00	3.37E+00	1.06E+02
Co-60	3.10E-01	9.77E+00	1.24E+01	3.92E+02
Ni-59	1.43E-02	4.50E-01	5.69E-01	1.79E+01
Ni-63	4.30E+00	1.36E+02	1.71E+02	5.39E+03
Zn-65	8.52E-01	2.69E+01	3.39E+01	1.07E+03
Sr-90	1.46E+02	4.59E+03	5.81E+03	1.83E+05
Tc-99	4.13E-01	1.30E+01	1.64E+01	5.19E+02
Sb-125	1.44E-01	4.54E+00	5.76E+00	1.82E+02
I-129	1.47E+02	4.64E+03	5.86E+03	1.85E+05
Cs-134	3.62E+00	1.14E+02	1.44E+02	4.55E+03
Cs-137	4.50E-01	1.42E+01	1.79E+01	5.64E+02
Sm-151	4.00E-02	1.26E+00	1.60E+00	5.05E+01
Eu-152	1.65E-01	5.20E+00	6.59E+00	2.08E+02
Eu-154	3.75E-01	1.18E+01	1.50E+01	4.74E+02
Eu-155	6.67E-01	2.10E+01	2.66E+01	8.40E+02
Pu-238	2.50E+02	7.90E+03	1.52E+04	4.79E+05
Pu-239	2.89E+02	9.11E+03	1.75E+04	5.52E+05
Pu-241	6.25E+00	1.97E+02	3.79E+02	1.19E+04
Am-241	2.96E+02	9.34E+03	1.79E+04	5.65E+05
Cm-242	1.11E+01	3.50E+02	6.77E+02	2.13E+04
Cm-243	1.99E+02	6.28E+03	1.21E+04	3.81E+05
Cm-244	1.54E+02	4.87E+03	9.33E+03	2.94E+05

NOTE 1 -Releases from the Reactor Building are considered ground releases.

These dose and dose rate factors apply to the maximum offsite X/Qs and D/Qs. Releasee from the Reactor Building are ground releases. The dose and dose rate factors or for the maximum organ dose when the Vegetable, Meat, Cow Milk, and Inhalation pathways are summed.

2.0 DEFINITIONS

This section lists definitions (Table 2.1.1) and dose equation variable names (Table 2.1.2) which are utilized in the VY ODCM.

TABLE 2.1.1
Definitions

1. Deposition Coefficient - DOQ or D/Q (i.e., deposition factor); is the average relative deposition per unit area at a given point in a given sector of an effluent species (particulate or radioiodine) considering depletion of a plume during transport by dry deposition and radioactive decay (See RG 1.109 and RG 1.111), units are in 1/m².
2. Dispersion Coefficient - XOQ or X/Q (i.e., dispersion factor); is the annual average effluent concentration (corrected as necessary for radioactive decay) at a specified distance and sector normalized by source strength (see RG 1.109 and RG 1.111), units are in sec/m³.
3. Groundwater – For purposes of the ODCM, groundwater is defined as subsurface water which is either shallow, deep or in bedrock layers. Shallow and deep groundwater wells are sampled to determine the flow rate and contamination concentrations of groundwater flowing to the Connecticut River above or on the bedrock layer. Bedrock groundwater wells, utilized for drinking water purposes both on and off the plant site, are monitored for radioactive contamination as part of the REMP.
4. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.
5. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time as specified is not part of the routine instrument calibration but will be checked once per operating cycle.
6. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.
7. Instrument Functional Test - An instrument functional test shall be:
 - a. Analog channels - the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.
 - b. Bistable channels - the injection of a signal into the sensor to verify the functionality including alarm and/or trip functions.

TABLE 2.1.1 (Continued)

8. Intercepted Groundwater – For the purposes of the ODCM, intercepted groundwater is defined as groundwater which has infiltrated Turbine Building sumps or trenches which have no contact with water from plant systems and groundwater removed from the saturated zone soil adjacent to or underlying structures, equipment, excavations, etc. to reduce groundwater infiltration.
9. Liquid Waste Discharge – For the purposes of the ODCM, liquid waste discharges are plant process water treated with the liquid radwaste system and discharged through the liquid radwaste effluent outfall, using the installed flow monitor and pre-discharge grab sample results.
10. Off-Site Dose Calculation Manual (ODCM) – A manual containing the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous effluent monitoring alarm/trip setpoints, and in the conduction of the environmental radiological monitoring program. The ODCM shall also contain (1) the Radioactive Effluent Controls (including the Radiological Environmental Monitoring) Program required by QAPM Appendix D, Section 2.5.B and (2) descriptions of the information that should be included in the annual Radioactive Effluent Release Report and Annual Radiological Environmental Operating Report required by QAPM Appendix D, Section 2.4.A and 2.4.B, respectively.
11. Site Boundary – The site boundary is shown in Plant Drawing 5920-6245.
12. Source Check - The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.
13. Streamtube(s) – Defined as flows of subsurface groundwater (having discrete width, depth and flow rate characteristics) in either the shallow or deep layers of permeable soils above the bedrock layer at the plant site. Streamtube flows are from west to east, towards the Connecticut River, and are assumed to discharge into the Connecticut River.
14. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%. The operating cycle interval is considered to be 18 months and the tolerance stated above is applicable.
15. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be functional. These tests unless otherwise stated in these specifications may be waived when the instrument,

TABLE 2.1.1 (Continued)

component, or system is not required to be functional, but these tests shall be performed on the instrument, component, or system prior to being required to be functional.

16. Deleted.
17. Vent/Purging – Vent/purging is the controlled process of discharging air or gas from the primary containment to control temperature, pressure, humidity, concentration or other operating conditions.

TABLE 2.1.2Summary of Variables

Variable		Definition	Units
C_{di}	=	Concentration of radionuclide "i" at the point of liquid discharge to an unrestricted area.	$\frac{\mu\text{Ci}}{\text{ml}}$
C_i	=	Concentration of radionuclide "i."	$\frac{\mu\text{Ci}}{\text{cc}}$
C_{pi}	=	Concentration, exclusive of noble gases, of radionuclide "i" from tank "p" at point of discharge to an unrestricted area.	$\frac{\mu\text{Ci}}{\text{ml}}$
C_{mi}	=	Concentration of radionuclide "i" in mixture at the monitor.	$\frac{\mu\text{Ci}}{\text{ml}}$
D_{cos}	=	Dose to critical organ from stack release.	mrem
D_{cog}	=	Dose to the critical organ from ground level release.	mrem
D_{mo}	=	Dose to the maximum organ.	mrem
D^S	=	Dose to skin from beta and gamma.	mrem
D_{tb}	=	Dose to the total body.	mrem
DF	=	Dilution factor.	ratio
DF_{min}	=	Minimum allowable dilution factor.	ratio
DFB_i	=	Total body gamma dose factor for nuclide "i."	$\frac{\text{mrem} - \text{m}^3}{\text{pCi} - \text{yr}}$
DFB_c	=	Composite total body dose factor.	$\frac{\text{mrem} - \text{m}^3}{\text{pCi} - \text{yr}}$
DFL_{itb}	=	Site-specific, total body dose factor for a liquid release of nuclide "i."	$\frac{\text{mrem}}{\text{Ci}}$
DFL_{imo}	=	Site-specific, maximum organ dose factor for a liquid release of nuclide "i."	$\frac{\text{mrem}}{\text{Ci}}$

TABLE 2.1.2 (Continued)

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."	$\frac{mrem - sec}{\mu Ci - yr}$
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."	$\frac{mrem - sec}{\mu Ci - yr}$
\dot{R}_{cos}	=	Critical organ dose rate due to particulates released from stack.	$\frac{mrem}{yr}$
\dot{R}_{cog}	=	Critical organ dose rate due to particulates released from ground.	$\frac{mrem}{yr}$
D/Q	=	Deposition factor for dry deposition of particulates.	$\frac{1}{m^2}$
F_d	=	Dilution Flow rate.	gpm
F_w	=	Waste Flow rate from liquid waste processing sample tank.	gpm
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" (10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 2)	$\frac{\mu Ci}{cc}$
Q_i	=	Release for radionuclide "i" from the point of interest.	curies
\dot{Q}_i	=	Release rate for radionuclide "i" at the point of interest.	$\frac{\mu Ci}{sec}$
\dot{Q}_i^{STP}	=	The tritium and particulate radionuclide "i" release rate from the plant stack.	$\frac{\mu Ci}{sec}$

TABLE 2.1.2 (Continued)

Variable		Definition	Units
\dot{Q}_i^{GLP}	=	The tritium and particulate radionuclide "i" release rate from ground level.	$\frac{\mu\text{Ci}}{\text{sec}}$
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 tritium and particulate radionuclide "i" from the plant stack.	curies
Q_i^{GLP}	=	The release of tritium and particulate radionuclide "i" from ground level.	curies
X/Q_s		Annual or long-term average undepleted atmospheric dispersion factor for stack release.	$\frac{\text{sec}}{\text{m}^3}$
X/Q_g		Annual or long-term average undepleted atmospheric dispersion factor for ground level release.	$\frac{\text{sec}}{\text{m}^3}$

3/4.0 EFFLUENT AND ENVIRONMENTAL CONTROLS

This section includes the effluent and environmental controls that were originally part of the Vermont Yankee Technical Specifications. These controls were relocated into the ODCM without any substantial changes, in accordance with NRC Generic Letter 89-01. Text and tables were reformatted to the style of the ODCM. The various controls were renumbered from the original numbering scheme of the Technical Specifications. A cross-reference of the old Technical Specifications section to the new ODCM section is presented in Table 1.1.8.

3/4.1 INSTRUMENTATION

3/4.1.1 Liquid Effluent Monitoring Instrumentation

Deleted.

TABLE 3.1.1

Liquid Effluent Monitoring Instrumentation

Deleted

TABLE 3.1.1 NOTATION

NOTE 1 - Deleted

NOTE 2 - Deleted

NOTE 3 - Deleted

NOTE 4 - Deleted

TABLE 4.1.1

Deleted

TABLE 4.1.1 NOTATION

Deleted

3/4.1 INSTRUMENTATION

3/4.1.2 Radioactive Gaseous Effluent Instrumentation

CONTROLS

- 3.1.2 The gaseous process and effluent monitoring instrumentation channels shall be functional in accordance with Control Table 3.1.2.

APPLICABILITY:

As shown in Table 3.1.2.

ACTION:

- b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels functional, take actions noted in Table 3.1.2.

SURVEILLANCE REQUIREMENTS

- 4.1.2.a Each gaseous process or effluent monitoring instrumentation channel shall be tested and calibrated as indicated in Table 4.1.2.

TABLE 3.1.2

Gaseous Effluent Monitoring Instrumentation

Instrument		Minimum Channels Functional	Notes
3.	Plant Stack		
a.	Particulate Sampler Filter	1	4, 5
b.	Sampler Flow Integrator (FI-17-156/157)	1	1, 5
c.	Stack Flow Rate Monitor (FI-108-22)	1	1, 5

TABLE 3.1.2 NOTATION

- NOTE 1 - With the number of channels functional less than required by the minimum channels functional requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.
- NOTE 2 - Deleted
- NOTE 3 - Deleted
- NOTE 4 - With the number of channels functional less than required by the minimum channels functional requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment.
- NOTE 5 - With the number of channels functional less than required by the minimum channels functional requirement, exert reasonable efforts to return the instrument(s) to functional status within 30 days.
- NOTE 6 - Deleted
- NOTE 7 - Deleted
- NOTE 8 - Deleted
- NOTE 9 - Deleted
- NOTE 10 - Deleted

TABLE 4.1.2

Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
3. Plant Stack				
a. Sampler Flow Integrator	Once each week	Not Applicable	Once each 18 months	Not Applicable
b. System Flow Rate Monitor	Once each day	Not Applicable	Once each 18 months	Once each quarter

TABLE 4.1.2 NOTATION

- (1) Deleted
- (2) Deleted
- (3) Deleted
- (4) Deleted

3/4.2 RADIOACTIVE LIQUID EFFLUENTS

3/4.2.1 Liquid Effluent Concentration

CONTROLS

- 3.2.1 The concentration of radioactive material in liquid effluents released in liquid waste effluents, intercepted groundwater released via liquid waste effluent line, storm drain or groundwater flowing to the Connecticut River from the site in radioactive concentrations above background (Unrestricted Areas for liquids is at the point of discharge from the plant discharge in Connecticut River) shall be limited to 10 times the concentrations specified in Appendix B to 10 CFR Part 20.1001 – 20.2402, Table 2, Column 2, for radionuclides other than noble gases.

APPLICABILITY:

At all times.

ACTION:

With the concentration of radioactive material in liquid effluents released to Unrestricted Areas exceeding the limits of Control 3.2.1, immediately take action to decrease the release rate of radioactive materials and/or increase the dilution flow rate to restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

- 4.2.1.a Radioactive material in liquid waste, intercepted groundwater releases, and subsurface groundwater flows to the Connecticut River shall be sampled and analyzed in accordance with requirements of Table 4.2.1.
- 4.2.1.b The results of the analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release to Unrestricted Areas are limited to the values in Control 3.2.1.

TABLE 4.2.1Radioactive Liquid Sampling and Analysis Program

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml) ^(a)
Batch Waste Processing Sample Tanks ^{(b)(k)}	Prior to each release Each Batch	Prior to each release Each Batch	Principal Gamma Emitters ^(d)	5×10^{-7}
	Prior to each release Each Batch	Once per month Composite ^(c)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Prior to each release Each Batch	Once per quarter Composite ^(c)	Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
Groundwater Interception Release Tanks ^(b)	Prior to each release Each Batch	Prior to each release Each Batch	Principal Gamma Emitters ^(d) H-3	Activity Analysis LLDs ^(e)
	Prior to each release Each Batch	Once per month Composite ^(c)	H-3 ^(e) Gross Alpha	Activity Analysis LLDs ^(e)
	Prior to each release Each Batch	Once per quarter Composite ^(c)	H-3 ^(e) Gamma Emitters ^(e) Ni-63 ^(e) Fe-55 ^(e) Sr-90 ^(e) Alpha Spec ^(e)	Activity Analysis LLDs ^(e)

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml) ^(a)
Storm Drain	Weekly Grab	Weekly	H-3	Activity Analysis LLDs ^(e)
Subsurface Groundwater Flows to the Connecticut River ^(f)	Monitoring Wells – according to the Groundwater Monitoring Plan ^{(g)(i)} Sentinel Wells- according to the Groundwater Monitoring Plan ^{(h)(j)}	Monitoring Wells – Annual ⁽ⁱ⁾ Sentinel Wells – Annual ^(j)	H-3 ^(e) Gamma Emitters ^(e) Ni-63 ^(e) Fe-55 ^(e) Sr-90 ^(e) Alpha Spec ^(e)	Activity Analysis LLDs ^(e)

TABLE 4.2.1 NOTATION

- a. The LLD is the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66 * S_b}{E * V * K * Y * e^{-\lambda * \Delta t}}$$

where:

LLD = the lower limit of detection as defined above (microcuries or picocuries/unit mass or volume)

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

E = the counting efficiency (counts/disintegration)

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

K = 2.22×10^6 disintegrations/minute/microcurie or 2.22 disintegration/minute/picocurie as applicable

Y = the fractional radiochemical yield (when applicable)

λ = the radioactive decay constant for the particular radionuclide (/minute)

Δt = the elapsed time between sample collection and analysis (minutes)

Typical values of E , V , Y and Δt can be used in the calculation. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples.

Analysis shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unavailable.

It should be recognized that the LLD is defined as a "before the fact" limit representing the capability of a measurement system and not as an "after the fact" limit for a particular measurement. This does not preclude the calculation of an "after the fact" LLD for a particular measurement based upon the actual parameters for the sample in question and appropriate decay correction parameters such as decay while sampling and during analysis.

TABLE 4.2.1 NOTATION
(Cont'd)

- b. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analysis, each batch shall be isolated and then thoroughly mixed to assure representative sampling.
- c. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.
- d. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Co-60, Zn-65, Cs-134, Cs-137 and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level, but as "not detected." When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Radioactive Effluent Release Report.
- e. At a minimum, each subsurface groundwater sample shall be analyzed for the following analytes. Lower Limit of Detection for each analyte is as follows:

STANDARD RADIONUCLIDES	LLD (PCI/L)
³ H	2000
⁵⁴ Mn	15
⁶⁰ Co	15
⁶⁵ Zn	30
⁹⁵ Zr	30
¹³⁴ Cs	15
¹³⁷ Cs	18

TABLE 4.2.1 NOTATION
(Cont'd)

If tritium or plant-generated gamma activity is positively detected, then a sample should be further analyzed for the presence of the following Hard-To-Detect (HTD) radionuclides, as a minimum, using the associated LLD values:

HTD RADIONUCLIDES	LLD (PCI/L)
^{55}Fe	110
^{63}Ni	530
^{90}Sr	3.5
Alpha Emitters	15

- f. ODCM Section 9.3 further defines the location and determination of flow through the plant site groundwater streamtubes.
- g. Monitoring wells used to measure flow and concentration of contaminants in streamtubes: GZ-1s, GZ-3s, GZ-4s, GZ-5s, GZ-13s, GZ-13d, GZ-14s, GZ-14d, GZ-18s, GZ-18d, GZ-19s and GZ-19d.
- h. Sentinel used to measure flow and concentration of contaminants in streamtubes: GZ-27s, GZ-26s, GZ-25s, GZ-23s and GZ-22d.
- i. The Groundwater Monitoring Plan is used to determine sample frequency of Monitoring wells based on analysis trends and risk of SSCs being monitored.
- j. The Groundwater Monitoring Plan is used to determine sample frequency of Sentinel wells. Sentinel wells with analysis results >LLD are sampled monthly and wells with analysis results < LLD are sampled according to the guidance in the procedure.
- k. At least two independent samples are analyzed in accordance with Control 4.2.1, and at least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valve line-up.

3/4.2 RADIOACTIVE LIQUID EFFLUENTS

3/4.2.2 Dose - Liquids

CONTROLS

3.2.2 The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released to Unrestricted Areas shall be limited to the following:

- a. During any calendar quarter:

 less than or equal to 1.5 mrem to the total body, and
 less than or equal to 5 mrem to any organ, and
- b. During any calendar year:

 less than or equal to 3 mrem to the total body, and
 less than or equal to 10 mrem to any organ.

APPLICABILITY:

At all times.

ACTION:

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to ODCM Section 10, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENTS

4.2.2 Cumulative dose contributions shall be determined in accordance with the methods in the ODCM at least once per month if releases during the period have occurred.

3/4.2 RADIOACTIVE LIQUID EFFLUENTS

3/4.2.3 Liquid Radwaste Treatment

CONTROLS

- 3.2.3 The liquid radwaste treatment system shall be used in its designed modes of operation to reduce the radioactive materials in the liquid waste prior to its discharge when the projected doses due to the liquid effluents released to Unrestricted Areas, when averaged with all other liquid releases over the last month, would exceed 0.06 mrem to the total body, or 0.2 mrem to any organ.

APPLICABILITY:

At all times.

ACTION:

With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, a Special Report that includes the information detailed in ODCM Section 10.4.1.

SURVEILLANCE REQUIREMENTS

- 4.2.3.a See Control 4.2.2.
- 4.2.3.b The liquid radwaste treatment system schematic is shown in ODCM Figure 9.1.

3/4.3 RADIOACTIVE GASEOUS EFFLUENTS

3/4.3.1 Gaseous Effluents Dose Rate

CONTROLS

3.3.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following:

- b. For tritium and radionuclides in particulate form with half-lives greater than 8 days; less than or equal to 1,500 mrem/yr to any organ.

APPLICABILITY:

At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately take action to decrease the release rate to within the limits of Control 3.3.1.

SURVEILLANCE REQUIREMENTS

4.3.1.b The dose rate due to tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Control 3.3.1 in accordance with the methods in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.3.1.

TABLE 4.3.1Radioactive Gaseous Waste Sampling And Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml) ^(a)
B. Containment Purge	Prior to each release/ Each Purge Grab Sample for Particulates	Prior to each release/ Each Purge	Principal Gamma Emitters ^(d,g)	1×10^{-9} ^(g)
C. Main Plant Stack	Continuous Sample	Once per month	H-3	1×10^{-6}
	Continuous ^(e)	Once per week ^(b) Particulate Sample	Principal Gamma Emitters ^(d,g)	1×10^{-11}
	Continuous ^(e)	Once per month Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous ^(e)	Once per quarter Composite Particulate Sample	Sr-90	1×10^{-11}

TABLE 4.3.1 NOTATION

- a. See footnote a. of Table 4.2.1.
- b. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after removal from samplers.
- c. Deleted
- d. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Co-60, Zn-65, Cs-134, Cs-137 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below LLD for the analyses should not be reported as being present at the LLD level for that nuclide, but as "not detected." When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Radioactive Effluent Release Report.
- e. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Controls 3.3.1 and 3.3.3.
- f. Deleted
- g. Lower Limit of Detection (LLD) applies only to particulate form radionuclides identified in Table Notation d. above.

3/4.3 RADIOACTIVE GASEOUS EFFLUENTS

3/4.3.2 Deleted

CONTROLS

3.3.2 Deleted

APPLICABILITY:

Deleted

ACTION:

Deleted

SURVEILLANCE REQUIREMENTS

4.3.2 Deleted

3/4.3 RADIOACTIVE GASEOUS EFFLUENTS

3/4.3.3 Dose –Radioactive Material in Particulate Form and Tritium

CONTROLS

3.3.3 The dose to a member of the public from tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the site to areas at and beyond the site boundary shall be limited to the following:

- a. During any calendar quarter:

less than or equal to 7.5 mrem to any organ, and
- b. During any calendar year:

less than or equal to 15 mrem to any organ.

APPLICABILITY:

At all times.

ACTION:

With the calculated dose from the release of tritium and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to ODCM Section 10.4.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENTS

4.3.3 Cumulative dose contributions for the total time period shall be determined in accordance with the methods in the ODCM at least once every month.

3/4.3 RADIOACTIVE GASEOUS EFFLUENTS

3/4.3.4 Deleted

3/4.3.5 Deleted

3/4.3.6 Deleted

3/4.3.7 Deleted

3/4.4 TOTAL DOSE

3/4.4.1 Total Dose (40 CFR 190)

CONTROLS

3.4.1 The dose or dose commitment to a member of the public* in areas at and beyond the Site Boundary from all station sources is limited to less than or equal to 25 mrem to the total body or any organ over a calendar year.

APPLICABILITY:

At all times.

ACTION:

With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Controls 3.2.2.a, 3.2.2.b, 3.3.3.a, or 3.3.3.b, calculations should be made, including direct radiation contributions from the station to determine whether the above limits of Control 3.4.1 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days a Special Report that includes the information detailed in ODCM Section 10.4.3.

SURVEILLANCE REQUIREMENTS

4.4.1.a Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Controls 4.2.2 and 4.3.3.

4.4.1.b Cumulative dose contributions from direct radiation from plant sources shall be determined in accordance with the methods in the ODCM. This requirement is applicable only under conditions set forth in Control 3.4.1 Action Statement.

* Note: For this Control, a member of the public may be taken as a real individual accounting for ones actual activities.

3/4.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.5.1 Environmental Monitoring Program

CONTROLS

3.5.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.5.1.

APPLICABILITY:

At all times.

ACTION:

- a. With the radiological environmental monitoring program not being conducted as specified in Tables 3.5.1 or 4.5.1, prepare and submit to the Commission, in the Annual Radiological Environmental Monitoring Report (per ODCM Section 10.2), a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.
- b. With the level of radioactivity as the result of plant effluents in an environmental sampling media at one or more locations specified in Control Table 3.5.1 exceeding the reporting levels of Control Table 3.5.2, prepare and submit to the Commission a Special Report within 30 days from receipt of the laboratory analysis (per ODCM Section 10.4.4).

SURVEILLANCE REQUIREMENTS

4.5.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.5.1 from the locations given in the ODCM and shall be analyzed pursuant to the requirements of Table 3.5.1 and the detection capabilities required by Table 4.5.1.

TABLE 3.5.1Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample Locations ^(a)	Sampling and Collection Frequency	Type and Frequency of Analysis
<p>1. AIRBORNE</p> <p>a. Particulates</p>	<p>Samples from 3 locations:</p> <p>1 sample from up valley, within 4 miles (~6.4 km) of Site Boundary. (major wind direction)</p> <p>1 sample from down valley, within 4 miles (~6.4 km) of Site Boundary. (major wind direction)</p> <p>1 sample from a control location.</p>	<p>Continuous operation of sampler with sample collection monthly or more frequently as required by dust loading.</p>	<p>Particulate sampler: Gross beta radioactivity analysis on each sample following filter change. ^(c) Composite (by location) ^(d) for gamma isotopic at least once per quarter.</p>

TABLE 3.5.1
(Cont'd)

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample Locations ^(a)	Sampling and Collection Frequency	Type and Frequency of Analysis
2. DIRECT RADIATION ^(b)	<p>Routine monitoring stations as follows:</p> <p>10 REMP stations (in representative meteorological sectors on land) located along the site boundary;</p> <p>Additional stations to be placed in special interest areas and control station areas.</p>	Quarterly.	<p>Gamma dose, at least once per quarter.</p> <p>Incident response TLDs de-dose only quarterly unless gaseous release Controls were exceeded in period.</p>

TABLE 3.5.1
(Cont'd)

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample Locations ^(a)	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface ^(e)	1 sample upstream.	Monthly grab sample.	Gamma isotopic analysis ^(d) of each sample. Tritium analysis of composite sample at least once per quarter.
	1 sample downstream.	Monthly grab sample.	Gamma isotopic analysis ^(d) of each sample. Tritium analysis of composite sample at least once per quarter.
b. Ground (potable – drinking water from bedrock wells)	1 sample from within 8 km (5 miles) distance.	Quarterly.	Gamma isotopic ^(d) and tritium analyses of each sample.
	1 sample from a control location.	Quarterly.	Gamma isotopic ^(d) and tritium analyses of each sample.
c. Sediment from Shoreline	1 sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis ^(d) of each sample.
	1 sample from north storm drain outfall.	Semiannually.	Gamma isotopic analysis ^(d) of each sample.

TABLE 3.5.1
(Cont'd)

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample Locations ^(a)	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Deleted			
b. Fish	1 sample of two recreationally important species in vicinity of plant discharge area.	Semiannually.	Gamma isotopic analysis ^(d) on edible portions.
	1 sample (preferably of same species) in areas not influenced by plant discharge.	Semiannually.	Gamma isotopic analysis ^(d) on edible portions.
c. Vegetation	1 grass sample at each air sampling station.	Quarterly when available.	Gamma isotopic analysis ^(d) of each sample.
	1 silage sample at each former milk sampling station (as available).	Quarterly when available.	Gamma isotopic analysis ^(d) of each sample.

TABLE 3.5.1 NOTATION

- a Specific parameters of distance and direction sector from the centerline of the reactor and additional descriptions where pertinent, shall be provided for each and every sample location in Table 3.5.1 in a table and figure(s) in the ODCM (Section 7). Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every reasonable effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to ODCM Section 10.2. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a Licensee Event Report and pursuant to ODCM Section 10.1, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).
- b One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a Thermoluminescent Dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 10 stations is not an absolute number. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.
- c Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thorium daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.
- d Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.
- e The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone.
- f Deleted.
- g Deleted.
- h Deleted.

TABLE 3.5.2

Reporting Levels For Radioactivity Concentrations In Environmental Samples^(a)
Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/Kg, wet)	Vegetation (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
H-3	$2 \times 10^{4(b)}$				
Mn-54	1×10^3		3×10^4		$3 \times 10^{3(c)}$
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-95	4×10^2				
Cs-134	30	10	1×10^3	1×10^3	
Cs-137	50	20	2×10^3	2×10^3	

- (a) Reporting levels may be averaged over a calendar quarter. When more than one of the radionuclides in Table 3.5.2 are detected in the sampling medium, the unique reporting requirements are not exercised if the following condition holds:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \dots < 1.0$$

When radionuclides other than those in Table 3.5.2 are detected and are the result of plant effluents, the potential annual dose to a member of the public must be less than or equal to the calendar year limits of Controls 3.2.2 and 3.3.1.

- (b) Reporting level for drinking water pathways. For nondrinking water pathways, a value of 3×10^4 pCi/l may be used.
- (c) Reporting level for individual grab samples taken at North Storm Drain Outfall only.
- (d) Deleted

TABLE 4.5.1Detection Capabilities For Environmental Sample Analysis^{(a)(c)}

Analysis ^(d)	Water (pCi/l)	Airborne Particulate or Gas (pCi/m ³)	Fish (pCi/Kg, wet)	Vegetation (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
Gross beta	4	0.01			
H-3	2000 ^(h)				
Mn-54	15		130		
Co-60	15		130		
Zn-65	30		260		
Zr-95	15				
Cs-134	15	0.05	130	60	150
Cs-137	18	0.06	150	80	180

TABLE 4.5.1 NOTATION

- (a) See Footnote (a) of Table 4.2.1.
- (b) Deleted
- (c) If the measured concentration minus the 5 sigma counting statistics is found to exceed the specified LLD, the sample does not have to be analyzed to meet the specified LLD.
- (d) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the listed nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to QAPM, Appendix D, Section 2.4.B and ODCM Section 10.2.
- (e) Deleted
- (f) Deleted.
- (g) Deleted
- (h) If no drinking water pathway exists, then a value of 3000 picocuries per liter may be used.

3/4.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.5.2 Land Use Census

CONTROLS

- 3.5.2 A land use census shall be conducted to identify the location of the nearest residence in each of the 16 meteorological sectors within a distance of five miles.

APPLICABILITY:

At all times.

ACTION:

- a. With a land use census identifying one or more locations which yield a calculated dose or dose commitment (via the same exposure pathway) at least 20 percent greater than at a location from which samples are currently being obtained in accordance with Control 3.5.1, add the new location(s) to the radiological environmental monitoring program within 30 days if permission from the owner to collect samples can be obtained, and sufficient sample volume is available. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.
- b. With the land census not being conducted as required above, prepare and submit to the Commission within 30 days a Special Report that includes information detailed in ODCM Section 10.4.5.

SURVEILLANCE REQUIREMENTS

- 4.5.2 The land use census shall be conducted at least once per year between the dates of June 1 and October 1 by either a door-to-door survey, aerial survey, or by consulting local agricultural authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to QAPM, Appendix D, Section 2.4.B and ODCM Section 10.2.

3/4.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.5.3 Interlaboratory Comparison Program

CONTROLS

- 3.5.3 Analyses shall be performed on referenced radioactive materials supplied as part of an Interlaboratory Program which has been approved by NRC.

APPLICABILITY:

At all times.

ACTION:

With analysis not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to ODCM Section 10.2

SURVEILLANCE REQUIREMENTS

- 4.5.3 A summary of the results of analyses performed as part of the above required Interlaboratory Program shall be included in the Annual Radiological Environmental Operating Report. NRC-approved interlaboratory programs utilized by environmental laboratories in processing Vermont Yankee samples shall be identified in the ODCM.

3/4.6 EFFLUENT AND ENVIRONMENTAL CONTROL BASES

INSTRUMENTATION

Gaseous Effluent Instrumentation (3.1.2)

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments are provided to ensure that the alarm/trip will occur prior to exceeding design bases dose rates identified in Control 3.3.1.

RADIOACTIVE EFFLUENTS

Liquid Effluents: Concentration (3.2.1)

This Control is provided to ensure that at any time the concentration of radioactive materials released in liquid waste effluents, intercepted groundwater released via liquid waste effluent line, storm drain or groundwater flowing to the Connecticut River from the site in radioactive contamination concentrations above background (Unrestricted Area for liquids is at the point of discharge from the plant discharge into Connecticut River) will not exceed 10 times the concentration levels specified in 10 CFR Part 20.1001-20.2402, Appendix B, Table 2, Column 2. These requirements provide operational flexibility, compatible with considerations of health and safety, which may temporarily result in releases higher than the absolute value of the concentration numbers in Appendix B, but still within the annual average limitation of the Regulation. Compliance with the design objective doses of Section II.A of Appendix I to 10 CFR Part 50 assure that doses are maintained ALARA, and that annual concentration limits of Appendix B to 10 CFR 20.1001-20.2402 will not be exceeded.

3/4.6 EFFLUENT AND ENVIRONMENTAL CONTROL BASES
(cont.)

Liquid Effluents: Dose (3.2.2)

This Control is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The requirements provide operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I, i.e., that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. In addition, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in potable drinking water that are in excess of the requirements of 40 CFR 141. No drinking water supplies drawn from the Connecticut River below the plant have been identified. The appropriate dose equations for implementation through requirements of the Specification are described in the Vermont Yankee Off-Site Dose Calculation Manual. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents were developed from the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", Revision 1, April 1977.

3/4.6 EFFLUENT AND ENVIRONMENTAL CONTROL BASES (cont.)

Liquid Radwaste Treatment (3.2.3)

The requirement that the appropriate portions of this system as indicated in the ODCM be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

Gaseous Effluents: Dose Rate (3.3.1)

The specified limits as determined by the methodology in the ODCM, restrict, at all times, the corresponding organ dose rate to <1500 mrem/yr to any organ. 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 licensee meets the dose objectives of Appendix I to 10 CFR 50.

Gaseous Effluents: Dose from Noble Gases (3.3.2)

Deleted

3/4.6 EFFLUENT AND ENVIRONMENTAL CONTROL BASES (cont.)

Gaseous Effluents: Dose from Tritium and Radionuclides in Particulate Form (3.3.3)

This Control is provided to implement the requirements of Section II.C, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation are the guides set forth in Section II.C of Appendix I. The requirements provide operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of the subject materials were also developed using the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for tritium and radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man, in areas at and beyond its site boundary. The pathways which were examined in the development of these specifications were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.6 EFFLUENT AND ENVIRONMENTAL CONTROL BASES (cont.)

Gaseous Radwaste Treatment (3.3.4) Deleted

Ventilation Exhaust Treatment (3.3.5) Deleted

Primary Containment (MARK I) (3.3.6) Deleted

Steam Jet Air Ejector (SJAE) (3.3.7) Deleted

Total Dose (40 CFR 190) (3.4.1)

This Control is provided to meet the dose limitations of 40 CFR Part 190 to Members of the Public in areas at and beyond the Site Boundary. The specification requires the preparation and submittal of a Specific Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. The Special Report will describe a course of action that should result in the limitation of the annual dose to a Member of the Public to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the Member of the Public is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.2203(a)(4), is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20. An individual is not considered a Member of the Public during any period in which he/she is engaged in carrying out any operation that subjects them to occupational exposures. For individuals in controlled areas who are considered Members of the Public per 10 CFR 20, the dose limits of 10 CFR 20.1301 apply since the licensee has the authority to control and limit access to these areas.

3/4.6 EFFLUENT AND ENVIRONMENTAL CONTROL BASES
(cont.)

Radiological Environmental Monitoring Program (3.5.1)

The radiological monitoring program required by this Control provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of member(s) of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

Ten years of plant operation, including the years prior to the implementation of the Augmented Off-Gas System, have amply demonstrated via routine effluent and environmental reports that plant effluent measurements and modeling of environmental pathways are adequately conservative. In all cases, environmental sample results have been two to three orders of magnitude less than expected by the model employed, thereby representing small percentages of the ALARA and environmental reporting levels. This radiological environmental monitoring program has therefore been modified as provided for by Regulatory Guide 4.1 (C.2.b), Revision 1, April 1975. Evaluation of plant gaseous effluents, meteorological conditions and potential accident scenaria have concluded that milk sampling is no longer required and silage and grass sampling have been instituted as an indicator of radionuclide deposition. Because of this change, the frequency of silage collection has been increased from annual to quarterly.

The detection capabilities required by Table 4.5.1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as a before-the-fact limit representing the capability of a measurement system and not as an after-the-fact limit for a particular measurement. This does not preclude the calculation of an after-the-fact LLD for a particular measurement based upon the actual parameters for the sample in question.

3/4.6 EFFLUENT AND ENVIRONMENTAL CONTROL BASES (cont.)

Land Use Census (3.5.2)

This Control is provided to ensure that changes in the use of areas at and beyond the site boundaries are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. The requirement of a garden census has been eliminated along with the food product monitoring requirement due to the substantial and widespread occurrence of dairy farming in the surrounding area which dominates the food uptake pathway.

The addition of new sampling locations to Control 3.5.1, based on the land use census, is limited to those locations which yield a calculated dose or dose commitment greater than 20 percent of the calculated dose or dose commitment at any location currently being sampled. This eliminates the unnecessary changing of the environmental radiation monitoring program for new locations which, within the accuracy of the calculation, contributes essentially the same to the dose or dose commitment as the location already sampled. The substitution of a new sampling point for one already sampled when the calculated difference in dose is less than 20 percent, would not be expected to result in a significant increase in the ability to detect plant effluent related nuclides. Due to the decay of I-131 and limited source terms available after permanent shutdown, milk sampling was discontinued, but sampling of vegetation and air sampling have remained in place.

Interlaboratory Comparison Program (3.5.3)

The requirement for participation in an intercomparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

Deleted

5.0 METHOD TO CALCULATE OFF-SITE LIQUID CONCENTRATIONS

Chapter 5 contains the basis for plant procedures that the plant operator requires to meet ODCM Control 3.2.1 which limits the total fraction of combined effluent concentration in liquid pathways, excluding noble gases, denoted here as, F_1^{ENG} at the point of discharge at any time (see Figure 9-1). F_1^{ENG} is limited to less than or equal to ten, i.e.,

$$F_1^{ENG} \leq 10$$

Evaluation of F_1^{ENG} and C_1^{NG} is required concurrent with the sampling and analysis program in Control Table 4.2.1.

5.1 Method to Determine F_1^{ENG}

Determine the total fraction of combined effluent concentrations at the point of discharge in liquid pathways (excluding noble gases), denoted F_1^{ENG} , as follows:

$$F_1^{ENG} = \sum_i \frac{C_{pi}}{ECL_i} \leq 10 \quad (5-1)$$

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

where:

F_1^{ENG} = Total sum of the fractions of each radionuclide concentration in liquid effluents (excluding noble gases) at the point of discharge to an unrestricted area, divided by each radionuclide's ECL value.

C_{pi} = Concentration at point of discharge to an unrestricted area of radionuclide “i”, except for dissolved and entrained noble gases, from any tank or other significant source, p, from which a discharge may be made (including the liquid waste sample tank, groundwater interception release tanks and any other significant source from which a discharge can be made) ($\mu\text{Ci/ml}$). This concentration can be calculated from: $C_{pi} = C_{TKi} \times F_{TK} / [F_{DIL} + F_{TK}]$ where: C_{TKi} equals the concentration of radionuclide i in the tank to be discharged ($\mu\text{Ci/ml}$); F_{DIL} is equal to the dilution flow provided by the waste dilution pumps, if used; F_{TK} equals the liquid waste discharge pump flow rate which regulates the rate at which liquid from a waste collection tank is discharged (gpm).

ECL_i = Annual average effluent concentration limits of radionuclide “i”, except for dissolved and entrained noble gases, from 10 CFR 20.1001-20.2402, Appendix B, Table 2, Column 2 ($\mu\text{Ci/ml}$).

5.2 Method to Determine Radionuclide Concentration for Each Liquid Effluent Pathway

5.2.1 Sample Tanks Pathways

C_{pi} is determined for each radionuclide above LLD from the activity in a representative grab sample of any of the sample tanks and the predicted flow at the point of discharge to an unrestricted area.

Most periodic batch releases are made from the waste sample tank. This tank serves to hold all the liquid wastes after they have been processed by ion exchange. Other periodic batch releases may also come from the intercepted groundwater tank. A batch release tank can be any collection device (e.g. bladder, pillow tank, etc.) that meets the discharge requirements of Table 4.2.1 Notation b.

The liquid waste sample tank is sampled and the contents analyzed for water quality and radioactivity. If the sample meets all the requirements, the contents of the tank may be re-used in the spent fuel pool or torus. If the sample does not meet all the requirements, the contents are recycled through the radwaste system or discharged. The groundwater intercept tank is sampled from the tank recirculation system, and the contents analyzed for water quality and radioactivity. If the sample meets the requirements for discharge, it may be discharged to the storm drain system.

Prior to discharge each sample tank is analyzed for tritium and dissolved and suspended gamma emitters.

5.2.2 Deleted

5.2.3 Deleted

5.2.4 Intercepted Groundwater Concentrations in Flowpaths to the Connecticut River

In order to minimize radioactive liquid waste, groundwater intruding into basement structures is intercepted from sub-slab groundwater wells. This water is environmentally derived and captured under basements to prevent additional cross contamination from plant systems and structures. The intercepted ground water is tanked, sampled and released via the liquid waste effluent line or to the storm system with outfalls to the Connecticut River. The Intercepted Groundwater Collection and Release System is isolated from all other plant systems containing liquids in order to prevent cross contamination and contains water collected from the plant environs, therefore continuous release monitoring is not required . Intercepted groundwater tanks are sampled and released as described in 5.2.1 Sample Tank Pathway.

5.2.5 Subsurface Contaminated Groundwater Concentrations in Flowpaths to the Connecticut River

The overall direction of groundwater flow at Vermont Yankee (VY) is towards the Connecticut River (west to east). Based on this understanding of site hydrogeologic conditions, the groundwater discharge rates from the developed portion of the site to the river are estimated using a streamtube approach based on Darcy's Law (see Section 9).

To estimate the groundwater concentration in each of the designated streamtubes, samples will be collected and analyzed according to requirements specified in Section 3 /4, Table 4.2.1. The concentrations shall then be determined using methods provided in Section 5.1.

6.0 OFF-SITE DOSE CALCULATION METHODS

Chapter 6 provides the basis for plant procedures required to meet the 10 CFR 50, Appendix I, ALARA dose objectives, and the 40 CFR 190 total dose limits to members of the public in unrestricted areas, as stated in the Radiological Effluent Controls (implementing the requirements of QAPM Appendix D, Section 2.5.B). A simple, conservative method (called Method I) is listed in Tables 1.1.2 to 1.1.4 and Table 1.1.6 for each of the Control requirements. Each of the Method I equations is presented, along with their bases in Sections 6.2, 6.3, 6.6, 6.9 and Section 6.11. In addition, reference is provided to more sophisticated but still conservative methods (called Method II) for use when more accurate results are needed. This chapter provides the methods, data, and reference material with which the operator can calculate the needed doses and dose rates. Setpoint methods for effluent monitor alarms are described in Chapter 8.

Demonstration of compliance with the dose limits of 40 CFR 190 is considered to be a demonstration of compliance with the 0.1 rem limit of 10 CFR 20.1301(a)(1) for members of the public in unrestricted areas (Reference 56 FR 23374, 3rd column).

6.1 Introductory Concepts

The Radiological Effluent Controls Program (QAPM Appendix D, Section 2.5.B) 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, refers 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 1500 mrem/year to any organ from tritium and radionuclides in particulate form. 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 10 CFR 50.

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

The dose D is the quantity calculated by the Chapter 6 dose equations. The D calculated by "Method I" equations is not necessarily the actual dose received by a real individual but usually provides an upper bound for a given release because of the conservative margin built into the dose factors and the selection and definition of critical receptors. The radioisotope specific dose factors in each "Method I" dose equation represent the greatest dose to any organ of any age group accounting for existing or potential pathways of exposure. The critical receptor assumed by "Method I" equations is typically a hypothetical individual whose behavior - in terms of location and intake - results in a dose which is expected to be higher than any real individual. The Method I equations employ five-year historical average atmospheric dispersion factors to define receptors of maximum impact. Method II allows for a more exact dose calculation for real individuals, if necessary, by considering only existing pathways of exposure, or actual concurrent meteorology with the recorded release (e.g., wind direction from on-site wind sock, representative off-site wind direction, or meteorology source, etc.).

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

Each of the methods to calculate dose or dose rate are presented in separate sections of Chapter 6, and are summarized in Tables 1.1.1 to 1.1.4 and Table 1.1.6. Each method has two levels of complexity and are called Method I and Method II. Method I is the simplest; generally a linear equation. Method II is a more detailed analysis which allows for use of site-specific factors and variable parameters to be selected to best fit the actual release conditions, within the bounds of the guidance provided.

The plant has both elevated and ground level gaseous release points: the main vent stack (elevated release) and the ground level release (e.g., Reactor Building). Therefore, total dose calculations for whole body, and the critical organ from gaseous releases will be the sum of the stack and ground level doses.

6.2 Method to Calculate the Total Body Dose from Liquid Releases

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

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

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

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

To assess the dose contribution from subsurface groundwater contaminated with plant-generated radionuclides, a dose evaluation shall be performed using Method I on a monthly basis. Radionuclide concentration averages and groundwater streamtube average flow rates shall be utilized to estimate the total plant-generated radioactive contaminants released for the previous monthly period.

6.2.1 Method I

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

$$D_{tb} = FC \sum_i Q_i DFL_{itb} \quad (6-1)$$

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

where:

FC = Flow Correction calculated by dividing the flow at the unrestricted area release point in gpm divided 20,000 gpm or release flow in ft³/sec divided by 44.6 ft³/sec.

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

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

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

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

6.2.2 Basis for Method I

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

Method I may be used to show that the effluent Controls which limit off-site total body dose from liquids (3.2.2 and 3.2.3) have been met for releases over the appropriate periods. Control 3.2.2 is based on the ALARA design objectives in 10 CFR 50, Appendix I Subsection II A. Control 3.2.3 is an "appropriate fraction," determined by the NRC, of that design objective (hereafter called the Objective). Control 3.4.1 is based on Environmental Standards for Uranium Fuel Cycle in 40 CFR 190 (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" (10 CFR 50, 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 ft³/sec was used with a mixing ratio of 0.0356 which corresponds to a minimum regulated river flow of 1250 cfs at the Vernon Dam just below the plant discharge outfall.* Tables 6.2.1 and 6.2.2 outline human consumption and environmental parameters used in the analysis. The resulting, site-specific, total body dose factors appear in Table 1.1.11.

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

$$\Delta D_{tb} = FC Q_i DFL_{itb} \left(\frac{\text{mrem}}{Ci} \right) \quad (6-2)$$

where:

FC = Flow Correction calculated by dividing the flow at the unrestricted area release point in gpm divided 20,000 gpm or release flow in ft³/sec divided by 44.6 ft³/sec

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

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

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

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

6.2.3 Method II

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

TABLE 6.2.1

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

VARIABLE			FOOD GROWN WITH CONTAMINATED WATER						
			POTABLE WATER	AQUATIC FOOD	SHORELINE ACTIVITY	VEGETABLES	LEAFY VEG.	MEAT	COW MILK
MP	Mixing Ratio		-	1.0	0.0356	0.0356	0.0356	0.0356	0.0356
TP	Transit Time	(HRS)	-	24.0	0.000	0.0000	0.0000	480.0	48.0
YV	Agricultural Productivity	(KG/M²)				2.0	2.0	2.0	2.0
P	Soil Surface Density	(KG/M²)				240.0	240.0	240.0	240.0
IRR	Irrigation Rate	(L/M²/HR)				0.152	0.152	0.152	0.152
TE	Crop Exposure Time	(HRS)				1440.0	1440.0	1440.0	1440.0
TH	Holdup Time	(HRS)				1440.0	24.0	2160.0	2160.0
QAW	Water Uptake Rate for Animal	(L/D)						50.0	60.0
QF	Feed Uptake Rate for Animal	(KG/D)						50.0	50.0
FI	Fraction of Year Crops Irrigated					0.5	0.5	0.5	0.5
	Location of Critical Receptor		Connecticut River Below Vernon Dam						

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TABLE 6.2.2

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

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

6.3 Method to Calculate Maximum Organ Dose from Liquid Releases

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

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

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

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

To assess the dose contribution from subsurface groundwater contaminated with plant-generated radionuclides, a dose evaluation shall be performed using Method I on a monthly basis. Radionuclide concentration averages and groundwater streamtube average flow rates shall be utilized to estimate the total plant-generated radioactive contaminants released for the monthly period.

6.3.1 Method I

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

$$D_{mo} = \sum_i FC Q_i DFL_{imo} \quad (6-3)$$

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

where:

FC = Flow Correction calculated by dividing the flow at the unrestricted area release point in gpm divided 20,000 gpm or release flow in ft³/sec divided by 44.6 ft³/sec.

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

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

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

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

6.3.2 Basis for Method I

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

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

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

(6-4)

$$\Delta D_{mo} = FC \cdot Q_i \cdot DFL_{imo}$$

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

where:

FC = Flow Correction calculated by dividing the flow at the unrestricted area release point in gpm divided 20,000 gpm or release flow in ft³/sec divided by 44.6 ft³/sec

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 10 CFR 20 concentrations (which limit public health consequences) a failure of Method I (i.e., the exposure of a real individual being underestimated) is improbable and the consequences of a failure are minimal.

6.3.3 Method II

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

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

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6.4.1 Method I

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6.4.2 Basis for Method I

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6.4.3 Method II

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6.5 Method to Calculate the Skin Dose Rate from Noble Gases

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6.5.1 Method I

Deleted

6.5.2 Basis For Method I

Deleted

6.5.3 Method II

Deleted

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

Effluent Control 3.3.1.b limits the dose rate to any organ, denoted \dot{R}_{co} , from all release sources of 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 particulates is commensurate with the time the particulate samplers are in service between changeouts (typically a week).

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

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

6.6.1 Method I

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

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

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

where:

\dot{Q}_i^{STP} = Stack activity release rate determination of radionuclide "i" (particulates with half-lives greater than 8 days, and tritium) in $\mu\text{Ci}/\text{sec}$. For $i = \text{Sr-90}$ or tritium, use the best estimates (such as most recent measurements).

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

For ground releases the critical organ dose rate from Tritium and Particulates with $T_{1/2}$ greater than 8 days is calculated as follows:

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

where:

\dot{Q}_i^{GLP} = Ground activity release rate determination of radionuclide "i" (particulates with half-lives greater than 8 days, and tritium) in $\mu\text{Ci}/\text{sec}$. For $i = \text{Sr-90}$, Fe-55 or tritium, use the best estimates (such as most recent measurements).

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

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

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

1. Normal operations (not emergency event), and
2. Tritium and particulate releases via either stack or ground level releases to the atmosphere.

6.6.2 Basis for Method I

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

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

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

$$\begin{aligned} \dot{D}_{cos} &= \sum_i Q_i \text{ DFG}_{ico} \\ (\text{mrem}) & \quad (\text{Ci}) \left(\frac{\text{mrem}}{\text{Ci}} \right) \end{aligned} \tag{6-17}$$

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

$$\begin{aligned} \dot{R}_{cos} &= 31.54 \sum_i \dot{Q}_i^{STP} \text{ DFG}_{sico} \\ \left(\frac{\text{mrem}}{\text{yr}} \right) & \quad \left(\frac{\text{Ci-sec}}{\mu\text{Ci-yr}} \right) \quad \left(\frac{\mu\text{Ci}}{\text{sec}} \right) \left(\frac{\text{mrem}}{\text{Ci}} \right) \end{aligned} \tag{6-18}$$

Equation 6-44 is written in the form:

$$\dot{R} = 31.54 \sum_i \dot{Q}_i^{STP} DFG_{sico} \left(\frac{\text{mrem}}{\text{yr}} \right) \left(\frac{\text{Ci} \cdot \text{sec}}{\mu\text{Ci} \cdot \text{yr}} \right) \left(\frac{\mu\text{Ci}}{\text{sec}} \right) \left(\frac{\text{mrem}}{\text{Ci}} \right) \quad (6-19)$$

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

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

The calculation of ground level release dispersion parameters are based on the location of the Reactor Building with respect to the site boundary that would experience the highest exposure.

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

6.6.3 Method II

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

6.7 Deleted

6.8 Deleted

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

Effluent Control 3.3.3 limits the critical organ dose to a Member of the Public from all release sources of Tritium and particulates with half-lives greater than 8 days (hereafter called "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 stack and ground level vent releases. The total critical organ dose limit of Control 3.3.3 is the total contribution from both ground level and stack releases occurring during the period of interest

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

6.9.1 Method I

$$D_{\text{cos}} = \sum_i Q_i^{\text{STP}} \text{DFG}_{\text{sico}} \quad (6-25)$$

(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.

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:

$$D_{\text{cog}} = \sum_i Q_i^{\text{GLP}} \text{DFG}_{\text{gico}} \quad (6-44)$$

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

Q_i^{GLP} = Total activity (Ci) released from ground level releases 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 $D_{\text{cos}} + D_{\text{cog}}$.

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

1. Normal operations (not emergency event),
2. P releases via the plant stack, ground releases (e.g., Reactor Building) to the atmosphere, and
3. Any other ground level continuous or batch release over any time period.

6.9.2 Basis for Method I

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

Method I may be used to show that the Control limit for off-site organ dose from gaseous releases in particulate form, including tritium (3.3.3) has been met for releases over the appropriate periods.

Method I was developed such that "the actual exposure of an individual ... is unlikely to be substantially underestimated" (10 CFR 50, Appendix I). The use below of a single "critical receptor" provides part of the conservative margin to the calculation of critical organ dose in Method I. Method II allows that actual individuals, with real behaviors, be taken into account for any given release. In fact, Method I was based on a Method II analysis of the critical receptor for the annual average conditions. For purposes of complying with the Control 3.3.3, maximum annual average atmospheric dispersion factors are appropriate for batch and continuous releases. That analysis was called the "base case"; it was then reduced to form Method I. The base case, the method of reduction, and the assumptions and data used are presented below.

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

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

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

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

Method I is more conservative than Method II in the region of the effluent dose Control limit because it is based on the following reduction of the base case. The dose factors DFG_{ico} used in Method I were chosen from the base case to be the highest of the set for that radionuclide. In effect each radionuclide is conservatively represented by its own critical age group and critical organ.

6.9.3 METHOD II

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

TABLE 6.9.1

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

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

*Regulatory Guide 1.109, Revision 1.

TABLE 6.9.1 (Continued)

Notes:

- (1) For Method II dose/dose rate analyses of identified radioactivity releases of less than one year, the soil exposure time for that release may be set at 8760 hours (1 year) for all pathways.
- (2) For Method II dose/dose rate analyses performed for releases occurring during the first or fourth calendar quarters, the fraction of time animals are assumed to be on pasture is zero (nongrowing season). For the second and third calendar quarters, the fraction of time on pasture (FP) will be set at 1.0. FP may also be adjusted for specific farm locations if this information is so identified and reported as part of the land use census.
- (3) For Method II analyses, the fraction of pasture feed while on pasture may be set to less than 1.0 for specific farm locations if this information is so identified and reported as part of the land use census.
- (4) For all Method II analyses, an absolute humidity value equal to 5.6 (gm/m³) shall be used to reflect conditions in the Northeast (Reference: Health Physics Journal, Vol. 39 (August), 1980; Page 318-320, Pergammon Press).
- (5) Variable T is a combination of variables TF and TS in Regulatory Guide 1.109, Revision 1.

TABLE 6.9.2

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

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

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

The gaseous effluent dose methods have been simplified by assuming an individual whose behavior and living habits inevitably lead to a higher dose than anyone else. The following exposure pathways to gaseous effluents listed in Regulatory Guide 1.109 (Reference A) have been considered for 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 meat.
5. Ingestion of milk.

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

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

6.10.1 Receptor Locations

Distances to the site boundary from the two evaluated gaseous release pathways (the Stack and Reactor Building) are provided in Table 6.10.2. Two 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 ground level air concentration (maximum depleted X/Q) of particulates for determining critical organ dose from inhalation; and
2. The point of maximum deposition (maximum D/Q) of 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 72,000 cfm. The location of the maximum ground level air concentration and deposition of particulates (WNW sector, 2150 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 tritium, and particulates, an animal was assumed to exist at the location of highest calculated ground level air concentration and deposition particulates as noted above. This location then conservatively bounds the deposition of particulates at all real animal locations.

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

6.10.2 Vermont Yankee Atmospheric Dispersion Model

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

XOQDOQ produces the following average atmospheric dispersion factors for each location:

1. Depleted X/Q dispersion factors for evaluating ground level concentrations of particulates;
2. D/Q deposition factors for evaluating dry deposition of elemental particulates.

The ground release (e.g., Reactor Building) depleted X/Q and D/Q factors were derived using Nuclear Regulatory Commission approved software XOQDOQ.

6.10.3 Long-Term Average Atmospheric Dispersion Factors for Receptors

Actual measured meteorological data for the five-year period, 2012 through 2016, were analyzed to determine all the values and locations of the maximum off-site long-term average atmospheric dispersion factors. Each dose and dose rate calculation incorporates the maximum applicable off-site long-term average atmospheric dispersion factor. The values used and their locations are summarized in Table 6.10.1. Table 6.10.1 also indicates which atmospheric dispersion factors are used to calculate the various doses or dose rates of interest.

TABLE 6.10.1

Atmospheric Dispersion Factors

Release Pathway	Dispersion Factor	Dose to Individual		
		Total Body	Skin	Critical Organ
Stack	X/Q Depleted (sec/m ³)			6.46E-07 (2414m WNW)
	D/Q (1/m ²)			2.4E-09 (1609m SSE)
Reactor Building	X/Q Depleted (sec/m ³)			4.0E-05 (402m NE)
	D/Q (1/m ²)			9.53E-08 (402m SSE)

TABLE 6.10.1 NOTATION

NOTE 1 – Releases from the Reactor Building are considered ground releases.

TABLE 6.10.2

Site Boundary Distances

<u>Downwind Sector</u>	<u>Stack Releases</u>	<u>Ground Releases</u>
N	400 m	529 m
NNE	350 m	468 m
NE	350 m	448 m
ENE	400 m	477 m
E	500 m	499 m
ESE	700 m	482 m
SE	750 m	512 m
SSE	850 m	555 m
S	385 m	419 m
SSW	300 m	575 m
SW	250 m	505 m
WSW	250 m	418 m
W	300 m	402 m
WNW	400 m	528 m
NW	550 m	917 m
NNW	550 m	831 m

TABLE 6.10.3

Recirculation Correction Factors

A. Stack Releases

<u>Sector</u>	<u>0.5 Mi</u>	<u>1.5 Mi</u>	<u>2.5 Mi</u>	<u>3.5 Mi</u>	<u>4.5 Mi</u>	<u>7.5 Mi</u>
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. Ground (e.g., Reactor Building) Release

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

6.11 Method to Calculate Direct Dose From Plant Operation

Effluent Control 3.4.1 (40 CFR 190) 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.

6.11.1 Deleted

6.11.2 Deleted

6.11.3 Deleted

6.11.4 Independent Spent Fuel Storage Installation Dose Contribution

The Independent Spent Fuel Storage Installation (ISFSI) has been constructed to provide a secure location for the long term storage of spent nuclear fuel bundles and Greater Than Class C waste. This installation is located just north of the Administrative and Radwaste Building within the Vermont Yankee Protected Area. The vendor, Holtec International, has prepared a study report to satisfy the requirements of 10 CFR 72.104 and this document is included as Reference H. The facility stores 58 Spent Fuel Dry Storage Casks (DSCs). An additional GTCC cask will be added when segmentation of the reactor internals is complete. The shielding analysis of the Independent Spent Fuel Storage Installation is provided in Reference H. The report analyzes the dose generated from a single cask as well as the dose generated from the 58 casks. The dose from each cask to the west site boundary (DR-53A Location) is monitored with REMP TLDs.

6.11.5 Total Direct Dose Summary

Estimates of direct exposure above background in areas at and beyond the site boundary can be determined from measurements made by environmental TLDs located as shown in Table 7.1 and Figure 7.4 that are part of the Radiological Environmental Monitoring Program. Alternatively, direct dose calculations from identified fixed sources on site can be used to estimate the off-site direct dose contribution where TLD information may not be applicable.

6.11.6 Deleted

6.12 Cumulative Doses

Cumulative Doses for a calendar quarter and a calendar year must be maintained to demonstrate a compliance with Controls 3.2.2 and 3.3.3 (10 CFR 50, Appendix I dose objectives). In addition, if the requirements of the Action Statement of Control 3.4.1 dictate, cumulative doses over a calendar year must be determined (demonstration of compliance with total dose, including direct radiation per requirements of 40 CFR 190). To ensure the limits are not exceeded, a running total must be kept for each release.

Demonstration of compliance with the dose limits of 40 CFR 190 is considered as demonstrating compliance with the 0.1 rem limit of 10 CFR 20.1301(a)(1) for members of the public in unrestricted areas.

7.0 ENVIRONMENTAL MONITORING PROGRAM

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

7.1 Intercomparison Program

All routine radiological analyses for environmental samples are performed at offsite environmental laboratories. The laboratories participate in several commercial inter-comparison programs in addition to an internal QC sample analysis program and the analysis of client-introduced QC sample programs. The external programs may include the Department of Energy – Mixed Analyte Performance Evaluation Program (MAPEP), Analytics Cross-Check Program - Environmental Inter-laboratory Cross-Check Program, and Environmental Resources Association - Environmental Radioactivity Performance Evaluation Program or other NRC-approved sources.

7.2 Airborne Pathway Monitoring

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

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

The Connecticut River Valley in the vicinity of the Vermont Yankee plant has a pronounced up- and down-valley wind flow. Based on five years of meteorological data, wind blows into the 3 “up-valley” sectors (N, NNW, and NW) 27 percent of the time, and the 4 “down-valley” sectors (S, SSE, SE, and ESE) 40 percent of the time, for a total “in-valley” time of 67 percent. This compares reasonably with the updated meteorological data (2012 – 2016) where upper wind data frequencies for the up-valley sectors (WNW, NW, NNW, N) account for 37 percent of the time, and the down-valley sectors (ESE, SE, SSE, S) account for 33 percent of the time.

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 0.75 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 the River Station Number 3-3 (AP/CF-11, SSE, 1.9 km). This station resides 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. This station has been in operation since the preoperational period.

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

7.3 Distances and Directions to Monitoring Stations

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

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

Table 7.1

Radiological Environmental Monitoring Stations⁽¹⁾

<u>Exposure Pathway and/or Sample Direction⁽⁵⁾</u>	<u>Sample Location and Designated Code⁽²⁾</u>		<u>Distance (km)⁽⁵⁾</u>	
1. AIRBORNE (Particulate)				
	AP-11	River Station No. 3-3	1.88	SSE
	AP-12	N. Hinsdale, NH	3.61	NNW
	AP-21	Spofford Lake ⁽⁹⁾	16.36	NNE
2. WATERBORNE				
a. Surface	WR-11	River Station No. 3-3	1.88	Downriver
	WR-21	Rt. 9 Bridge ⁽⁹⁾	11.83	Upriver
b. Ground	WG-11	Plant Well	0.24	On-Site
	WG-12	Vernon Nursing Well	2.13	SSE
	WG-22	Copeland Well ⁽⁹⁾	13.73	N
c. Sediment	SE-11	Shoreline Downriver	0.57	SSE
From	SE-12	North Storm	0.13	E
Shoreline		Drain Outfall ⁽³⁾		
3. INGESTION				
a. Deleted				
b. Mixed	TG-11	River Station No. 3-3	1.88	SSE
Grasses	TG-12	N. Hinsdale, NH	3.61	NNW
	TG-21	Spofford Lake ⁽⁹⁾	16.36	NNE
c. Silage	TC-11	Miller Farm	0.82	W
	TC-18	Blodgett Farm	3.60	SE
	TC-22	Franklin Farm ⁽⁹⁾	9.73	WSW
d. Fish	FH-11	Vernon Pond	(6)	(6)
	FH-21	Rt. 9 Bridge ⁽⁹⁾	11.83	Upriver

TABLE 7.1 (Continued)

<u>Exposure Pathway and/or Sample Direction⁽⁵⁾</u>	<u>Sample Location and Designated Code⁽²⁾</u>	<u>Distance (km)⁽⁵⁾</u>	
4. DIRECT RADIATION			
	DR-1 River Station No. 3-3	1.61	SSE
	DR-2 N. Hinsdale, NH	3.88	NNW
	DR-5 Spofford Lake ⁽⁹⁾	16.53	NNE
	DR-6 Vernon School	0.52	WSW
	DR-7 Site Boundary	0.28	W
	DR-8 Site Boundary	0.25	SSW
	DR-43 Site Boundary	0.44	SSE
	DR-45 Site Boundary	0.12	NE
	DR-46 Site Boundary	0.28	NNW
	DR-53A West Cornfield	0.39	WSW

- (1) Sample locations are shown on Figures 7.1 to 7.6.
- (2) Station Nos. 10 through 19 are indicator stations. Station Nos. 20 through 29 are control stations (except the direct radiation stations).
- (3) To be sampled and analyzed semiannually.
- (4) Deleted
- (5) Distance and direction from the center of the Reactor Building for direct radiation monitors; from the plant stack for all others.
- (6) Fish samples are collected from anywhere in Vernon Pond, which is adjacent to the plant (see Figure 7-1).
- (7) Deleted.
- (8) Deleted.
- (9) Control stations



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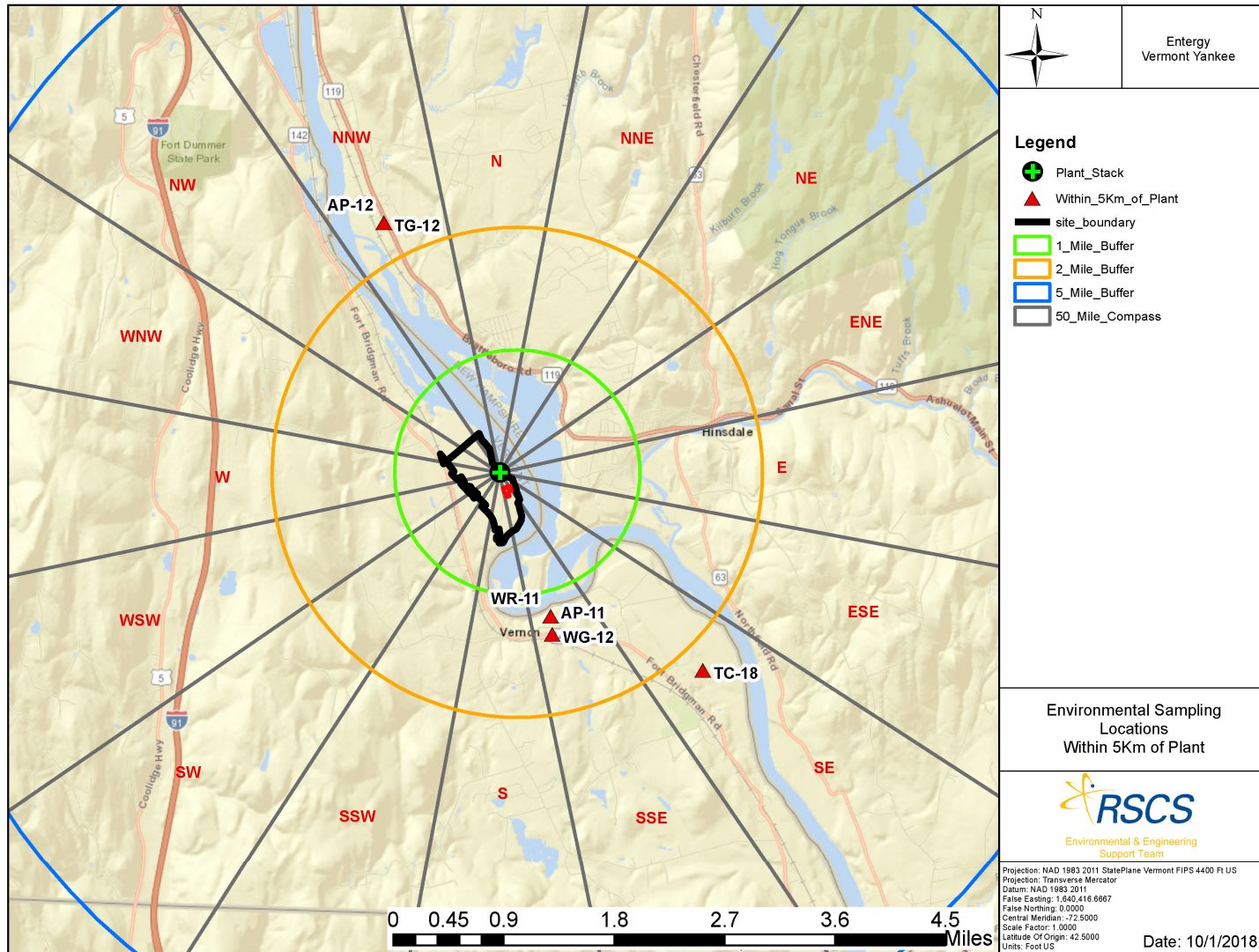


Figure 7-2 Environmental Sampling Locations Within 5 Km of Plant

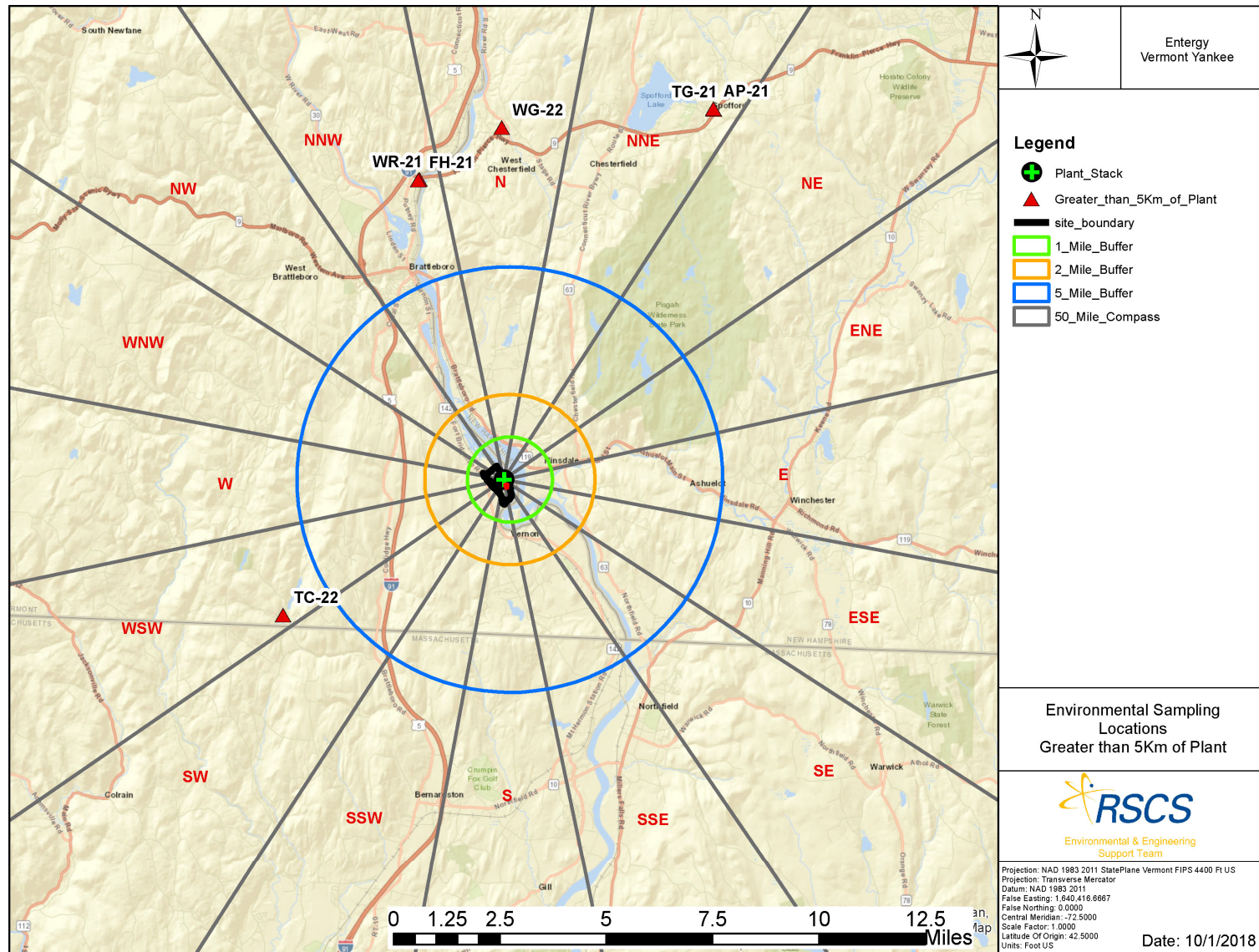


Figure 7-3 Environmental Sampling Locations Greater than 5 Km from Plant

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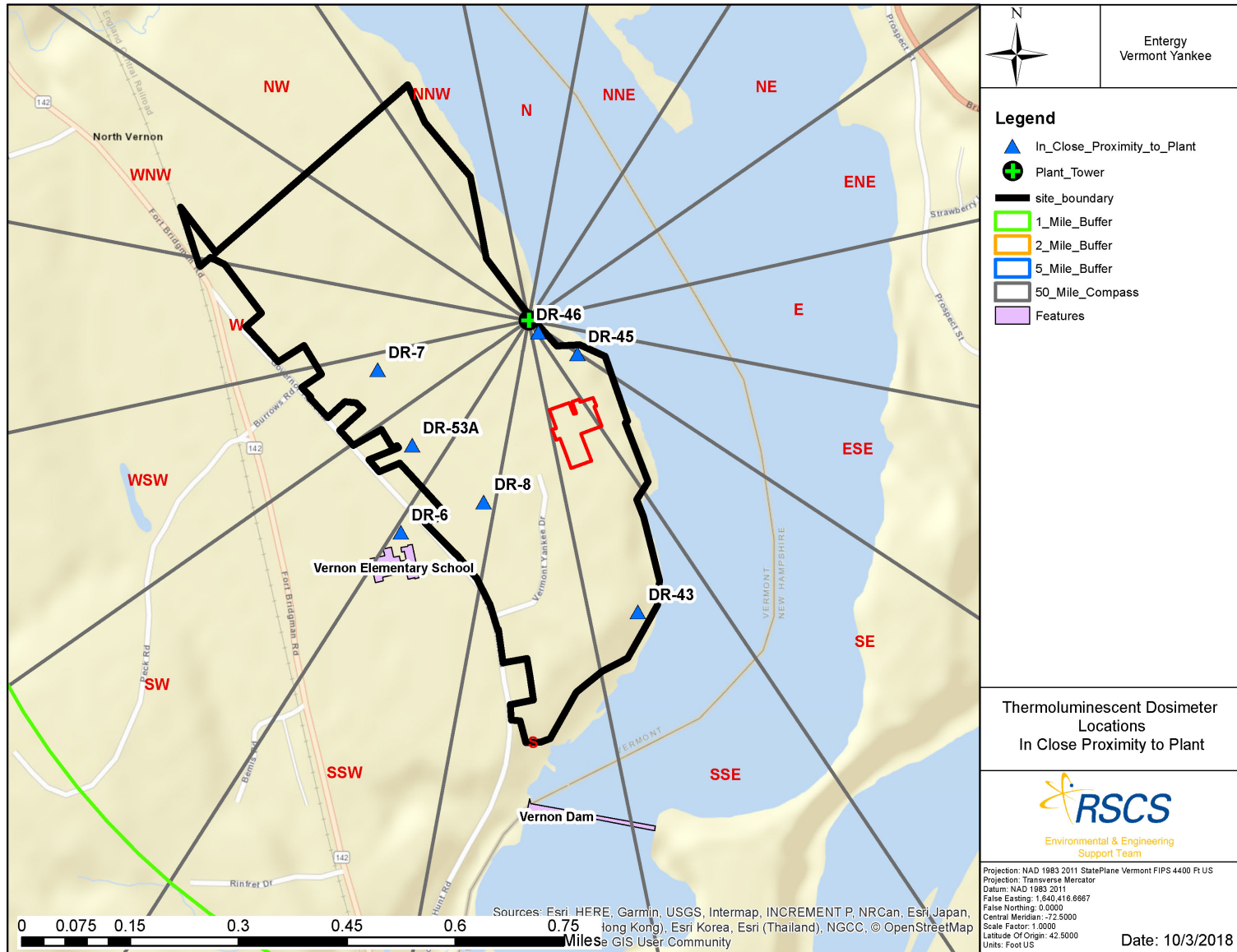


Figure 7-4 TLD Locations in Close Proximity to Plant

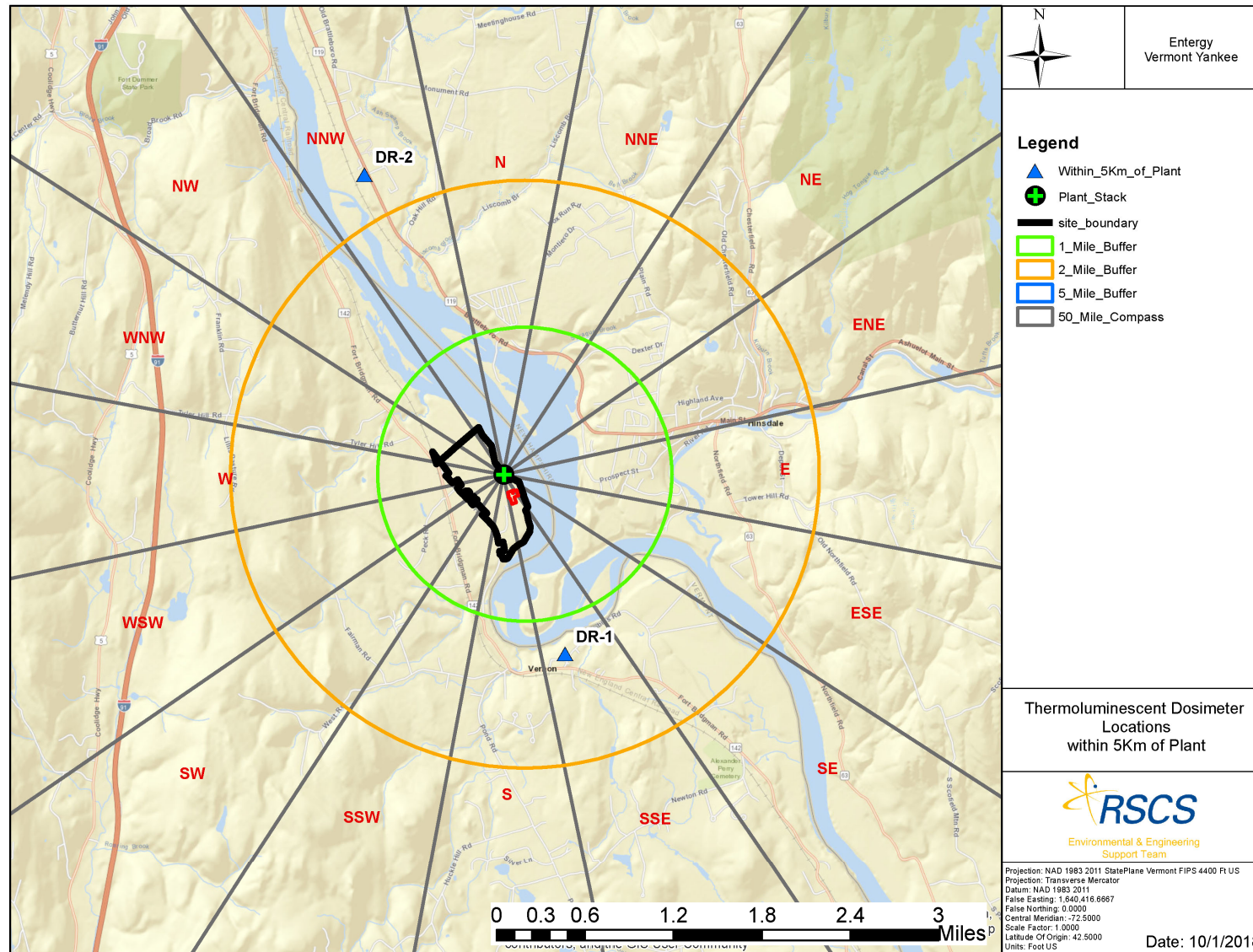


Figure 7-5 TLD Locations Within 5 Km of Plant

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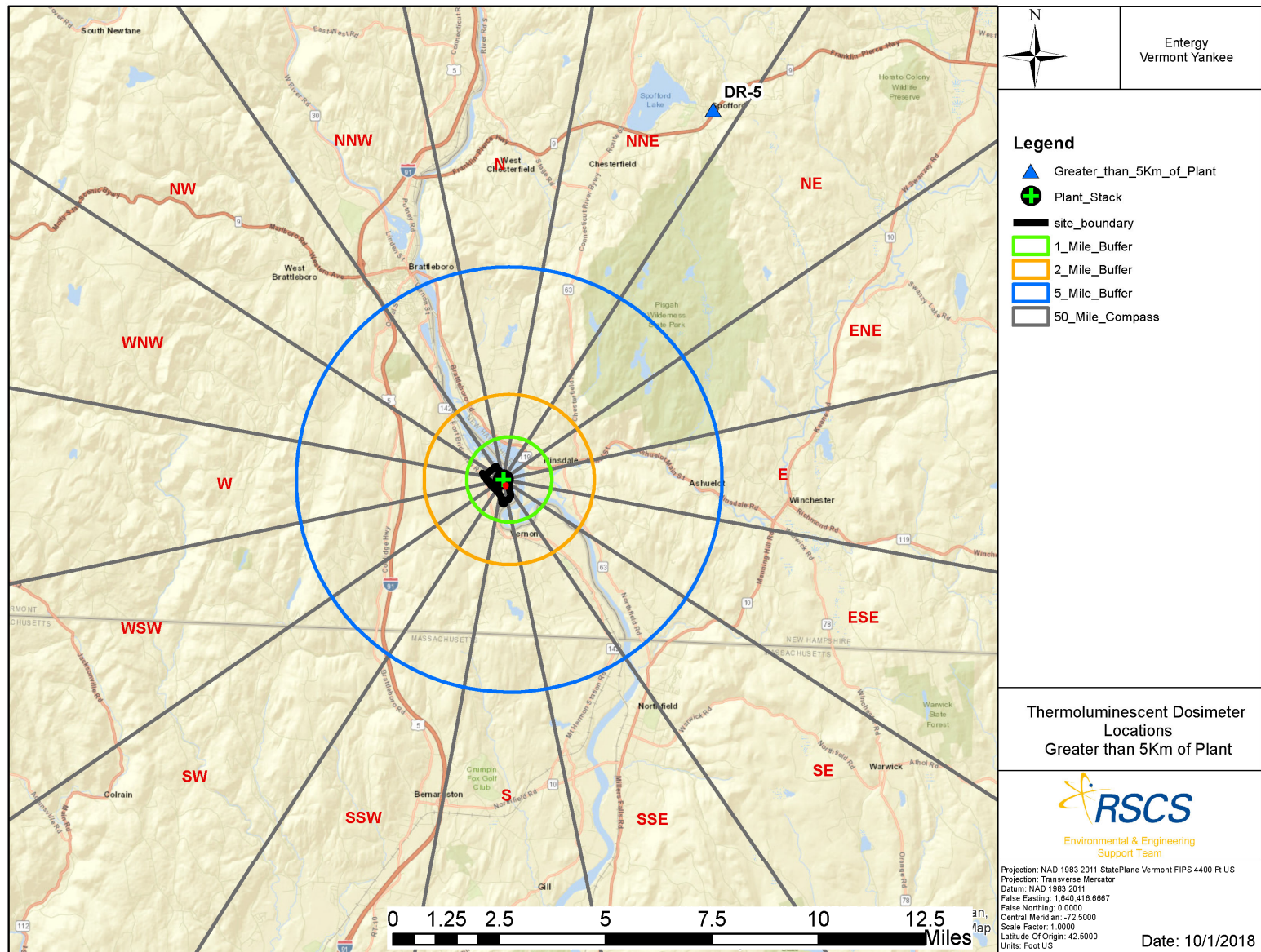


Figure 7-6 TLD Locations Greater Than 5 Km from Plant

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8.0 SETPOINT DETERMINATIONS

Chapter 8 contains the basis for plant procedures used to meet the setpoint requirements of the Radioactive Effluent Instrumentation Controls. Control 3.1.2 outlines the radioactive gaseous effluent instrumentation. All installed liquid effluent monitoring instrumentation has been abandoned.

8.1 Deleted

8.1.1 Deleted

8.1.2 Deleted

8.2 Gaseous Effluent Instrumentation Setpoints

Control 3.1.2 requires that the radioactive gaseous effluent instrumentation in Control Table 3.1.2 have their alarm setpoints set to ensure that Control 3.3.1 is not exceeded.

8.2.1 Deleted

8.2.2 Deleted

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

Figure 9-1 shows the normal (design) radioactive liquid effluent streams, radiation monitors, and the appropriate Liquid Radwaste Treatment System. Figure 9-2 shows the normal (design) gaseous effluent systems, radiation monitors, and the appropriate Gaseous Radwaste Treatment System. Figure 9-3 shows the normal subsurface shallow groundwater stream tube configuration for the plant site. Figure 9-4 shows the normal subsurface deep groundwater stream tube configuration for the plant site.

9.1 In-Plant Radioactive Liquid Effluent Pathways

The Liquid Radwaste System has been modified for the permanently shutdown and defueled SAFSTOR configuration. The installed plant Liquid Radwaste System has been permanently isolated from the Liquid Waste Discharge line and replaced by a Liquid Waste Processing System as shown in Figure 9-1. The Liquid Waste Processing System consists of particulate filter(s) and demineralizers, a sample tank and pump with recirculation, neutralization and sampling capabilities. The sample tank is filled from the Torus and/or Turbine Building Intercepted Groundwater. Typically, water is processed through particulate filters and/or demineralizers prior to filling the sample tank. The liquid wastes collected in the Sample Tank are handled on a batch basis. Prior to discharge, the Sample Tank is recirculated and sampled as required by Table 4.2.1 and the contents analyzed for radioactivity and water purity. A release is allowed once it is determined that the activity in the liquid wastes will not exceed control release limits. The Sample Tank is isolated from all inputs when discharging through the Liquid Waste Discharge Line.

A discharge from the Sample Tank is accomplished by first isolating inputs, starting the pump and opening the necessary valves. The release rate in the discharge line is between 0 and 100 gpm (typically 70 gpm) and discharge volume is monitored by the Liquid Waste Flow Meter.

During discharge the Sample Tank is isolated from the Torus, Intercepted Groundwater and processing system demineralizers and filters preventing the possibility of discharging uncharacterized liquid wastes. The Liquid Waste Discharge Monitor (RM-17-350) has been removed from service since pre-discharge sampling requirements and controls equivalent to those required when the monitor was out of service have been permanently implemented. The monitor is no longer required due to the low radioactivity of SAFSTOR liquids and the simplified Liquid Waste Processing configuration. Discharge occurs to outfall S/N-001 in the Connecticut River via the existing or a replacement liquid waste discharge line.

The Circulating Water System has been removed from service and permanently isolated from the Discharge Structure preventing the possibility of a liquid release from this system. The Circulating Water System Radiation Monitor (RM-17-359) has been removed from service.

9.2 In-Plant Radioactive Gaseous Effluent Pathways

The gaseous radwaste system includes subsystems that dispose of gases from the 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.

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.

9.3 Subsurface Groundwater Pathways to the Connecticut River

9.3.1 Overview

As presented in the Hydrogeologic Investigation Report prepared by GZA GeoEnvironmental, Inc. (See ODCM Reference Section) the overall direction of groundwater flow at the Vermont Yankee plant site is towards the Connecticut River. Based on this understanding of site hydrogeologic conditions, the groundwater discharge rates from the developed portion of the site to the river, has been estimated using a steam tube approach based on Darcy's Law.

9.3.2 Geology

The site geology consists of a discontinuous layer of engineered fill material underlain by glaciolacustrine/glaciofluvial deposits. The total overburden thickness varies from approximately 30 to 80 feet. The upper-most soil deposit is fill, generally consisting of silty sand. Directly underlying the fill is an upper sand unit which consists of fine to medium sand with various amounts of silt. This unit is underlain by a confining silt layer which appears fairly continuous in areas east of the AOG Building (approximately 1 to 16 feet thick) but pinches out towards the north. Groundwater flux towards the river through this silt unit is likely to be negligible due to its low permeability. Consequently, the silt layer was not included in the

groundwater flux calculation presented herein. Below the silt layer and in the vicinity of the Connecticut River, is a finer-grained lower sand unit. This lower sand deposit appears to be of limited lateral extent, extending from approximately the intake structure southerly to the discharge structure.

The bedrock below the overburden is reported to be biotite gneiss. Pre-construction seismic testing of the bedrock indicates that it is generally hard and massive. The depth of bedrock in the release area, east of the AOG building, is approximately 55 to 80 feet below the plant grade of approximately 251.5 ft NGVD, with bedrock generally sloping towards the river. Groundwater movement through the bedrock in the vicinity of the river appears to be in an upward direction, toward the river.

9.3.3 Streamtube Method

To estimate the groundwater flow through the developed area of the site containing systems, structures and components (SSCs) which may carry radionuclides, a series of twelve streamtubes in the shallow overburden and five streamtubes in the deep overburden were delineated. The streamtubes show groundwater flow direction and were drawn perpendicular to the groundwater contours, which were developed from synoptic groundwater elevation data collected on December 15, 2010. The boundaries of the streamtubes represent groundwater flow lines, with groundwater flow generally parallel to these lines within the streamtubes. Based on frequent groundwater level measurements recorded in the monitoring wells at the site, the overall groundwater flow pattern and hydraulic gradients do not vary significantly at the site, and thus the location and shape of the streamtubes are anticipated to be relatively stable. The location and configuration of these streamtubes are shown on Figures 9-3 and 9-4. In the shallow overburden, stream-tubes were generally centered on individual downgradient monitoring wells (perimeter wells) extending across the site from the off-gas stack (well GZ-27) southerly to approximately the area of the discharge structure (well GZ-5). In the deep overburden, streamtubes extend over the lateral extent of the lower sand deposit (from well GZ-18D southerly to well GZ-19D), and were centered on the deep downgradient wells.

9.3.4 Cross Sectional Flow Area

The cross sectional area (A) of flow for each streamtube is based on the streamtube width and the thicknesses of the saturated transmissive overburden units within the streamtube. For the shallow overburden streamtubes, the saturated thickness is estimated based on the difference between the measured groundwater table elevation and the elevation of the bottom of the upper sand unit depending on the specific geological conditions present within the screened portion of the well. If a significant thickness of saturated fill is present, the cross sectional area must also

include the thickness of any saturated upper sand present under the fill. For the deep overburden stream-tubes, the flow area was computed using the thickness of the fully saturated lower sand unit.

9.3.5 Hydraulic Conductivity

As presented in the GZA's May 2011 Hydrogeologic Investigation report, hydraulic conductivity (K) testing was performed in monitoring wells at the Site to characterize the hydrogeologic properties of the overburden soils. Based on this data, the geometric mean K for each major hydrogeologic unit is presented in Table 5.2 of the GZA report. These values, combined with other input parameters presented herein, are used to calculate groundwater flow rates through each streamtube. A summary of the geometric mean K values for each major hydrogeologic unit is presented below.

Hydrogeologic Unit		Geometric Mean K (ft/day)
Shallow Overburden	Fill	0.3
	Upper Sand (East of Power Block)	4
	Upper sand (North of Power Block)	12
Deep Overburden	Lower Sand	1.2

9.3.6 Streamtube Parameters

A summary of selected streamtube parameters used to calculate groundwater flow rates (as discussed further below) is presented in the following table:

Streamtube	Key Well within Streamtube	Approximate Width (ft)	Hydrogeologic Unit(s) Present within Streamtube	K (ft/day)
Shallow Overburden Streamtubes				
ST-1S	GZ-27S	76	Upper Sand – (North of Power Block)	12
ST-2S	GZ-26S	215	Upper Sand – (North of Power Block)	12
ST-3S	GZ-25S	260	Upper Sand – (North of Power Block)	12
ST-4S	GZ-1S	200	Upper Sand – (North of Power Block)	12
ST-5S	GZ-18S	160	Fill	0.3
ST-6S	GZ-13S	50	Fill	0.3
			Upper Sand – (East of Power Block)	4
ST-7S	GZ-23S	85	Fill	0.3
			Upper Sand – (East of Power Block)	4
ST-8S	GZ-3S	85	Fill	0.3
			Upper Sand – (East of Power Block)	4
ST-9S	GZ-14S	105	Fill	0.3
			Upper Sand – (East of Power Block)	4
ST-10S	GZ-4S	90	Fill	0.3
			Upper Sand – (East of Power Block)	4
ST-11S	GZ-19S	95	Fill	0.3
			Upper Sand – (East of Power Block)	4
ST-12S	GZ-5S	125	Fill	0.3
			Upper Sand – (East of Power Block)	4
Deep Overburden Streamtubes				
ST-1D	GZ-18D	75	Lower Sand	1.2
ST-2D	GZ-13D	100	Lower Sand	1.2
ST-3D	GZ-22D	110	Lower Sand	1.2
ST-4D	GZ-14D	110	Lower Sand	1.2
ST-5D	GZ-19D	150	Lower Sand	1.2

9.3.6 Hydraulic Gradient

For each streamtube, average hydraulic gradients (i) are calculated based on quarterly groundwater elevation contours developed from quarterly groundwater level measurements in the monitoring wells.

9.3.7 Calculated Groundwater Flow Rate

Based on the above parameter estimates, the quarterly groundwater flow in each streamtube is computed using the following form of the Darcy's equation for fluid flow through porous media:

$$Q = KiA$$

Where:

- Q = flow (ft³/day)
- K = hydraulic conductivity (ft/day)
- i = hydraulic gradient (ft/ft)
- A = cross-sectional area of flow (width times depth) (ft²)

The quarterly groundwater flow rates are utilized to determine monthly estimates of flow rate for the dose calculations required in Section 6 of the ODCM. The quarterly groundwater flow rates are also averaged over the reporting year to account for variations in seasonal precipitation, and thus the associated seasonal variation groundwater flux to estimate annual radiation doses. To conservatively account for potential uncertainties and the heterogeneity inherent in geologic materials such as those at the plant site, a factor of safety may be applied to the estimated flow to provide a more conservative estimate for the groundwater flow through the subsurface.

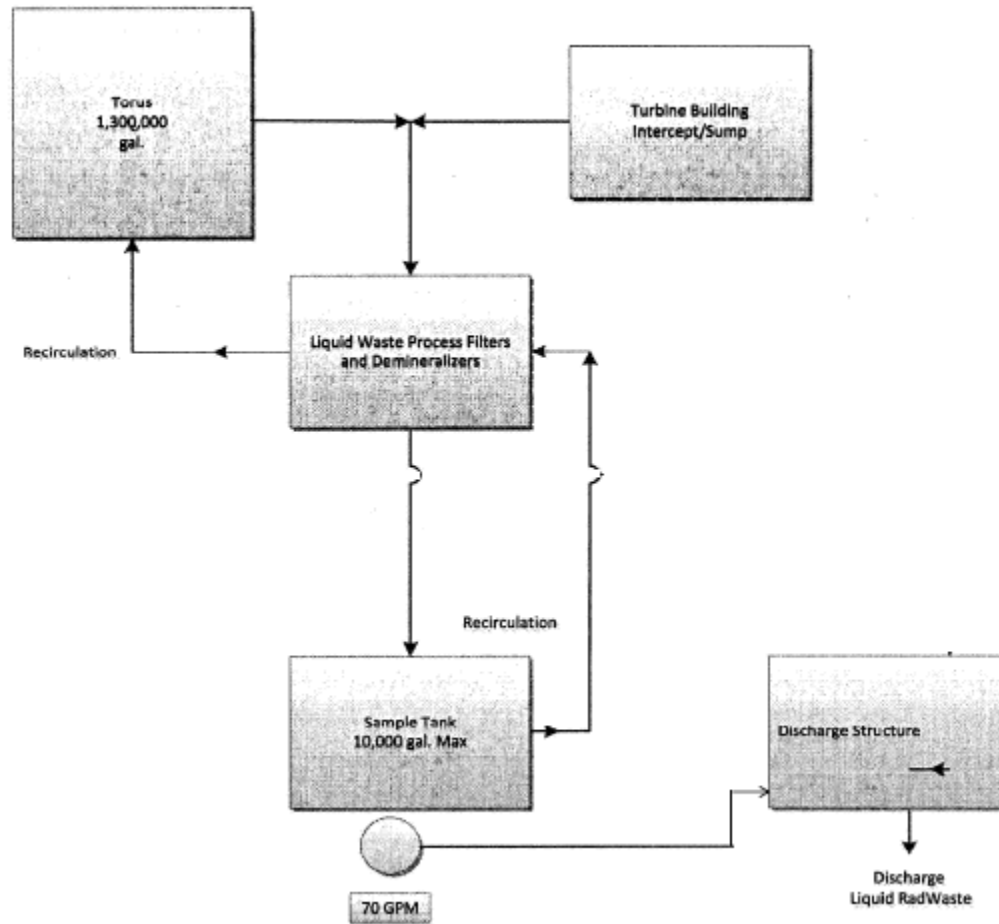


Figure 9.1: Radioactive Liquid Effluent Streams, Radiation Monitors, and Radwaste Treatment System at Vermont Yankee*

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

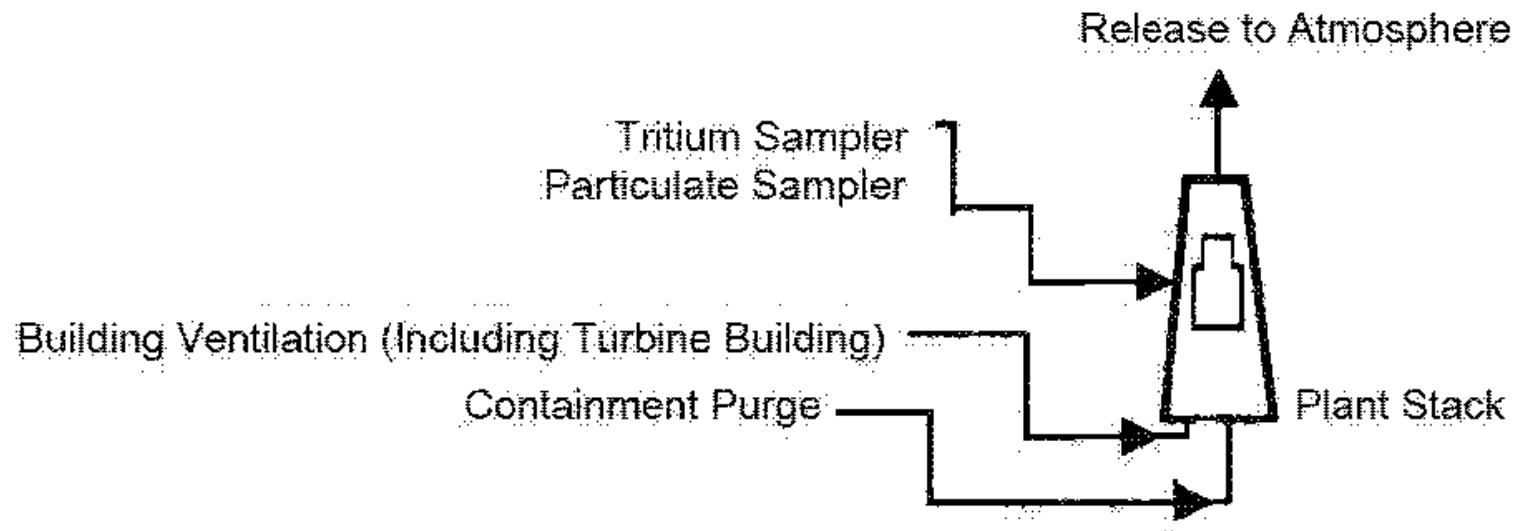


Figure 9.2: Radioactive Gaseous Effluent Streams, Radiation Monitors, and Radwaste Treatment System at Vermont Yankee*

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

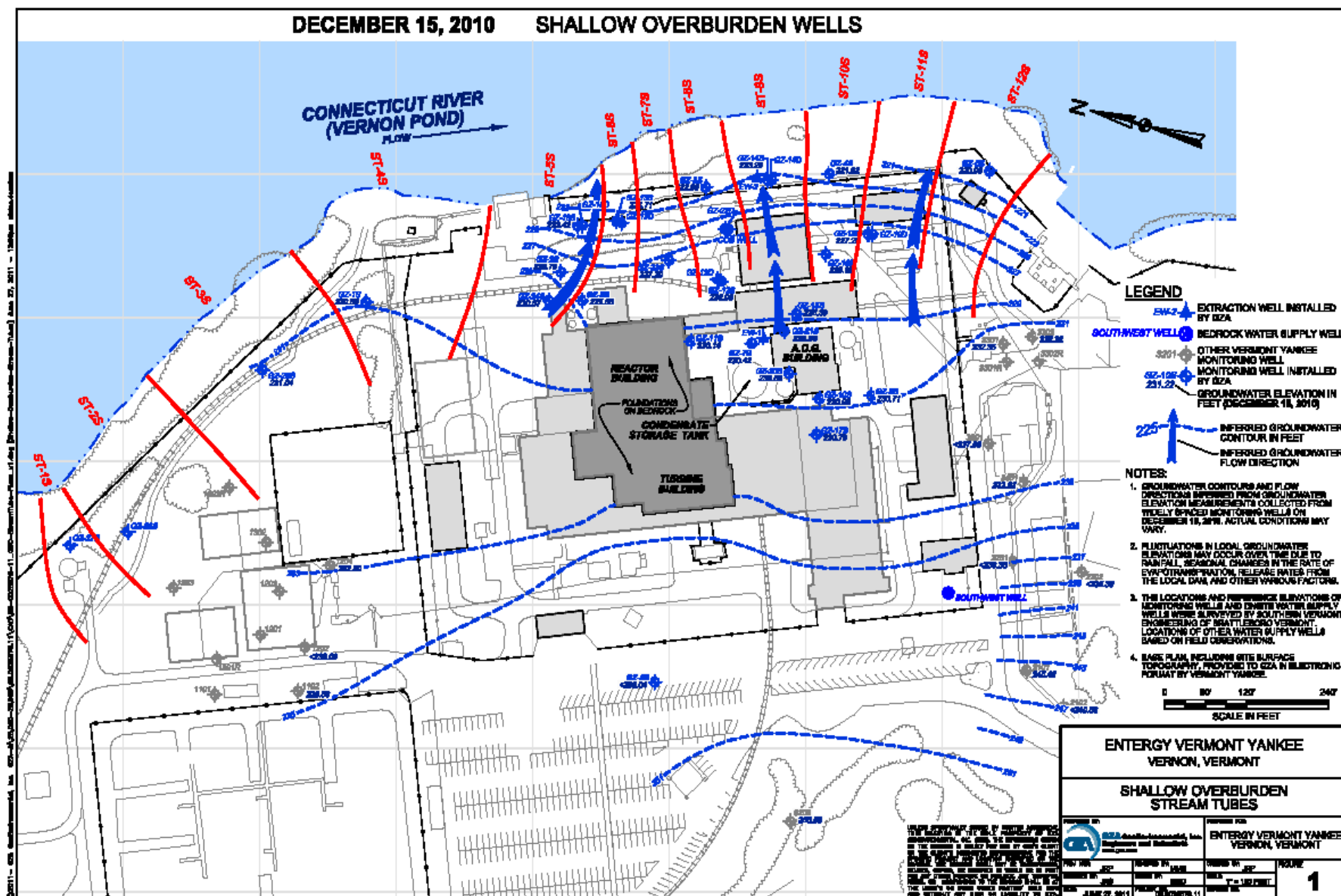


Figure 9.3: Subsurface Shallow Groundwater Streamtubes from the Plant Site to the Connecticut River

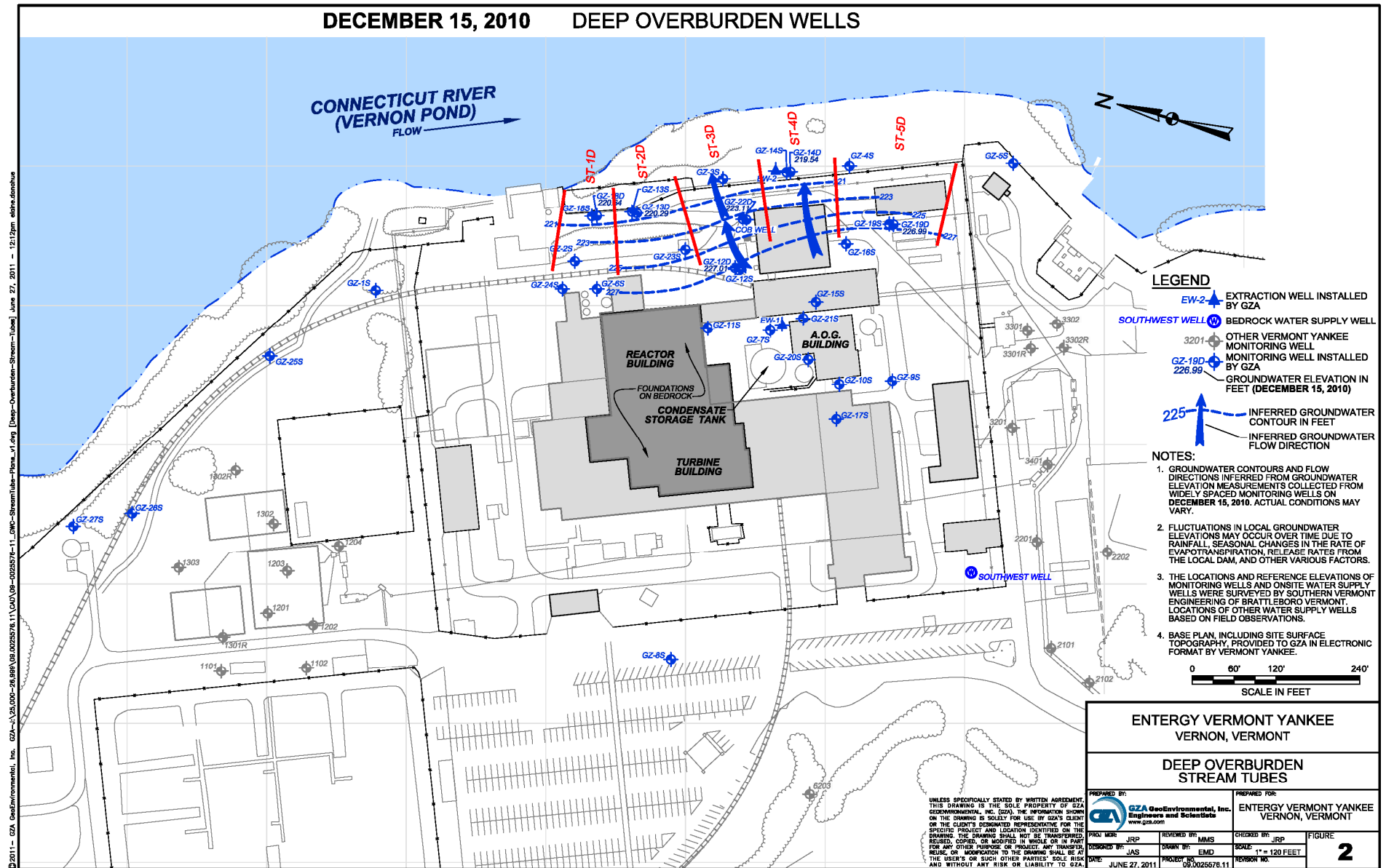


Figure 9.4: Subsurface Deep Groundwater Streamtubes from the Plant Site to the Connecticut River

10.0 UNIQUE REPORTING REQUIREMENTS

10.1 Annual Radioactive Effluent Release Report

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

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

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

With the limits of Control 3.4.1 being exceeded during the calendar year, the Radioactive Effluent Release Report shall also include an assessment of radiation doses to the likely most exposed real member(s) of the public from reactor releases (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

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

With the quantity of radioactive material in any outside tank exceeding the limit of Section 4 of the Defueled Safety Analysis Report, describe the events leading to this condition in the next Radioactive Effluent Release Report.

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

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

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

10.2 Environmental Radiological Monitoring

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

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

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of all radiological environmental samples taken during the report period pursuant to Table 7-1 and Figures 7-1 through 7-6. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

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

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

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

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

10.3 ISFSI Reporting Requirements

In accordance with 10 CFR 72.44(d)(3), the Annual Independent (Interim) Spent Fuel Storage Installation Radioactive Effluent Control Program Report (AISFSIRECPR) will be generated and issued by February 28th of each year.

Since it has been determined by Holtec International in their Final Safety Analysis Report (Reference I) that the Holtec HI-STORM 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are no radioactive effluents from the SFSC.

In light of the information presented in the previous paragraphs, the AISFSIRECPR, to be issued by February 28th of each year, shall state that no radioactive effluents were discharged from the Independent (Interim) Spent Fuel Storage Installation and therefore no ISFSI-specific monitoring program is in place at Vermont Yankee and there are no ISFSI-specific data to report for the previous calendar year reporting period.

10.4 Special Reports

Special reports shall be submitted to the NRC Document Control Desk and Regional Office within the time period specified for each report.

10.4.1 Liquid Effluents (Controls 3.2.2 and 3.2.3)

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

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

- (1) explanation of why liquid radwaste was being discharged without treatment, identification of any non-functional equipment or subsystems, and the reasons for the non-functionality;
- (2) action(s) taken to restore the non-functional equipment to functional status; and
- (3) summary description of action(s) taken to prevent a recurrence.

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10.4.2 Gaseous Effluents (Control 3.3.3)

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

10.4.3 Total Dose (Control 3.4.1)

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

10.4.4 Radiological Environmental Monitoring (Control 3.5.1)

With the level of radioactivity as the result of plant effluents in an environmental sampling media at one or more of the locations specified in Control Table 3.5.1 exceeding the reporting levels of Control Table 3.5.2, prepare and submit to the Commission within 30 days from the receipt of the Laboratory Analyses a special report which includes an evaluation of any release conditions, environmental factors or other factors which caused the limits of Control Table 3.5.2 to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents, however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

10.4.5 Land Use Census (Control 3.5.2)

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

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

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

- A. Shall be reported to the commission in the Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Independent Safety Review (ISR). The discussion of each change shall contain:
 1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;
 2. Sufficient detailed information to support the reason for the change without benefit of additional or supplemental information;
 3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;
 4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;
 5. An evaluation of the change, which shows the expected maximum exposures to member(s) of the public at the site boundary and to the general population that differ from those previously estimated in the license application and amendments thereto;
 6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;
 7. An estimate of the exposure to plant operating personnel as a result of the change; and
 8. Documentation of the fact that the change was reviewed and found acceptable by ISR.
- B. Shall become effective upon review and acceptance by ISR and approval by the manager responsible for overall operational activities.

REFERENCES

- A. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," U.S. Nuclear Regulatory Commission, Revision 1, October 1977.
- B. XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Plants, NUREG/CR-2919, 1982.
- C. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," U.S. Nuclear Regulatory Commission, Rev. 1, July 1977.
- D. National Bureau of Standards, "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure," Handbook 69, June 5, 1959.
- E. Slade, D. H., "Meteorology and Atomic Energy - 1968, USAEC, July 1968.
- F. Lowder, W. M., P. D. Raft, and G. dePlanque Burke, "Determination of N-16 Gamma Radiation Fields at BWR Nuclear Power Stations," Health and Safety Laboratory, Energy Research and Development Administration, Report No. 305, May 1976.
- G. Letter from Charles L. Miller of the United States Nuclear Regulatory Commission to John F. Schmidt of the Nuclear Energy Institute, dated December 26, 1995.
- H. "Dose vs Distance From HI-STORM 100S Containing MPC-68 and MPC-68M for Vermont Yankee ISFSI" Holtec Report No: HI-2146076, Holtec Project No: 2347, Approved on 04/13/2017. (VY-RPT-17-0002)
- I. Certificate of Compliance, Holtec International, Final Safety Analysis Report, Amendment 2, 06/07/05. (HI-STORM 1-13)
- J. Certificate of Compliance, Holtec International, Final Safety Analysis Report, Amendment 10, 05/31/16 (HI-STORM 14-58)
- K. AREVA Calculation #32-9068362-000 "Vermont Yankee Site Boundary Direct Dose From N-16 Methodology", April 4, 2008, Mark Strum and John Hamawi.
- L. "Hydrogeologic Investigation of Tritium in Groundwater, Vermont Yankee Nuclear Power Station, Vernon, Vermont" File No. 09.0025576.11, dated May, 2011. Prepared for Entergy Vermont Yankee by GZA, GeoEnvironmental, Inc. One Edgewater Drive, Norwood MA 02062
- M. "Vermont Yankee Shut-Down Environmental Radionuclides of Concern and Off-Site Dose Calculation Manual Changes", RSCS TSD No. 16-041

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APPENDIX B

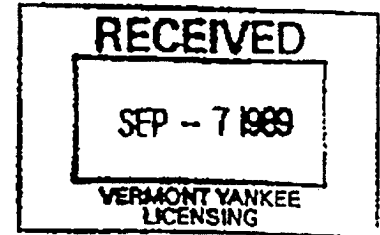
Approval of Criteria for Disposal of Slightly Contaminated
Septic Waste On-Site at Vermont Yankee

Revision 9 Date 3/2/90



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 30, 1989



Docket No. 50-271

Mr. L. A. Tremblay
Licensing Engineering
Vermont Yankee Nuclear Power Corporation
Engineering Office
580 Main Street
Bolton, Massachusetts 01740-1398

Dear Mr. Tremblay:

SUBJECT: APPROVAL UNDER 10 CFR 20.302(a) OF PROCEDURES FOR DISPOSAL OF
SLIGHTLY CONTAMINATED SEPTIC WASTE ON SITE AT VERMONT YANKEE (TAC
NO. 73776)

REFERENCE: (a) June 28, 1989 letter from R. W. Capstick to US NRC Document
Control Desk, including Attachment I and Attachment II.
(b) Final Environmental Statement related to the operation of
Vermont Yankee Nuclear Power Station, dated July 1972.

In reference (a) Vermont Yankee Nuclear Power Corporation (Vermont Yankee, or the licensee) submitted an application for disposal of licensed material on site. This disposal was not previously considered by the staff in the Vermont Yankee Final Environmental Statement (FES), reference (b). This extensive application, prepared in accordance with 10 CFR 20.302(a), contains a detailed description of the licensed material, thoroughly analyzes and evaluates the information pertinent to the effects on the environment of the proposed disposal of the licensed material, and commits the licensee to follow specific procedures to minimize the risk of unexpected or hazardous exposures. In the FES, the NRC staff considered the potential effects on the environment of licensed material from operation of the plant and, in the assessment of the total radiological impact of the Vermont Yankee Station concluded that: "...operation of the Station will contribute only an extremely small increment to the radiation dose that area residents receive from natural background. Since fluctuations of the background dose may be expected to exceed the increment contributed by the plant, the dose will be immeasurable in itself and will constitute no meaningful risk to be balanced against the benefits of the plant."

Revision 9 Date 3/2/90

Mr. L. A. Tremblay

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August 30, 1989

Since the disposal proposed by the licensee involves licensed material containing less than 0.1 percent of the radioactive materials, primarily cobalt-60 and cesium-137, already considered acceptable in the FES, and involves exposure pathways much less significant than those considered in the FES, we consider the site-specific application (Reference (a)) for Vermont Yankee Nuclear Power Station to have insignificant radiological impact. We accept the commitments and evaluations of the licensee, documented in reference (a), as further assurance that the proposed disposal procedures will have a negligible effect on the environment and the general population in comparison to normal background radiation.

LAI | In conclusion, we find the licensee's procedures with commitments as
855 | documented in reference (a) to be acceptable, provided that reference (a) is
| permanently incorporated into the licensee's Offsite Dose Calculation Manual
| (ODCM) as an Appendix, and future modifications of reference (a) be reported
| to NRC in accordance with licensee commitments regarding ODCM changes.

Pursuant to 10 CFR 51.22(c)(9), no environmental assessment is required. This completes our review under TAC No.73776.

Sincerely,

Morton B. Fairtile

Morton B. Fairtile, Project Manager
Project Directorate I-3 Division of
Reactor Projects I/II Office of
Nuclear Reactor Regulation

cc: See next page

Revision 9 Date 3/2/90

Mr. L. A. Tremblay

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Revision 9 Date 3/2/90

Mr. L. A. Tremblay

- 4 -

cc:

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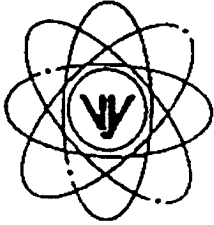
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Adjudicatory File (2)
Atomic Safety and Licensing Board Panel Docket
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

(25)

Revision 9 Date 3/2/90

VERMONT YANKEE NUCLEAR POWER CORPORATION



Ferry Road, Brattleboro, VT 05301-7002

June 28, 1989
BVY 89-59

REPLY TO
ENGINEERING OFFICE
580 MAIN STREET
BOLTON, MA 01740
(508) 778-6711

United States Nuclear Regulatory Commission
Washington, DC 20555

Attention: Document Control Desk

Reference: License No. DPR-28 (Docket No. 50-271).

Subject: Request to Routinely Dispose of Slightly Contaminated Septic Waste in
Accordance with 10 CFR 20.302(a)

Dear Sir:

In accordance with the criteria of the Code of Federal Regulations, Title 10, Section 20.302(a) (10CFR20.302(a)), enclosed please find the subject application for the disposal of very low level radioactive waste materials. Vermont Yankee Nuclear Power Corporation (Vermont Yankee) hereby requests NRC approval of the proposed procedures for the disposal of slightly contaminated septic waste generated at the Vermont Yankee Nuclear Power Plant in Vernon, Vermont.

This application specifically requests approval to dispose of septic tank waste, contaminated at minimal levels, which have been or might be generated through the end of station operations at the Vermont Yankee Nuclear Power Plant. The proposed method of disposal is for the on-site land spreading in designated areas in compliance with State of Vermont health code requirements for septic waste. Disposal of this waste in the manner proposed, rather than at a 10 CFR Part 61 licensed facility would save Vermont Yankee not only substantial cost, but also valuable disposal site space which would then be available for wastes of higher radioactivity levels. Disposal as radioactive waste would require treatment of the biological aspects of the septage and solidification to a stable waste form, thereby increasing the volume substantially.

A radiological assessment and proposed operational controls, based upon the continued on-site disposal of septic waste as presently contained in the plant's septic tanks, are detailed in Attachments 1 and 2. Based upon this analysis, Vermont Yankee requests approval to dispose of septic tank waste on-site by land spreading in such a manner that the radioactivity concentration limit in any batch of septage to be spread does not exceed one-tenth of the MPC values listed in 10 CFR 20, Appendix B, Table II; and the combined radiological impact for all disposal operations shall be limited to a total body or organ dose of a maximally exposed member of the public of less than one mrem/year (less than 5 mrem/year to an inadvertent intruder).

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United States Nuclear Regulatory Commission
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Page 2

Due to our expected need to utilize the proposed methodology of land application of septic waste on-site during the spring of 1990, we request your review and approval of this proposed disposal method by the end of the first quarter of 1990.

We trust that the information contained in the submittal is sufficient; however, should you have any questions or require further information concerning this matter, please contact this office

Very truly yours,

VERMONT YANKEE NUCLEAR POWER CORPORATION

A handwritten signature in black ink, appearing to read "Robert W. Capstick, Jr.", written in a cursive style.

Robert W. Capstick, Jr.
Licensing Engineer

MSS/emd

Enclosures

cc: USNRC - Region I
USNRC - Resident Inspector, VTNPS

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ATTACHMENT 1

VERMONT YANKEE NUCLEAR POWER PLANT

APPLICATION FOR APPROVAL TO ROUTINELY DISPOSE OF
SEPTIC WASTE WITH MINIMAL LEVELS OF RADIOACTIVITY

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ATTACHMENT 1

VERMONT YANKEE NUCLEAR POWER PLANT

Application for Approval to Routinely Dispose of Septic Waste With Minimal Levels of Radioactivity

1.0 INTRODUCTION

Vermont Yankee Nuclear Power Corporation (Vermont Yankee) requests approval, pursuant to 10CFR20.302(a), of a method proposed herein for the routine disposal of slightly contaminated septic tank waste. Vermont Yankee proposes to dispose of this waste by spreading it on designated areas within the plant's site boundary fence. This application addresses specific information requested in 10CFR20.302(a).

2.0 WASTE STREAM DESCRIPTION

The waste involved in this application consists of residual solids and water associated with the sewage collection system at Vermont Yankee. The plant's sewage systems are of the septic tank and disposal field type. The two systems servicing the majority of the plant's sanitary waste are identified as (1) main septic system and (2) the south sewage disposal system.

The main septic system (design flow capacity 4,950 gallons/day) consists of a wastewater lift station, septic tank, and dual alternating disposal fields located on the north side of the plant. This system services the main complex of buildings central to the plant and processes approximately 3,500 gallons of wastewater per day. The septic tank, shown in Figure (1), will typically contain 9,250 gallons of septage.

The south sewage disposal system is a newly-installed (January 1989) pressurized mound system, which is used in lieu of the construction office building (COB) holding tank that had previously serviced the lavatory facilities on the south end of the plant. The new system is composed of a septic tank (5,700 gallon capacity, see Figure 2), pumping station, and pressurized mound disposal field. When dosing the field, a force main pressurizes the disposal field's piping system with the septic tank effluent, which distributes throughout the field. The south sewage disposal system has

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the design flow capacity to process 4,607 gallons of wastewater per day. The system is typically loaded at approximately 2,500 gallons per day during normal plant operations. Figure (3) indicates diagrammatically the flow of both potable and wastewater throughout Vermont Yankee.

Both the main septic system and the south sewage disposal system's septic tanks collect waste from the plant's lavatories, showers, kitchens, and janitorial facilities outside the Radiological Control Area (RCA). No radioactivity is intentionally discharged to either of the septic systems. However, plant investigations into the source of low levels of contamination found in septic waste have identified that very small quantities of radioactive materials, which are below detection limits for radioactivity releases from the RCA, are carried out of the control area on individuals and accumulate in the septic waste collection tanks by way of floor wash water, showers, and hand washing. As a means of minimizing the transport of radioactive materials into the septic collection tanks, the primary source of the radioactivity (i.e., floor wash water) is now poured through a filter bag to remove suspended solids and dirt before the water is released into a janitorial sink.

The majority of the radioactivity found in waste sludge has been associated with the main septic tank. Grab samples of sludge from the bottom of the COB and main septic tank were analyzed by gamma spectroscopy with the following results of plant-related radionuclides:

	<u>Isotope</u>	<u>Activity Concentration ±1 Sigma (pCi/kg Wet)</u>
COB Sludge (June 8, 1988)	Cs-137	10.3 ± 1.8
	Co-60	45.4 ± 3.1
Main Tank Sludge (June 8, 1988)	Mn-54	39.3 ± 4.3
	Co-60	853.0 ± 12.0
	Zn-65	52.7 ± 8.2
	Cs-134	13.0 ± 2.2
	Cs-137	120.7 ± 5.2

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The principle radionuclide is Cobalt-60, which accounts for 79% of the plant related activity in the septage samples. In comparison to in-plant smear samples taken for 10CFR61 waste characterizations, the septage sample from the main tank correlates very close with the distribution of radionuclides identified in-plant as shown below:

Relative Isotopic Distributions

<u>Isotope</u>	<u>In-Plant Smears</u>	<u>Main Tank Sludge</u>
Mn-54	3.6%	3.6%
Co-60	81.5	79.1
Zn-65	3.8	4.9
Cs-134	0.4	1.2
Cs-137	10.3	11.2

Additional analyses of the main tank septage showed that the liquid portion of the collected sample did not contain any plant-related activation or fission products, and that essentially all of the activity in the waste was associated with the solid sludge fraction. The average density of the collected sludge was found to be approximately equal to that of water, with a wet to dry ratio of 25.4 to 1.

Both the liquid and solid fractions of the main tank septic waste were also analyzed for strontium with no detectable activity found. The liquid portion of the waste sample was also analyzed for tritium with no activity above the minimum detectable levels found. Appendix A to Attachment 2 contains the laboratory analysis reports of the samples taken from the COB and main septic tanks.

Prior to identification of the plant-related radioactivity in septage waste, the COB holding tank was being pumped on the average of twice per week, with the sludge and waste liquid transported off-site primarily to the Brattleboro, Vermont, sewage treatment facility. Waste from the main septic tank was being pumped and transported off-site for disposal on the average of twice per year.

With the replacement of the COB holding tank by the new south sewage disposal system, and the requested implementation of on-site land disposal of accumulated septic waste, the frequency of collection tank pump-outs with land application of the waste is expected to be once per year. With the past pump-out frequency of the main tank being every six months, the accumulation of sludge at the bottom of the tank was well below its design capacity. During the 1988 sample collections, it was estimated that the sludge thickness was less than 1 foot of its 6-foot depth. However, for conservatism in the radiological evaluations, it is assumed that the sludge layer in the main septic tank and south disposal tank occupies 30% of their combined design volume, and that the frequency of pump-outs is semiannual as opposed to the expected annual cycle. Also, as noted above from laboratory analyses of the sludge layer taken from the bottom of the main tank, the average density of the tank contents is approximately equal to that of water, with a wet-to-dry ratio of 25.4 to 1. Hence, the weight of solids (W_{sol}) being disposed of is estimated, for purposes of this bounding dose assessment, to be approximately:

$$\begin{aligned} W_{sol} &= 14,950 \text{ [gal]} \times 3,785.4 \text{ [cc/gal]} \times 10^{-3} \text{ [kg/cc]} \\ &\quad \times 0.30 \text{ [solids fraction]} \times (1/25.4) \text{ [dry/wet ratio]} \\ &\sim 700 \text{ [kg]} \text{ per pump-out of both tanks} \end{aligned}$$

or, 1,400 kg of dry solids per year.

3.0 DISPOSAL METHOD

Approval of this application will allow Vermont Yankee to dispose of septage by utilization of a technique of land spreading or surface injection in a manner consistent with all applicable state of Vermont health regulations regarding disposal of septic waste. Details of the chemical and biological controls necessary to satisfy state health code requirements are provided in Reference 5.

The septage will be spread or surface injected on land areas owned by Vermont Yankee and situated within the plant's site boundary. Transportation of the septage waste to the disposal areas will involve pumping from one of the septic waste collection tanks (i.e., main septic tank, COB holding tank,

new replacement COB septic tank, or from any other on-site septic waste collection point) into an enclosed truck-mounted tank. The enclosed tank truck is used to prevent spillage while in transit to the disposal areas. The septage will be transported to one of the two disposal sites designated for land application for septage from Vermont Yankee, and applied at a fixed rate based on either limitations imposed by the state of Vermont for heavy metals or organic content of the waste, or on the radioactivity content such that projected maximum individual doses will not exceed established dose objectives.

3.1 Septic Waste Disposal Procedure

Gamma isotopic analysis of septic waste shall be made prior to each disposal by obtaining a representative sample from each tank prior to pump-out. At least two septic waste samples will be collected from each tank to be pumped by taking a volumetric column of sludge and waste water which allows for analysis of the solid's distribution and content from top to bottom of each tank. The weight percent of solid content of the collected waste will be determined and applied to the gamma isotopic analysis in order to estimate the total radioactivity content of each tank to be pumped and spread on designated disposal fields.

These gamma isotopic analyses of the representative samples will be performed at the environmental Technical Specification lower limit of detection (LLD) requirements for liquids (see Technical Specification Table 4.9.3) in order to document the estimation of radiological impact from septage disposal.

The radionuclide concentrations and total radioactivity identified in the septage will be compared to the concentration and total curie limits established herein prior to disposal. The methodology and limits associated with determining compliance with the disposal dose and activity criteria are described in Attachment 2. If the concentration and total activity limits are met, compliance with the dose assessment criteria will have been demonstrated since the radiological analysis (Section 4.5 and Attachment 2) was based on evaluating the exposure to a maximally exposed individual and inadvertent intruder after the accumulation of twenty years of periodic semiannual

spreading of the septic waste on a single (2 acre) plot within one of the designated disposal areas. If the activity limit per disposal area is projected to be exceeded, the appropriate exposure pathways as described in Section 4.5 will be evaluated prior to each additional application, or a separate plot within the designated disposal area will be utilized.

Annually, for years in which disposal occurs, the potential dose impact from disposal operations conducted during the year, including the impact from previous years, will be performed and results reported in the plant's Semiannual Radioactive Effluent Release Report which is filed after January 1. All exposures will be assessed utilizing the methodology described in Attachment 2.

The established dose criteria requires that all applications of septage within the approved designated disposal areas shall be limited to ensure the dose to a maximally-exposed individual be maintained less than 1 mrem/year to the whole body and any organ, and the dose to the inadvertent intruder be maintained less than 5 mrem/year. The total activity based on the measured radionuclide distribution for any single disposal plot is not expected to exceed the following:

<u>Isotope</u>	Maximum Accumulated Radioactivity Allowed
	<u>Per Acre Q_{lim} [μCi]</u>
Mn-54	1.4
Co-60	120.0
Zn-65	1.4
Cs-134	0.7
Cs-137	46.5

If any of the above radionuclides are projected to exceed the indicated activity values, then dose calculations will be performed prior to spreading, in accordance with the methods detailed in Section 4.2.2 of Attachment 2, to make the determination that the dose limit criteria will not be exceeded.

The concentration of radionuclides in any tank of septic waste to be disposed of will also be limited to a combined Maximum Permissible Concentration of Water (MPC) (as listed in 10CFR, Part 20, Appendix B, Table II, Column 2) ratio of less than or equal to 0.1.

For radiological control, each application of septage will be applied on the designated land area by approved plant procedure which adheres to the following assumptions which were used in developing the dose impact:

- o During surface spreading or injection, the septage, and any precipitation falling onto or flowing onto the disposal field, shall not overflow the perimeter of the designated area.
- o Septage shall not be surface spread or injected into the top 6-inch soil layer within 300 feet from any drinking water well supply.
- o Septage shall not be surface spread closer than 300 feet from the nearest dwelling or public building (or within 100 feet if injected into the top 6-inch surface layer).
- o Septage shall not be surface spread closer than 50 feet (or within 25 feet if injected into the top 6-inch surface layer) from any roads or site boundary adjacent to land areas.
- o Septage shall not be surface spread within 100 feet (or within 50 feet if injected into the top 6-inch surface layer) of any surface water (rivers, streams, drainage ditches).
- o Low areas of the approved fields, subject to seasonally high groundwater levels, are excluded from the septage application.

In addition to the radiological controls to limit the total accumulation of radioactive materials released by septic waste spreading, state of Vermont health code requirements will be followed to ensure the protection of the public and environment from chemical and biological hazards. The application rate and acreage will be determined prior to each

disposal operation. This will vary with the chemical composition of the septage, the percent solids, and the radioactive concentrations.

3.2 Administrative Procedures

Complete records of each disposal will be maintained. These records will include the concentration of radionuclides in the septage, the total volume of septic waste disposed, the total activity in each batch as well as total accumulated on the disposal plot at time of spreading, the plot on which the septage was applied, and the results of any dose calculations required.

The annual disposal of septage on each of the approved plot areas will be limited to within the established dose, activity, and concentration criteria noted above, in addition to limitations dictated by chemical and biological conditions. Dose guidelines, and concentration and activity limits, will be maintained within the appropriate values as detailed in Attachment 2.

Any farmer using land which has been used for the disposal of septic waste will be notified of any applicable restrictions placed on the site due to the land spreading or injection of waste.

4.0 EVALUATION OF ENVIRONMENTAL IMPACT

4.1 Site Characteristics

4.1.1 Site Topography

The proposed disposal sites consist of two fields located on the Vermont Yankee Nuclear Power Plant site, which is located on the west bank of the Connecticut River in southwestern Vermont at latitude 42 degrees, 47 minutes north and longitude 72 degrees 31 minutes west. Both fields are on plant property within the site boundary and surrounded by a chain link fence.

Site A contains an approximate eight-acre parcel of usable land centered approximately 2,200 feet northwest of the Reactor Building. Site B contains about two acres and is centered approximately 1,700 feet south of the Reactor Building. The usable acreage of both the north and south disposal fields is restricted to those areas which have no slopes greater than five percent to limit surface runoff. A radiological assessment based on the 1988 measured radioactivity concentrations in sludge has determined that a single two-acre plot would be sufficient for the routine disposal of septage for twenty years without exceeding the dose criteria to maximum exposed individual or inadvertent intruder. As a result, the eight-acre field to the northwest could be divided into four disposal plots, with the two-acre site at the south end of the plant site, providing a fifth plot. A portion of the United States Geological Survey topographic map (Brattleboro quadrangle), showing the plant site, is presented in the Final Safety Analysis Report (FSAR) as Figure 2.5-1. A plan map showing the plant site and the disposal sites is given on Figure 4.

The sites are located along a glacial terrace on the west side of the Connecticut River. This terrace extends about 3,000 feet west rising gently and then more abruptly to a higher terrace and then to dissected uplands. Distance to the east from the disposal sites to the river is at least 100 feet if septage is disposed of by surface spreading within the designated areas, or 50 feet if septage is injected directly into the soil.

Relief of the proposed disposal sites is low, with elevation ranging between 250 feet and 265 feet (msl). Mean water surface elevation of the adjacent river is about 220 feet.

The topographic character of the site and surrounding area is compatible with this use. The spreading of septage at these locations will have no effect on the topography of the area.

4.1.2 Site Geology

Profiles of site exploratory borings are shown in the FSAR in Figures 2.5-8 through 2.5-11. Current site characteristics as determined from a recent detailed site investigation can be found in Reference 5.

Composition of surfacial materials is compatible with the proposed use of the site for septic waste disposal.

4.2 Area Characteristics

4.2.1 Meteorology

The site area experiences a continental-type climate with some modification due to the marine climate which prevails at the Atlantic seacoast to the east. Annual precipitation averages 43 inches and is fairly evenly distributed in each month of the year.

Potential impacts on septic waste disposal include occasional harsh weather: ice storms, severe thunderstorms, heavy rains due to hurricanes, the possibility of a tornado, and annual snowfall of from 30 to 118 inches per year. In addition, frozen ground can occur for up to 4 months of the year.

Septage spreading will be managed by written procedure such that material which is spread or a mix of that material with precipitation will not overflow the perimeter of the disposal site.

Additional information on meteorology of the site can be found in Section 2.3 of the Final Safety Analysis Report.

4.2.2 Hydrology

Hydrology of the site and local area is tied closely to flow in the adjacent Connecticut River. River flow is controlled by a series of hydroelectric and flood-control dams including the Vernon Dam which is about 3,500 feet downstream of the site.

All local streams drain to the Connecticut River and the site is in the direct path of natural groundwater flow from the local watershed easterly toward the river. Site groundwater level is influenced by both precipitation and changes in the level of ponding of the Connecticut River behind the Vernon Dam due to natural flow or dam operation.

Flood flows on the Connecticut are controlled by numerous dams including five upstream of the site. Elevation of the 100-year flood is about 228 ft (msl); and, thus, well below the elevation of the proposed site which ranges from about 250 to 265 feet (msl). The 100-year flood level is based on information presented in References (1) and (2).

Septage disposal by means of land spreading on the proposed site will have no adverse impact on area hydrology.

Further information about site hydrology is in Section 2.4 of the FSAR.

4.3 Water Usage

4.3.1 Surface Water

The adjacent Connecticut River is used for hydroelectric power, for cooling water for the Vermont Yankee plant, as well as for a variety of recreational purposes such as fishing and boating. The Connecticut River is not used as a potable water supply within 50 miles downstream of the plant.

Locally, water from natural springs are used for domestic and farm purposes. FSAR Table 2.4.5 and Figure 2.4-2 show springs used within a 1-mile radius of the site. FSAR Table 2.4.4 and Figure 2.4-1 show water supplies with surface water sources which are within a ten-mile radius of the site.

There will be no impact on surface water usage or quality as a result of septage disposal due to the required separation distances between surface waters and the disposal plots.

4.3.2 Groundwater

Based on a review of groundwater measurements in various site borings presented in the FSAR and References 3 and 5, an upper estimate of groundwater levels at the plant is about 240 feet. Considering the proximity of the Connecticut River and Vernon Pond, with a mean water surface elevation of 220 feet, this estimate for the groundwater level appears to be reasonable. Given the topography of the proposed disposal sites, it is highly unlikely that the groundwater level will be within 3 feet of the disposal area surface elevation. Prior to each application of septic waste to a disposal plot, the groundwater level in nearby test wells will be determined and no application will be allowed if the groundwater level in the vicinity of the disposal plot is found to be less than 3 feet.

Groundwater provides potable water for public wells as shown in FSAR Table 2.4.5 and Figure 2.4-1. Groundwater flow in the vicinity of the proposed disposal sites is towards the Connecticut River. There are no drinking-water wells located between the site and the river. Therefore, it is highly unlikely that any drinking water wells could be affected by septage disposal. FSAR Figure 2.4-2 and Table 2.4-5 present information on private wells near the plant.

The Vermont Yankee on-site wells provide water for plant use. This supply is routinely monitored for radioactive contamination.

To quantify the impact of septage disposal on the Connecticut River, a conservative groundwater/radionuclide travel time analysis was performed. For an assumed average travel distance of 200 feet from the disposal site to the river, a groundwater travel time of 408 days was estimated from Darcy's Law. This estimate is based on a permeability for the glacial till of 10 gpd/ft^2 , a hydraulic gradient of 0.11 ft/ft, and a soil porosity of 0.3. This analysis conservatively assumed that the septage placed on the ground was immediately available to the groundwater. In practice, a minimum of 3 feet separation between groundwater and the surface will be required at time of application of the septic waste.

Due to ionic adsorption of the radionuclides on solid particles in the groundwater flow regime, most radionuclides travel at only a small fraction of the groundwater velocity. For the radionuclides present in the sludge, retardation coefficients were developed from NUREG/CR-3130 (Reference 4). Retardation coefficients for Co-60, Cs-137, and Cs-134 were directly obtained from NUREG/CR-3130. The coefficients for Zn-65 and Mn-54 were conservatively estimated using NUREG/CR-3130 as a guide. The radionuclides, their half-lives, retardation coefficients, and their travel time to the river are summarized in Table 1.

TABLE 1
Radionuclide Travel Times

<u>Radionuclide</u>	<u>Half Life</u>	<u>Retardation Coefficient</u>	<u>Travel Time to River</u>
Co-60	5.3 years	860	961 years
Cs-137	30.2 years	173	193 years
Cs-134	2.1 years	173	193 years
Zn-65	244 days	3	1,224 days
Mn-54	312 days	3	1,224 days

The radiological impact on the river for the radionuclides reaching the river under this conservative analysis is discussed in Attachment 2. Water usage of the Connecticut River downstream from the disposal area is limited to drinking water for dairy cows, irrigation of vegetable crops, and irrigation of cow and cattle fodder.

Based on the assessments noted above, it is concluded that groundwater sources will not be adversely impacted as a result of septage disposal on the proposed site.

4.4 Land Use

Both the eight-acre and two-acre sites proposed for the disposal areas are currently part of the Vermont Yankee Nuclear Power Plant Site inside the plant's site boundary which is enclosed by a chain link fence. It is

undeveloped except for transmission line structures which traverse a portion of the northern disposal area. Development potential is under the control of Vermont Yankee. At present, the eight-acre site on the north end of the plant property is used by a local farmer for the growing of feed hay for use with his dairy herd. No curtailment of this activity as a result of the low levels of radioactivity in septage will be necessary.

Utilization of the proposed sites for septic waste disposal will result in no impact on adjacent land or properties because of the separation of the disposal plots from off-site properties, the general movement of groundwater toward the river and away from adjacent land areas, and the very low levels of radioactive materials contained in the waste. Administrative controls on spreading and the monitoring of disposal area conditions will provide added assurance that this proposed practice will not impact adjacent properties.

4.5 Radiological Impact

In addition to state of Vermont limits imposed on septage spreading, based on nutrient and heavy metal content, the amount of septage applied on each of the proposed disposal plots will also be procedurally controlled to insure doses are maintained within the stated limits. These limits are based on NRC Nuclear Reactor Regulation (NRR) staff proposed guidance (described in AIF/NESP-037, August 1986). The proposed dose criteria require that the maximally exposed member of the general public receive a dose less than 1 mrem/year to the whole body or any organ due to the disposal material, and less than 5 mrem/year to an inadvertent intruder.

To assess the doses received by the maximally-exposed individual and the inadvertent intruder, six potential pathways have been identified. These include:

- (a) Standing on contaminated ground,
- (b) Inhalation of resuspended radioactivity,

- (c) Ingestion of leafy vegetables,
- (d) Ingestion of stored vegetables,
- (f) Ingestion of meat, and
- (g) Ingestion of milk.

The liquid pathway was also evaluated and determined to be insignificant. Both the maximum individual and inadvertent intruder are assumed to be exposed to these pathways with difference between the two related to the occupancy time. The basic assumptions used in the radiological analyses include:

- (a) Exposure to the ground contamination and to resuspended radioactivity is for a period of 104 hours per year during Vermont Yankee active control of the disposal sites, and continuous thereafter. The 104-hour interval being representative of a farmer's time on a plot of land (4 hours per week for 6 months).
- (b) The septic tanks are emptied every two to three years. (The assumed practice is to pump septic tanks once per year. The actual practice may be to pump septic tanks every two to three years.)
- (c) The tank radioactivity remains constant at the currently determined level. To account for the uncertainty associated with the counting statistics, the measured activity concentrations listed in Section 2 were increased by 3 sigmas. That is, the activity concentrations employed in dose assessment and the total radioactivity content per pump-out (at 700 kg of solids per batch) are as follows:

<u>Isotope</u>	<u>Upper-Bound Activity Concentration [pCi/kg dry]</u>	<u>Upper-Bound Activity Content [Ci/tankful]</u>
Mn-54	1,348	9.436E-07
Co-60	23,060	1.614E-05
Zn-65	1,620	1.134E-06
Cs-134	322	2.254E-07
Cs-137	4,100	2.870E-06

- (d) The radiation source corresponds to the accumulation of radioactive material on a single plot (two-acre) within the proposed disposal sites over a period of 20 years (40 applications at 6-month intervals). (In actuality, the proposed sites will accommodate more than one disposal plot, and, in practice, more than one plot will most probably be used with an application frequency of once per year.)
- (e) For the analysis of the radiological impact during Vermont Yankee active control of the disposal sites, all dispersed radioactive material remains on the surface and forms a source of unshielded radiation. (In practice, the septic waste will be either surface spread or directly injected within the top 6 inches of the disposal plot, in which case, the radioactive material will be mixed with the soil. This, in effect, would reduce the ground plane source of exposure by a factor of about four due to self-shielding.)
- (f) No radioactive material is dispersed directly on crops for human or animal consumption, crop contamination being only through root uptake.
- (g) The deposition on crops of resuspended radioactivity is insignificantly small.

- (h) Pathway data and usage factors used in the analysis are the same as those used in the plant's ODCM assessment of the off-site radiological impact from routine releases, with the exception that the fraction of stored vegetables grown on the disposal plots was conservatively increased from 0.76 to 1.0 (at present no vegetable crops for direct human consumption are grown on any of the proposed disposal plots).
- (i) It is conservatively assumed that Vermont Yankee relinquishes control of the disposal sites after the fortieth pump-out (i.e., the above source term applies also for the inadvertent intruder).
- (j) For the analysis of the impact after Vermont Yankee control of the sites is relinquished, the radioactive material is plowed under and forms a uniform mix with the top six inches of soil; but, nonetheless, undergoes resuspension at the same rate as surface contamination.

From radiological impact assessments associated with the disposal of septage on different plot sizes (Attachment 2), it was determined that a single two-acre plot within the disposal sites would accommodate the 1 mrem/year prescribed dose to the critical organ of the maximally exposed individual for a period of up to 20 years, as well as the 5 mrem/year prescribed dose to the inadvertent intruder after control is assumed to be relinquished. The calculated potential radiation exposures following the spreading of 40 combined (main septic system and south disposal system) tankfuls (at six-month intervals) on a single two-acre plot are as follows:

<u>Control of Disposal Sites</u>	<u>Radiation Exposure</u>	<u>Individual/Organ</u>
Controlled by VYNPS (Maximum Exposed Individual)	0.1 mrem/yr 0.2 mrem/yr Maximum	Child/Whole Body Child/Liver
Uncontrolled (Inadvertent Intruder)	1.3 mrem/yr 3.9 mrem/yr Maximum	Adult/Whole Body Teenager/Lung

The individual pathway contributions to the total dose at the end of the 20-year accumulation of waste deposited on a single two-acre plot are as listed below:

Pathway-Dependent Critical Organ Doses

<u>Pathway</u>	Maximally Exposed Individual/Organ (Child/Liver) (mrem/year)	Inadvertent Intruder Critical Individual/Organ (Teenager/Lung) (mrem/year)
Ground Irradiation	0.0576	1.16
Inhalation	0.00122	2.74
Stored Vegetables	0.0913	0.00601
Leafy Vegetable	0.00467	0.00040
Milk Ingestion	0.0421	0.00229
Meat Ingestion	<u>0.00249</u>	<u>0.00012</u>
TOTAL	0.1994	3.909

In addition, an isotopic breakdown of the critical organ dose results listed above is shown in the following table:

Isotopic Breakdown of Maximum Radiation Exposures

<u>Description</u>	<u>Isotope</u>	<u>Radioactivity [μCi/2 Acres]</u>	<u>Exposure [mrem/yr]</u>
During Vermont Yankee control of the disposal sites. Maximally Exposed Individual/Organ: Child/Liver	Mn-54	2.831	0.000436
	Co-60	235.3	0.0559
	Zn-65	2.801	0.0230
	Cs-134	1.457	0.00231
	Cs-137	92.59	<u>0.118</u>
	TOTAL		0.199
After Vermont Yankee control of sites is relinquished. Inadvertent Intruder Critical Individual/ Organ: Teenager/Lung	Mn-54	2.831	0.0144
	Co-60	235.3	3.76
	Zn-65	2.801	0.00983
	Cs-134	1.457	0.000505
	Cs-137	92.59	<u>0.1247</u>
	TOTAL		3.91

Of interest are also derived dose conversion factors which provide a means of ensuring septage disposal operations within the prescribed radiological guidelines. The critical-organ (worst-case) all-pathway values per acre are as follows:

All-Pathway Critical-Organ Dose Conversion Factors
During Vermont Yankee Control of Disposal Sites

<u>Isotope</u>	<u>Individual/Organ</u>	<u>Exposure [mrem/yr-μCi/acre]</u>
Mn-54	Adult/GE-LLI	3.74E-4
Co-60	Teenager/Lung	7.14E-4
Zn-65	Child/Liver	1.64E-2
Cs-134	Child/Liver	3.18E-3
Cs-137	Child/Bone	2.66E-3

The calculational methodology and details of the radiological assessment and proposed operational controls on total activity and concentration of waste to be disposed are presented in Attachment 2.

5.0 RADIATION PROTECTION

The disposal operation will follow the applicable Vermont Yankee procedures to maintain doses as low as reasonably achievable and within the specified dose and release concentration criteria.

REFERENCES

1. Flood Insurance Study, Vernon, Vermont, Windham County, FEMA, Community No. 500137, July 25, 1980.
2. Flood Insurance Study, Town of Hinsdale, New Hampshire, Cheshire County, FEMA, Community No. 330022, October 15, 1980.
3. Vermont Yankee Well Development Evaluation by Wagner, Heindel, and Noyes, Inc. July 10, 1986.
4. NUREG/CR-3130, Influence of Leach Rate and Other Parameters on Groundwater Migration, by Dames & Moore, February 1983.
5. Vermont Yankee Nuclear Power Corporation On-Site Septage Disposal Plan, by Wagner, Heindel, and Noyes, Incorporated, June 1989.

FIGURE 1
MAIN SEPTIC TANK

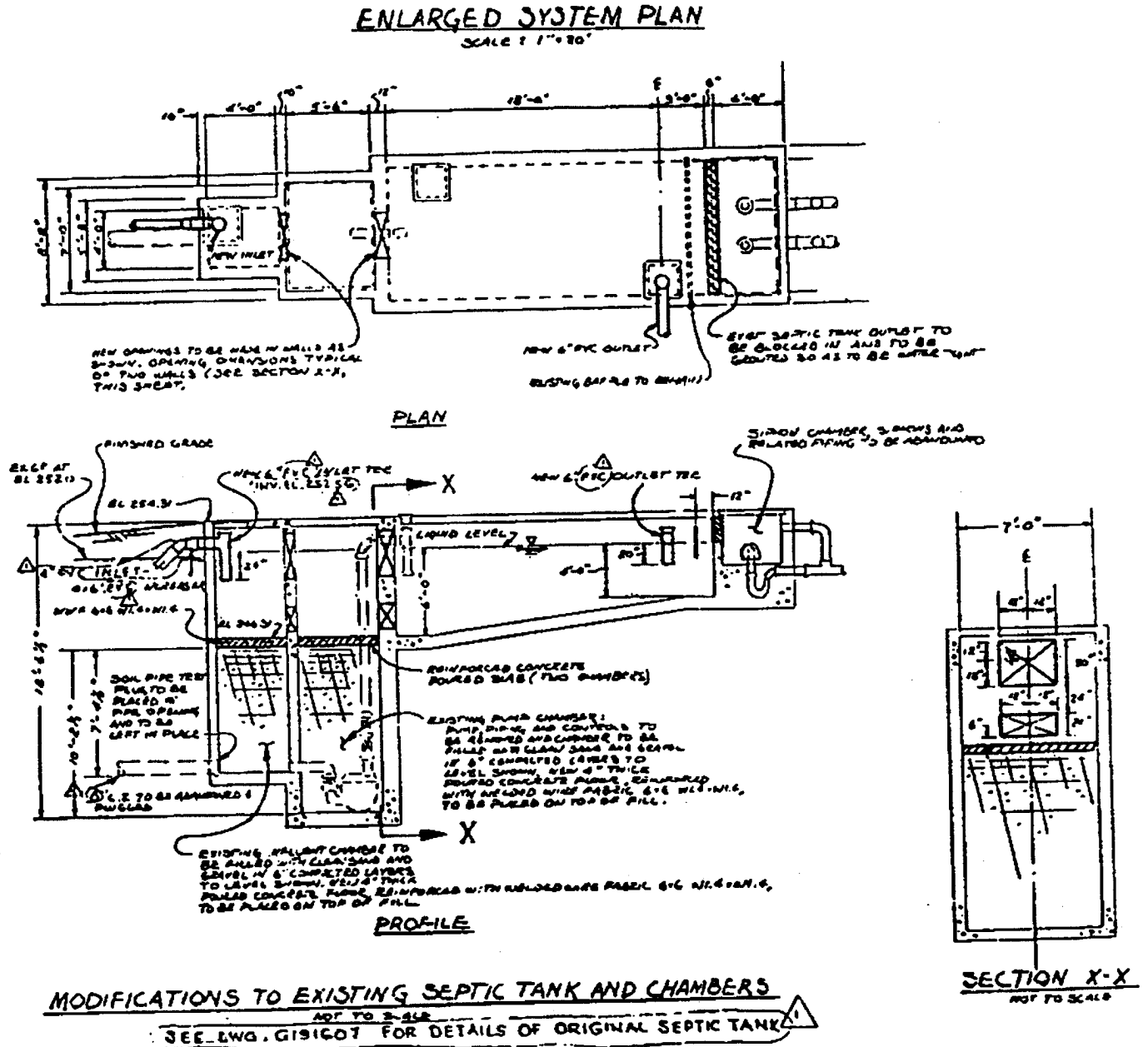


FIGURE 2

SOUTH DISPOSAL SYSTEM

SEPTIC TANK

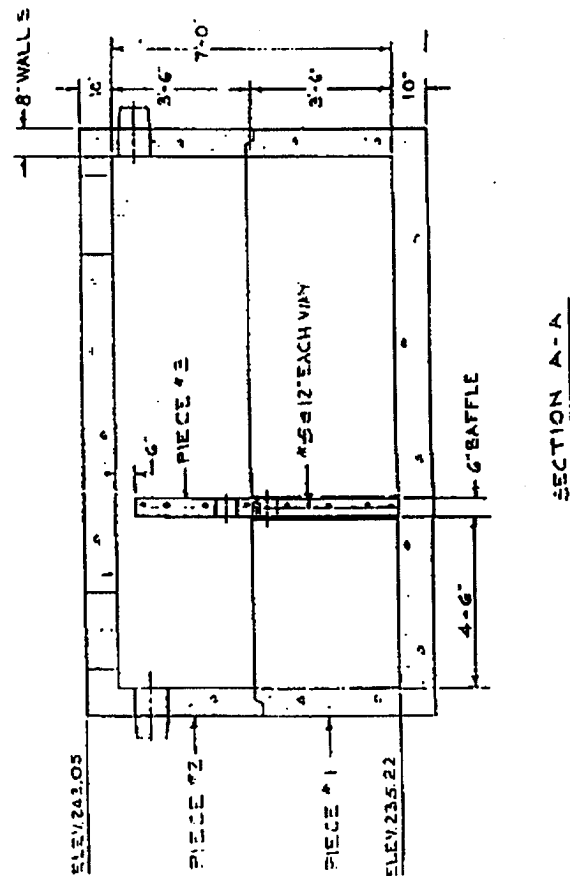
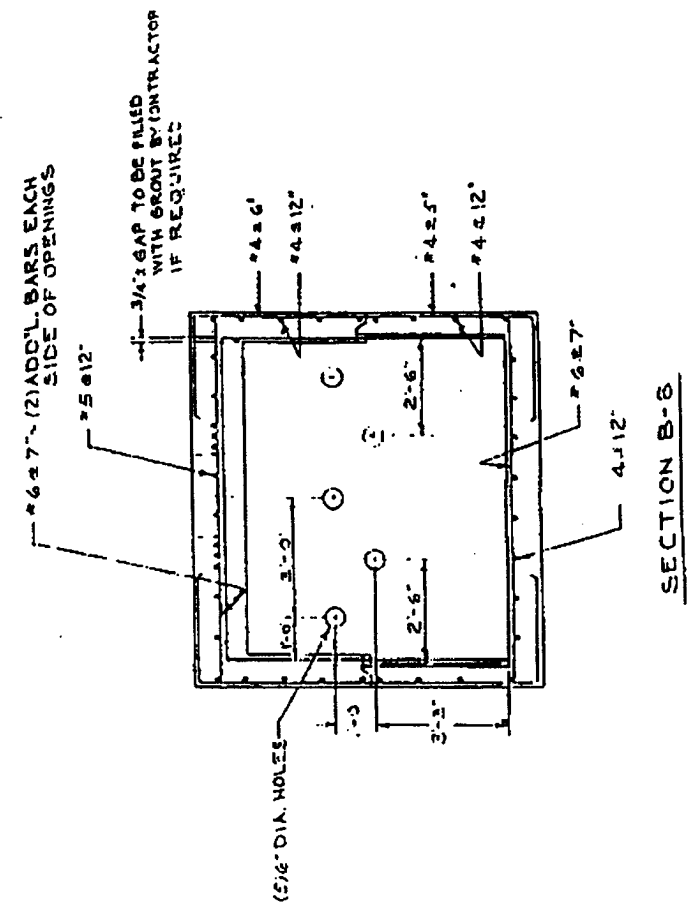
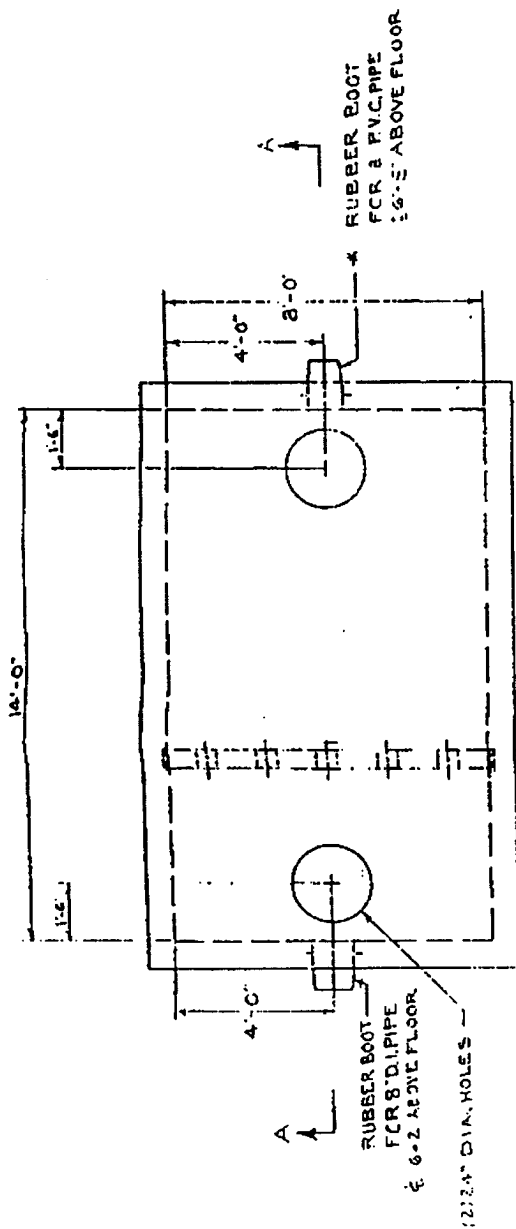
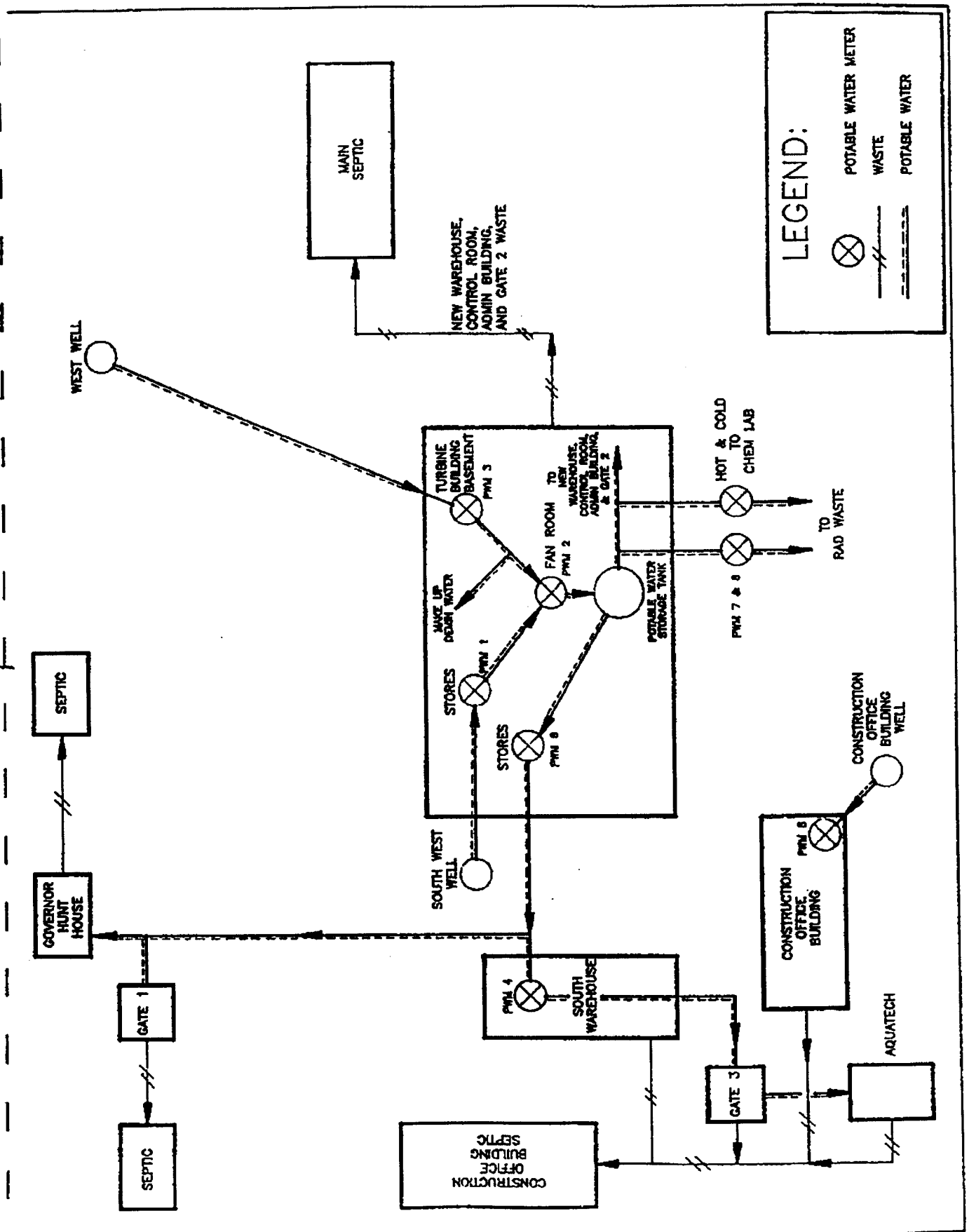
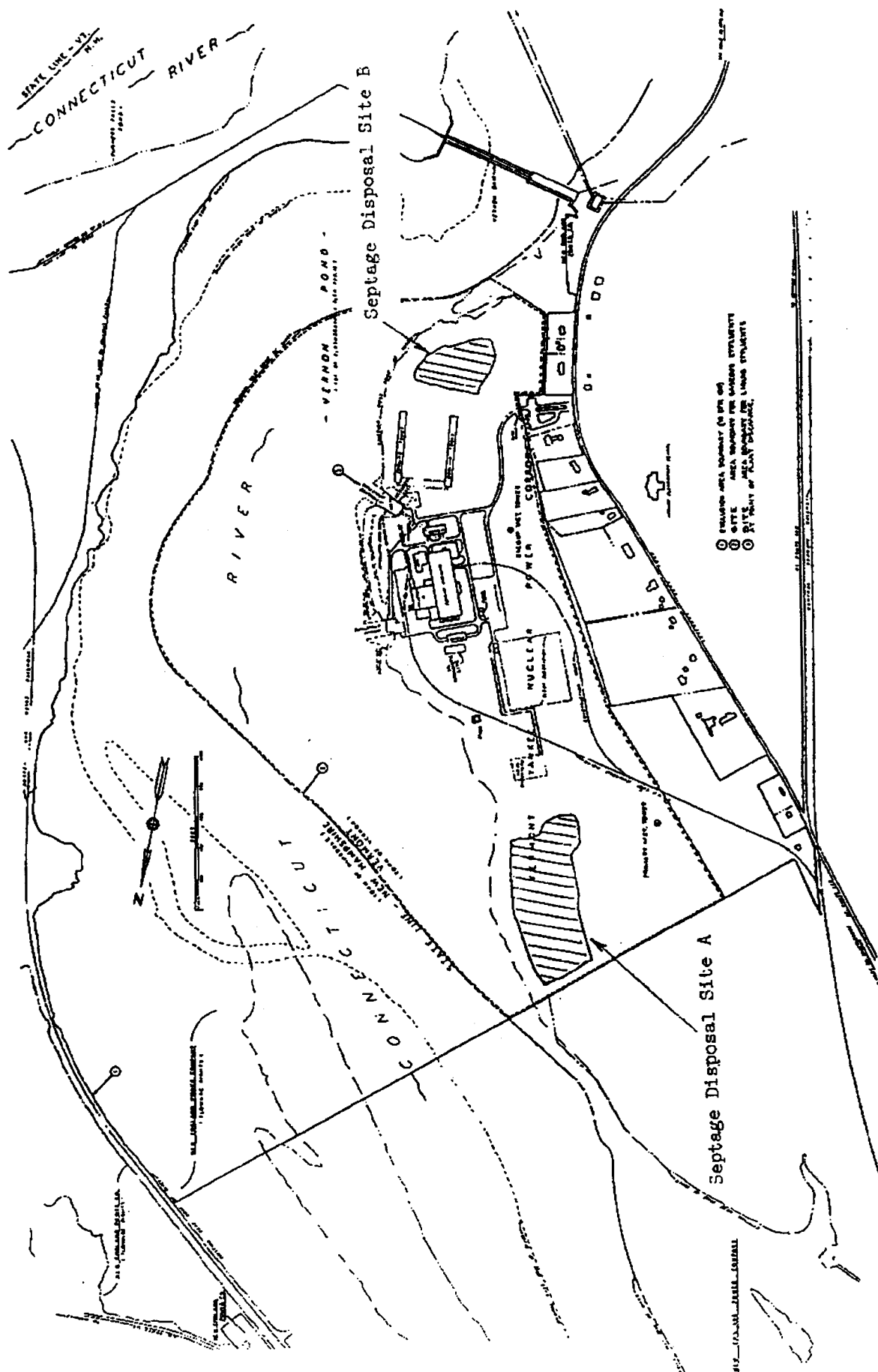


FIGURE 3
POTABLE AND WASTE WATER FLOW





(This attachment to Appendix B is incorporated into the ODCM by reference due to size. A complete copy is on file with Vermont Yankee Document Control as part of Correspondence Letter BVY 89-59.)

ATTACHMENT 2

VERMONT YANKEE NUCLEAR POWER PLANT

RADIOLOGICAL ASSESSMENT OF

ON-SITE DISPOSAL OF SEPTIC WASTE

and

PROPOSED PROCEDURAL CONTROLS TO ENSURE

COMPLIANCE WITH RADIOLOGICAL LIMITS

Revision 9 Date 3/2/90

APPENDIX D

ASSESSMENT OF SURVEILLANCE CRITERIA
FOR GAS RELEASES FROM
WASTE OIL INCINERATION

Revision 16 Date 10/28/93

APPENDIX D

ASSESSMENT OF SURVEILLANCE CRITERIA FOR GAS
RELEASES FROM WASTE OIL INCINERATIONINTRODUCTION:

The Nuclear Regulatory Commission amended its regulations (10CFR20) in a Federal Register Notice (Vol. 57, No. 235; page 57649 / Monday, December 7, 1992) that permitted the on-site incineration of contaminated waste oil generated at licensed nuclear power plants without the need to amend existing operating licenses. This action will help to ensure that the limited capacity of licensed low level waste disposal facilities is used efficiently while maintaining releases from operating nuclear power plants at levels which are "as low as reasonably achievable." Incineration of this class of waste must be in full compliance with the Commission's current regulations that restrict the release of radioactive materials to the environment. Any other applicable Federal, State, or local requirements that relate to the toxic or hazardous characteristics of the waste oil would also have to be satisfied.

Incineration of waste oil is to be carried out under existing effluent limits, recordkeeping and reporting requirements. Specifically, licensees must comply with the effluent release limitations of 10CFR Part 20, and Part 50; Appendix I. This includes the site gaseous pathway dose and dose rate limits contained in the plant's Technical Specifications (Section 3.8). The dose contribution to members of the public resulting from the on-site incineration of contaminated waste oil must not cause the total dose or dose rate from all effluent sources to exceed the dose or concentration limits imposed by 10CFR20, 10CFR50; Appendix I, and the Radiological Effluent Technical Specifications (RETS). It is expected that the actual contribution to public exposures caused by waste oil burning will be a small fraction of the site's effluent limits, as well as a small portion of the total releases from the site.

SOURCE DESCRIPTION

Contaminated waste oil suitable for on-site incineration can be burned in the Waste Oil Burner located in the North Warehouse. The burner has its own exhaust stack situated on the roof of the warehouse. However, due to the short height of the exhaust stack above the roof line, this release point is considered to be a ground level point source for modeling discharges to the environment. In addition, the building wake effects from the North Warehouse are assumed to be independent of the larger Turbine Hall/Reactor Building complex due to its distance from these main structures. Consequently, the relatively small size of the North Warehouse leads to

Revision 16 Date 10/28/93

meteorological dispersion factors that are conservative with respect to the dispersion factors for the main plant structures.

The waste oil burner is rated to process oil at a 2 gal./hour from a 500 gallon day tank. The offgas flow rate for the burner is rated as 199 cfm. This provides an air to oil dilution during the incineration of 44,800.

WASTE OIL SAMPLING/SURVEILLANCE REQUIREMENTS

The oil burner stack is not equipped with continuous air monitoring or sampling capability for the direct determination of radiological effluent releases during the incineration process. As a consequence, sampling and analysis of the waste oil prior to its incineration is necessary to project the dose and dose rate consequences of burning contaminated oil. Calculations of projected dose from the incineration of total quantity of oil to be added to the Waste Oil Burn Day Tank for each series of burns will be performed in accordance with the methods in the ODCM and compared to the accumulated site total dose for that period before initiation of incineration. Dose rate determinations will be determined by averaging the projected dose for the quantity of radioactivity determined to be present in the oil over the expected duration of the burn necessary to incinerate the total volume to be added to the Day Tank. Inherent in this determination is the assumption that all radioactivity found to be present in each batch of oil will be released to the atmosphere during the incineration. No retention of activity in the combustion chamber is assumed in calculating the offsite radiological impact.

Normal sampling and analysis methods for gaseous release streams cannot be applied directly to liquids (waste oil). Therefore, the sampling and analysis requirements for liquids as identified in Technical Specification Table 4.8.1 shall be used to determine the level of contamination in waste oil. The stated Lower Limits of Detection (LLD) given on Table 4.8.1 provide assurance that undetectable levels of contamination up to the LLD values will not result in a significant dose impact to the maximum offsite receptor. If waste oil was burned continuously for an entire calendar quarter, and the radionuclides listed in the ODCM Dose Conversion Factor Table 1.1-12 were assumed to be present in the oil at the LLD values specified in Technical Specification Table 4.8.1, the resultant maximum organ dose would amount to only 0.28% of the ALARA quarterly limit of 7.5 mrem.

The principle limitation in the incineration of waste oil is that the site release limits contained in RETS, and implemented by the ODCM methodology, shall not be exceeded. The use of the liquid LLDs on waste oil sample analyses provide sufficient sensitivity to ensure that site dose limits will not be exceeded as a consequence of burning slightly contaminated oil.

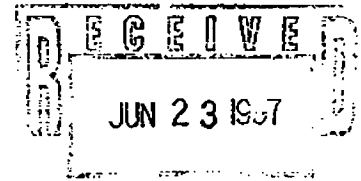
APPENDIX F

APPROVAL PURSUANT TO 10CFR20.2002 FOR ONSITE DISPOSAL OF COOLING TOWER SILT



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 18, 1997
NVY 97-85



D.A. REID

LAI 1130, 1210, 1193

Mr. Donald A. Reid
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
Ferry Road
Brattleboro, VT 05301

SUBJECT: REVISED SAFETY EVALUATION - APPROVAL PURSUANT TO 10 CFR 20.2002
FOR ONSITE DISPOSAL OF COOLING TOWER SILT - VERMONT YANKEE NUCLEAR
POWER STATION (TAC NO. M96371)

Dear Mr. Reid:

By letter dated August 30, 1995, Vermont Yankee Nuclear Power Corporation (VYNPC) requested approval, pursuant to 10 CFR 20.2002, for the onsite disposal of slightly contaminated silt material removed from Vermont Yankee Nuclear Power Station's (Vermont Yankee's) cooling towers. In a safety evaluation (SE) dated March 4, 1996, the NRC staff approved the proposed silt disposal. However, because of discrepancies VYNPC identified between the safety evaluation and VYNPC's letter of August 30, 1995, VYNPC postponed implementation of the silt disposal until resolution of the discrepancies. By letter dated August 2, 1996, VYNPC informed the NRC staff of the discrepancies and requested that the SE be revised accordingly. Recognizing the discrepancies, the NRC staff has prepared the enclosed SE to resolve the discrepancies and to replace the SE of March 4, 1996.

LAI 1130 | The NRC staff's approval of VYNPC's silt disposal request is granted provided the enclosed replacement SE is permanently incorporated into Vermont Yankee's Offsite Dose Calculation Manual as an appendix. Any modification to the proposed action that may be considered in the future must have prior NRC staff approval

Pursuant to the provisions of 10 CFR Part 51, the Commission has published in the Federal Register an Environmental Assessment and Finding of No Significant Impact (61 FR 6662).

Revision 21 Date 08/14/97

D. Reid

- 2 -

If you have any further questions regarding this matter, please contact Mr. Kahtan Jabbour at (301) 415-1496.

Sincerely,

A handwritten signature in black ink, appearing to read 'P. D. Milano', with a long horizontal flourish extending to the right.

Patrick D. Milano, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: Safety Evaluation

cc w/encl: See next page

Revision 21 Date 08/14/97

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Revision 21 Date 08/14/97

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**UNITED STATES
NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO ONSITE DISPOSAL OF SLIGHTLY CONTAMINATED COOLING TOWER SILT

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated August 30, 1995, Vermont Yankee Nuclear Power Corporation (VYNPC) requested approval for the onsite disposal of slightly contaminated silt material removed from Vermont Yankee Nuclear Power Station's (Vermont Yankee's) cooling towers. In a safety evaluation (SE) dated March 4, 1996, the NRC staff approved the proposed silt disposal. However, because of discrepancies between the SE and VYNPC's letter of August 30, 1995, VYNPC postponed implementation of the silt disposal until resolution of the discrepancies. By letter dated August 2, 1996, VYNPC informed the NRC staff of the discrepancies and requested that the SE be revised accordingly. Recognizing the discrepancies, the NRC staff has prepared this SE to resolve the discrepancies and to replace the SE of March 4, 1996.

2.0 BACKGROUND

VYNPC has previously obtained NRC staff approval of the onsite disposal of very-low-level radioactive material similar to the proposed silt disposal. By letter dated June 28, 1989, VYNPC proposed the onsite disposal of slightly contaminated septic waste material by land application at Vermont Yankee. By letter dated August 30, 1989, the NRC staff approved this request pursuant to 10 CFR 20.302 (now 10 CFR 20.2002). The NRC staff considered this site-specific application for Vermont Yankee to have insignificant radiological impact because the proposed septic waste material disposal involved licensed material containing less than 0.1 percent of the radioactive material, primarily cobalt-60 and cesium-137, already considered acceptable in the Final Environmental Statement (FES) of July 1972, and involved exposure pathways much less significant than those in the FES. In addition, the proposed septic waste material disposal satisfied the following applicable boundary conditions for the disposal of licensed material:

- a. The whole body dose to the hypothetical maximally exposed individual must be less than 1.0 mrem/year.

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- b. Doses to the whole body and any organ of an inadvertent intruder from the probable pathways of exposure are less than 5 mrem/year.
- c. The disposal must be at the same site.

Following the NRC staff's approval on August 30, 1989, VYNPC implemented the disposal of the contaminated septic waste material as proposed.

By letter dated August 30, 1995, VYNPC requested that the previous authorization for the onsite disposal of very-low-level radioactive material be amended to permit the onsite disposal of slightly contaminated silt material, within the boundary conditions of the previously approved septic waste material disposal.

3.0 EVALUATION

In its letter of August 30, 1995, VYNPC stated that the proposed silt disposal method is the same as the previously approved septic waste disposal method, and utilizes land spreading in the same onsite areas approved for septic waste disposal. The volume of silt proposed for onsite disposal consists of 14,000 cubic feet (396 cubic meters) accumulated through August 1995 plus approximately 4,000 cubic feet (113 cubic meters) to be removed from the cooling towers during each 18-month operating cycle. The activity contained in the currently accumulated silt, based on samples taken by VYNPC in June 1995, is 0.193 millicuries, principally from 0.034 millicuries of cobalt-60 and 0.159 millicuries of cesium-137. The activity contained in the additional silt to be removed from the cooling towers each 18-month operating cycle is anticipated to be 0.059 millicuries, principally from 0.012 millicuries of cobalt-60 and 0.047 millicuries of cesium-137.

VYNPC's radiological assessment enclosed with its August 30, 1995, letter demonstrates that the combined radiological impact for all onsite disposal operations, the proposed disposal of silt and the previously approved disposal of septic waste material, will continue to meet the applicable boundary conditions (given above) for the disposal of licensed material. Therefore, the proposed onsite disposal of slightly contaminated silt is acceptable.

As discussed in VYNPC's letter of August 2, 1996, if the onsite disposal of cooling tower silt or septic waste material would result in exceeding the applicable boundary conditions (given above), then VYNPC must obtain prior NRC staff approval of the disposal. In addition, VYNPC made the following commitments:

- a. VYNPC will report in the Annual Radiological Effluent Release Report a list of the radionuclides present and the total radioactivity associated with the onsite disposal activities at Vermont Yankee.

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- AI 1130 | b. VYNPC will maintain records of radionuclide concentrations and total activity associated with onsite disposal activities at Vermont Yankee in accordance with 10 CFR 50.75(g).

4.0 CONCLUSION

The NRC staff finds that the radiological conditions at the Vermont Yankee site (see attachment) that would result from the onsite disposal of slightly contaminated silt material, as proposed by VYNPC pursuant to 10 CFR 20.2002, and the previously approved onsite disposal of slightly contaminated septic waste material, are within the applicable boundary conditions (given above) for the disposal of licensed material. Therefore, the proposed onsite disposal of slightly contaminated silt removed from Vermont Yankee's cooling towers is acceptable.

LAI 1130 | VYNPC is required to permanently incorporate this SE into the Vermont Yankee Offsite Dose Calculation Manual as an Appendix to document the the radioactive material onsite disposal activities approved for Vermont Yankee, and VYNPC's related commitments regarding reporting and record keeping. Any additional modification of VYNPC's disposal activities which go beyond those proposed in the August 30, 1995, submittal, and are not addressed above must have prior NRC staff approval. In addition, any onsite disposal of cooling tower silt or septic waste material that would result in exceeding the applicable boundary conditions (given above), must also have prior NRC staff approval.

Principal Contributors: J. Minns
D. Dorman
C. Harbuck

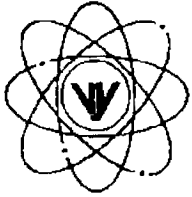
Date: June 18, 1997

Attachment: Vermont Yankee Site Area Map

Revision 21 Date 08/14/97



VERMONT YANKEE
NUCLEAR POWER CORPORATION



Ferry Road, Brattleboro, VT 05301-7002

RESPONSIBILITY

LAS 1044 CLOSED

REPLY TO
ENGINEERING OFFICE
580 MAIN STREET
BOLTON, MA 01740
(508) 779-6711

August 30, 1995
BVY 95-97

United States Nuclear Regulatory Commission
Washington, DC 20555

ATTN: Document Control Desk

References: (1) License No. DPR-28 (Docket No. 50-271)
(2) Letter from R. W. Capstick, Vermont Yankee, to USNRC, "Request to Routinely Dispose of Slightly Contaminated waste in Accordance with 10CFR20.302(a)", BVY 89-59, June 18, 1989.
(3) Letter from M. B. Fairtile, USNRC, to L. A. Tremblay, Vermont Yankee, "Approval Under 10 CFR 20.302(a) of Procedures for Disposal of Slightly Contaminated Septic Waste on Site at Vermont Yankee (TAC No. 73776)", dated August 30, 1989.

Subject: Request to Amend Previous Approval Granted Under 10 CFR 20.302(a) for Disposal of Contaminated Septic Waste

In accordance with the criteria of the Code of Federal Regulations, Title 10, Section 20.2002 (previously cited 10CFR20.302(a)), enclosed please find the subject application to amend the previously granted approval (Reference 3) to dispose of slightly contaminated septic waste on site at Vermont Yankee by expanding the allowable waste stream to include slightly contaminated Cooling Tower silt material.

This application specifically requests approval to dispose of Cooling Tower silt deposits, contaminated at minimal levels, which have been or might be generated through the end of station operations at the Vermont Yankee Nuclear Power Plant. The proposed silt disposal method is the same as the septic waste disposal method requested in Reference 2 and approved in Reference 3. The disposal method utilizes on site land spreading in the same designated areas used for septic waste. Disposal of this waste in the manner proposed, rather than holding it for future disposal at a 10CFR Part 61 licensed facility when access to one becomes available, will save substantial costs and valuable disposal site space for waste of higher radioactivity levels.

A radiological assessment and proposed operational controls based on continued on site disposal of accumulated river silt removed from the basins of the plant's mechanical draft cooling towers is contained in Enclosure A. The assessment demonstrates that the dose impact expected from the disposal of silt removed from the cooling towers during normal maintenance will not exceed the dose limits already imposed for septic waste disposal. The combined radiological impact for all on site disposal operations shall be limited to a total body or organ dose of a maximally exposed member of the public of less than one mrem/year during the period of active Vermont Yankee control of the site, or less than five mrem/year to an inadvertent intruder after termination of active site control. Enclosure B contains a copy of the original assessment and disposal procedures for

Revision 21 Date 08/14/97

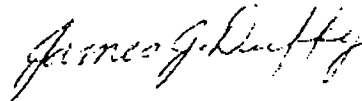
septic waste (References 2 and 3) for your use and reference in evaluating the proposed amendment.

Upon receipt of your approval, Enclosure A will be incorporated into the Vermont Yankee ODCM.

LAI 1044 closed | We trust that the information contained in the submittal is sufficient, however, should you have any questions or require further information concerning this matter, please contact this office.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION



James J. Duffy
Licensing Engineer

Enclosures A & B

c: USNRC Region I Administrator (Letter and Enclosure A)
USNRC Resident Inspector - VYNPS (Letter and Enclosure A)
USNRC Project Manager - VYNPS (Letter and Enclosure A)

Revision 21 Date 08/14/97

ENCLOSURE A

VERMONT YANKEE NUCLEAR POWER PLANT

ASSESSMENT OF ROUTINE DISPOSAL OF COOLING TOWER SILT IN
AREAS ON SITE PREVIOUSLY DESIGNATED FOR SEPTIC WASTE DISPOSAL

Revision 21 Date 08/14/97

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3.1 Silt Disposal Procedure Requirements	9
3.2 Administrative Procedure Requirements	12
4.0 EVALUATION OF ENVIRONMENTAL IMPACTS	13
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VERMONT YANKEE NUCLEAR POWER PLANT

Assessment of Routine Disposal of Cooling Tower Silt in Areas On-Site Previously Designated for Septic Waste Disposal

1.0 INTRODUCTION:

Vermont Yankee Nuclear Power Corporation (Vermont Yankee) requested from the NRC in 1989 permission to routinely dispose of slightly contaminated septic waste in designated on-site areas in accordance with 10CFR20.302(a). The NRC responded to this request on August 30, 1989 by granting approval of the proposed procedures for on-site disposal of septic waste concluding that the commitments, as documented in our request, were acceptable provided that our request and analysis be permanently incorporated into the plant's Offsite Dose Calculation Manual (ODCM). Revision 9 to the ODCM (Appendix B) incorporated the assessment and approval of the methods utilized for on-site disposal of slightly contaminated sewage sludge.

In addition to the previously identified solids content of septic waste as a source of environmental, low level radioactive contaminated material, cooling tower silt deposits resulting from the settling of solids from river water passing through the mechanical draft cooling tower system have been identified to also contain low levels of plant-specific radionuclides. Periodic removal of the silt from the cooling tower basins is a necessary maintenance practice to insure operability of the cooling system. However, due to the presence of by-product materials in the silt, proper disposal requirements must be applied to insure that the potential radiological impact is within acceptable limits.

This assessment of silt disposal expands the original septic waste disposal assessment to include earthen type materials (cooling tower silt deposits) while maintaining the original radiological assessment modeling and dose limit criteria that have been approved for septic waste spreading on site. This assessment demonstrates that cooling tower silt can be disposed of in the same manner and under the same dose limit criteria as previously approved for septic waste in Appendix B to the Vermont Yankee ODCM. Implementation of the following commitments as an amendment to the original 10 CFR

Part 20.302(a) approval for septic waste shall be incorporated into the Vermont Yankee ODCM upon approval by the NRC.

2.0 WASTE STREAM DESCRIPTION:

The waste involved in this assessment is residual solids (silt) collected in the basins of the plant's mechanical draft cooling tower system. The silt consists of organic and inorganic sediments and earthen type materials that have settled from the cooling water flow taken from the Connecticut River as it passes through the towers. As a result of de-sludging the tower basins in 1993, an estimated 14,000 cubic feet of silt was accumulated on site. Clean-out operations will also occur periodically to ensure continued system operability. Sample analysis performed to the plant's environmental lower limits of detection requirements, as contained in Technical Specification Table 4.9.3., has identified Cobalt-60 and Cesium-137 in low concentrations as being present in silt collected in 1993.

The cooling towers are located at the southern end of the plant facility complex but are not directly connected to any system in the plant that contains radioactivity. The postulated mechanism of how plant-related radionuclides have been introduced into the cooling system silt assumes that past routine effluents discharged from nearby plant gaseous release points were entrained in the large mechanically-induced air flow that is pulled through the towers as a heat exchange medium. The cooling water flow provides a scrubbing action as it breaks up into fine water droplets due to the splash pans of the towers. This scrubbing action washes any airborne particulates out of the air. Over long periods of operation, any radioactivity removed from the air flow could buildup to measurable levels in silt that settles out in the basins at the bottom of the towers.

Table 1 lists the analyses of twenty-one samples collected from the silt pile removed from the cooling tower basins. Radioactivity measurements, averaged over all the samples, indicate that the silt material can be characterized as containing approximately 50 pCi/kg (dry wt.) of Cobalt-60 and 198 pCi/kg (dry wt.) of Cesium-137. Eight of the samples indicated no positive Cobalt-60 above a minimum detectable level achieved for the analysis.

Table 1

Cooling Tower Silt Radioactivity
(1993 samples*)

Sample #	Co-60 (pCi/kg dry)	Cs-137 (pCi/kg dry)
G12759	53	144
G12758	72	172
G12757	<14	201
G12756	<17	245
G12755	73	206
G12754	<16	165
G12753	<27	240
G12752	79	181
G12751	<29	180
G12750	35	107
G12749	59	171
G12748	<19	205
G12747	<38	209
G12746	< 7	218
G12745	50	241
G12744	40	220
G12743	68	264
G12742	45	195
G12724	71	115
G12723	104	264
G13940	126	218
Average:	50	198
Max.	126	264
Min.	< 7	107
Standard deviation:	30	42

* Average wet to dry sample weight ratio equal to 1.6. Dry weight silt density equal to 1.3 gm/cc.

For the purpose of estimating the total activity in the silt pile, the less than values in Table 1 are included as positives in the calculation of the average radioactivity concentration.

Cobalt-60, due to its relatively short half life, is typically associated with plant operations when measured in the near environment. However, Cesium-137 when measured in the environment may have a background component that is not related to power plant operations. Past weapons testing fallout has imposed a man made background level of Cesium-137 in New England soils and sediments that can vary over several hundred pCi/kg. The plant's Environmental Monitoring Program has shown that Connecticut River sediment in the vicinity of Vermont Yankee averages about 123 pCi/kg (dry wt.) of Cesium-137 (Table 2) with no plant related detectable level of Cobalt-60. The value of 123 pCi/kg may represent an estimate of background level of Cesium-137 in sediment that would be subject to entrainment in cooling water flow that enters the plant. In comparing the measured levels of Cesium-137 on Table 1 with the past river sediment level, the average concentration in the cooling tower silt is higher than that of the river sediment data but does fall within the observed range of recorded sediment Cs-137. The river sediment Cesium-137 concentration averages about 62% of the concentration value detected in the tower silt. For purposes of this assessment of plant-related dose impact from the on-site disposal of silt material, it is conservatively assumed that all detectable Cs-137 in cooling tower silt is directly related to plant operations. No background component is subtracted from the measured values for this case study since only a single sampling location (down stream) is included in the Environmental Monitoring Program which may not fully describe the true background levels in the region.

The total radioactivity for the current 14,000 cubic feet of silt collected on site can be estimated by multiplying this volume by its "as is" density of 2.1 gm/cc (i.e. 1.3 gm/cc dry weight density x 1.6 wet/dry weight ratio) and then conservatively assume that the measured average dry weight radioactivity concentrations for Cobalt and Cesium would be the same as in the collected silt. Multiplying the average Cobalt-60 and Cesium-137 concentration in silt by the mass of the collected material produces estimates (Table 3) of total radioactivity that was removed from the cooling tower basin in September 1993.

Table 2

Cesium-137 in
Connecticut River Sediments*

Date	Cs-137 (pCi/kg dry)
05/24/94	61
10/13/93	85
06/02/93	60
10/15/92	137
05/20/92	176
10/24/91	178
05/16/91	230
10/25/90	84
05/16/90	62
10/04/89	<174
05/26/89	179
10/12/88	115
05/12/88	62
Average:	132
Max.	230
Min.	60
Standard deviation:	56

* Samples collected as part of the Vermont Yankee Radiological Environmental Monitoring Program (REMP) for river sediment sample location SE-11.

Table 3

Estimated Total Radioactive Material for 1993 Tower Clean-Out

	Volume of Silt (ft ³)	Mass (kg)	Average Concentration (pCi/kg)	Total Act. (as of 11/93) (uCi)	Decayed Act. (as of 6/95) (uCi)
Co-60	14,000	8.32E+5	50	42	34
Cs-137	14,000	8.32E+5	198	165	159

In addition to 14,000 cubic feet of silt already accumulated, it is anticipated that periodic maintenance work in cleaning out the cooling tower basins will generate approximately 4,000 cubic feet of new silt material over each successive 18 month operating cycle. Assuming the same level of plant-related radioactivity concentration that was originally observed, the additional amounts of radioactivity that will require on site disposal following each refueling cycle can also be estimated. Table 4 lists an estimate of the total radioactivity that might be present at each 18 month clean-out cycle.

Table 4

Estimated Total Radioactive Material for Each 18 Month Maintenance Cycle

	Volume of Silt (ft ³)	Mass (kg)	Average Concentration (pCi/kg)	Total Activity (uCi)
Co-60	4,000	2.38E+5	50	12
Cs-137	4,000	2.38E+5	198	57

3.0 DISPOSAL METHOD:

The method of silt disposal shall utilize a technique of land spreading in a manner consistent with the current commitments for the on-site disposal of septic waste as approved by the NRC and implemented as Appendix B of the Vermont Yankee ODCM (Reference 1). The same land areas designated and approved for septic waste disposal shall be used for the placement of silt removed from the cooling tower basins. Determination of the radiological dose impact shall also be made based on the same models and pathway assumptions as indicated in Appendix B of the ODCM.

3.1 Silt Disposal Procedure Requirements:

Gamma isotopic analysis of silt samples shall be made prior to each disposal by obtaining representative composite samples in sufficient numbers to characterize the material removed from the cooling tower basins. Each gamma isotopic analysis shall be required to achieve the environmental lower limits of detection as indicated for sediment on Table 4.9.3 of the Vermont Yankee Technical Specifications.

The estimation of total radioactivity to be disposed of shall be made based on the average of all composite sample analyses. The estimation of total radioactivity and projected dose impact shall be made prior to placing the collected silt on the designated disposal plots. The dose impact from each disposal operation shall be included with all past septic waste and silt spreading operations to ensure that the appropriate dose limits are not exceeded on any waste disposal area for the combination of all past operations.

The established dose criteria requires that all applications of earthen type materials within the approved designated disposal areas shall be limited to ensure that dose to a maximally exposed individual (during the Vermont Yankee control period) be maintained less than 1 mrem/year to the whole body and any organ, and the dose to an inadvertent intruder following termination of site control be maintained less than 5 mrem/year to the whole body and any organ.

The limits on concentrations of radionuclides as addressed in Appendix B to the ODCM for septic waste (i.e., each tank of septic sludge to be disposed are limited to a combined MPC ratio of less than 0.1) were included to ensure proper control was in place to address the situation of small quantities of relatively high concentration material. This limitation does not directly apply to silt deposits since the silt is handled as dewatered sediments as opposed to liquid slurries of septic waste.

For dry, earthen type material such as silt, a specific radionuclide concentration limit shall be applied in place of the septic waste liquid MPC ratio. No soil associated with a sample analysis that identifies a plant-related radionuclide in excess of the concentration limits of Table 5 will be permitted regardless of the total pathway dose assessment determined for the quantity material under consideration. For the case where more than one radionuclide is detected, the sum of the ratio rule will be applied. The measured concentration of each radionuclide divided by its limiting concentration value shall be added with the sum of all fractions equal to or less than 1. This limiting condition will prevent small volumes of relatively high specific radioactivity from being spread on the disposal plots, and therefore reduce the potential for creating unexpected hot spots of concentrated material.

Table 5 lists, by radionuclide, soil concentration values that would generate an annual external effective dose equivalent of 25 mrem/year if it were assumed that an individual continuously stood on an infinite plane of soil contaminated to a depth of 15 cm. The assumptions of an infinite plane and continuous occupancy are conservative for situations where the amount of contaminated soil identified would not provide for a 15 cm soil depth over an extended surface area and where disposal site access is limited. Twenty-five mrem/year was selected as a reference value based on the fact that it was a suitable fraction of the NRC annual dose limit (100 mrem/year per 10 CFR Part 20.1301) applied to members of the public from all station sources. The 25 mrem/year also equals the EPA dose limit from 40 CFR Part 190 which would apply to real members of the public offsite and allow for credit to be taken in accounting for actual usage patterns such as occupancy time. The external dose factors provided on Table 5 were derived from Table E-2 of NUREG/CR-5512 (Reference 3).

Table 5

Dry Soil Maximum Concentration Values

Radionuclide	Soil Concentration pCi/kg (equal to 25 mrem/yr)
Cr-51	1.51E+05
Mn-54	5.50E+03
Fe-59	3.83E+03
Co-58	4.70E+03
Co-60	1.82E+03
Zn-65	7.85E+03
Zr-95	6.18E+03
Ag-110m	1.66E+03
Sb-124	2.51E+03
Cs-134	2.95E+03
Cs-137	8.13E+03
Ce-141	7.85E+04
Ce-144	8.75E+04

Assumptions include infinite planar distribution, uniform depth distribution to 15 cm, soil density at 1.625E+06 gm/m³ and external direct dose pathway only with a 100% occupancy factor.

3.2 Administrative Procedure Requirements:

Dry silt material shall be dispersed using typical agricultural dry bulk surface spreading practices only in approved disposal areas on site.

Complete records of each disposal will be maintained. These records will include the concentration of radionuclides detected in the silt, an estimate of the total volume of silt disposed of, the total radioactivity in each disposal operation as well as the total accumulated on each disposal plot at the time of the spreading, the plot on which the silt was applied, and the results of any dose calculations or maximum allowable accumulated activity determinations required to demonstrate that the dose limits imposed on these land spreading operations have not been exceeded. The determination of the total radioactivity and dose calculations shall also include all past septic waste and silt disposal operations that placed low level radioactive material on the designated disposal plots.

The periodic disposal of silt on each of the approved land spreading areas will be limited to within the same established dose and radioactivity criteria that have been approved for septic waste disposal.

Concentration limits that are applied to the disposal of earthen type materials (dry soil) shall restrict the placement of small volumes of material that have relatively high concentrations of radioactivity such that direct exposure could not exceed a small proportion (25%) of the annual dose limits to members of the public that is contained in 10 CFR Part 20.1301.

Any farmer leasing land used for the disposal of silt deposits will be notified of the applicable restrictions placed on the site due to the land spreading of low level contaminated material. These restrictions are the same as detailed for septic waste spreading as given in Reference 1.

4.0 EVALUATION OF ENVIRONMENTAL IMPACTS:

4.1 Site Characteristics

The designated disposal sites consist of two fields located on the Vermont Yankee Nuclear Power Plant site. Both fields are on the plant property within the site boundary security fence. Site A contains an approximate ten-acre parcel of land centered approximately 2,000 feet northwest of the Reactor Building. Site B consist of approximately two acres and is centered approximately 1,500 feet south of the Reactor Building. These are the same land parcels approved by the NRC for the land disposal of septic waste and are described in detail in Reference 1 along with the boundary restrictions for the placement of contaminated material.

Radiological assessments of septic waste disposal have determined that a single two-acre plot would be sufficient for the routine disposal of that waste stream over a 20-year period without exceeding the dose criteria to a maximum exposed individual or inadvertent intruder. As a result, the ten-acre field to the northwest can be divided into five disposal plots, with the two-acre site at the south end of the plant site providing a sixth plot. It is therefore concluded that there is sufficient space within the already approved disposal plots to accommodate additional material from the cooling tower basins along with the septic waste without the likelihood of exceeding the approved dose limit criteria.

Since the residual organic and inorganic solids associated with river sediment (silt) are similar to the sand and residual organic material remaining after decomposition of septic waste that is removed from the plant's septic tanks, the conclusions of no significant environmental (non-radiological) impact associated with the disposal of septic waste are not changed by the addition of another earthen type material, namely silt.

4.2 Radiological Impact:

The amount of cooling tower silt, in combination with any septic waste disposals, will be procedurally controlled to insure doses are maintained within the prior approved limits (Reference 1). These limits are based on

past NRC proposed guidance (described in AIF/NESP-037), August 1986). The dose criteria require that the maximally exposed member of the general public receive a dose less than 1 mrem/year to the whole body or any organ due to the disposed material and less than 5 mrem/year to the whole body or any organ of an inadvertent intruder.

To assess the doses received by the maximally exposed individual and inadvertent intruder resulting from silt spreading, the same pathway modeling, assumptions and dose calculation methods as approved for septic disposal are used. These dose models implement the methodologies and dose conversion factors as provided in Regulatory Guide 1.109 (Reference 2).

Six potential pathways have been identified and include:

- (a) Standing on contaminated ground,
- (b) Inhalation of resuspended radioactivity,
- (c) Ingestion of leafy vegetables,
- (d) Ingestion of stored vegetables,
- (f) Ingestion of meat, and
- (g) Ingestion of cow's milk

Based on the septic waste evaluations, the liquid pathway was determined to be insignificant.

Both the maximum individual and inadvertent intruder are assumed to be exposed to these pathways with the difference between them related to occupancy time. The basic assumptions used in the radiological analyses include:

- (a) Exposure to ground contamination and resuspended radioactivity is for a period of 104 hours per year during the Vermont Yankee active control of the disposal sites and continuous thereafter. The 104-hour interval is representative of a farmer's time spent on a plot of land (4 hours per week for 6 months).
- (b) For the purpose of projecting and illustrating the magnitude of dose impacts over the remaining life of the plant, it is assumed that the current concentration levels of activity detected in silt remain

constant. Table 1 indicates the measured radioactivity levels for Cobalt-60 and Cesium-137 first noted in silt material.

(c) The maximum radiation source corresponds to the accumulation of radioactive material on a single plot (two acre) within the approved disposal sites over a period of 13 operating cycles. This extends over the next 18 years until after the operating license expires in 2012. The initial application (referenced to June 1995) consists of 14,000 cubic feet of silt collected in 1993 along with the first periodic clean out of the tower basins that adds an additional 4,000 cubic feet. All subsequent applications of 4,000 cubic feet occur at 18-month intervals.

(d) For the analysis of the radiological impact during the Vermont Yankee active control of the disposal sites, no plowing is assumed to take place and all dispersed radioactive material remains on the surface forming a source of unshielded direct radiation.

(e) No radioactive material is dispersed directly on crops for human or animal consumption. Crop contamination is only through root uptake.

(f) The deposition on crops of suspended radioactivity is insignificantly small.

(g) Pathway data and usage factors used in the analysis are the same as those used in the plant's ODCM assessment of off-site radiological impact from routine releases with the exception that the fraction of stored vegetables grown on the contaminated land was conservatively increased from 0.76 to 1.0 (at present no vegetable crops for human consumption are grown on any of the approved disposal plots).

(h) It is conservatively assumed that Vermont Yankee relinquishes control of the disposal sites after the operating license expires in 2012 (i.e., the source term accumulated on a single, 2-acre disposal plot applies also for the inadvertent intruder).

(i) For the analysis of the impact after Vermont Yankee control of the site is relinquished, the radioactive material is plowed under and forms a uniform mix with the top six inches of soil; but nonetheless, undergoes resuspension in air at the same rate as the unplowed surface contamination. For direct ground plane exposure the self shielding due to the six-inch plow layer reduces the surface dose rate by about a factor of four.

The dose models and methods used to generate deposition values and accumulated activity over the operating life of the plant are documented in Attachment 2 to Reference 1. Based on the measured concentrations and silt volumes noted in Section 2.0 above, the total radioactivity that remains on the disposal plots after the operating license expires is estimated on Table 6.

Table 6

Projected Radioactivity Buildup Due to Silt Spreading

Nuclide	Contribution from initial 14,000 ft ³ (uCi)	Accumulation from 13 cycles at 4,000 ft ³ /ea. (uCi)	Total Remaining in year 2013 (uCi)
Cobalt-60	3.2	61.9	65.1
Cesium-137	104.9	500.5	605.4

The calculated potential radiation exposure following the spreading of all silt material anticipated to be generated through the remainder of the operating license on a single, two-acre plot is provided on Table 7.

Table 7

Dose Impact Due to Continued Spreading to End of License

Disposal Site Access	Radiation Exposure	Individual/Organ
Controlled by VYNPS (max. exposed individual)	0.228 mrem/yr 0.820 mrem/yr max.	adult/whole body child/bone
Uncontrolled by VYNPS (inadvertent intruder after license termination)	1.46 mrem/yr 2.41 mrem/yr max.	adult/whole body child/bone

The individual pathway contributions to the total dose due to continued silt spreading are shown on Table 8.

Table 8

Pathway-Dependent Critical Organ Doses

Pathway	Maximally exposed Individual/Organ (Child/Bone) (mrem/year)	Inadvertent Intruder Critical Individual/Organ (Child/Bone) (mrem/year)
Ground Irradiation	0.0474	0.957
Inhalation	0.00814	0.685
Stored Vegetables	0.528	0.528
Leafy Vegetables	0.0265	0.0265
Milk Ingestion	0.201	0.201
Meat Ingestion	<u>0.00833</u>	<u>0.00833</u>
Total:	0.82	2.41

In addition, the isotopic breakdown of the critical organ doses listed above (Table 8) for the two detected radionuclides is seen to be:

Table 9

Isotopic Breakdown of Maximum Radiation Exposures

Description	Isotope	Radioactivity (uCi/2 acres)	Dose (mrem/yr)	Percent of total
During control of disposal sites Max. organ: child/bone	Cs-137	605.4	0.805	98.2
	Co-60	65.1	<u>0.0144</u>	1.8
			0.82	
Termination of disposal site Max. organ: child/bone	Cs-137	605.4	2.12	88.0
	Co-60	65.1	<u>0.29</u>	12.0
			2.41	

For comparison to the total dose calculated assuming the continued disposal of silt removed from the tower basins through the end of the operating license, the dose from just the original 14,000 cubic feet collected is shown on Table 10.

Table 10

Dose Impact Due to Single (14,000 ft³) Silt Spreading

Disposal Site Access	Radiation Exposure	Individual/Organ
Controlled by VYNPS (max. exposed individual in 1995)	0.064 mrem/yr 0.219 mrem/yr max.	adult/whole body child/bone
Uncontrolled by VYNPS (inadvertent intruder after license termination)	0.224 mrem/yr 0.393 mrem/yr max.	adult/whole body child/bone

Table 10 shows that the application of the silt material initially collected (14,000 cubic feet) accounts for about 27 percent of the maximum individual organ dose during the control period as compared to the scenario of continued periodic silt spreading over the balance of the operating license. This illustrates that dose impacts from the material currently collected are well below the acceptance criteria of limiting the dose from any two-acre plot to no more than 1 mrem/year during the control period and 5 mrem/year after termination of the license, and is expected to remain below the acceptance criteria throughout the plant life. If unexpected buildup of radioactivity in future silt clean-out operations were to occur, the option for use of alternate disposal plots remains available to ensure that the impact from any single, two-acre plot stays within the acceptance criteria.

Also of interest are derived dose conversion factors which provide a means of ensuring that septic and silt disposal operations remain within the prescribed radiological guidelines noted above. The critical organ (worst case) and whole body dose factors for all pathways on a per acre bases are

given on Table 11 for periods during Vermont Yankee control of the disposal site and on Table 12 for post control periods associated with the inadvertent intruder scenario. The dose conversion factors have been expanded to include other potential radionuclide beyond the original five that were addressed in Reference 1. This provides a means to assess other nuclides if future disposal operations identify additional radionuclides not previously observed. The development of these additional nuclide dose conversion factors utilize the same modeling and pathway assumptions as used to derive the factors for the original five radionuclides identified in septic waste. The models for these site and pathway-specific dose factors are those in Regulatory Guide 1.109 (Reference 2) and are described in detail in Attachment II to Reference 1.

Table 11

All-Pathway Critical Organ / Whole Body Dose Conversion Factors
During Vermont Yankee Control of Disposal Sites

Nuclide	Individual/Organ	Critical Organ Dose Factor	Whole Body Dose Factor
		(mrem/yr per uCi/acre)	
Cr-51	Teen/Lung	1.14E-05	5.76E-06
Mn-54	Adult/GI-LLI	3.75E-04	1.93E-04
Fe-55	Child/Bone	6.45E-06	1.06E-06
Fe-59	Teen/Lung	4.61E-04	2.13E-04
Co-58	Teen/Lung	3.27E-04	2.01E-04
Co-60	Teen/Lung	7.17E-04	5.31E-04
Zn-65	Child/Liver	1.64E-02	1.03E-02
Zr-95	Teen/Lung	4.47E-04	1.34E-04
Ag110m	Teen/GI-LLI	1.32E-02	5.24E-04
Sb-124	Teen/Lung	8.34E-04	3.54E-04
Cs-134	Child/Liver	3.18E-03	1.28E-03
Cs-137	Child/Bone	2.66E-03	7.02E-04
Ce-141	Teen/Lung	1.54E-04	1.50E-05
Ce-144	Teen/Lung	6.00E-04	2.44E-05

Table 12

All-Pathway Critical Organ / Whole Body Dose Conversion Factors
Post Vermont Yankee Control of Disposal Sites
(Inadvertent Intruder)

Nuclide	Individual/Organ	Critical Organ Dose Factor	Whole Body Dose Factor
		(mrem/yr per uCi/acre)	
Cr-51	Teen/Lung	5.89E-04	1.19E-04
Mn-54	Teen/Lung	1.02E-02	3.12E-03
Fe-55	Teen/Lung	3.50E-04	2.27E-05
Fe-59	Teen/Lung	2.55E-02	4.43E-03
Co-58	Teen/Lung	1.59E-02	3.72E-03
Co-60	Teen/Lung	3.19E-02	9.09E-03
Zn-65	Child/Liver	1.89E-02	1.25E-02
Zr-95	Teen/Lung	2.93E-02	2.99E-03
Ag110m	Teen/Lung	3.59E-02	9.53E-03
Sb-124	Teen/Lung	4.73E-02	7.04E-03
Cs-134	Child/Liver	1.21E-02	9.36E-03
Cs-137	Child/Bone	6.98E-03	3.85E-03
Ce-141	Teen/Lung	1.21E-02	3.44E-04
Ce-144	Teen/Lung	5.00E-02	1.52E-03

5.0 RADIATION PROTECTION:

The disposal operation of silt material from the cooling tower basins will follow the applicable Vermont Yankee procedures to maintain doses as low as reasonably achievable and within the specific dose criteria as previously approved for septic waste disposal (Reference 1).

6.0 CONCLUSIONS:

Silt collected from the cooling tower basins is an earthen type material that is similar in characteristics to septic waste residual solids with respect to the radiological pathway behavior and modeling and can be disposed of through on-site land spreading on the same disposal plots as previously evaluated and approved for septic waste disposal. The radiological assessment of low level contaminated silt shows that the projected dose from the on-site periodic spreading of this material will have no significant dose impact to members of the public and can be maintained below the approved dose limitations already in place for septic waste.

7.0 REFERENCES:

- (1) Vermont Yankee ODCM, Appendix B; "Approval of Criteria for Disposal of Slightly Contaminated Septic Water On-Site at Vermont Yankee". (Included NRC approval letter dated August 30, 1989, VY request for approval dated June 28, 1989 with Attachments I and II).
- (2) USNRC Regulatory Guide 1.109, Rev. 1; "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", dated October 1977.
- (3) NUREG/CR-5512, Vol. 1, "Residual Radioactive Contamination From Decommissioning", Final Report, dated October 1992.

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APPENDIX G

MAXIMUM PERMISSIBLE CONCENTRATIONS (MPCs)
IN AIR AND WATER ABOVE NATURAL BACKGROUND
TAKEN FROM 10CFR20.1 TO 20.602, APPENDIX B

APPENDIX G

With the implementation of the revised Part 20 to Title 10 of the Code of Federal Regulations (10CFR20.1001-20.2401), the Maximum Permissible Concentrations (MPCs) that were part of the old 10CFR20 were replaced by a new Appendix B to 10CFR20 for the limits that apply to effluents released to unrestricted areas. However, MPC values were also used and accepted as licensing conditions for the control of radioactive materials in situations other than those directly covered by the requirements of the regulations. One example is the on-site disposal of septic waste which used the MPC values as one criteria of acceptability for land spreading. Appendix B to the ODCM references 10CFR20.1-20.601 Appendix B MPCs as concentration criteria for this disposal option.

With the final publication of the revised 10CFR20.1001-20.2401, the original MPC tables of the old 10CFR20 are no longer in print. As such, this appendix to the ODCM is added to provide a reference source for the MPC values contained in the original Appendix B to 10CFR20.1-20.601 for those conditions that still refer to these requirements.

APPENDIX G

MAXIMUM PERMISSIBLE CONCENTRATIONS (MPCs)
(FROM 10CFR20.1 TO 20.602, APPENDIX B)

APPENDIX B TO §§20.1 - 20.602
 CONCENTRATIONS IN AIR AND WATER ABOVE NATURAL BACKGROUND

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Actinium (89)	Ac 227	S I	2×10^{-12} 3×10^{-11}	6×10^{-5} 9×10^{-3}	8×10^{-14} 9×10^{-13}	2×10^{-6} 3×10^{-4}
	Ac 228	S I	8×10^{-8} 2×10^{-8}	3×10^{-3} 3×10^{-3}	3×10^{-9} 6×10^{-10}	9×10^{-5} 9×10^{-5}
Americium (95)	Am 241	S I	6×10^{-12} 1×10^{-10}	1×10^{-4} 8×10^{-4}	2×10^{-13} 4×10^{-12}	4×10^{-6} 3×10^{-5}
	Am 242m	S I	6×10^{-12} 3×10^{-10}	1×10^{-4} 3×10^{-3}	2×10^{-13} 9×10^{-12}	4×10^{-6} 9×10^{-5}
	Am 242	S I	4×10^{-8} 5×10^{-8}	4×10^{-3} 4×10^{-3}	1×10^{-9} 2×10^{-9}	1×10^{-4} 1×10^{-4}
	Am 243	S I	6×10^{-12} 1×10^{-10}	1×10^{-4} 8×10^{-4}	2×10^{-13} 4×10^{-12}	4×10^{-6} 3×10^{-5}
	Am 244	S I	4×10^{-6} 2×10^{-5}	1×10^{-1} 1×10^{-1}	1×10^{-7} 8×10^{-7}	5×10^{-3} 5×10^{-3}
Antimony	Sb 122	S I	2×10^{-7} 1×10^{-7}	8×10^{-4} 8×10^{-4}	6×10^{-9} 5×10^{-9}	3×10^{-5} 3×10^{-5}
	Sb 124	S I	2×10^{-7} 2×10^{-8}	7×10^{-4} 7×10^{-4}	5×10^{-9} 7×10^{-10}	2×10^{-5} 2×10^{-5}
	Sb 125	S I	5×10^{-7} 3×10^{-8}	3×10^{-3} 3×10^{-3}	2×10^{-8} 9×10^{-10}	1×10^{-4} 1×10^{-4}
Argon (18)	A37	Sub ²	6×10^{-3}	1×10^{-4}
	A41	Sub	2×10^{-6}	4×10^{-8}
Arsenic (33)	As 73	S I	2×10^{-6} 4×10^{-7}	1×10^{-2} 1×10^{-2}	7×10^{-8} 1×10^{-8}	5×10^{-4} 5×10^{-4}
	As 74	S I	3×10^{-7} 1×10^{-7}	2×10^{-3} 2×10^{-3}	1×10^{-8} 4×10^{-9}	5×10^{-5} 5×10^{-5}
	As 76	S I	1×10^{-7} 1×10^{-7}	6×10^{-4} 6×10^{-4}	4×10^{-9} 3×10^{-9}	2×10^{-5} 2×10^{-5}
	As 77	S I	5×10^{-7} 4×10^{-7}	2×10^{-3} 2×10^{-3}	2×10^{-8} 1×10^{-8}	8×10^{-5} 8×10^{-5}

Revision 21 Date 08/14/97

APPENDIX B TO §§20.1 - 20.602
CONCENTRATIONS IN AIR AND WATER ABOVE NATURAL BACKGROUND

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Astatine (85)	At 211	S I	7×10^{-9} 3×10^{-8}	5×10^{-5} 2×10^{-3}	2×10^{-10} 1×10^{-9}	2×10^{-6} 7×10^{-5}
Barium (56)	Ba 131	S I	1×10^{-6} 4×10^{-7}	5×10^{-3} 5×10^{-3}	4×10^{-8} 1×10^{-8}	2×10^{-4} 2×10^{-4}
	Ba 140	S I	1×10^{-7} 4×10^{-8}	8×10^{-4} 7×10^{-4}	4×10^{-9} 1×10^{-9}	3×10^{-5} 2×10^{-5}
Berkelium (97)	Bk 249	S I	9×10^{-10} 1×10^{-7}	2×10^{-2} 2×10^{-2}	3×10^{-11} 4×10^{-9}	6×10^{-4} 6×10^{-4}
	Bk 250	S I	1×10^{-7} 1×10^{-6}	6×10^{-3} 6×10^{-3}	5×10^{-9} 4×10^{-8}	2×10^{-4} 2×10^{-4}
Beryllium (4)	Be 7	S I	6×10^{-6} 1×10^{-6}	5×10^{-2} 5×10^{-2}	2×10^{-7} 4×10^{-8}	2×10^{-3} 2×10^{-3}
Bismuth (83)	Bi 206	S I	2×10^{-7} 1×10^{-7}	1×10^{-3} 1×10^{-3}	6×10^{-9} 5×10^{-9}	4×10^{-5} 4×10^{-5}
	Bi 207	S I	2×10^{-7} 1×10^{-8}	2×10^{-3} 2×10^{-3}	6×10^{-9} 5×10^{-10}	6×10^{-5} 6×10^{-5}
	Bi 210	S I	6×10^{-9} 6×10^{-9}	1×10^{-3} 1×10^{-3}	2×10^{-10} 2×10^{-10}	4×10^{-5} 4×10^{-5}
	Bi 212	S I	1×10^{-7} 2×10^{-7}	1×10^{-2} 1×10^{-2}	3×10^{-9} 7×10^{-9}	4×10^{-4} 4×10^{-4}
Bromine (35)	Br 82	S I	1×10^{-6} 2×10^{-7}	8×10^{-3} 1×10^{-3}	4×10^{-8} 6×10^{-9}	3×10^{-4} 4×10^{-5}
Cadmium (48)	Cd 109	S I	5×10^{-8} 7×10^{-8}	5×10^{-3} 5×10^{-3}	2×10^{-9} 3×10^{-9}	2×10^{-4} 2×10^{-4}
	Cd 115m	S I	4×10^{-8} 4×10^{-8}	7×10^{-4} 7×10^{-4}	1×10^{-9} 1×10^{-9}	3×10^{-5} 3×10^{-5}
	Cd 115	S I	2×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}	8×10^{-9} 6×10^{-9}	3×10^{-5} 4×10^{-5}
Calcium (20)	Ca 45	S I	3×10^{-8} 1×10^{-7}	3×10^{-4} 5×10^{-3}	1×10^{-9} 4×10^{-9}	9×10^{-6} 2×10^{-4}
	Ca 47	S I	2×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}	6×10^{-9} 6×10^{-9}	5×10^{-5} 3×10^{-5}

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[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Californium (98)	Cf 249	S	2×10^{-12}	1×10^{-4}	5×10^{-14}	4×10^{-6}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	2×10^{-5}
	Cf 250	S	5×10^{-12}	4×10^{-4}	2×10^{-13}	1×10^{-5}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	3×10^{-5}
	Cf 251	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
Carbon (6)	Cf 252	S	6×10^{-12}	2×10^{-4}	2×10^{-13}	7×10^{-6}
		I	3×10^{-11}	2×10^{-4}	1×10^{-12}	7×10^{-6}
	Cf 253	S	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
		I	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
	Cf 254	S	5×10^{-12}	4×10^{-6}	2×10^{-13}	1×10^{-7}
		I	5×10^{-12}	4×10^{-6}	2×10^{-13}	1×10^{-7}
Cerium (58)	C 14 (CO ₂)	S	4×10^{-6}	2×10^{-2}	1×10^{-7}	8×10^{-4}
		Sub	5×10^{-5}	1×10^{-6}
	Ce 141	S	4×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
Cesium (55)	Ce 143	I	2×10^{-7}	3×10^{-3}	5×10^{-9}	9×10^{-5}
		S	3×10^{-7}	1×10^{-3}	9×10^{-9}	4×10^{-5}
	Ce 144	S	1×10^{-8}	3×10^{-4}	3×10^{-10}	1×10^{-5}
		I	6×10^{-9}	3×10^{-4}	2×10^{-10}	1×10^{-5}
	Cs 131	S	1×10^{-5}	7×10^{-2}	4×10^{-7}	2×10^{-3}
		I	3×10^{-6}	3×10^{-2}	1×10^{-7}	9×10^{-4}
Cesium (55)	Cs 134m	S	4×10^{-5}	2×10^{-1}	1×10^{-6}	6×10^{-3}
		I	6×10^{-6}	3×10^{-2}	2×10^{-7}	1×10^{-3}
	Cs 134	S	4×10^{-8}	3×10^{-4}	1×10^{-9}	9×10^{-6}
		I	1×10^{-8}	1×10^{-3}	4×10^{-10}	4×10^{-5}
	Cs 135	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	9×10^{-8}	7×10^{-3}	3×10^{-9}	2×10^{-4}
Cesium (55)	Cs 136	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	9×10^{-5}
		I	2×10^{-7}	2×10^{-3}	6×10^{-9}	6×10^{-5}
Cesium (55)	Cs 137	S	6×10^{-8}	4×10^{-4}	2×10^{-9}	2×10^{-5}
		I	1×10^{-8}	1×10^{-3}	5×10^{-10}	4×10^{-5}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)	Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)
Chlorine (17)	Cl 36	S I	4×10^{-7} 2×10^{-8}	2×10^{-3} 2×10^{-3}	1×10^{-8} 8×10^{-10}	8×10^{-5} 6×10^{-5}
	Cl 38	S I	3×10^{-6} 2×10^{-6}	1×10^{-2} 1×10^{-2}	9×10^{-8} 7×10^{-8}	4×10^{-4} 4×10^{-4}
Chromium (24)	Cr 51	S I	1×10^{-5} 2×10^{-6}	5×10^{-2} 5×10^{-2}	4×10^{-7} 8×10^{-8}	2×10^{-3} 2×10^{-3}
Cobalt (27)	Co 57	S I	3×10^{-6} 2×10^{-7}	2×10^{-2} 1×10^{-2}	1×10^{-7} 6×10^{-9}	5×10^{-4} 4×10^{-4}
	Co 58m	S I	2×10^{-5} 9×10^{-6}	8×10^{-2} 6×10^{-2}	6×10^{-7} 3×10^{-7}	3×10^{-3} 2×10^{-3}
	Co 58	S I	8×10^{-7} 5×10^{-8}	4×10^{-3} 3×10^{-3}	3×10^{-8} 2×10^{-9}	1×10^{-4} 9×10^{-5}
	Co 60	S I	3×10^{-7} 9×10^{-9}	1×10^{-3} 1×10^{-3}	1×10^{-8} 3×10^{-10}	5×10^{-5} 3×10^{-5}
Copper (29)	Cu 64	S I	2×10^{-6} 1×10^{-6}	1×10^{-2} 6×10^{-3}	7×10^{-8} 4×10^{-8}	3×10^{-4} 2×10^{-4}
Curium (96)	Cm 242	S I	1×10^{-10} 2×10^{-10}	7×10^{-4} 7×10^{-4}	4×10^{-12} 6×10^{-12}	2×10^{-5} 2×10^{-5}
	Cm 243	S I	6×10^{-12} 1×10^{-10}	1×10^{-4} 7×10^{-4}	2×10^{-13} 3×10^{-12}	5×10^{-6} 2×10^{-5}
	Cm 244	S I	9×10^{-12} 1×10^{-10}	2×10^{-4} 8×10^{-4}	3×10^{-13} 3×10^{-12}	7×10^{-6} 3×10^{-5}
	Cm 245	S I	5×10^{-12} 1×10^{-10}	1×10^{-4} 8×10^{-4}	2×10^{-13} 4×10^{-12}	4×10^{-6} 3×10^{-5}
	Cm 246	S I	5×10^{-12} 1×10^{-10}	1×10^{-4} 8×10^{-4}	2×10^{-13} 4×10^{-12}	4×10^{-6} 3×10^{-5}
	Cm 247	S I	5×10^{-12} 1×10^{-10}	1×10^{-4} 6×10^{-4}	2×10^{-13} 4×10^{-12}	4×10^{-6} 2×10^{-5}
	Cm 248	S I	6×10^{-13} 1×10^{-11}	1×10^{-5} 4×10^{-5}	2×10^{-14} 4×10^{-13}	4×10^{-7} 1×10^{-6}
	Cm 249	S I	1×10^{-5} 1×10^{-5}	6×10^{-2} 6×10^{-2}	4×10^{-7} 4×10^{-7}	2×10^{-3} 2×10^{-3}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)	Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)
Dysprosium (66)	Dy 165	S I	3×10^{-6} 2×10^{-6}	1×10^{-2} 1×10^{-2}	9×10^{-8} 7×10^{-8}	4×10^{-4} 4×10^{-4}
	Dy 166	S I	2×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}	8×10^{-9} 7×10^{-9}	4×10^{-5} 4×10^{-5}
Einsteinium (99)	Es 253	S I	8×10^{-10} 6×10^{-10}	7×10^{-4} 7×10^{-4}	3×10^{-11} 2×10^{-11}	2×10^{-5} 2×10^{-5}
	Es 254m	S I	5×10^{-9} 6×10^{-9}	5×10^{-4} 5×10^{-4}	2×10^{-10} 2×10^{-10}	2×10^{-5} 2×10^{-5}
	Es 254	S I	2×10^{-11} 1×10^{-10}	4×10^{-4} 4×10^{-4}	6×10^{-13} 4×10^{-12}	1×10^{-5} 1×10^{-5}
	Es 255	S I	5×10^{-10} 4×10^{-10}	8×10^{-4} 8×10^{-4}	2×10^{-11} 1×10^{-11}	3×10^{-5} 3×10^{-5}
Erbium (68)	Er 169	S I	6×10^{-7} 4×10^{-7}	3×10^{-3} 3×10^{-3}	2×10^{-8} 1×10^{-8}	9×10^{-5} 9×10^{-5}
	Er 171	S I	7×10^{-7} 6×10^{-7}	3×10^{-3} 3×10^{-3}	2×10^{-8} 2×10^{-8}	1×10^{-4} 1×10^{-4}
Europium (63)	Eu 152 (T/2=9.2 hrs)	S I	4×10^{-7} 3×10^{-7}	2×10^{-3} 2×10^{-3}	1×10^{-8} 1×10^{-8}	6×10^{-5} 6×10^{-5}
	Eu 152 (T/2=13 yrs)	S I	1×10^{-8} 2×10^{-8}	2×10^{-3} 2×10^{-3}	4×10^{-10} 6×10^{-10}	8×10^{-5} 8×10^{-5}
	Eu 154	S I	4×10^{-9} 7×10^{-9}	6×10^{-4} 6×10^{-4}	1×10^{-10} 2×10^{-10}	2×10^{-5} 2×10^{-5}
	Eu 155	S I	9×10^{-8} 7×10^{-8}	6×10^{-3} 6×10^{-3}	3×10^{-9} 3×10^{-9}	2×10^{-4} 2×10^{-4}
Fermium (100)	Fm 254	S I	6×10^{-8} 7×10^{-8}	4×10^{-3} 4×10^{-3}	2×10^{-9} 2×10^{-9}	1×10^{-4} 1×10^{-4}
	Fm 255	S I	2×10^{-8} 1×10^{-8}	1×10^{-3} 1×10^{-3}	6×10^{-10} 4×10^{-10}	3×10^{-5} 3×10^{-5}
	Fm 256	S I	3×10^{-9} 2×10^{-9}	3×10^{-5} 3×10^{-5}	1×10^{-10} 6×10^{-11}	9×10^{-7} 9×10^{-7}
Fluorine (9)	F 18	S I	5×10^{-6} 3×10^{-6}	2×10^{-2} 1×10^{-2}	2×10^{-7} 9×10^{-8}	8×10^{-4} 5×10^{-4}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Gadolinium (64)	Gd 153	S I	2×10^{-7} 9×10^{-8}	6×10^{-3} 6×10^{-3}	8×10^{-9} 3×10^{-9}	2×10^{-4} 2×10^{-4}
	Gd 159	S I	5×10^{-7} 4×10^{-7}	2×10^{-3} 2×10^{-3}	2×10^{-8} 1×10^{-8}	8×10^{-5} 8×10^{-5}
Gallium (31)	Ga 72	S I	2×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}	8×10^{-9} 6×10^{-9}	4×10^{-5} 4×10^{-5}
Germanium (32)	Ge 71	S I	1×10^{-5} 6×10^{-6}	5×10^{-2} 5×10^{-2}	4×10^{-7} 2×10^{-7}	2×10^{-3} 2×10^{-3}
Gold (79)	Au 196	S I	1×10^{-6} 6×10^{-7}	5×10^{-3} 4×10^{-3}	4×10^{-8} 2×10^{-8}	2×10^{-4} 1×10^{-4}
	Au 198	S I	3×10^{-7} 2×10^{-7}	2×10^{-3} 1×10^{-3}	1×10^{-8} 8×10^{-9}	5×10^{-5} 5×10^{-5}
	Au 199	S I	1×10^{-6} 8×10^{-7}	5×10^{-3} 4×10^{-3}	4×10^{-8} 3×10^{-8}	2×10^{-4} 2×10^{-4}
Hafnium (72)	Hf 181	S I	4×10^{-8} 7×10^{-8}	2×10^{-3} 2×10^{-3}	1×10^{-9} 3×10^{-9}	7×10^{-5} 7×10^{-5}
Holmium (67)	Ho 166	S I	2×10^{-7} 2×10^{-7}	9×10^{-4} 9×10^{-4}	7×10^{-9} 6×10^{-9}	3×10^{-5} 3×10^{-5}
Hydrogen (1)	H3	S I Sub	5×10^{-6} 5×10^{-6} 2×10^{-3}	1×10^{-1} 1×10^{-1}	2×10^{-7} 2×10^{-7} 4×10^{-5}	3×10^{-3} 3×10^{-3}
Indium (49)	In 113m	S I	8×10^{-6} 7×10^{-6}	4×10^{-2} 4×10^{-2}	3×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}
	In 114m	S I	1×10^{-7} 2×10^{-8}	5×10^{-4} 5×10^{-4}	4×10^{-9} 7×10^{-10}	2×10^{-5} 2×10^{-5}
	In 115m	S I	2×10^{-6} 2×10^{-6}	1×10^{-2} 1×10^{-2}	8×10^{-8} 6×10^{-8}	4×10^{-4} 4×10^{-4}
	In 115	S I	2×10^{-7} 3×10^{-8}	3×10^{-3} 3×10^{-3}	9×10^{-9} 1×10^{-9}	9×10^{-5} 9×10^{-5}
Iodine (53)	I 125	S I	5×10^{-9} 2×10^{-7}	4×10^{-5} 6×10^{-3}	8×10^{-11} 6×10^{-9}	2×10^{-7} 2×10^{-4}
	I 126	S I	8×10^{-9} 3×10^{-7}	5×10^{-5} 3×10^{-3}	9×10^{-11} 1×10^{-8}	3×10^{-7} 9×10^{-5}
	I 129	S I	2×10^{-9} 7×10^{-8}	1×10^{-5} 6×10^{-3}	2×10^{-11} 2×10^{-9}	6×10^{-8} 2×10^{-4}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Iodine (53) (Continued)	I 131	S I	9×10^{-9} 3×10^{-7}	6×10^{-5} 2×10^{-3}	1×10^{-10} 1×10^{-8}	3×10^{-7} 6×10^{-5}
	I 132	S I	2×10^{-7} 9×10^{-7}	2×10^{-3} 5×10^{-3}	3×10^{-9} 3×10^{-8}	8×10^{-6} 2×10^{-4}
	I 133	S I	3×10^{-8} 2×10^{-7}	2×10^{-4} 1×10^{-3}	4×10^{-10} 7×10^{-9}	1×10^{-6} 4×10^{-5}
	I 134	S I	5×10^{-7} 3×10^{-6}	4×10^{-3} 2×10^{-2}	6×10^{-9} 1×10^{-7}	2×10^{-5} 6×10^{-4}
	I 135	S I	1×10^{-7} 4×10^{-7}	7×10^{-4} 2×10^{-3}	1×10^{-9} 1×10^{-8}	4×10^{-6} 7×10^{-5}
Iridium (77)	Ir 190	S I	1×10^{-6} 4×10^{-7}	6×10^{-3} 5×10^{-3}	4×10^{-8} 1×10^{-8}	2×10^{-4} 2×10^{-4}
	Ir 192	S I	1×10^{-7} 3×10^{-8}	1×10^{-3} 1×10^{-3}	4×10^{-9} 9×10^{-10}	4×10^{-5} 4×10^{-5}
	Ir 194	S I	2×10^{-7} 2×10^{-7}	1×10^{-3} 9×10^{-4}	8×10^{-9} 5×10^{-9}	3×10^{-5} 3×10^{-5}
Iron (26)	Fe 55	S I	9×10^{-7} 1×10^{-6}	2×10^{-2} 7×10^{-2}	3×10^{-8} 3×10^{-8}	8×10^{-4} 2×10^{-3}
	Fe 59	S I	1×10^{-7} 5×10^{-8}	2×10^{-3} 2×10^{-3}	5×10^{-9} 2×10^{-9}	6×10^{-5} 5×10^{-5}
Krypton (36)	Kr 85m	Sub	6×10^{-6}	1×10^{-7}
	Kr 85	Sub	1×10^{-5}	3×10^{-7}
	Kr 87	Sub	1×10^{-6}	2×10^{-8}
	Kr 88	Sub	1×10^{-6}	2×10^{-8}
Lanthanum (57)	La 140	S I	2×10^{-7} 1×10^{-7}	7×10^{-4} 7×10^{-4}	5×10^{-9} 4×10^{-9}	2×10^{-5} 2×10^{-5}
Lead (82)	Pb 203	S I	3×10^{-6} 2×10^{-6}	1×10^{-2} 1×10^{-2}	9×10^{-8} 6×10^{-8}	4×10^{-4} 4×10^{-4}
	Pb 210	S I	1×10^{-10} 2×10^{-10}	4×10^{-6} 5×10^{-3}	4×10^{-12} 8×10^{-12}	1×10^{-7} 2×10^{-4}
	Pb 212	S I	2×10^{-8} 2×10^{-8}	6×10^{-4} 5×10^{-4}	6×10^{-10} 7×10^{-10}	2×10^{-5} 2×10^{-5}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)	Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)
Lutetium (71)	Lu 177	S	6×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Manganese (25)	Mn 52	S	2×10^{-7}	1×10^{-3}	7×10^{-9}	3×10^{-5}
		I	1×10^{-7}	9×10^{-4}	5×10^{-9}	3×10^{-5}
	Mn 54	S	4×10^{-7}	4×10^{-3}	1×10^{-8}	1×10^{-4}
		I	4×10^{-8}	3×10^{-3}	1×10^{-9}	1×10^{-4}
	Mn 56	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
Mercury (80)	Hg 197m	S	7×10^{-7}	6×10^{-3}	3×10^{-8}	2×10^{-4}
		I	8×10^{-7}	5×10^{-3}	3×10^{-8}	2×10^{-4}
	Hg 197	S	1×10^{-6}	9×10^{-3}	4×10^{-8}	3×10^{-4}
		I	3×10^{-6}	1×10^{-2}	9×10^{-8}	5×10^{-4}
	Hg 203	S	7×10^{-8}	5×10^{-4}	2×10^{-9}	2×10^{-5}
		I	1×10^{-7}	3×10^{-3}	4×10^{-9}	1×10^{-4}
Molybdenum (42)	Mo 99	S	7×10^{-7}	5×10^{-3}	3×10^{-8}	2×10^{-4}
		I	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
Neodymium (60)	Nd 144	S	8×10^{-11}	2×10^{-3}	3×10^{-12}	7×10^{-5}
		I	3×10^{-10}	2×10^{-3}	1×10^{-11}	8×10^{-5}
	Nd 147	S	4×10^{-7}	2×10^{-3}	1×10^{-8}	6×10^{-5}
		I	2×10^{-7}	2×10^{-3}	8×10^{-9}	6×10^{-5}
	Nd 149	S	2×10^{-6}	8×10^{-3}	6×10^{-8}	3×10^{-4}
		I	1×10^{-6}	8×10^{-3}	5×10^{-8}	3×10^{-4}
Neptunium (93)	Np 237	S	4×10^{-12}	9×10^{-5}	1×10^{-13}	3×10^{-6}
		I	1×10^{-10}	9×10^{-4}	4×10^{-12}	3×10^{-5}
	Np 239	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	7×10^{-7}	4×10^{-3}	2×10^{-8}	1×10^{-4}
Nickel (28)	Ni 59	S	5×10^{-7}	6×10^{-3}	2×10^{-8}	2×10^{-4}
		I	8×10^{-7}	6×10^{-2}	3×10^{-8}	2×10^{-3}
	Ni 63	S	6×10^{-8}	8×10^{-4}	2×10^{-9}	3×10^{-5}
		I	3×10^{-7}	2×10^{-2}	1×10^{-8}	7×10^{-4}
	Ni 65	S	9×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}

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[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)	Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)
Niobium (Columbium) (41)	Nb 93m	S I	1×10^{-7} 2×10^{-7}	1×10^{-2} 1×10^{-2}	4×10^{-9} 5×10^{-9}	4×10^{-4} 4×10^{-4}
	Nb 95	S I	5×10^{-7} 1×10^{-7}	3×10^{-3} 3×10^{-3}	2×10^{-8} 3×10^{-9}	1×10^{-4} 1×10^{-4}
	Nb 97	S I	6×10^{-6} 5×10^{-6}	3×10^{-2} 3×10^{-2}	2×10^{-7} 2×10^{-7}	9×10^{-4} 9×10^{-4}
Osmium (76)	Os 185	S I	5×10^{-7} 5×10^{-8}	2×10^{-3} 2×10^{-3}	2×10^{-8} 2×10^{-9}	7×10^{-5} 7×10^{-5}
	Os 191m	S I	2×10^{-5} 9×10^{-6}	7×10^{-2} 7×10^{-2}	6×10^{-7} 3×10^{-7}	3×10^{-3} 2×10^{-3}
	Os 191	S I	1×10^{-6} 4×10^{-7}	5×10^{-3} 5×10^{-3}	4×10^{-8} 1×10^{-8}	2×10^{-4} 2×10^{-4}
	Os 193	S I	4×10^{-7} 3×10^{-7}	2×10^{-3} 2×10^{-3}	1×10^{-8} 9×10^{-9}	6×10^{-5} 5×10^{-5}
Palladium (46)	Pd 103	S I	1×10^{-6} 7×10^{-7}	1×10^{-2} 8×10^{-3}	5×10^{-8} 3×10^{-8}	3×10^{-4} 3×10^{-4}
	Pd 109	S I	6×10^{-7} 4×10^{-7}	3×10^{-3} 2×10^{-3}	2×10^{-8} 1×10^{-8}	9×10^{-5} 7×10^{-5}
Phosphorus (15)	P 32	S I	7×10^{-8} 8×10^{-8}	5×10^{-4} 7×10^{-4}	2×10^{-9} 3×10^{-9}	2×10^{-5} 2×10^{-5}
Platinum (78)	Pt 191	S I	8×10^{-7} 6×10^{-7}	4×10^{-3} 3×10^{-3}	3×10^{-8} 2×10^{-8}	1×10^{-4} 1×10^{-4}
	Pt 193m	S I	7×10^{-6} 5×10^{-6}	3×10^{-2} 3×10^{-2}	2×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}
	Pt 193	S I	1×10^{-6} 3×10^{-7}	3×10^{-2} 5×10^{-2}	4×10^{-8} 1×10^{-8}	9×10^{-4} 2×10^{-3}
	Pt 197m	S I	6×10^{-6} 5×10^{-6}	3×10^{-2} 3×10^{-2}	2×10^{-7} 2×10^{-7}	1×10^{-3} 9×10^{-4}
	Pt 197	S I	8×10^{-7} 6×10^{-7}	4×10^{-3} 3×10^{-3}	3×10^{-8} 2×10^{-8}	1×10^{-4} 1×10^{-4}

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[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Plutonium (94)	Pu 238	S I	2×10^{-12} 3×10^{-11}	1×10^{-4} 8×10^{-4}	7×10^{-14} 1×10^{-12}	5×10^{-6} 3×10^{-5}
	Pu 239	S I	2×10^{-12} 4×10^{-11}	1×10^{-4} 8×10^{-4}	6×10^{-14} 1×10^{-12}	5×10^{-6} 3×10^{-5}
	Pu 240	S I	2×10^{-12} 4×10^{-11}	1×10^{-4} 8×10^{-4}	6×10^{-14} 1×10^{-12}	5×10^{-6} 3×10^{-5}
	Pu 241	S I	9×10^{-11} 4×10^{-8}	7×10^{-3} 4×10^{-2}	3×10^{-12} 1×10^{-9}	2×10^{-4} 1×10^{-3}
	Pu 242	S I	2×10^{-12} 4×10^{-11}	1×10^{-4} 9×10^{-4}	6×10^{-14} 1×10^{-12}	5×10^{-6} 3×10^{-5}
	Pu 243	S I	2×10^{-6} 2×10^{-6}	1×10^{-2} 1×10^{-2}	6×10^{-8} 8×10^{-8}	3×10^{-4} 3×10^{-4}
	Pu 244	S I	2×10^{-12} 3×10^{-11}	1×10^{-4} 3×10^{-4}	6×10^{-14} 1×10^{-12}	4×10^{-6} 1×10^{-5}
Polonium (84)	Po 210	S I	5×10^{-10} 2×10^{-10}	2×10^{-5} 8×10^{-4}	2×10^{-11} 7×10^{-12}	7×10^{-7} 3×10^{-5}
Potassium (19)	K42	S I	2×10^{-6} 1×10^{-7}	9×10^{-3} 6×10^{-4}	7×10^{-8} 4×10^{-9}	3×10^{-4} 2×10^{-5}
Praseodymium (59)	Pr 142	S I	2×10^{-7} 2×10^{-7}	9×10^{-4} 9×10^{-4}	7×10^{-9} 5×10^{-9}	3×10^{-5} 3×10^{-5}
	Pr 143	S I	3×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}	1×10^{-8} 6×10^{-9}	5×10^{-5} 5×10^{-5}
Promethium (61)	Pm 147	S I	6×10^{-8} 1×10^{-7}	6×10^{-3} 6×10^{-3}	2×10^{-9} 3×10^{-9}	2×10^{-4} 2×10^{-4}
	Pm 149	S I	3×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}	1×10^{-8} 8×10^{-9}	4×10^{-5} 4×10^{-5}
Protoactinium (91)	Pa 230	S I	2×10^{-9} 8×10^{-10}	7×10^{-3} 7×10^{-3}	6×10^{-11} 3×10^{-11}	2×10^{-4} 2×10^{-4}
	Pa 231	S I	1×10^{-12} 1×10^{-10}	3×10^{-5} 8×10^{-4}	4×10^{-14} 4×10^{-12}	9×10^{-7} 2×10^{-5}
	Pa 233	S I	6×10^{-7} 2×10^{-7}	4×10^{-3} 3×10^{-3}	2×10^{-8} 6×10^{-9}	1×10^{-4} 1×10^{-4}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Radium (88)	Ra 223	S I	2×10^{-9} 2×10^{-10}	2×10^{-5} 1×10^{-4}	6×10^{-11} 8×10^{-12}	7×10^{-7} 4×10^{-6}
	Ra 224	S I	5×10^{-9} 7×10^{-10}	7×10^{-5} 2×10^{-4}	2×10^{-10} 2×10^{-11}	2×10^{-6} 5×10^{-6}
	Ra 226	S I	3×10^{-11} 5×10^{-11}	4×10^{-7} 9×10^{-4}	3×10^{-12} 2×10^{-12}	3×10^{-8} 3×10^{-5}
	Ra 228	S I	7×10^{-11} 4×10^{-11}	8×10^{-7} 7×10^{-4}	2×10^{-12} 1×10^{-12}	3×10^{-8} 3×10^{-5}
Radon (86)	Rn 220	S	3×10^{-7}	1×10^{-8}
	Rn 222 ³		3×10^{-8}	3×10^{-9}
Rhenium (75)	Re 183	S I	3×10^{-6} 2×10^{-7}	2×10^{-2} 8×10^{-3}	9×10^{-8} 5×10^{-9}	6×10^{-4} 3×10^{-4}
	Re 186	S I	6×10^{-7} 2×10^{-7}	3×10^{-3} 1×10^{-3}	2×10^{-8} 8×10^{-9}	9×10^{-5} 5×10^{-5}
	Re 187	S I	9×10^{-6} 5×10^{-7}	7×10^{-2} 4×10^{-2}	3×10^{-7} 2×10^{-8}	3×10^{-3} 2×10^{-3}
	Re 188	S I	4×10^{-7} 2×10^{-7}	2×10^{-3} 9×10^{-4}	1×10^{-8} 6×10^{-9}	6×10^{-5} 3×10^{-5}
Rhodium (45)	Rh 103m	S I	8×10^{-5} 6×10^{-5}	4×10^{-1} 3×10^{-1}	3×10^{-6} 2×10^{-6}	1×10^{-2} 1×10^{-2}
	Rh 105	S I	8×10^{-7} 5×10^{-7}	4×10^{-3} 3×10^{-3}	3×10^{-8} 2×10^{-8}	1×10^{-4} 1×10^{-4}
Rubidium (37)	Rb 86	S I	3×10^{-7} 7×10^{-8}	2×10^{-3} 7×10^{-4}	1×10^{-8} 2×10^{-9}	7×10^{-5} 2×10^{-5}
	Rb 87	S I	5×10^{-7} 7×10^{-8}	3×10^{-3} 5×10^{-3}	2×10^{-8} 2×10^{-9}	1×10^{-4} 2×10^{-4}
Ruthenium (44)	Ru 97	S I	2×10^{-6} 2×10^{-6}	1×10^{-2} 1×10^{-2}	8×10^{-8} 6×10^{-8}	4×10^{-4} 3×10^{-4}
	Ru 103	S I	5×10^{-7} 8×10^{-8}	2×10^{-3} 2×10^{-3}	2×10^{-8} 3×10^{-9}	8×10^{-5} 8×10^{-5}
	Ru 105	S I	7×10^{-7} 5×10^{-7}	3×10^{-3} 3×10^{-3}	2×10^{-8} 2×10^{-8}	1×10^{-4} 1×10^{-4}
	Ru 106	S I	8×10^{-8} 6×10^{-9}	4×10^{-4} 3×10^{-4}	3×10^{-9} 2×10^{-10}	1×10^{-5} 1×10^{-5}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Samarium (62)	Sm 147	S I	7×10^{-11} 3×10^{-10}	2×10^{-3} 2×10^{-3}	2×10^{-12} 9×10^{-12}	6×10^{-5} 7×10^{-5}
	Sm 151	S I	6×10^{-8} 1×10^{-7}	1×10^{-2} 1×10^{-2}	2×10^{-9} 5×10^{-9}	4×10^{-4} 4×10^{-4}
	Sm 153	S I	5×10^{-7} 4×10^{-7}	2×10^{-3} 2×10^{-3}	2×10^{-8} 1×10^{-8}	8×10^{-5} 8×10^{-5}
Scandium (21)	Sc 46	S I	2×10^{-7} 2×10^{-8}	1×10^{-3} 1×10^{-3}	8×10^{-9} 8×10^{-10}	4×10^{-5} 4×10^{-5}
	Sc 47	S I	6×10^{-7} 5×10^{-7}	3×10^{-3} 3×10^{-3}	2×10^{-8} 2×10^{-8}	9×10^{-5} 9×10^{-5}
	Sc 48	S I	2×10^{-7} 1×10^{-7}	8×10^{-4} 8×10^{-4}	6×10^{-9} 5×10^{-9}	3×10^{-5} 3×10^{-5}
Selenium (34)	Se 75	S I	1×10^{-6} 1×10^{-7}	9×10^{-3} 8×10^{-3}	4×10^{-8} 4×10^{-9}	3×10^{-4} 3×10^{-4}
Silicon (14)	Si 31	S I	6×10^{-6} 1×10^{-6}	3×10^{-2} 6×10^{-3}	2×10^{-7} 3×10^{-8}	9×10^{-4} 2×10^{-4}
Silver (47)	Ag 105	S I	6×10^{-7} 8×10^{-8}	3×10^{-3} 3×10^{-3}	2×10^{-8} 3×10^{-9}	1×10^{-4} 1×10^{-4}
	Ag 110m	S I	2×10^{-7} 1×10^{-8}	9×10^{-4} 9×10^{-4}	7×10^{-9} 3×10^{-10}	3×10^{-5} 3×10^{-5}
	Ag 111	S I	3×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}	1×10^{-8} 8×10^{-9}	4×10^{-5} 4×10^{-5}
Sodium (11)	Na 22	S I	2×10^{-7} 9×10^{-9}	1×10^{-3} 9×10^{-4}	6×10^{-9} 3×10^{-10}	4×10^{-5} 3×10^{-5}
	Na 24	S I	1×10^{-6} 1×10^{-7}	6×10^{-3} 8×10^{-4}	4×10^{-8} 5×10^{-9}	2×10^{-4} 3×10^{-5}
Strontium (38)	Sr 85m	S I	4×10^{-5} 3×10^{-5}	2×10^{-1} 2×10^{-1}	1×10^{-6} 1×10^{-6}	7×10^{-3} 7×10^{-3}
	Sr 85	S I	2×10^{-7} 1×10^{-7}	3×10^{-3} 5×10^{-3}	8×10^{-9} 4×10^{-9}	1×10^{-4} 2×10^{-4}
	Sr 89	S I	3×10^{-8} 4×10^{-8}	3×10^{-4} 8×10^{-4}	3×10^{-10} 1×10^{-9}	3×10^{-6} 3×10^{-5}
	Sr 90	S I	1×10^{-9} 5×10^{-9}	1×10^{-5} 1×10^{-3}	3×10^{-11} 2×10^{-10}	3×10^{-7} 4×10^{-5}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)	Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)
Strontium (38) (Continued)	Sr 91	S I	4×10^{-7} 3×10^{-7}	2×10^{-3} 1×10^{-3}	2×10^{-8} 9×10^{-9}	7×10^{-5} 5×10^{-5}
	Sr 92	S I	4×10^{-7} 3×10^{-7}	2×10^{-3} 2×10^{-3}	2×10^{-8} 1×10^{-8}	7×10^{-5} 6×10^{-5}
Sulfur (16)	S 35	S I	3×10^{-7} 3×10^{-7}	2×10^{-3} 8×10^{-3}	9×10^{-9} 9×10^{-9}	6×10^{-5} 3×10^{-4}
Tantalum (73)	Ta 182	S I	4×10^{-8} 2×10^{-8}	1×10^{-3} 1×10^{-3}	1×10^{-9} 7×10^{-10}	4×10^{-5} 4×10^{-5}
Technetium (43)	Tc 96m	S I	8×10^{-5} 3×10^{-5}	4×10^{-1} 3×10^{-1}	3×10^{-6} 1×10^{-6}	1×10^{-2} 1×10^{-2}
	Tc 96	S I	6×10^{-7} 2×10^{-7}	3×10^{-3} 1×10^{-3}	2×10^{-8} 8×10^{-9}	1×10^{-4} 5×10^{-5}
	Tc 97m	S I	2×10^{-6} 2×10^{-7}	1×10^{-2} 5×10^{-3}	8×10^{-8} 5×10^{-9}	4×10^{-4} 2×10^{-4}
	Tc 97	S I	1×10^{-5} 3×10^{-7}	5×10^{-2} 2×10^{-2}	4×10^{-7} 1×10^{-8}	2×10^{-3} 8×10^{-4}
	Tc 99m	S I	4×10^{-5} 1×10^{-5}	2×10^{-1} 8×10^{-2}	1×10^{-6} 5×10^{-7}	6×10^{-3} 3×10^{-3}
	Tc 99	S I	2×10^{-6} 6×10^{-8}	1×10^{-2} 5×10^{-3}	7×10^{-8} 2×10^{-9}	3×10^{-4} 2×10^{-4}
Tellurium (52)	Te 125m	S I	4×10^{-7} 1×10^{-7}	5×10^{-3} 3×10^{-3}	1×10^{-8} 4×10^{-9}	2×10^{-4} 1×10^{-4}
	Te 127m	S I	1×10^{-7} 4×10^{-8}	2×10^{-3} 2×10^{-3}	5×10^{-9} 1×10^{-9}	6×10^{-5} 5×10^{-5}
	Te 127	S I	2×10^{-6} 9×10^{-7}	8×10^{-3} 5×10^{-3}	6×10^{-8} 3×10^{-8}	3×10^{-4} 2×10^{-4}
	Te 129m	S I	8×10^{-8} 3×10^{-8}	1×10^{-3} 6×10^{-4}	3×10^{-9} 1×10^{-9}	3×10^{-5} 2×10^{-5}
	Te 129	S I	5×10^{-6} 4×10^{-6}	2×10^{-2} 2×10^{-2}	2×10^{-7} 1×10^{-7}	8×10^{-4} 8×10^{-4}
	Te 131m	S I	4×10^{-7} 2×10^{-7}	2×10^{-3} 1×10^{-3}	1×10^{-8} 6×10^{-9}	6×10^{-5} 4×10^{-5}
	Te 132	S I	2×10^{-7} 1×10^{-7}	9×10^{-4} 6×10^{-4}	7×10^{-9} 4×10^{-9}	3×10^{-5} 2×10^{-5}

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air (μCi/ml)	Col. 2 Water (μCi/ml)	Col. 1 Air (μCi/ml)	Col. 2 Water (μCi/ml)
Terbium (65)	Tb 160	S	1x10 ⁻⁷	1x10 ⁻³	3x10 ⁻⁹	4x10 ⁻⁵
		I	3x10 ⁻⁸	1x10 ⁻³	1x10 ⁻⁹	4x10 ⁻⁵
Thalium (81)	Tl 200	S	3x10 ⁻⁶	1x10 ⁻²	9x10 ⁻⁸	4x10 ⁻⁴
		I	1x10 ⁻⁶	7x10 ⁻³	4x10 ⁻⁸	2x10 ⁻⁴
	Tl 201	S	2x10 ⁻⁶	9x10 ⁻³	7x10 ⁻⁸	3x10 ⁻⁴
		I	9x10 ⁻⁷	5x10 ⁻³	3x10 ⁻⁸	2x10 ⁻⁴
	Tl 202	S	8x10 ⁻⁷	4x10 ⁻³	3x10 ⁻⁸	1x10 ⁻⁴
		I	2x10 ⁻⁷	2x10 ⁻³	8x10 ⁻⁹	7x10 ⁻⁵
	Tl 204	S	6x10 ⁻⁷	3x10 ⁻³	2x10 ⁻⁸	1x10 ⁻⁴
		I	3x10 ⁻⁸	2x10 ⁻³	9x10 ⁻¹⁰	6x10 ⁻⁵
Thorium (90)	Th 227	S	3x10 ⁻¹⁰	5x10 ⁻⁴	1x10 ⁻¹¹	2x10 ⁻⁵
		I	2x10 ⁻¹⁰	5x10 ⁻⁴	6x10 ⁻¹²	2x10 ⁻⁵
	Th 228	S	9x10 ⁻¹²	2x10 ⁻⁴	3x10 ⁻¹³	7x10 ⁻⁶
		I	6x10 ⁻¹²	4x10 ⁻⁴	2x10 ⁻¹³	1x10 ⁻⁵
	Th 230	S	2x10 ⁻¹²	5x10 ⁻⁵	8x10 ⁻¹⁴	2x10 ⁻⁶
		I	1x10 ⁻¹¹	9x10 ⁻⁴	3x10 ⁻¹³	3x10 ⁻⁵
	Th 231	S	1x10 ⁻⁶	7x10 ⁻³	5x10 ⁻⁸	2x10 ⁻⁴
	I	1x10 ⁻⁶	7x10 ⁻³	4x10 ⁻⁸	2x10 ⁻⁴	
	Th 232	S	3x10 ⁻¹¹	5x10 ⁻⁵	1x10 ⁻¹²	2x10 ⁻⁶
		I	3x10 ⁻¹¹	1x10 ⁻³	1x10 ⁻¹²	4x10 ⁻⁵
	Th natural	S	6x10 ⁻¹¹	6x10 ⁻⁵	2x10 ⁻¹²	2x10 ⁻⁶
		I	6x10 ⁻¹¹	6x10 ⁻⁴	2x10 ⁻¹²	2x10 ⁻⁵
	Th 234	S	6x10 ⁻⁸	5x10 ⁻⁴	2x10 ⁻⁹	2x10 ⁻⁵
		I	3x10 ⁻⁸	5x10 ⁻⁴	1x10 ⁻⁹	2x10 ⁻⁵
Thulium (69)	Tm 170	S	4x10 ⁻⁸	1x10 ⁻³	1x10 ⁻⁹	5x10 ⁻⁵
		I	3x10 ⁻⁸	1x10 ⁻³	1x10 ⁻⁹	5x10 ⁻⁵
	Tm 171	S	1x10 ⁻⁷	1x10 ⁻²	4x10 ⁻⁹	5x10 ⁻⁴
		I	2x10 ⁻⁷	1x10 ⁻²	8x10 ⁻⁹	5x10 ⁻⁴
Tin (50)	Sn 113	S	4x10 ⁻⁷	2x10 ⁻³	1x10 ⁻⁸	9x10 ⁻⁵
		I	5x10 ⁻⁸	2x10 ⁻³	2x10 ⁻⁹	8x10 ⁻⁵
	Sn 125	S	1x10 ⁻⁷	5x10 ⁻⁴	4x10 ⁻⁹	2x10 ⁻⁵
		I	8x10 ⁻⁸	5x10 ⁻⁴	3x10 ⁻⁹	2x10 ⁻⁵

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Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)	Col. 1 Air ($\mu\text{Ci/ml}$)	Col. 2 Water ($\mu\text{Ci/ml}$)
Tungsten (Wolfram) (74) .	W 181	S I	2×10^{-6} 1×10^{-7}	1×10^{-2} 1×10^{-2}	8×10^{-8} 4×10^{-9}	4×10^{-4} 3×10^{-4}
	W 185	S I	8×10^{-7} 1×10^{-7}	4×10^{-3} 3×10^{-3}	3×10^{-8} 4×10^{-9}	1×10^{-4} 1×10^{-4}
	W 187	S I	4×10^{-7} 3×10^{-7}	2×10^{-3} 2×10^{-3}	2×10^{-8} 1×10^{-8}	7×10^{-5} 6×10^{-5}
Uranium (92)	U 230	S I	3×10^{-10} 1×10^{-10}	1×10^{-4} 1×10^{-4}	1×10^{-11} 4×10^{-12}	5×10^{-6} 5×10^{-6}
	U 232	S I	1×10^{-10} 3×10^{-11}	8×10^{-4} 8×10^{-4}	3×10^{-12} 9×10^{-13}	3×10^{-5} 3×10^{-5}
	U 233	S I	5×10^{-10} 1×10^{-10}	9×10^{-4} 9×10^{-4}	2×10^{-11} 4×10^{-12}	3×10^{-5} 3×10^{-5}
	U 234	S ⁴ I	6×10^{-10} 1×10^{-10}	9×10^{-4} 9×10^{-4}	2×10^{-11} 4×10^{-12}	3×10^{-5} 3×10^{-5}
	U 235	S ⁴ I	5×10^{-10} 1×10^{-10}	8×10^{-4} 8×10^{-4}	2×10^{-11} 4×10^{-12}	3×10^{-5} 3×10^{-5}
	U 236	S I	6×10^{-10} 1×10^{-10}	1×10^{-3} 1×10^{-3}	2×10^{-11} 4×10^{-12}	3×10^{-5} 3×10^{-5}
	U 238	S ⁴ I	7×10^{-11} 1×10^{-10}	1×10^{-3} 1×10^{-3}	3×10^{-12} 5×10^{-12}	4×10^{-5} 4×10^{-5}
	U 240	S I	2×10^{-7} 2×10^{-7}	1×10^{-3} 1×10^{-3}	8×10^{-9} 6×10^{-9}	3×10^{-5} 3×10^{-5}
	U-natural	S ⁴ I	1×10^{-10} 1×10^{-10}	1×10^{-3} 1×10^{-3}	5×10^{-12} 5×10^{-12}	3×10^{-5} 3×10^{-5}
Vanadium (23)	V 48	S I	2×10^{-7} 6×10^{-8}	9×10^{-4} 8×10^{-4}	6×10^{-9} 2×10^{-9}	3×10^{-5} 3×10^{-5}
Xenon (54)	Xe 131m	Sub	2×10^{-5}	4×10^{-7}
	Xe 133	Sub	1×10^{-5}	3×10^{-7}
	Xe 133m	Sub	1×10^{-5}	3×10^{-7}
	Xe 135	Sub	4×10^{-6}	1×10^{-7}
Ytterbium (70)	Yb 175	S I	7×10^{-7} 6×10^{-7}	3×10^{-3} 3×10^{-3}	2×10^{-8} 2×10^{-8}	1×10^{-4} 1×10^{-4}
Yttrium (39)	Y 90	S I	1×10^{-7} 1×10^{-7}	6×10^{-4} 6×10^{-4}	4×10^{-9} 3×10^{-9}	2×10^{-5} 2×10^{-5}

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APPENDIX B TO §§20.1 - 20.602
CONCENTRATIONS IN AIR AND WATER ABOVE NATURAL BACKGROUND

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)	Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)
	Y 91m	S I	2×10^{-5} 2×10^{-5}	1×10^{-1} 1×10^{-1}	8×10^{-7} 6×10^{-7}	3×10^{-3} 3×10^{-3}
	Y 91	S I	4×10^{-8} 3×10^{-8}	8×10^{-4} 8×10^{-4}	1×10^{-9} 1×10^{-9}	3×10^{-5} 3×10^{-5}
	Y 92	S I	4×10^{-7} 3×10^{-7}	2×10^{-3} 2×10^{-3}	1×10^{-8} 1×10^{-8}	6×10^{-5} 6×10^{-5}
	Y 93	S I	2×10^{-7} 1×10^{-7}	8×10^{-4} 8×10^{-4}	6×10^{-9} 5×10^{-9}	3×10^{-5} 3×10^{-5}
Zinc (30)	Zn 65	S I	1×10^{-7} 6×10^{-8}	3×10^{-3} 5×10^{-3}	4×10^{-9} 2×10^{-9}	1×10^{-4} 2×10^{-4}
	Zn 69m	S I	4×10^{-7} 3×10^{-7}	2×10^{-3} 2×10^{-3}	1×10^{-8} 1×10^{-8}	7×10^{-5} 6×10^{-5}
	Zn 69	S I	7×10^{-6} 9×10^{-6}	5×10^{-2} 5×10^{-2}	2×10^{-7} 3×10^{-7}	2×10^{-3} 2×10^{-3}
Zirconium (40)	Zr 93	S I	1×10^{-7} 3×10^{-7}	2×10^{-2} 2×10^{-2}	4×10^{-9} 1×10^{-8}	8×10^{-4} 8×10^{-4}
	Zr 95	S I	1×10^{-7} 3×10^{-8}	2×10^{-3} 2×10^{-3}	4×10^{-9} 1×10^{-9}	6×10^{-5} 6×10^{-5}
	Zr 97	S I	1×10^{-7} 9×10^{-8}	5×10^{-4} 5×10^{-4}	4×10^{-9} 3×10^{-9}	2×10^{-5} 2×10^{-5}
Any single radionuclide not listed above with decay mode other than alpha emission or spontaneous fission and with radioactive half-life less than 2 hours.	Sub	1×10^{-6}	3×10^{-8}
Any single radionuclide not listed above with decay mode other than alpha emission or spontaneous fission and with radioactive half-life greater than 2 hours.		3×10^{-9}	9×10^{-5}	1×10^{-10}	3×10^{-6}
Any single radionuclide not listed above which decays by alpha emission or spontaneous fission.		6×10^{-13}	4×10^{-7}	2×10^{-14}	3×10^{-8}

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¹Soluble (S); Insoluble (I).

²"Sub" means that values given are for submersion in a semispherical infinite cloud of airborne material.

³These radon concentrations are appropriate for protection from radon-222 combined with its short-lived daughters. Alternatively, the value in Table I may be replaced by one-third (1/3) "working level." (A "working level" is defined as any combination of short-lived radon-222 daughters, polonium-218, lead-214, bismuth-214 and polonium-214, in one liter of air, without regard to the degree of equilibrium, that will result in the ultimate emission of 1.3×10^5 MeV of alpha particle energy.) The Table II value may be replaced by one-thirtieth (1/30) of a "working level." The limit on radon-222 concentrations in restricted areas may be based on an annual average.

⁴For soluble mixtures of U-238, U-234 and U-235 in air chemical toxicity may be the limiting factor. If the percent by weight-enrichment of U-235 is less than 5, the concentration value for a 40-hour work week, Table I, is 0.2 milligrams uranium per cubic meter of air average. For any enrichment, the product of the average concentration and time of exposure during a 40-hour work week shall not exceed 8×10^{-3} SA μ Ci-hr/ml, where SA is the specific activity of the uranium inhaled. The concentration value for Table II is 0.007 milligrams uranium per cubic meter of air. The specific activity for natural uranium is 6.77×10^{-7} curies per gram U. The specific activity for other mixtures of U-238, U-235 and U-234, if not known, shall be:

$$SA = 3.6 \times 10^{-7} \text{ curies/gram U} \quad \text{U-depleted}$$

$$SA = (0.4 + 0.38 E + 0.0034 E^2) \times 10^{-6} \quad E \geq 0.72$$

where E is the percentage by weight of U-235, expressed as percent.

NOTE: In any case where there is a mixture in air or water of more than one radionuclide, the limiting values for purposes of this Appendix should be determined as follows:

1. If the identity and concentration of each radionuclide in the mixture are known, the limiting values should be derived as follows: Determine, for each radionuclide in the mixture, the ratio between the quantity present in the mixture and the limit otherwise established in Appendix B for the specific radionuclide when not in a mixture. The sum of such ratios for all the radionuclides in the mixture may not exceed "1" (i.e., "unity").

EXAMPLE: If radionuclides A, B, and C are present in concentrations C_A , C_B , and C_C , and if the applicable MPC's are MPC_A , MPC_B , and MPC_C respectively, then the concentrations shall be limited so that the following relationship exists:

$$(C_A/MPC_A) + (C_B/MPC_B) + (C_C/MPC_C) \leq 1$$

2. If either the identity or the concentration of any radionuclide in the mixture is not known, the limiting values for purposes of Appendix B shall be:

- For purposes of Table I, Col. 1 - 6×10^{-13}
- For purposes of Table I, Col. 2 - 4×10^{-7}
- For purposes of Table II, Col. 1 - 2×10^{-14}
- For purposes of Table II, Col. 2 - 3×10^{-8}

3. If any of the conditions specified below are met, the corresponding values specified below may be used in lieu of those specified in paragraph 2 above.

a. If the identity of each radionuclide in the mixture is known but the concentration of one or more of the radionuclides in the mixture is not known, the concentration limit for the mixture is the limit specified in Appendix "B" for the radionuclide in the mixture having the lowest concentration limit; or

b. If the identity of each radionuclide in the mixture is not known, but it is known that certain radionuclides specified in Appendix "B" are not present in the mixture, the concentration limit for the mixture is the lowest concentration limit specified in Appendix "B" for any radionuclide which is not known to be absent from the mixture; or

c. Element (atomic number) and isotope	Table I		Table II	
	Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)	Col. 1 Air ($\mu\text{Ci}/\text{ml}$)	Col. 2 Water ($\mu\text{Ci}/\text{ml}$)
If it is known that Sr 90, I 125, I 126, I 129, I 131 (I 133, Table II only), Pb 210, Po 210, At 211, Ra 23, Ra 224, Ra 226, Ac 227, Ra 228, Th 230, Pa 231, Th 232, Th-nat, Cm 248, Cf 254, and Fm 256 are not present	9×10^{-5}	3×10^{-6}
If it is known that Sr 90, I 125, I 126, I 129 (I 131, I 133, Table II only), Pb 210, Po 210, Ra 223, Ra 226, Ra 228, Pa 231, Th-nat, Cm 248, Cf 254, and Fm 256 are not present	6×10^{-5}	2×10^{-6}
If it is known that Sr 90, I 129 (I 125, I 126, I 131, Table II only), Pb 210, Ra 226, Ra 228, Cm 248, and Cf 254 are not present	2×10^{-5}	6×10^{-7}
If it is known that (I 129, Table II only), Ra 226, and Ra 228 are not present	3×10^{-6}	1×10^{-7}
If it is known that alpha-emitters and Sr 90, I 129, Pb 210, Ac 227, Ra 228, Pa 230, Pu 241, and Bk 249 are not present	3×10^{-9}	1×10^{-10}
If it is known that alpha-emitters and Pb 210, Ac 227, Ra 228, and Pu 241 are not present	3×10^{-10}	1×10^{-11}
If it is known that alpha-emitters and Ac 227 are not present	3×10^{-11}	1×10^{-12}
If it is known that Ac 227, Th 230, Pa 231, Pu 238, Pu 239, Pu 240, Pu 242, Pu 244, Cm 248, Cf 249 and Cf 251 are not present	3×10^{-12}	1×10^{-13}

4. If a mixture of radionuclides consists of uranium and its daughters in ore dust prior to chemical separation of the uranium from the ore, the values specified below may be used for uranium and its daughters through radium-226, instead of those from paragraphs 1, 2, or 3 above.

a. For purposes of Table I, Col. 1 - 1×10^{-10} $\mu\text{Ci}/\text{ml}$ gross alpha activity; or 5×10^{-11} $\mu\text{Ci}/\text{ml}$ natural uranium or 75 micrograms per cubic meter of air natural uranium.

b. For purposes of Table II, Col. 1 - 3×10^{-12} $\mu\text{Ci}/\text{ml}$ gross alpha activity; 2×10^{-12} $\mu\text{Ci}/\text{ml}$ natural uranium; or 3 micrograms per cubic meter of air natural uranium.

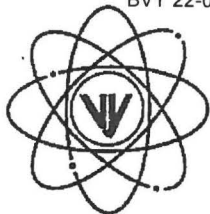
5. For purposes of this note, a radionuclide may be considered as not present in a mixture if (a) the ratio of the concentration of that radionuclide in the mixture (C_A) to the concentration limit for that radionuclide specified in Table II of Appendix "B" (MPC_A) does not exceed 1/10, (i.e. $C_A/\text{MPC}_A \leq 1/10$) and (b) the sum of such ratios for all the radionuclides considered as not present in the mixture does not exceed 1/4 i.e.

$$(C_A/\text{MPC}_A + C_B/\text{MPC}_B \dots + \leq 1/4).$$

Revision 21 Date 08/14/97

Appendix H

1. "Request to Amend Previous Approvals Granted Under 10CFR20.302(a) for Disposal of Contaminated Septic Waste and Cooling Tower Silt to Allow for Disposal of Contaminated Soil", dated June 23rd, 1999, BVY 99-80
2. "Supplement to Request to Amend Previous Approvals Granted Under 10CFR20.30(a) to Allow for Disposal of Contaminated Soil", dated January 4th, BVY 00-02
3. "Vermont Yankee Nuclear Power Station, Request to Amend Previous Approvals Granted under 10CFR20.302(a) to Allow for Disposal of Contaminated Soil (TAC No. MA5950), dated June 15th, 2000, NVY 00-58



VERMONT YANKEE NUCLEAR POWER CORPORATION

185 Old Ferry Road, Brattleboro, VT 05301-7002
(802) 257-5271

June 23, 1999
BVY 99-80

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

References:

- (a) Letter from R.W. Capstick, Vermont Yankee, to USNRC, "Request to Routinely Dispose of Slightly Contaminated Waste in Accordance with 10CFR20.302(a)", BVY 89-59, June 28, 1989
- (b) Letter from M.B. Fairtile, USNRC, to L. A. Trembley, Vermont Yankee, "Approval Under 10CFR20.302(a) of Procedures for Disposal of Slightly Contaminated Septic Waste on Site at Vermont Yankee (TAC No. 73776)", NVY 89-189, dated August 30, 1989
- (c) Letter from J.J. Duffy, Vermont Yankee, to USNRC, "Request to Amend Previous Approval Granted Under 10CFR20.302(a) for Disposal of Contaminated Septic Waste", BVY 95-97, dated August 30, 1995
- (d) Letter from S.A. Varga, USNRC, to D. A. Reid, Vermont Yankee, "Approval Pursuant to 10CFR20.2002 for Onsite Disposal of Cooling Tower Silt - Vermont Yankee Nuclear Power Station (TAC No. M93420)", NVY 96-39, dated March 4, 1996
- (e) Letter from P.D. Milano, USNRC, to D. A. Reid, Vermont Yankee, "Revised Safety Evaluation - Approval Pursuant to 10CFR20.2002 for Onsite Disposal of Cooling Tower Silt - Vermont Yankee Nuclear Power Station (TAC No. M96371)", NVY 97-85, dated June 18, 1997

Subject: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
Request to Amend Previous Approvals Granted Under 10 CFR 20.302(a) for Disposal of Contaminated Septic Waste and Cooling Tower Silt to Allow for Disposal of Contaminated Soil

In accordance with 10CFR 20.2002 (previously 10CFR20.302(a)), Vermont Yankee submits this application to amend the previously granted approvals to dispose of slightly contaminated septic waste and cooling tower silt on-site. This application expands the allowable waste stream to include slightly contaminated soil generated as a residual by-product of on-site construction activities.

This application specifically requests approval to dispose of soil contaminated at minimal levels, which has been or might be generated through end of station operations at the Vermont Yankee Nuclear Power Plant. The proposed soil disposal method is the same as the septic waste and cooling tower silt disposal methods requested in References (a) and (c), and approved in References (b) and (e). The disposal method utilizes on-site land spreading in the same designated areas used for septic waste and cooling tower silt. Disposal of this waste in the manner proposed, rather than holding it for future disposal at a 10CFR Part 61 licensed facility will save substantial costs and reserve valuable disposal site space for waste of higher radioactivity levels.

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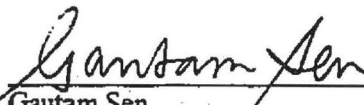
A radiological assessment and proposed operational controls for the inclusion of additional earthen material (soil) for on-site disposal with septic waste and cooling tower silt is provided in Attachment A. The assessment demonstrates that the dose impact expected from the existing accumulation of approximately 25.5 cubic meters of soil, in total with all past waste spreading operations, will not approach the dose limits already imposed for septic and cooling tower silt disposal. In addition to the existing accumulated soil, VY also requests that any future low level contaminated soil that might be generated as a by-product of plant construction and maintenance activities be allowed to be disposed of in the same manner provided the approved acceptance dose criteria are met. All soil analyses will be to environmental lower limits of detection. The results of all disposal operations will also be reported in the Annual Radioactive Effluent Release Report. The combined radiological impact for all on-site disposal operations will continue to be limited to a total body or organ dose of a maximally exposed member of the public of less than one mrem/year during the period of active Vermont Yankee control of the site, or less than five mrem/year to an inadvertent intruder after termination of active site control.

The Vermont Yankee Off-Site Dose Calculation Manual (ODCM) contains a copy of the original assessment and NRC approval for septic waste disposal (References a and b) and the previous amendment for cooling tower silt (References c and e). Upon receipt of your approval, the information contained in Attachment 1 as well as the basis for approval will be incorporated into the ODCM.

We trust that the information contained in the submittal is sufficient. However, should you have any questions or require further information concerning this matter, please contact Mr. Jim DeVincentis at 802-258-4236.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION



Gautam Sen
Licensing Manager

Attachment

cc: USNRC Region I Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
VT Department of Public Service

SUMMARY OF VERMONT YANKEE COMMITMENTS

BVY NO.: 99-80

The following table identifies commitments made in this document by Vermont Yankee. Any other actions discussed in the submittal represent intended or planned actions by Vermont Yankee. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager of any questions regarding this document or any associated commitments.

COMMITMENT	COMMITTED DATE OR "OUTAGE"
None	

VYAPF 0058.04 (Sample)
AP 0058 Original
Page 1 of 1

H-4

Revision <u>29</u> Date <u>1/11/02</u>
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Docket No. 50-271
BVY 99-80

Attachment 1

Vermont Yankee Nuclear Power Station

Assessment of On-site Disposal of Contaminated Soil by Land Spreading

Revision 29 Date 1/11/02

BVY 99-80 / Attachment 1 / Page 1

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1.0 INTRODUCTION

1.1 Background

In 1989, Vermont Yankee Nuclear Power Corporation requested approval from the NRC to routinely dispose of slightly contaminated septic waste in designated on-site areas in accordance with 10CFR20.302(a). Approval from the NRC was granted on August 30, 1989 and the information was permanently incorporated into the Offsite Dose Calculation Manual (ODCM) as Appendix B.

In 1995, Vermont Yankee Nuclear Power Corporation requested that the previous authorization for on-site disposal of very low-level radioactive material in septic waste be amended to permit the on-site disposal of slightly contaminated cooling tower silt material. Approval from the NRC was granted on June 18, 1997 and the information was permanently incorporated into the ODCM as Appendix F.

In 1994, approximately 25.5 m³ of excess soil was generated during on-site construction activities. Sampling of the soil revealed low levels of radioactivity that were similar in radionuclides and activity to the septic waste and cooling tower silts previously encountered. An evaluation determined that the soil could be managed in similar fashion as the septic waste and cooling tower silts; however, prior approval from the NRC would be required under 10 CFR 20.2002 (formerly 20.302(a)).

1.2 Objective

The objective of this assessment is to present the data and radiological evaluation to demonstrate that the proposed disposition of the soil will meet the existing boundary conditions as approved by the NRC for septic waste and cooling tower silt. The boundary conditions established for disposal of the septic waste and cooling tower silts on the designated plots are:

The dose to the whole body or any organ of a hypothetical maximally exposed individual must be less than 1.0 mrem/yr.

Doses to the whole body and any organ dose of an inadvertent intruder from the probable pathways of exposure are less than 5 mrem/yr.

Disposal operations must be at one of the approved on-site locations.

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2.0 WASTE DESCRIPTION

The soil that is the subject of this evaluation was derived from excavations resulting from activities associated with a new security fence along the plant's Protected Area boundary. The volume of soil generated was approximately 25.5 m^3 , and is typical of fill material containing light to dark brown poorly sorted soils with some small stones, and includes small incidental pieces of asphalt. The soil was removed from its original location by shovel, backhoe and front-end loader, and placed into dump trucks for transport to the location between the cooling towers where it was deposited on the ground surface and covered to prevent erosion. This location was selected because it was away from areas routinely occupied by plant staff, and could easily be controlled. The most probable source of the low levels of radioactive contamination is the presence of below detectable removable contamination redistributed by foot traffic from inside the plant to walkways and parking areas. Subsequent surface runoff carries the contamination to nearby exposed soil near the Protected Area boundary where it had accumulated over time to low level detectable concentrations.

In April 1995, a total of 20 composite soil samples were collected to characterize the volume. Composites were obtained by collecting a grab sample from one side, the top and the opposite side at equal distances along the length of the pile, then combining the three into one sample. Soil samples were sent to the Yankee Atomic Environmental Laboratory for analysis and counted to environmental lower limits of detection required of environmental media. Results of the analyses are presented in Table 1. Analytical results are provided for when the samples were collected and decay corrected to the present. The results identified both Cs-137 and Co-60 in most of the composite samples, which verified that plant-related radioactivity was present in the soil.

For the purpose of estimating the total activity in the soil pile, the actual analytical result was used for those samples that were less than the MDC to calculate the average radioactivity concentration.

The mass of soil (dry) was estimated by multiplying the total in-situ volume (25.5 m^3) by its wet density, $1.47\text{E}+03 \text{ kg/m}^3$, and then dividing by the wet:dry ratio of 1.12, thus yielding a mass of $3.35\text{E}+04 \text{ kg}$ (dry). The mass of the soil was then multiplied by the average Co-60 and Cs-137 concentrations measured in the soil to obtain the total activity of each radionuclide in the 25.5 m^3 . Table 2 presents the estimated total radioactivity in the 25.5 m^3 of soil at the time of sample collection and analysis, and decay corrected to the present.

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3.0 SOIL DISPOSAL AND ADMINISTRATIVE PROCEDURE REQUIREMENTS

The method of soil disposal will use the technique of land spreading in a manner consistent with the current commitments for the on-site disposal of septic waste and cooling tower silts as approved by the NRC. The accumulation of radioactivity on the disposal plot for this soil spreading operation will be treated as if cooling tower silt or septic waste was being disposed of since the characteristics of all these residual solids are similar (earthen-type matter). The south field (approximately 1.9 acres in size) designated and approved for septic waste and cooling tower silts disposal has been used for all past disposal operations, and will be used for the placement of this soil. Determination of the radiological dose impact has been made based on the same models and pathway assumptions used in the previous submittals.

Dry soil material will be dispersed using typical agricultural dry bulk surface spreading practices in approved disposal areas on-site. Incidental pieces of asphalt and stones that were picked up with the soil from area where paving ran along the fence line will be screened out before the soil is spread and disposed of as radioactive material at an off-site licensed facility if detectable radioactivity is found.

Records of the disposal that will be maintained include the following:

- (a) The radionuclide concentrations detected in the soil (measured to environmental lower limits of detection).
- (b) The total volume of soil disposed of.
- (c) The total radioactivity in the disposal operation as well as the total accumulated on each disposal plot at the time of spreading.
- (d) The plot on which the soil was applied.
- (e) Dose calculations or maximum allowable accumulated activity determinations required to demonstrate that the dose limits imposed on the land spreading operations have not been exceeded.

To ensure that the addition of the soil containing the radioactivity will not exceed the boundary conditions, the total radioactivity and dose calculation will include all past disposals of septic waste and cooling tower silt containing low-level radioactive material on the designated disposal plots. In addition, concentration limits applied to the disposal of earthen type materials (dry soil) restrict the placement of small volumes of materials that have relatively high radioactivity concentrations.

Any farmer leasing land used for the disposal of soil will be notified of the applicable restrictions placed on the site due to the spreading of low level contaminated material. These restrictions are the same as detailed for the previously approved septic waste spreading application.

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4.0 EVALUATION OF ENVIRONMENTAL IMPACTS

4.1 Site Characteristics

The designated disposal site is located on the Vermont Yankee Nuclear Power Plant site and is within the site boundary security fence. The south field consists of approximately 1.9 acres and is centered approximately 1500 feet south of the Reactor Building. This parcel of land has been previously approved by the NRC for the land disposal of septic waste and cooling tower silt.

4.2 Radiological Impact

The amount of radioactivity added to the south field soil is procedurally controlled to ensure that doses are maintained within the prior approved limits of the boundary conditions.

To assess the dose received (after spreading the soil) by the maximally exposed individual during the period of plant controls over the property, and to an inadvertent intruder after plants controls of access ends, the same pathway modeling, assumptions and dose calculation methods as approved for septic and cooling tower silt disposal were used. These dose models implement the methods and dose conversion factors as provided in Regulatory Guide 1.109.

The following six potential pathways were identified and included in the analysis:

- (a) Standing on contaminated ground.
- (b) Inhalation of resuspended radioactivity.
- (c) Ingestion of leafy vegetables.
- (d) Ingestion of stored vegetables.
- (e) Ingestion of meat.
- (f) Ingestion of cow's milk.

Both the maximum individual and inadvertent intruder are assumed to be exposed to these pathways; the difference between them is due to the occupancy time. The basic assumptions used in the radiological analyses include:

- (a) Exposure to ground contamination and re-suspended radioactivity is for a period of 104 hours per year during the Vermont Yankee active control of the disposal sites and continuous thereafter. The 104-hour interval is representative of a farmer's time spent on a plot of land (4 hours per week for 6 months).

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- (b) For the purpose of projecting and illustrating the magnitude of dose impacts over the remaining life of the plant, it is assumed that the concentration levels of activity as of April 1, 1999 remain constant. Table 1 indicates the measured radioactivity levels for Co-60 and Cs-137 first noted in the soil, and decay corrected to April 1, 1999.
- (b) For the analysis of the radiological impact during the Vermont Yankee active control of the disposal sites until 2013, no plowing is assumed to take place and all dispersed radioactive material remains on the surface forming a source of unshielded direct radiation.
- (c) The crop exposure time was changed from 2160 hours to 0 hours to reflect the condition that no radioactive material is dispersed directly on crops for human or animal consumption. Crop contamination is only through root uptake.
- (d) The deposition on crops of re-suspended radioactivity is insignificant.
- (e) The pathway data and usage factors used in the analysis are the same as those used in the Vermont Yankee's ODCM assessment of off-site radiological impact from routine releases, with the following exceptions. The fraction of stored vegetables grown on the contaminated land was conservatively increased from 0.76 to 1.0 (at present no vegetable crops for human consumption are grown on any of the approved disposal plots). Also, the soil exposure time to account for buildup was changed from the standard 15 years to 1 year.
- (f) It is conservatively assumed that Vermont Yankee relinquishes control of the disposal sites after the operating license expires in 2012 (i.e., the source term accumulated on a single disposal plot applies also for the inadvertent intruder).
- (g) For the analysis of the impact after Vermont Yankee control of the site is relinquished, the radioactive material is plowed under and forms a uniform mix with the top six inches of the soil; but nonetheless, undergoes re-suspension in the air at the same rate as the unplowed surface contamination. However, for direct ground plane exposure the self-shielding due to the six-inch plow layer reduces the surface dose rate by about a factor of four.

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As shown in the previous submittals, in which the concentrations of Co-60 and Cs-137 in septic waste exceed those identified in the soil, the liquid pathway was found to be an insignificant contributor to the dose. Therefore, the liquid pathway is not considered in this analysis.

The dose models and methods used to generate deposition values and accumulated activity over the operating life of the plant are documented the ODCM. Table 3 presents the radioactivity that currently exists on the south field after the last spreading event which occurred on September 28, 1998 (total elapsed time from September 28, 1998 to April 1, 1999 is 184 days). In addition, the total activity on the south field is presented assuming the addition of the 25.5 m³ of soil subject of this evaluation.

The total activity that would be present on south field at license termination (i.e., total elapsed time of 14 years post April 1, 1999, or 2013), assuming no future additions of material containing radioactivity after disposal of the proposed soil volume was also evaluated and is presented in Table 4.

In order to demonstrate compliance with the boundary conditions, the critical organ and whole body dose from all pathways to a maximally exposed individual during Vermont Yankee control, and to the inadvertent intruder were calculated. The dose calculations were performed using the dose conversion factors presented in Table 5 and 6 below which were obtained from the ODCM. The contribution to dose from Co-60 and Cs-137 to the whole body and organ at the present and at license expiration is presented in Table 7. The present and future total dose impact from the south field with and without spreading of the soil is presented in Table 8.

These results demonstrate that disposal of the approximately 25.5 m³ of accumulated soil will be well within the accepted dose limit criteria of 1 mrem/yr to any organ or whole body during the control period, and 5 mrem/yr to an inadvertent intruder after control of the site is assumed to be relinquished. This analysis shows that significant dose margin still exists on the approved disposal plots to accommodate potential future spreading operations.

5.0 RADIOLOGICAL PROTECTION

The disposal operation of soil piles will follow the applicable Vermont Yankee procedures to maintain doses as low as reasonably achievable and within the specific dose criteria as previously approved for septic waste and cooling tower silt disposal.

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6.0 CONCLUSIONS

Soil generated from on-site construction activities reflects an earthen type material similar in characteristics to septic waste residual solids and cooling tower silt with respect to the radiological pathway behavior and modeling. Based on the similarity in characteristics between the proposed soil volume and waste streams that have already been approved for disposal, and the evaluation of the incremental dose impact, it is concluded that disposal of the approximately 25.5 m³ of existing soil through on-site land spreading will meet the boundary conditions specified in the ODCM. That is, with respect to the addition of the approximately 25.5 m³ soil pile to the existing radioactivity already spread on the south field:

1. Total doses to the whole body and critical organ to the hypothetically maximally exposed individual were estimated as 3.00E-02 mrem/yr and 1.04E-01, respectively, which are less than the prescribed 1.0 mrem/yr.
2. Total doses to the whole body and critical organ of an inadvertent intruder from the probable pathways of exposure were estimated as 1.13E-01 mrem/yr and 2.21E-01 mrem/yr, respectively, which are less than 5 mrem/yr.
3. The disposal is assumed to take place on the south field that is the same site approved for disposal of septic waste and cooling tower silts.

If the soil were spread on an approved plot which had not yet been used for disposal, the dose impact from the approximately 25.5 m³ of soil alone would at present be 4.17E-03 mrem/yr whole body and a maximum organ dose of 1.46E-02 mrem/yr. In addition, for the inadvertent intruder, the whole body dose would be 1.60E-02 mrem/yr, and a maximum organ dose of 3.11E-02 mrem/yr. Each of these doses also meet the boundary conditions specified in the ODCM.

7.0 REFERENCES

- (1) Vermont Yankee ODCM, Rev 23, Appendix B, "Approval of Criteria for Disposal of Slightly Contaminated Septic Waste On-Site at Vermont Yankee".
- (2) Vermont Yankee ODCM, Rev 23, Appendix F, "Approval Pursuant to 10CFR20.2002 for On-Site Disposal of Cooling Tower Silt".
- (3) USNRC Regulatory Guide 1.109, Rev 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 40, Appendix I", dated October 1997.

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Table 1
Radioanalytical Results of Composite Soil Samples

Sample ID	Cs-137 (pCi/kg)		Co-60 (pCi/kg)	
	April, 1995	April, 1999	April, 1995	April, 1999
G22716	234	213	49**	29
G22717	522	476	143	84
G22718	337	307	37**	22
G22719	291	265	111	66
G22720	348	317	47**	28
G22721	135	123	73	43
G22722	107	98	82	48
G22723	222	203	140	83
G22724	180	164	92	54
G22725	269	245	118	70
G22726	810	739	114	67
G22727	378	345	106	63
G22728	810	739	124	73
G22729	376	343	62	37
G22730	331	302	87	51
G22731	253	231	5**	3
G22732	150	137	58	34
G22733	247	225	105	62
G22734	326	297	54**	32
G22735	235	214	100	59
<i>Average</i>	328	299	85	50
<i>Maximum Value</i>	810	739	143	84
<i>Minimum Value</i>	107	98	5	3
<i>Standard Deviation</i>	191	174	37	22

* Average wet to dry sample weight ratio equal to 1.12. Average wet density equal to 1.47 gm/cc.

** The apparent response of the gamma isotopic analysis which was less than Minimum Detectable Concentration.

Table 2
Estimated Total Radioactivity in Soil Volume

Radionuclide	Volume of Soil (m ³)	Mass (kg-dry)	Average Concentration (pCi/kg-dry)		Total Activity (μCi)	
			April, 1995	April, 1999	April, 1995	April, 1999
Cs-137	25.5	3.35E+04	328	299	11.0	10.0
Co-60	25.5	3.35E+04	85	50	2.8	1.7

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Table 3
Total Activity on South Field After Last Spreading Event

Radionuclide	Total activity after last spreading event ($\mu\text{Ci/acre}$)	Total activity decay corrected to April 1, 1999 ($\mu\text{Ci/acre}$)	Total activity after proposed soil disposal ($\mu\text{Ci/acre}$)
Mn-54	0.17	0.11	0.11
Co-60	5.93	5.55	6.44
Zn-65	0.074	0.044	0.044
Cs-137	32.27	31.90	37.16

Table 4
Total Projected Radioactivity on South Field Remaining at License Expiration

Radionuclide	Total Activity as of License Expiration ($\mu\text{Ci/acre}$)
Mn-54	8.9E-07
Co-60	0.89
Zn-65	7.6E-09
Cs-137	26.33

Table 5
All Pathway Critical Organ/Whole Body Dose Conversion Factors During Vermont Yankee Control of Disposal Sites

Radionuclide	Individual/Organ	Critical Organ Dose Factor (mrem/yr per $\mu\text{Ci/acre}$)	Whole Body Dose Factor (mrem/yr per $\mu\text{Ci/acre}$)
Mn-54	Adult/GI-LLI	3.75E-04	1.93E-04
Co-60	Teen/Lung	7.17E-04	5.31E-04
Zn-65	Child/Liver	1.64E-02	1.03E-02
Cs-137	Child/Bone	2.66E-03	7.02E-04

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Table 6
All Pathway Critical Organ/Whole Body Dose Conversion Factors Post Vermont
Yankee Control of Disposal Sites (Inadvertent Intruder)

Radionuclide	Individual/Organ	Critical Organ Dose Factor (mrem/yr per $\mu\text{Ci}/\text{acre}$)	Whole Body Dose Factor (mrem/yr per $\mu\text{Ci}/\text{acre}$)
Mn-54	Teen/Lung	1.02E-02	3.12E-03
Co-60	Teen/Lung	3.19E-02	9.09E-03
Zn-65	Child/Liver	1.89E-02	1.25E-02
Cs-137	Child/Bone	6.98E-03	3.85E-03

Table 7
Dose Contribution from Co-60 and Cs-137 in 25.5 m³ Soil Volume after Land
Spreading

Case	Present Dose Impact¹ (Maximally exposed individual)		Future Dose Impact² (Inadvertent Intruder)	
	Dose (mrem/yr)	Individual/ Organ	Dose (mrem/yr)	Individual/ Organ
Cobalt-60	4.75E-04 6.42E-04	Whole body Max. Organ	1.29E-03 4.51E-03	Whole body Max. Organ
Cesium-137	3.69E-03 1.40E-02	Whole body Max. Organ	1.47E-02 2.66E-02	Whole body Max. Organ

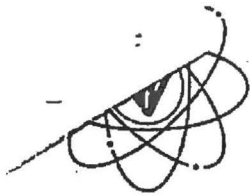
¹ Based on inventory of Co-60 of 0.895 $\mu\text{Ci}/\text{acre}$ and Cs-137 of 5.26 $\mu\text{Ci}/\text{acre}$ in April 1, 1999.

² Based on inventory of Co-60 of 0.141 $\mu\text{Ci}/\text{acre}$ and Cs-137 of 3.82 $\mu\text{Ci}/\text{acre}$ in April 1, 2013.

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Table 8
Present and Future Dose Impact Due to the Soil Spreading for Two Cases

Case	Present Dose Impact (Maximally exposed individual)		Future Dose Impact (Inadvertent Intruder)	
	Dose (mrem/yr)	Individual/ Organ	Dose (mrem/yr)	Individual/ Organ
<i>Case One</i> South Field as it currently exists	2.58E-02 8.96E-02	Whole body Max. Organ	9.70E-02 1.89E-01	Whole body Max. Organ
<i>Case Two</i> South Field if disposal of soil volume is approved	3.00E-02 1.04E-02	Whole body Max. Organ	1.13E-01 2.21E-01	Whole body Child/Bone
Increase in dose impact from disposal of soil	4.17E-03 1.46E-02	Whole body Max. Organ	1.60E-02 3.11E-02	Whole body Max. Organ



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January 4, 2000
BVY 00-02

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

References: (a) Letter, VYNPC to USNRC, "Request to Amend Previous Approvals Granted Under 10 CFR 20.302(a) for Disposal of Contaminated Septic Waste and Cooling Tower Silt to Allow for Disposal of Contaminated Soil," BVY 99-80, dated June 23, 1999

Subject: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
Supplement to Request to Amend Previous Approvals Granted
Under 10 CFR 20.302(a) to Allow for Disposal of Contaminated Soil

Reference (a) provided Vermont Yankee's application to amend the previously granted approvals to dispose of slightly contaminated septic waste and cooling tower silt on-site to include slightly contaminated soil generated as a residual by-product of on-site construction activities. The request was to allow the disposal of approximately 25.5 cubic meters of waste that has been accumulated to date and to allow for disposal of future waste from construction related activities.

Based on discussions with USNRC staff, additional information related to the estimated volume and dose consequences of the proposed future material was needed to complete your review. Attachment 1 has been revised accordingly to include the information requested. Attachment 1 supercedes the evaluation previously submitted.

We trust that the information will allow you to complete your review of our submittal. However, should you have any questions or require further information concerning this matter, please contact Mr. Jim DeVincentis at 802-258-4236.

Sincerely,

Vermont Yankee Nuclear Power Corporation



Gautam Sen
Licensing Manager

Attachment

cc: USNRC Region I Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
VT Department of Public Service

Docket No. 50-271

BVY 00-02

Attachment 1

Vermont Yankee Nuclear Power Station

Assessment of On-site Disposal of Contaminated Soil by Land Spreading

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1.0 INTRODUCTION

1.1 Background

In 1989, Vermont Yankee Nuclear Power Corporation requested approval from the NRC to routinely dispose of slightly contaminated septic waste in designated on-site areas in accordance with 10CFR20.302(a). Approval from the NRC was granted on August 30, 1989 and the information was permanently incorporated into the Offsite Dose Calculation Manual (ODCM) as Appendix B.

In 1995, Vermont Yankee Nuclear Power Corporation requested that the previous authorization for on-site disposal of very low-level radioactive material in septic waste be amended to permit the on-site disposal of slightly contaminated cooling tower silt material. Approval from the NRC was granted on June 18, 1997 and the information was permanently incorporated into the ODCM as Appendix F.

In 1994, approximately 25.5 m³ of excess soil was generated during on-site construction activities. Subsequent sampling and analysis of the soil revealed low levels of radioactivity that were similar in radionuclides and activity to the septic waste and cooling tower silts previously encountered. An evaluation determined that the soil could be managed in similar fashion as the septic waste and cooling tower silts; however, prior approval from the NRC would be required under 10 CFR 20.2002 (formerly 20.302(a)).

1.2 Objective

The purpose of this assessment is to present the data and radiological evaluation to demonstrate that the proposed disposition of the soil (i.e., on-site disposal via land spreading on designated fields) will meet the existing boundary dose conditions as approved by the NRC for septic waste and cooling tower silt. The boundary conditions established for disposal of the septic waste and cooling tower silt on designated plots are:

1. The dose to the whole body or any organ of a hypothetical maximally exposed individual must be less than 1.0 mrem/yr.
2. Doses to the whole body and any organ of an inadvertent intruder from the probable pathways of exposure are less than 5 mrem/yr.
3. Disposal operations must be at one of the approved on-site locations.

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2.0 WASTE DESCRIPTION

The existing accumulation of contaminated soil was derived from excavation activities associated with the construction of a new security fence along the plant's Protected Area boundary. The volume of soil generated was approximately 25.5 m^3 , and is typical of fill material containing light to dark brown poorly sorted soils with some small stones, and includes small incidental pieces of asphalt. The soil was removed from its original location by shovel, backhoe and front-end loader, and placed into dump trucks for transport to a location between the cooling towers where it was deposited on the ground surface and covered to prevent erosion. This location was selected because it was away from areas routinely occupied by plant staff, and could easily be controlled. The most probable source of the low levels of radioactive contamination is the presence of below detectable removable contamination redistributed by foot traffic from inside the plant to walkways and parking areas. Subsequent surface runoff carries the contamination to nearby exposed soil near the Protected Area boundary where it had accumulated over time to low level detectable concentrations.

In April 1995, a total of 20 composite soil samples were collected to characterize the accumulated volume. Composites were obtained by taking a grab sample from opposite sides of the pile and the top at equal distances along its length. These three grab samples were then combined into one composite sample. Soil samples were sent to the Yankee Atomic Environmental Laboratory for analysis and counted to environmental lower limits of detection required of environmental media. Results of the analyses are presented in Table 1. For estimating the total activity in the soil pile, the actual analytical result was used for those samples that were less than the MDC to calculate the average radioactivity concentration. Analytical results are provided for both the times when the samples were collected as well as decay corrected to the present (7/15/99). The results identified both Cs-137 and Co-60 in most of the composite samples, which verified that plant-related radioactivity, was present in the soil.

The mass of accumulated soil (dry) was estimated by multiplying the total in-situ volume (25.5 m^3) by its wet density, $1.47\text{E}+03 \text{ kg/m}^3$, and then dividing by the wet:dry ratio of 1.12, thus yielding a mass of $3.35\text{E}+04 \text{ kg}$ (dry). The mass of the soil was then multiplied by the average measured Co-60 and Cs-137 concentrations to obtain the total activity of each radionuclide in the 25.5 m^3 . Table 2 presents the estimated total radioactivity in the 25.5 m^3 volume at the time of sample collection and analysis, and decay corrected to the date of the most recent disposal (septic waste) spreading operation (i.e., July 15, 1999).

In addition to the existing 25.5 m^3 (900 ft^3) of soil included in this request, it is anticipated that the need to dispose of very low-level contaminated soil will occur in the future. Each spring, approximately 28.3 m^3 (1000 ft^3) of road and walkway sand spread during the winter season is swept up from inside the Protected Area. This material is subject to the same contamination mechanisms that are believed to have lead to the observed contamination in the construction fill removed from within the Protected Area in the past. For purposes of

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evaluating the radiological impact of potential future soil disposals, it is assumed that an additional 28.3 m³ per year of sand / soil is contaminated at the same concentration levels as originally observed (April 1995) in the currently collected 25.5m³ of soil. It is also assumed that this material is placed on the same approved disposal field used for all past septic and cooling tower silt disposal operations. Table 3 shows the estimated amount of radioactivity associated with the annual disposal of the 28.3m³ of soil. It is assumed that this material is disposed of each year for the next 14 years (until the end of plant operating license in 2013) on the same field (South Disposal Plot) along with the continued application of septic waste and cooling tower silt.

Table 4 shows a record of the actual amount of septic/silt material that has been spread on south field for the past 10 years. A review of the actual waste disposal operations show that the annual average radioactivity content placed on the 1.9 acre South field from septic and silt disposals are as follows:

Mn-54	0.147	uCi/year
Co-60	2.58	uCi/year
Zn-65	0.269	uCi/year
Cs-134	0.010	uCi/year
Cs-137	6.21	uCi/year

The maximum radioactivity inventory resulting from the accumulated buildup of past and projected future disposal operations (i.e., septic waste, cooling tower silt, plus the existing 25.5m³ of accumulated soil along with a projected annual addition of 28.3m³ of soil each year until the termination of the operating license) is shown on Table 5.

3.0 SOIL DISPOSAL AND ADMINISTRATIVE PROCEDURE REQUIREMENTS

The method of soil disposal will use the technique of land spreading in a manner consistent with the current commitments for the on-site disposal of septic waste and cooling tower silts as approved by the NRC. The accumulation of radioactivity on the disposal plot for this soil spreading operation will be treated as if cooling tower silt or septic waste was being disposed of since the characteristics of all these residual solids are similar (earthen-type matter). The south field (approximately 1.9 acres in size) has been designated and approved for septic waste and cooling tower silt disposal and has been used for all past disposal operations, and is expected to be used for the placement of soil. Determination of the radiological dose impact has been made based on the same models and pathway assumptions used in the previous submittals.

Dry soil material will be dispersed using typical agricultural dry bulk surface spreading practices in approved disposal areas on-site. Incidental pieces of asphalt and stones that are picked up with the soil will be screened out before the soil is spread and disposed of as radioactive material at an off-site licensed facility if detectable radioactivity is found.

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Records of the disposal that will be maintained include the following:

- (a) The radionuclide concentrations detected in the soil (measured to environmental lower limits of detection).
- (b) The total volume of soil disposed of.
- (c) The total radioactivity in the disposal operation as well as the total accumulated on each disposal plot at the time of spreading.
- (d) The plot on which the soil was applied.
- (e) Dose calculations or maximum allowable accumulated activity determinations required to demonstrate that the dose limits imposed on the land spreading operations have not been exceeded.

To ensure that the addition of soil containing low levels of radioactivity will not exceed the boundary dose limits, each new spreading operation will require an estimate of total radioactivity that includes all past disposals of septic waste, cooling tower silt and soil material on the designated disposal plots. This will be compared with the boundary dose limits or equivalent radioactivity limits on a per acre basis. In addition, concentration limits applied to the disposal of earthen type materials (dry soil) restrict the placement of small volumes of materials that have relatively high radioactivity concentrations.

Any farmer leasing land used for the disposal of soil (or other approved waste) will be notified of the applicable restrictions placed on the site due to the spreading of low level contaminated material. These restrictions are the same as detailed for the previously approved septic waste spreading application.

4.0 EVALUATION OF ENVIRONMENTAL IMPACTS

4.1 Site Characteristics

All designated disposal sites are located on the Vermont Yankee Nuclear Power Plant site and are within the site boundary security fence. The south field consists of approximately 1.9 acres and is centered approximately 1500 feet south of the Reactor Building. This parcel of land has been previously approved by the NRC for the land disposal of septic waste and cooling tower silt, and is the only portion of the approved disposal areas which has been utilized to-date for the spreading of contaminated material. For estimating the maximum future radiological impact, it is assumed in the analysis that all future disposal operations will continue to use the South field as the disposal plot.

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4.2 Radiological Impact

The amount of radioactivity added to the south field is procedurally controlled to ensure that doses are maintained within the prior approved limits of the boundary conditions.

To assess the dose received by the maximally exposed individual during the period of plant controls over the property, and to an inadvertent intruder after it is assumed plant access controls end, the same pathway modeling, assumptions and dose calculation methods as approved for septic and cooling tower silt disposal were used. These dose models implement the methods and dose conversion factors as provided in Regulatory Guide 1.109, Revision 1 (1977).

The following six potential pathways were identified and included in the analysis:

- (a) Standing on contaminated ground.
- (b) Inhalation of resuspended radioactivity.
- (c) Ingestion of leafy vegetables.
- (d) Ingestion of stored vegetables.
- (e) Ingestion of meat.
- (f) Ingestion of cow's milk.

As shown in the previous application for septic waste disposal, the liquid pathway was found to be an insignificant contributor to the dose for the radionuclides identified fixed in the soil type matrixes associated these waste forms. Therefore, the liquid pathway is not considered in this analysis.

Both the maximum individual and inadvertent intruder are assumed to be exposed to these pathways, the difference between them being due to the occupancy time. The basic assumptions used in the radiological analyses include:

- (a) Exposure to ground contamination and re-suspended radioactivity is for a period of 104 hours per year during the Vermont Yankee active control of the disposal sites and continuous thereafter. The 104-hour interval is representative of a farmer's time spent on a plot of land (4 hours per week for 6 months).
- (b) For the purpose of projecting and illustrating the magnitude of dose impact over the remaining life of the plant, it is assumed that future disposals of septic and silt material will be placed annually on the same field at the annual average radioactivity levels observed for these waste streams over the past ten years. The future disposals will also consist of the additional 28.3 m³ (1000 ft³) annual volume of new soil at the same radioactivity concentrations observed at the time of collection of the existing 25.5 m³ soil volume. The maximum individual dose impact from the buildup of disposed material

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occurs at the same time (2013) for both the Control Period and Intruder scenarios.

- (c) For the analysis of the radiological impact during the Vermont Yankee active control of the disposal sites until 2013, no plowing is assumed to take place and all dispersed radioactive material remains on the surface forming a source of unshielded direct radiation.
- (d) The crop exposure time was changed from 2160 hours to 0 hours to reflect the condition that no radioactive material is dispersed directly on crops for human or animal consumption. Crop contamination is only through root uptake.
- (e) The deposition on crops of re-suspended radioactivity is insignificant.
- (f) Most of the pathway data and usage factors used in the analysis are the same as those used in the Vermont Yankee's ODCM assessment of off-site radiological impact from routine releases. The fraction of stored vegetables grown on the contaminated land was conservatively increased from 0.76 to 1.0 (at present no vegetable crops for human consumption are grown on any of the approved disposal plots). For each year's spreading operations, the soil exposure time to account for buildup was changed from the standard 15 years to 1 year.
- (g) It is conservatively assumed that Vermont Yankee relinquishes control of the disposal sites after the operating license expires in 2013 (i.e., the source term accumulated on a single disposal plot applies also for the inadvertent intruder).
- (h) For the analysis of the impact after Vermont Yankee control of the site is relinquished, the radioactive material is plowed under and forms a uniform mix with the top six inches of the soil; but nonetheless, undergoes re-suspension in the air at the same rate as the unplowed surface contamination. However, for direct ground plane exposure the self-shielding due to the six-inch plow layer reduces the surface dose rate by about a factor of four.

The dose models and methods used to generate deposition values and accumulated activity over the operating life of the plant are documented in the Vermont Yankee ODCM. The total activity that would be present on south field at the end of the operating period (i.e., total elapsed time of 14 years post July 15, 1999, or 2013) from the buildup of all waste streams (i.e., septic, cooling tower silt and soil) is presented in Table 5.

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In order to evaluate the dose impact associated with the different disposal streams, a dose assessment was performed for the following four disposal scenarios:

- (I) Impact from past septic and silt spreading only – Table 7
- (II) Impact from past septic and silt spreading, plus a single 25.5m³ soil disposal operation for the existing accumulated soil – Table 8
- (III) Impact from past septic and silt disposals along with the existing 25.5 m³ of accumulated soil and postulated future annual soil disposal volumes (28.3 m³ /yr). –Table 9
- (IV) Impact from past septic and silt disposals plus annual projected disposals of septic, silt and soil. – Table 10

For each scenario, the critical organ and whole body dose from all pathways to a maximally exposed individual for both the Vermont Yankee control period and the inadvertent intruder situation were calculated. The dose calculations were performed using the dose conversion factors presented in Table 5, which were obtained from the Vermont Yankee ODCM, Appendix F, "Approval Pursuant to 10CFR20.2002 for On-Site Disposal of Cooling Tower Silt."

A summary of the calculated dose impact associated with the four different scenarios is shown in Table 11. These results demonstrate that disposal of the 25.5 m³ of accumulated soil will be well within the accepted dose limit criteria of 1 mrem/yr to any organ or whole body during the control period, and 5 mrem/yr to an inadvertent intruder. In addition, if continued soil spreading is necessary, the resulting dose is expected to also remain below the established limits even assuming the annual application of already approved disposal media (i.e., septic waste and cooling tower silt).

5.0 RADIOLOGICAL PROTECTION

The disposal operation of the soil will follow the applicable Vermont Yankee procedures to maintain doses as low as reasonably achievable and within the specific dose criteria as previously approved for septic waste and cooling tower silt disposal.

6.0 CONCLUSIONS

Soil generated from on-site construction and maintenance activities constitutes an earthen type material similar in characteristics to septic waste residual solids and cooling tower silt with respect to the radiological pathway behavior and modeling. Based on the similarity in characteristics between the proposed soil volume and waste streams that have already been approved for disposal and the evaluation of the incremental dose impact, it is concluded that the disposal of the existing 25.5 m³ and the projected 28.3 m³/year of soil through on-site land spreading will meet the existing NRC approved boundary dose conditions specified in

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the Vermont Yankee ODCM (see Appendix B for Septic Waste Disposal). That is, with respect to the addition of the initial 25.5 m³ of soil along with the projected 28.3 m³/year of soil and the projected future disposal of septic and silt waste to the existing radioactivity already spread on the south field:

1. Total doses to the whole body and critical organ of the hypothetically maximally exposed individual were estimated as 1.15E-01 mrem/yr and 4.03E-01 mrem/yr, respectively, which are less than the prescribed 1.0 mrem/yr limit during the period of active site control.
2. Total doses to the whole body and critical organ of an inadvertent intruder from the probable pathways of exposure were estimated as 7.57E-01 mrem/yr and 1.17 mrem/yr, respectively, which are less than 5 mrem/yr limit associated with an intruder scenario following assumed loss of site access control as the end of the operating license.
3. For purposes of projecting maximum impact, all disposals (past and future) are assumed to take place on the south disposal plot.

Therefore, the disposition of the present 25.5 m³ and the projected 28.3 m³/year of soil will continue to meet the existing boundary conditions as approved by the NRC for septic waste and cooling tower silt.

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7.0 REFERENCES

- (1) Vermont Yankee ODCM, Rev 23, Appendix B, "Approval of Criteria for Disposal of Slightly Contaminated Septic Waste On-Site at Vermont Yankee".
- (2) Vermont Yankee ODCM, Rev 23, Appendix F, "Approval Pursuant to 10CFR20.2002 for On-Site Disposal of Cooling Tower Silt".
- (3) USNRC Regulatory Guide 1.109, Rev 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 40, Appendix I", dated October 1977.

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Table 1
Radioanalytical Results of Composite Samples Taken from 25.5m³ Soil Pile

Sample ID	Cs-137 (pCi/kg)		Co-60 (pCi/kg)		
	April, 1995	July 15, 1999	April, 1995	July 15, 1999	
G22716	234	212.2	49	28.0	*
G22717	522	473.4	143	81.7	
G22718	337	305.7	37	21.1	*
G22719	291	263.9	111	63.4	
G22720	348	315.6	47	26.8	*
G22721	135	122.4	73	41.7	
G22722	107	97.0	82	46.8	
G22723	222	201.4	140	80.0	
G22724	180	163.3	92	52.6	
G22725	269	244.0	118	67.4	
G22726	810	734.7	114	65.1	
G22727	378	342.8	106	60.6	
G22728	810	734.7	124	70.8	
G22729	376	341.0	62	35.4	
G22730	331	300.2	87	49.7	
G22731	253	229.5	5	2.9	*
G22732	150	136.0	58	33.1	
G22733	247	224.0	105	60.0	
G22734	326	295.7	54	30.8	*
G22735	235	213.1	100	57.1	
<hr/>					
Average	328	298	85	49	
Maximum	810	735	143	82	
Minimum	107	97	5	3	
Std Dev.	186	169	36	20	

*The apparent response of the gamma isotopic analysis was less than the Minimum Detectable Concentration.

Table 2
Estimated Total Radioactivity in 25.5m³ Accumulated Soil

Nuclide	Volume of Soil (m ³)	Soil Mass (kg - dry)	Average Concentration (pCi/kg - dry)		Total Activity (uCi)	
			April 1995	July 15, 1999	April 1995	July 15, 1999
Cs-137	25.5	3.35E+04	328	298	11.0	10.0
Co-60	25.5	3.35E+04	85	49	2.8	1.6

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Table 3
Estimated Total Radioactivity in Future Soil Additions

Nuclide	Volume of soil (m ³)	Soil Mass (kg-dry)	Average Concentration (pCi/kg-dry assuming (April, 1995 concentrations))	Total Activity (uCi/yr)
Cs-137	28.3	3.72E+04	328	12.8
Co-60	28.3	3.72E+04	85	3.16

Table 4
Record of Septic and Silt Radioactive Material Spread Each Year on the South Field

Year	Spreading Date	Material Type	Mn-54 (uCi/acre)	Co-60 (uCi/acre)	Zn-65 (uCi/acre)	Cs-134 (uCi/acre)	Cs137 (uCi/acre)	Ce-141 (uCi/acre)
1990	10/31/90	Septage	0.00	3.89	0.00	0.00	0.26	0.00
1990	11/20/90	Septage	0.17	2.03	0.41	0.00	0.29	1.40E-08
1991		no spreading	0.00	0.00	0.00	0.00	0.00	0.00
1992	10/19/92	septage	0.11	1.73	0.52	0.05	0.32	0.006
1993	10/14/93	septage	0.05	1.41	0.21	0.00	0.30	0.00
1994	06/14/94	septage	0.08	0.43	0.00	0.00	0.09	0.00
1995	06/29/95	septage	0.00	0.88	0.00	0.00	0.00	0.00
1996		no spreading	0.00	0.00	0.00	0.00	0.00	0.00
1997	06/18/97	septage	0.12	1.00	0.00	0.00	0.19	0.00
1998	07/30/98	septage	0.14	0.72	0.09	0.00	0.12	0.00
1998	09/28/98	Silt	0.00	0.00	0.00	0.00	30.87	0.00
1999	07/15/99	Septage	0.11	1.47	0.20	0.00	0.25	0.00
Average Activity/yr (uCi/acre):			0.08	1.36	0.14	0.01	3.27	0.01
Average Activity (uCi/yr) to be disposed of on 1.9 acre field			0.147	2.58	0.269	0.010	6.21	0.001 not significant

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Table 5**Total Projected Radioactivity Remaining on South Field after License Termination**

Nuclide	Accum. Activity in Silt & Septic @ Year 2013 (uCi)	Accum. Activity in Soil @ year 2013 (uCi)	Accum. Activity Total All Paths @ year 2013 (uCi)	Accum. Activity Total All Paths @ Year 2013 (uCi/acre)*
Mn-54	0.26	-	0.26	0.14
Co-60	19.68	21.83	41.51	21.85
Zn-65	0.42	-	0.42	0.22
Cs-137	119.03	154.74	273.78	144.09
Cs-134	0.04	-	0.04	0.02

* The total activity is assumed to be spread on the 1.9 acre South field to generate the uCi/acre value.

Table 6**All Pathway Critical Organ/Whole Body Dose Conversion Factors**

Radionuclide	Individual/Organ	<u>During VY Control</u>		<u>Post VY Control (Intruder Scenario)</u>	
		Critical Organ Dose Factor (mrem/yr per <u>uCi/acre</u>)	Whole Body Dose Factor (mrem/yr per <u>uCi/acre</u>)	Critical Organ Dose Factor (mrem/yr per <u>uCi/acre</u>)	Whole Body Dose Factor (mrem/yr per <u>uCi/acre</u>)
Mn-54	Adult/GI-LLI	3.75E-04	1.93E-04	1.02E-02	3.12E-03
Co-60	Teen/Lung	7.17E-04	5.31E-04	3.19E-02	9.09E-03
Zn-65	Child/Liver	1.62E-02	1.03E-02	1.89E-02	1.25E-02
Cs-137	Child/Bone	2.66E-03	7.02E-04	6.98E-03	3.85E-03

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Table 7 (Scenario I)

Dose Impact from Past Septic and Silt Spreading on South Field (as of 7/15/99)

Control Scenario:

	Half-Life (Years)	All Other Spreadings To Date (uCi/acre)	Maximum Organ Dose Factor (mrem/yr/uCi/acre)	Whole Body Dose Factor (mrem/yr/uCi/acre)	Maximum Organ Dose (mrem/yr)	Whole Body Dose (mrem/yr)
Mn-54	0.86	0.20	3.75E-04	1.93E-04	7.35E-05	3.78E-05
Co-60	5.27	6.86	7.17E-04	5.31E-04	4.92E-03	3.64E-03
Zn-65	0.67	0.23	1.62E-02	1.03E-02	3.77E-03	2.40E-03
Cs-137	30.17	31.92	2.66E-03	7.02E-04	8.49E-02	2.24E-02
Total Dose:					9.37E-02	2.85E-02
Dose Limit:					1	1
% of Limit:					9.37%	2.85%

Intruder Scenario:

	Half-Life (Years)	All Other Spreadings To Date (uCi/acre)	Activity on Plot Decayed to Year 2013 (uCi/acre)	Maximum Organ Dose Factor (mrem/yr/uCi/acre)	Whole Body Dose Factor (mrem/yr/uCi/acre)	Maximum Organ Dose (mrem/yr)	Whole Body Dose (mrem/yr)
Mn-54	0.86	0.20	2.31E-06	1.02E-02	3.12E-03	2.36E-08	7.22E-09
Co-60	5.27	6.86	1.08E+00	3.19E-02	9.09E-03	3.46E-02	9.85E-03
Zn-65	0.67	0.23	1.15E-07	1.89E-02	1.25E-02	2.17E-09	1.43E-09
Cs-137	30.17	31.92	2.31E+01	6.98E-03	3.85E-03	1.62E-01	8.91E-02
Total Dose:						1.96E-01	9.90E-02
Dose Limit:						5	5
% of Limit:						3.92%	1.98%

Table 8 (Scenario II)

Dose Impact from Past Septic/Silt Spreading and Single 25.5m³ Soil Disposal

Control Scenario:

	Half-Life (Years)	All Spreadings to Date (uCi/acre)	Maximum Organ Dose Factor (mrem/yr/uCi/acre)	Whole Body Dose Factor (mrem/yr/uCi/acre)	Maximum Organ Dose (mrem/yr)	Whole Body Dose (mrem/yr)
Mn-54	0.86	0.196	3.75E-04	1.93E-04	7.35E-05	3.78E-05
Co-60	5.27	7.70	7.17E-04	5.31E-04	5.52E-03	4.09E-03
Zn-65	0.67	0.233	1.62E-02	1.03E-02	3.77E-03	2.40E-03
Cs-137	30.17	37.19	2.66E-03	7.02E-04	9.89E-02	2.61E-02
Total Dose:					1.08E-01	3.26E-02
Dose Limit:					1	1
% of Limit:					10.83%	3.26%

Intruder Scenario:

Under Scenario:	All Spreadings	Activity on Plot	Maximum Organ	Whole Body	Maximum	Whole Body	
	Half-Life	to Date	Decayed to 2013	Dose Factor	Dose Factor	Organ Dose	Dose
	(Years)	(uCi/acre)	(uCi/acre)	(mrem/yr/uCi/acre)	(mrem/yr/uCi/acre)	(mrem/yr)	(mrem/yr)
Mn-54	0.86	0.196	2.31E-06	1.02E-02	3.12E-03	2.36E-08	7.22E-09
Co-60	5.27	7.70	1.22E+00	3.19E-02	9.09E-03	3.88E-02	1.11E-02
Zn-65	0.67	0.233	1.15E-07	1.89E-02	1.25E-02	2.17E-09	1.43E-09
Cs-137	30.17	37.19	2.70E+01	6.98E-03	3.85E-03	1.88E-01	1.04E-01
Total Dose						2.27E-01	1.15E-01
Dose Limit:						5	5
% of Limit:						4.54%	2.30%



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 15, 2000

NVY 00-58

Mr. Samuel L. Newton
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, VT 05301

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION, REQUEST TO AMEND
PREVIOUS APPROVALS GRANTED UNDER 10 CFR 20.302(a) TO ALLOW
FOR DISPOSAL OF CONTAMINATED SOIL (TAC NO. MA5950)

Dear Mr. Newton:

By letter dated June 23, 1999, as supplemented on January 4, 2000, you submitted a request to amend a previously approved application granted by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 20.2002 (previously 10 CFR 20.302) to allow the addition of slightly contaminated soil and soil/sand material to the list of already approved materials (i.e., septic waste and cooling tower silt) for on-site disposal via land spreading on designated fields.

We have completed our review of your proposal and find it to be acceptable because the previously approved bounding conditions will continue to be met.

Pursuant to the provisions of 10 CFR Part 51, the NRC has published an Environmental Assessment and Finding of No Significant Impact in the *Federal Register* on June 15, 2000 (65 FR 37583).

Sincerely,

A handwritten signature in black ink, appearing to read "J. A. Zwolinski", is written over the "Sincerely," text.

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: Safety Evaluation

cc w/encl: See next page

Revision 29 Date 1/11/02

H-37

Vermont Yankee Nuclear Power Station

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
VERMONT YANKEE NUCLEAR POWER STATION
DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated June 23, 1999, as supplemented on January 4, 2000, Vermont Yankee Nuclear Power Corporation (the licensee), submitted a request to amend a previously approved application granted by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 20.2002 (previously 10 CFR 20.302) to allow the addition of slightly contaminated soil and soil/sand material to the list of already approved materials (i.e., septic waste and cooling tower silt) for on-site disposal via land spreading on designated fields.

In 1989, pursuant to 10 CFR 20.302 (current 10 CFR 20.2002), the licensee received approval from the NRC to routinely dispose of contaminated septic waste in designated on-site areas. In 1997, the NRC amended the approved on-site disposal application to also include contaminated cooling tower silt material.

In this 10 CFR 20.2002 amendment application, the licensee identified 25.5 cubic meters of soil to be disposed of on-site immediately, and approximately 28.3 cubic meters of soil/sand material to be disposed of on an annual basis until the expiration of the plant's operating license in 2013. The 25.5 cubic meters of contaminated soil were generated as a result of on-site construction activities. The anticipated 28.3 cubic meters of soil/sand material will be generated from the annual winter spreading of sand on roads and walkways at the plant site. The licensee has performed a comprehensive radiological evaluation that includes all of the anticipated materials (i.e., the current 25.5 cubic meters and the 28.3 cubic meters generated annually thereafter). The licensee's evaluation shows that the soil/sand can be managed on-site in the same manner as the septic waste and cooling tower silt (i.e., by land spreading on designated fields).

2.0 EVALUATION

The licensee will dispose of the soil and future soil/sand material using a land spreading technique consistent with its current commitments for on-site disposal of septic waste and cooling tower silts previously approved by the NRC. The licensee will continue to use the designated and approved areas of their property (approximately 1.9 acres in size) which currently receives the septic waste and cooling tower silts. Determination of the radiological dose impact of the new material has been made based on the same dose assessment models and pathway assumptions used in the previously approved submittals.

The licensee will procedurally control and maintain records of all disposals. The following information will be recorded:

1. The radionuclide concentrations detected in the material (measured to radiation levels consistent with the licensee's radiological environmental monitoring program).
2. The total volume of material disposed.
3. The total radioactivity in the disposal operation as well as the total radioactivity accumulated on each disposal plot at the time of spreading.
4. The plot on which the material was applied.
5. Dose calculations or maximum allowable accumulated activity determinations required to demonstrate that the dose condition values imposed (i.e., imposed by this 10 CFR 20.2002 application) on the land spreading operation have not been exceeded.

The bounding dose conditions for the on-site disposals are as follows:

1. The annual dose to the whole body or any organ of a hypothetical maximally exposed individual must be less than 1.0 mrem.
2. Annual doses to the whole body and any organ of an inadvertent intruder from the probable pathways of exposure must be less than 5 mrem.
3. Disposal operations must be at one of the approved on-site locations.

To ensure that the addition of new material containing low levels of radioactivity will not exceed the bounding dose conditions, for each new spreading operation the licensee will calculate an estimate of the total radioactivity applied to the designated disposal plots. These calculated estimates will include all past disposals of septic waste, cooling tower silt, soil and soil/sand material on the designated disposal plots. This will be compared with the bounding dose condition value or equivalent radioactivity value on a per acre basis. In addition, concentration limits will be applied to the disposed material to restrict the placement of small volumes of material that may have relatively high radioactivity concentrations.

The licensee assessed the dose that may be received by the maximally exposed individual during the period of plant control over the property, and to an inadvertent intruder after plant access control ends using the same pathway modeling, assumptions, and dose calculation methods that were previously approved by the NRC for the septic waste and cooling tower silt disposals. The dose models are based on the guidance in NRC Regulatory Guide 1.109, Revision 1 (1977).

The licensee's dose assessment is as follows:

1. Total annual doses to the whole body and critical organ of the hypothetically maximally exposed individual were estimated to be 0.115 mrem and 0.403 mrem, respectively. These values are less than the prescribed annual dose condition value of 1.0 mrem for the time period of active site control.

2. Total annual doses to the whole body and critical organ of an inadvertent intruder from the probable pathways of exposure were estimated to be 0.757 mrem and 1.17 mrem. These values are less than the prescribed annual dose condition value of 5.0 mrem for the time period after active site control.
3. The dose calculations are based on projecting the maximum potential impact of all disposals (past and future) on the designated disposal plot of land.

3.0 CONCLUSION

The staff finds the licensee's proposal to dispose of the low-level radioactive soil and soil/sand material, pursuant to 10 CFR 20.2002, in the same manner, location, and within the bounding dose conditions as the materials (i.e., septic waste and cooling tower silt) previously approved by the NRC to be acceptable because the bounding conditions will continue to be met.

Principal Contributor: S. Klementowicz

Date: June 15, 2000

Appendix I

1. "Request to Amend Previous Approval Granted Pursuant to 10CFR20.2002 for Disposal of Contaminated Soil", dated September 11th, 2000, BVY 00-71
2. Vermont Yankee Nuclear Power Station – Safety Evaluation for an Amendment to an Approved 10CFR20.2002 Application (TAC No. MA9972)", dated June 26th, 2001, NVY 01-66

VERMONT YANKEE NUCLEAR POWER CORPORATION

185 OLD FERRY ROAD, PO BOX 7002, BRATTLEBORO, VT 05302-7002
(802) 257-5271

September 11, 2000
BVY 00-71

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

References:

- (a) Letter, VYNPC to USNRC, "Request to Amend Previous Approvals Granted under 10 CFR 20.302(a) for Disposal of Contaminated Septic Waste and Cooling Tower Silt to Allow for Disposal of Contaminated Soil," BVY 99-80, dated June 23, 1999.
- (b) Letter, VYNPC to USNRC, "Supplement to Request to Amend Previous Approvals Granted under 10 CFR 20.302(a) to Allow for Disposal of Contaminated Soil," BVY 00-02, dated January 4, 2000.
- (c) Letter, USNRC to VYNPC, "Vermont Yankee Nuclear Power Station, Request to Amend Previous Approvals Granted under 10 CFR 20.302(a) to Allow for Disposal of Contaminated Soil (TAC No. MA5950)," NVY 00-58, dated June 15, 2000.
- (d) Letter, USNRC to VYNPC, "Revised Safety Evaluation -- Approval Pursuant to 10 CFR 20.2002 for Onsite Disposal of Cooling Tower Silt -- Vermont Yankee Nuclear Power Station (TAC No. M96371)," NVY 97-85, dated June 18, 1997.

**Subject: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
Request to Amend Previous Approval Granted Pursuant to
10 CFR 20.2002 for Disposal of Contaminated Soil**

In accordance with 10 CFR 20.2002 (previously 10 CFR 20.302(a)), Vermont Yankee (VY) submits this application to amend the previously granted approval to dispose of slightly contaminated soil. This application expands the allowable waste stream to include slightly contaminated soil generated as a residual by-product of other types of on-site construction activities.

In References (a) and (b), VY requested approval to dispose of approximately 25.5 m³ of accumulated soil that was generated due to construction activities. In addition, it was requested that VY be allowed to dispose of approximately 28.3 m³ of soil that is spread annually on station roads and walkways during the winter. NRC acceptance is documented in Reference (c).

This application specifically requests approval to dispose of contaminated soil that is created due to other on-site construction related activities including but not limited to design change implementation and land maintenance.

VERMONT YANKEE NUCLEAR POWER CORPORATION

BVY 00-71/ Page 2 of 2

In addition, VY requests that NRC's review recognize that, although VY indicated in Reference (b) that the south disposal field (approximately 1.9 acres in size) is currently expected to be used for disposal of the subject material, VY is also authorized to use the alternate north disposal field (approximately 10 acres in size). Approval to use both the north and south fields for disposal was granted in Reference (d). VY's radiological impact assessments have conservatively assumed all of the disposal activities occur on the smaller south field to maximize potential calculated doses. These assessments bound the situation where a portion of the land spreading occurs on the north field.

VY will continue to limit the total activity spread, from approximately 28.3 m³ of soil generated each year, to within the limits assumed in the radiological assessment previously submitted in Reference (b).

A radiological assessment and proposed operational controls for the inclusion of the additional material for on-site disposal was provided in Reference (b). The assessment demonstrates that the dose impact expected from the proposed activity, in total with all past waste spreading operations, will not approach the dose limits already imposed for septic and cooling tower silt disposal. All soil analyses will be to environmental lower limits of detection.

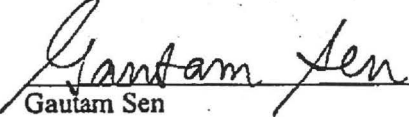
The results of all disposal operations will be reported in the Annual Radioactive Effluent Release Report. The combined radiological impact, for all on-site disposal operations, will continue to be limited to a total body or organ dose of a maximally exposed member of the public of less than one mrem/year during the period of active VY control of the site, or less than five mrem/year to an inadvertent intruder after termination of active site control.

Upon receipt of your approval, this request as well as the basis for approval will be incorporated into the Off-Site Dose Calculation Manual.

We trust that the information contained in the submittal is sufficient. However, should you have any questions or require further information concerning this matter, please contact Mr. Jim DeVincentis at 802-258-4236.

Sincerely,

Vermont Yankee Nuclear Power Corporation


Gautam Sen
Licensing Manager

cc: USNRC Region I Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
VT Department of Public Service

SUMMARY OF VERMONT YANKEE COMMITMENTS

BVY NO.: 00-71

The following table identifies commitments made in this document by Vermont Yankee. Any other actions discussed in the submittal represent intended or planned actions by Vermont Yankee. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager of any questions regarding this document or any associated commitments.

COMMITMENT	COMMITTED DATE OR "OUTAGE"
None	N/A

VYAPF 0058.04
AP 0058, Revision 1
Page 1 of 1



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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2-012 / Enclosure 2 / Page 277 of 338

June 26, 2001

NAVY 01-66

Mr. Michael A. Balduzzi
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
P.O. Box 7002
Brattleboro, VT 05302-7002

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - SAFETY EVALUATION
FOR AN AMENDMENT TO AN APPROVED 10 CFR 20.2002 APPLICATION
(TAC NO. MA9972)

Dear Mr. Balduzzi:

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review of the Vermont Yankee Nuclear Power Corporation (VYNPC) request dated September 11, 2000, to amend an approved 10 CFR 20.2002 (previously 10 CFR 20.302) application dated June 23, 1999, as supplemented on January 4, 2000. The licensee requested NRC approval to allow the addition of slightly contaminated soil resulting from on-site construction-related activities, including but not limited to, design change implementation and land maintenance, to the list of already approved materials (i.e., septic waste, cooling tower silt and soil/sand from roads and walkways) for on-site disposal.

Based on our review, we find the proposed changes to be acceptable because the previously approved bounding conditions will continue to be met. The enclosure to this letter provides our safety evaluation of VYNPC's application.

Pursuant to the provisions of 10 CFR Part 51, the NRC has published an Environmental Assessment and finding of No Significant Impact in the *Federal Register* on June 14, 2001 (66 FR 32399).

Sincerely,

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: Safety Evaluation

cc w/enc: See next page

Vermont Yankee Nuclear Power Station

cc:

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated September 11, 2000, Vermont Yankee Nuclear Power Corporation (VYNPC/ licensee) submitted a request to amend a Title 10 of the *Code of Federal Regulations* (10 CFR) Section 20.2002 (former 10 CFR 20.302) application, dated June 23, 1999, as supplemented on January 4, 2000, that was approved by the U.S. Nuclear Regulatory Commission (NRC). This amendment will allow the addition of slightly contaminated soil resulting from on-site construction-related activities, including but not limited to, design change implementation and land maintenance, to the list of already approved materials (i.e., septic waste, cooling tower silt and soil/sand from roads and walkways) for on-site disposal via land spreading on designated disposal fields.

In 1989, pursuant to 10 CFR 20.302 (current 10 CFR 20.2002), the licensee received approval from the NRC to routinely dispose of contaminated septic waste in designated on-site areas. In 1997, the NRC amended the approved on-site disposal application to also include contaminated cooling tower silt material. In 2000, the NRC amended the approved on-site disposal application to also include a one-time disposal of slightly contaminated soil and an annual disposal of 28.3 cubic meters of slightly contaminated soil/sand material.

In this 10 CFR 20.2002 amendment application, the licensee requested that slightly contaminated soil resulting from on-site construction-related activities be disposed of on-site on an annual basis until the end of the plant's operating license in 2013. The anticipated annual volume of soil generated by on-site construction, as identified by the licensee, combined with the soil/sand generated from the annual winter spreading of sand on roads and walkways at the plant site will not exceed 28.3 cubic meters. This volume is the same volume that was approved in the January 4, 2000, request. The licensee performed a comprehensive radiological evaluation which included the annual disposal of 28.3 cubic meters of soil and soil/sand materials, and shows that these materials can be managed on-site in the same manner as the septic waste and cooling tower silt (i.e., land spreading on designated fields).

2.0 EVALUATION

The licensee will dispose of the future soil material using a land spreading technique consistent with the current commitments for on-site disposal of septic waste, cooling tower silts and sand/soil material previously approved by the NRC. The licensee will continue to use the

-2-

designated and approved areas of their property which include approximately 1.9 acres, which currently receive the septic waste, cooling tower silts and soil/sand material, and approximately 10 acres which have not been previously used for disposal. Determination of the radiological dose impact of the new material has been made based on the same dose assessment models and pathway assumptions used in the previously approved submittals.

The licensee will procedurally control and maintain records of all disposals. The following information will be recorded:

1. The radionuclide concentrations detected in the material (measured to radiation levels consistent with the licensee's radiological environmental monitoring program);
2. The total volume of material disposed;
3. The total radioactivity in the disposal operation as well as the total radioactivity accumulated on each disposal plot at the time of spreading;
4. The plot on which the material was applied;
5. Dose calculations or maximum allowable accumulated activity determinations required to demonstrate that the dose condition values imposed (i.e., imposed by the approved 10 CFR 20.2002 application dated June 23, 1999) on the land spreading operation have not been exceeded.

The bounding dose conditions for the on-site disposals are as follows:

1. The annual dose to the whole body or any organ of a hypothetical maximally exposed individual must be less than 1.0 mrem.
2. Annual doses to the whole body and any organ of an inadvertent intruder from the probable pathways of exposure must be less than 5 mrem.
3. Disposal operations must be at one of the approved on-site locations.
4. Total annual combined volume of soil and soil/sand materials must not exceed 28.3 cubic meters.

To ensure that the addition of new material containing low levels of radioactivity will not exceed the bounding dose conditions, for each new spreading operation the licensee will calculate an estimate of the total radioactivity that includes all past disposals of septic waste, cooling tower silt, and soil/sand and soil material on the designated disposal plots. This will be compared with the bounding dose condition value or equivalent radioactivity value on a per acre basis.

The licensee assessed the dose from soil and soil/sand material that may be received by the maximally exposed individual during the period of plant control over the property, and to an inadvertent intruder after plant access control ends using the same pathway modeling, assumptions, and dose calculation methods that were previously approved by the NRC for the septic waste and cooling tower silt disposals. The dose models are based on the guidance in NRC Regulatory Guide 1.109, Revision 1 (1977).

The licensee's dose assessment is as follows:

1. Total annual doses to the whole body and critical organ of the hypothetically maximally exposed individual were estimated to be 0.115 mrem and 0.403 mrem respectively. These values are less than the prescribed annual dose condition value of 1.0 mrem for the time period of active site control.
2. Total annual doses to the whole body and critical organ of an inadvertent intruder from the probable pathways of exposure were estimated to be 0.757 mrem and 1.17 mrem. These values are less than the prescribed annual dose condition value of 5.0 mrem for the time period after active site control.
3. The dose calculations are based on projecting the maximum potential impact, of all disposals (past and future) on the approved disposal site.

3.0 CONCLUSION

The staff finds the licensee's proposal to dispose of the low-level radioactive soil material, pursuant to 10 CFR 20.2002, in the same manner, location, and within the bounding dose conditions as the materials (i.e., septic waste, cooling tower silt and soil/sand from roads and walkways) previously approved by the NRC to be acceptable.

The licensee has committed to permanently incorporate this modification into their Offsite Dose Calculation Manual.

Principal Contributor: A. Hayes

Date: June 26, 2001

APPENDIX J

1. "Request to Amend Previous Approval Granted Pursuant to 10CFR20.2001 for Increase of the Annual Volume Limit and One-time Spreading of Current Inventory," dated October 4, 2004, BVY 04-110.....2
2. "Safety Evaluation of Request to Amend Previous Approvals Granted Pursuant to 10CFR20.2002 – Vermont Yankee Nuclear Power Station (TAC No. MC5104)" dated July 19, 2005, NVY 05-090.....51

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October 4, 2004

Docket No. 50-271

BYV 04-110

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

References:

- a) Letter, VYNPC to USNRC, "Request to Amend Previous Approval Granted Pursuant to 10 CFR 20.2002 for Disposal of Contaminated Soil", BYV 00-71, dated September 11, 2000.
- b) Letter, USNRC to VYNPC, "Vermont Yankee Nuclear Power Station – Safety Evaluation for an Amendment to an Approved 10CFR 20.2002 Application (TAC No. MA9972)", NVC 01-66, Dated June 26, 2001.
- c) Letter, VYNPC to USNRC, "Supplement to Request to Amend Previous Approvals Granted under 10 CFR 20.302(a) to Allow for Disposal of Contaminated Soil", BYV 00-02, dated January 4, 2000.

**Subject: Vermont Yankee Nuclear Power Station
Request to Amend Previous Approval Granted Pursuant
to 10CFR20.2002 for Increase of the Annual Volume
Limit and One-time Spreading of Current Inventory**

In accordance with 10CFR20.2002 (previously 10CFR20.302(a)), Entergy Nuclear Operations, Inc. (ENO) submits this application to amend the previously granted Vermont Yankee (VY) approval to dispose of slightly contaminated soil. This application requests an increase of the current annual volume limit of 28.3 cubic meters of soil as specified in the previous approval (Reference (b)) to a new volume limit of 150 cubic meters of soil. This application also requests permission to spread the current inventory of approximately 528 cubic meters of soil as described in Attachment A in a one-time spreading activity following receipt of your approval.

ENO will continue to limit the total activity spread each year to remain within the limits specified in the radiological assessment previously submitted in Reference (c).

A radiological assessment of the impact of spreading the current inventory of soils and sediments located at VY is provided in Attachment A. The assessment concludes that:

- a) There is significant capacity remaining in the South Disposal Plot to continue to accept additional earthen materials for land spreading without exceeding established dose limitations.

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- b) The existing inventory of waste soils in storage can be placed on the South Disposal Plot without exceeding dose impact limits previously established for a single disposal field.
- c) The continued use of the South Disposal Plot will not exceed the limiting dose criteria established in the VY Offsite Dose Calculation Manual.
- d) The approved dose impact methodology used to determine compliance with the on-site spreading dose limits are not driven by the volume of waste material disposed of, but by the total radioactivity content of the material that is spread over a fixed disposal plot area (1.9 acres for the South Disposal Plot). The dose modeling conservatively assumes that all radioactivity spread on the field remains in the top 15 centimeter surface layer, even after subsequent additions are placed on the same field area. Existing limits on the concentration of radioactivity in waste media provides protection from small volumes of "hot" or high specific activity materials from being spread on the disposal field.

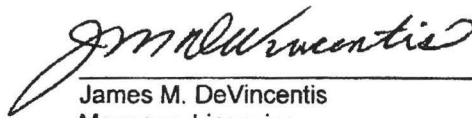
The results of all disposal operations will continue to be reported in the Annual Radioactive Effluent Release Report. The combined radiological impact, for all on-site disposal operations, will continue to be limited to a total body or organ dose of a maximally exposed member of the public of less than one mrem/year during the period of active VY control of the site, or less than five mrem/year to an inadvertent intruder after termination of active site control.

Upon receipt of your approval, this request as well as the basis for approval will be incorporated into the VY Offsite Dose Calculation Manual.

There are no new commitments being made in this submittal.

We trust that the information contained in the submittal is sufficient. However, should you have any questions or require further information concerning this matter, please contact me at (802) 258-4236.

Sincerely,


 James M. DeVincentis
 Manager, Licensing
 Vermont Yankee Nuclear Power Station

Attachment (1)

cc: USNRC Regional Administrator – Region 1
 USNRC Resident Inspector – VYNPS
 USNRC Project Manager – VYNPS
 Vermont Department of Public Service

Appendix J Original
 Off-Site Dose Calculation Manual
 Page 3 of 57

APPENDIX J (Continued)

Attachment to BVY 04-110
Docket No. 50-271

**Assessment of On-Site Disposal of Contaminated
Stored Soils by Land Spreading**

Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station

Appendix J Original
Off-Site Dose Calculation Manual
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1.0 EVALUATION OBJECTIVE

Current restrictions on the annual volume of slightly contaminated soil (1000 ft^3 or 28.3 m^3) that can be disposed of on-site (ODCM, Appendix I, Reference 1), coupled with several plant facility construction projects in recent years, has resulted in the accumulation of a back-log of low level contaminated earthen material that is awaiting to be dispositioned by land spreading on previously approved on-site disposal areas.

The objective of this assessment is to present the data and formal evaluation to demonstrate that the proposed one time disposal of the existing accumulated backlog of soil / sand materials (as of November 2003) without regards to the annual soil volume limit, will meet the existing dose objective boundary conditions as approved by the NRC for septic waste, Cooling Tower silt and other earthen type materials (Reference 1), even if use of the same disposal field for future spreading is assumed to continue over the remaining plant operating license.

The established dose based boundary conditions (NRC approved) for disposal and accumulation of low-level contaminated septic waste, Cooling Tower silts and soil/sand mixes on designated plots within the VY site boundary will continue to be applied without change. These dose limit criteria are taken from Appendix B of Reference 1, and are:

- The dose to the whole body or any organ of a hypothetical maximally exposed individual must be less than 1.0 mrem/yr during the period that VY has active control over the disposal plots (plant operating life).
- The doses to the whole body and any organ of an inadvertent intruder following the period of active plant control over the property from the probable pathways of exposure is less than 5 mrem/yr.
- Disposal operations must be at one of the approved on-site locations.

1.1 Background

In 1989, Vermont Yankee Nuclear Power Corporation requested from the NRC permission to routinely dispose of slightly contaminated septic waste in designated on-site areas in accordance with 10CFR20.302(a). Approval from the NRC was granted on August 30, 1989, provided that the request and analysis be permanently incorporated into the plant's Offsite Dose Calculation Manual (ODCM). Revision 9 to the ODCM (Appendix B) incorporated the assessment and the approval of methods utilized for on-site disposal of slightly contaminated sewage sludge by land spreading. The approval allowed for the existing septic inventory to be disposed of on-site along with future quantities anticipated to be generated as part of routine system maintenance. For purposes of demonstrating that future addition of waste materials could be added to the disposal plots, the radiological analysis projected an annual generation rate of about 18,600 gallons of sewage containing about 1400 kg of solid materials that might require on-site spreading. NRC permission for these future disposals was granted as long as both the projected dose (for both current and all past disposal operations) and radionuclide concentration limits ($\leq 10\%$ of the 10CFR20, Appendix B, Table II, concentration values) are satisfied. No specific limit on annual volume of septic waste that could be disposed of was included in the approval.

In 1995, Vermont Yankee requested from the NRC that the previous authorization for on-site disposal of septic waste be amended to permit the on-site disposal of slightly contaminated Cooling

APPENDIX J (Continued)

Tower silt material. The application analyzed the expected radiological impact from both the existing inventory at that time of about 14,000 ft³ (~396 m³) of accumulated silt, along with an operating cycle (18 months) generation rate of about 4000 ft³ (~113 m³). The NRC returned their safety evaluation, dated March 4, 1996, granting approval for the proposed silt disposal. Similar to the sewage waste disposal, NRC acceptance required that all disposal operations be conducted such that both the projected dose (for both current and all past disposal operations) and radionuclide concentration limits are satisfied. The soil concentration limits (for any sample) were based on limiting external annual dose to 25 mrem assuming continuous occupancy on an infinite plot at that concentration uniformly spread to a 15 cm depth. No specific limit on annual volume of Cooling Tower silt that could be disposed of was included in the approval. The NRC also required that any further modification to the proposed action have prior NRC staff approval.

In 1999 (with a supplemental filing in 2000), Vermont Yankee filed a third request under 10 CFR 20.2002 with the NRC to amend the previously approved applications for on-site land disposal of slightly contaminated earth type materials (septic sludge and Cooling Tower silt) to include approximately 900 ft³ (25.5 m³) of accumulated contaminated soil generated during construction activities within the VY Protected Area. Sampling of the soil revealed low levels of radioactivity that were similar in radionuclides and activity levels to the septic waste and Cooling Tower silts previously encountered. The request to the NRC for this additional material also indicated that additional amounts of contaminated soil / sand associated with road sweepings following winter sanding of road and walkways in the Protected Area could result in an estimated 1000 ft³ per year (28.3 m³ per year) that might need to be disposed of as slightly contaminated materials.

The NRC requested that the initial submittal (1999) of the soil spreading 20.2002 application include an analysis that evaluated projected future additions of an estimated annual volume of soil being added to the designated disposal plots. This information was required if Vermont Yankee intended to use the 20.2002 soil disposal application for approval to dispose of potential future volumes (i.e., not just a one-time disposal application) of low level contaminated soil in the same manner as already approved for septic waste and Cooling Tower silt. Vermont Yankee revised its application by adding an analysis for a projected annual volume of 1000 ft³, or equivalently 28.3 m³, of contaminated soil starting in the year 2000 and continuing on a yearly basis until end of plant license in 2013. At the end of the projected disposal stream, the accumulated buildup of contamination from all sources (septic waste, Cooling Tower silt and soil / sand mixes) on the disposal field was evaluated for both the dose impact to the critical receptor at the end of the control period and the assumed intruder. These dose impacts were found acceptable when compared against the original on-site spreading dose acceptance criteria of 1 mrem/yr (Control Period) and 5 mrem/yr (Intruder Scenario). The 1000 ft³ (28.3 m³) annual generation rate of soil was based on plant staff estimates that approximately that amount of soil and sand is collected from road and walkway sweepings inside the Protected Area following each year's winter clean-up as part of routine maintenance. This is the only type of earthen materials that has a specific annual volume limit associated with it in addition to the projected dose and concentration limits associated with the disposal of septic waste and Cooling Tower silt. No volume estimate for unidentified future site excavation and construction activities was provided.

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2.0 SUMMARY OF RESULTS

The evaluation of the radiological impact of all past, accumulated storage inventory, and projected future waste spreading operations on a single disposal plot (1.9 acres at South end of site) have indicated the following results and conclusions:

- All past spreading of septic waste, Cooling Tower silt, and soil/sand mixes through the end of 2003 have resulted in a maximum organ dose to a critical receptor (control Period use) that accounts for only 13.7% of the 1 mrem/year limit. With respect to the Intruder dose limit of 5 mrem/year, all past spreadings through the end of 2003 account for only 7.2% of the maximum organ dose to the limiting receptor at the end of plant license.
- The impact from the projected spreading of the existing waste materials in storage is estimated to account for only 5.3% of the 1 mrem/year Control Period dose limit, or only 17.7% of the same limit when all past spreadings are combined with the materials currently in storage (as of the end of November 2003). The maximum Intruder organ dose from all past spreading and stored materials is 0.456 mrem/yr, or 9.1% of the 5 mrem/yr Intruder scenario dose limit. These results indicate that the existing inventory of waste materials in storage can be placed on the South disposal plot without exceeding dose impact limits previously established for a single disposal field.
- Assuming the same annual average generation rate of radioactivity in waste materials (septage/silt/soil) that has been observed over the last fourteen years is added to all past waste spreadings and stored soil commitments, the projected dose at the end of the current plant licensing period (year 2013) yields a limiting maximum critical receptor dose (for either the control period or inadvertent intruder) equivalent to only 25.3% of the most restrictive annual dose limit (associated with the 1 mrem/year limit for the maximum organ during the Control Period). This finding demonstrates that the continued use of the South disposal plot, even with the addition of 18,653 ft³ of slightly contaminated soil currently in storage, will not exceed the approved limiting dose criteria established in the ODCM.
- The dose impact methodology used to determine compliance with the on-site spreading dose limits are not driven by the volume of waste material disposed of, but by the total radioactivity content of the material. Existing limits on the concentration of radioactivity in waste media provides protection from small volumes of "hot" or high specific activity materials from being spread on the disposal field.

APPENDIX J (Continued)

3.0 METHOD OF EVALUATION

The method of evaluating the impacts from the on-site spreading is the same as used and approved in the original exemption request made to the NRC under 10CFR20.302 for septic waste and which has been applied in all subsequent amendment requests for additional types of earthen materials to be disposed by land spreading on-site. The pathway and dose models found in Regulatory Guide 1.109 (Reference 2) are employed in performing the radiological dose impact assessment. The application of the dose models begins with the characterization of the waste materials that are to be subject to the dose evaluation.

3.1 Waste Characterization

The existing accumulated soil / sand that has created a backlog of accumulated material over the last several years is identified on Table 1, along with the estimated volume and origination of material. The soil materials were primarily derived from excavation activities associated with the construction of new security fences along the plant's Protected Area boundary and the construction of new plant installations associated with the capability to perform hydrogen water chemistry treatment of the plant's coolant system. Also included are road treatment sands used for winter traction inside the protected area.

The soil / sand mix is typical of fill material containing light to dark brown poorly sorted soils with some small stones, and may include small incidental pieces of asphalt. The soil was removed from its original location by shovel, backhoe and front-end loader, and placed into dump trucks for transport to the temporary storage area located between the Cooling Towers where it was deposited on the ground surface and covered to prevent erosion. This location was selected because it was away from areas routinely occupied by plant staff, and could easily be controlled. The most probable source of the low levels of radioactive contamination is due to the presence of below detectable removable contamination redistributed by foot traffic from inside the plant to walkways and parking areas. Subsequent surface runoff carries the contamination to nearby exposed soil near the Protected Area boundary where it accumulates over time to low-level detectable concentrations. Down wash and deposition of particulate activity released from the plant's Primary Vent Stack as part of routine gaseous emissions may also have contributed to the low levels of detectable activity.

For potential future disposal volumes of sand and soil, the current volume limit (1000 ft³ [28.3 m³], ODCM Appendix H, Reference 1) was based on the expected rate of road sand used for winter road and walkway traction, but did not anticipate or reflect the potential for future site construction activities that could excavate soils on-site that also contain low levels of plant related radioactivity. The present inventory of stored soil / sand between the Cooling Towers includes approximately 18,653 ft³ (528 m³) of material of which only 3.3% or 616 ft³ (17.4 m³) originated as roadway sweepings. In accordance with Appendixes B and F of the ODCM, disposal of septic waste and Cooling Tower silt material is not limited by an annual volume limit but by total dose impact related to the radioactivity content of the silt and the concentration of radioactivity contained within it. Currently, only soils / sand mixtures have an annual on-site disposal limit equal to 1000 ft³ (28.3 m³).

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Table 1
Inventory of Contaminated Soil Piles
In Storage
(December 2003)

Pile description	Overall Length	Overall width	Max Height	Estimated Volume
	ft	ft	inches	ft ³
(1) 2002-01: Security Fence Upgrade	75	21	44	4,679
(2) 2002-02: Security Fence Upgrade	75	28	60	9,000
(3) 2002-03: Security Fence Upgrade	39	22	42	2,457
(4) Sand Sweepings (inside protected area)	16	11	42	616
(5) 2001 HWC Solls Excavation	38	11	42	931
(6) 1996 Soil Remnants from Fence Upgrade	24	6.4	84	970
total Vol. (ft ³) =				18,653

3.2 Soil Disposal and Administrative Procedure Requirements

The method of soil/sand disposal of the existing backlog inventory will use the technique of land spreading in a manner consistent with the current commitments for the on-site disposal of septic waste and Cooling Tower silts as approved by the NRC and implemented in Appendices B and F of the Vermont Yankee ODCM (Reference1). The accumulation of radioactivity on the disposal plot for this proposed soil spreading operation will be treated as if Cooling Tower silt or septic waste was being disposed of since the characteristics of all these residual solids are similar (earthen-type matter). The South field (approximately 1.9 acres in size) has been used for all past disposal operations and is expected to be used for the placement of the existing backlog (see Table 1 above) of approximately 18,653 ft³ (528 m³) of soil and all annual projected future disposals of septic waste, Cooling Tower silt, and low-level contaminated soil volumes through the end of the plant's current operating license (year 2013). Determination of the radiological dose impact has been made based on the same models and pathway assumptions as indicated in Appendix B of the Vermont Yankee ODCM and approved as part of the original disposal analysis application for septic waste.

Both the existing accumulated and future potential soil material will be dispersed using typical agricultural dry bulk surface spreading practices in approved disposal areas on site. Incidental pieces of asphalt and large stones that are picked up with the soil will be screened out before the soil/sand is spread.

Records of the disposal that will be maintained include the following (as prescribed in Reference 1, Appendix B):

- the radionuclide concentrations detected in the soil/sand (measured to environmental lower limits of detection)
- the total volume of material disposed of
- the total radioactivity in the disposal operation as well as the total accumulated on each disposal plot at the time of spreading
- the plot on which the soil was applied, and

APPENDIX J (Continued)

- (e) dose calculations or maximum allowable accumulated activity determinations required to demonstrate that the dose limits imposed on the land spreading operations have not been exceeded.

To ensure that the addition of the soil containing the radioactivity will not exceed the boundary conditions, the total radioactivity and dose calculation will include all past disposal operations of septic waste, Cooling Tower silt and soil/sand decay-corrected to the date of the latest spreading placed on the designated disposal plots. In addition, concentration limits applied to the disposal of earthen type materials (dry soil) restrict the placement of small volumes of materials that have relatively high radioactivity concentrations.

Any farmer leasing land used for the disposal of soil will be notified of the applicable restrictions placed on the site due to the spreading of low level contaminated material. These restrictions are the same as detailed for septic waste spreading as given in Reference 1.

The disposal operation of the soil piles will follow the applicable Vermont Yankee procedures to maintain doses as low as reasonably achievable and within the specific dose criteria as previously approved for septic and Cooling Tower silt waste disposal.

3.3 Disposal Plot Characteristics

All designated disposal sites (six different plots) are located on the Entergy Nuclear Northeast Vermont Yankee plant site and are within the site boundary security fence. The South field consists of approximately 1.9 acres and is centered approximately 1500 feet South of the Reactor Building. This field has been the only one of the NRC approved fields that has actually been utilized for this purpose to-date. It is anticipated that future disposal operations will also utilize the South field since sufficient margin in comparison to the approved dose limit criteria still exists for anticipated waste disposal of the existing backlog of soil now in storage, plus all expected future disposals of septic waste, Cooling Tower silt and soil / sand mixes assuming the same observed generation rates (see Tables 13 through 19) persist to the end of the plant license in 2013.

In addition to the South field, the north end of the site has an additional ten acre parcel centered approximately 2,000 feet northwest of the Reactor Building. Prior assessments have demonstrated that a single plot of about 2 acres is sufficient to meet routine or expected disposal needs. Therefore the northern site could be subdivided into 5 plots if additional capacity was needed.

3.4 Radiological Impact Methodology

The amount of radioactivity added to any of the disposal fields is procedurally controlled to insure that doses are maintained within the prior approved limits of the boundary conditions (see Section 1.0 above).

To assess the dose received (after the spreading of the existing 18,653 ft³ [528 m³] along with both past recorded disposal applications, plus projected future applications) by the maximally exposed individual during the period of plant control, and to an inadvertent intruder after plant control of access ends (reference year of 2013), the same pathway modeling, assumptions and dose calculation methods as approved for septic waste, Cooling Tower silt and past soil / sand disposals are used.

APPENDIX J (Continued)

These dose models implement the methods and dose conversion factors as provided in Regulatory Guide 1.109 (Reference 2).

The following six potential pathways were identified and included in the analysis:

- (a) Standing on contaminated ground,
- (b) Inhalation of resuspended radioactivity,
- (c) Ingestion of leafy vegetables,
- (d) Ingestion of stored vegetables,
- (e) Ingestion of meat, and
- (f) Ingestion of cow's milk

Both the maximum individual and inadvertent intruder are assumed to be exposed to these pathways, with the difference between them being the occupancy time. The basic assumptions used in the radiological analyses include:

- (a) Direct exposure to ground contamination and inhalation of resuspended radioactivity from the ground by movement of air is for a period of 104 hours per year during the Vermont Yankee active control of the disposal sites and continuous thereafter. The 104-hour interval is representative of a farmer's time spent on a plot of land (4 hours per week for 6 months). The resuspension factor for soil material on the ground back into the air is taken as $1.0\text{E-}05$ based on an assumption that the disposal field will display characteristics similar to semiarid grassland experimental results. [NUREG-75/014; WASH-1400, "Reactor Safety Study", Appendix VI, Table VI E-3; USNRC, October 1975]
- (b) For the purpose of projecting and illustrating the magnitude of dose impacts over the remaining life of the plant, it is assumed that future disposals of septic, silt and soil material will be placed annually on the same field at the annual average radioactivity levels observed for these waste streams over the past fourteen years. The future disposals will also consist of the annual average radioactivity content observed in the accumulated soil/sand materials collected over the last several years that involved site facility construction projects that has lead to the existing backlog. The maximum individual dose impact from the buildup of disposed material occurs at the same time (2013) for both the Control Period and Intruder scenarios.
- (c) For the analysis of the radiological impact during the Vermont Yankee active control of the disposal sites until 2013, no plowing is assumed to take place and all dispersed radioactive material remains on the surface forming a source of unshielded direct radiation.
- (d) The crop exposure time was changed from 2160 hours to 0 hours to reflect the condition that no radioactive material is dispersed directly on crops for human or animal consumption. Crop contamination is only through root uptake.
- (e) The deposition on crops of resuspended radioactivity is insignificant.

APPENDIX J (Continued)

- (f) Most of the pathway data and usage factors used in the analysis are the same as those used in the Vermont Yankee's ODCM assessment of off-site radiological impacts from routine releases. The fraction of stored vegetables grown on the contaminated land was conservatively increased from 0.76 to 1.0 (at present no vegetable crops for human consumption are grown on any of the approved disposal plots). Also, the soil exposure time to account for buildup was changed from the standard 15 years (given in Reference 2) to 1 year.
- (g) It is conservatively assumed that Vermont Yankee relinquishes control of the disposal sites after the current operating license expires in 2013 (i.e., the source term accumulated on a single disposal plot applies also for the inadvertent intruder at that time).
- (h) For the analysis of the impact after Vermont Yankee control of the site is relinquished, the radioactive material is plowed under and forms a uniform mix with the top six inches of the soil, but, nonetheless, undergoes resuspension in the air at the same rate as the unplowed surface contamination. However, for direct ground plane exposure the self-shielding due to the six-inch plow layer reduces the surface dose rate by about a factor of four.

As shown in Reference 1 (Appendix B) for the original analysis in septic waste, the liquid transport and exposure pathway was found to be an insignificant contributor to the dose. Restrictions on the placement of the disposal plots put them at significant distances from wet lands, potable well supplies, and surface waters (Connecticut River). Therefore, the liquid pathway is not considered in this analysis.

The dose models and methods used to generate deposition values and accumulated activity over the operating life of the plant are documented in Reference 1 (Appendixes B and F). Table 18 presents the radioactivity that currently exists on the South field after the last spreading event, which occurred on November 4, 2003. Table 18 also indicates the residual radioactivity that would remain on South field at the end of the plant operating period (2013) if no additional disposals were to take place.

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4.0 ASSUMPTIONS AND INPUTS

- 1) The volume of the accumulated soil / sand currently in storage between the Cooling Towers (as of December, 2003) was estimated from field measurements of length, width, height and general shape of each pile taken in 2003. The estimated volume of each pile is summarized on Table 1.
- 2) The radioactivity content of each pile was determined by averaging the numerous grab samples (typically 30 samples per pile) collected for characterization of the soil material collected. Appendixes A through F provide the individual results of positive analysis for plant related radionuclides. Laboratory analyses were performed either by Vermont Yankee or the AREVA-Framatome (formerly the Yankee Atomic / Duke Engineering) Environmental laboratory with samples counted with respect to the NRC environmental LLD requirements as indicated in the VY ODCM.
- 3) Appendix G provides the total accumulated radioactivity on the South disposal Plot (1.9 acres) decayed to the date of the last spreading of waste materials of November 4, 2003.
- 4) Dose Conversion Factors (DCF) specific to the land spreading of materials at Vermont Yankee were taken from the VY ODCM, Appendix F, Tables 11 and 12 (Reference 1). These DCF's were based on the same dosimetric models and input parameters as used in the original analysis of septic waste spreading at VY and which was approved by the NRC for inclusion in the ODCM as Appendix B. Section 3.2 of this calculation provides an outline of the key aspects of the dose model and assumptions used. The following (Tables 2 & 3) listing notes these site specific dose conversion factors for both the Control Period and the Intruder scenario for the nuclides detected as positive in one or more of the sample analyses.

Table 2
Site Specific Control Period Dose Conversion Factors

Isotope	Individual/Organ	Max Organ	Whole Body	Half-Life	Decay Constant (λ) Yr ⁻¹
		DCF Control (mrem/yr- μ Ci/acre)	DCF Control (mrem/yr- μ Ci/acre)		
Mn-54	Adult/GI-LLI	3.75E-04	1.93E-04	3.125E+02	8.113E-01
Co-60	Teen/Lung	7.17E-04	5.31E-04	1.925E+03	1.315E-01
Zn-65	Child/Liver	1.64E-02	1.03E-02	2.438E+02	1.038E+00
Cs-134	Child/Liver	3.18E-03	1.28E-03	7.531E+02	3.356E-01
Cs-137	Child/Bone	2.66E-03	7.02E-04	1.102E+04	2.290E-02
Ce-141	Teen/Lung	1.54E-04	1.50E-05	3.250E+01	7.788E+00
Ce-144	Teen/Lung	6.00E-04	2.44E-05	284.6	8.888E-01

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Table 3
Site Specific Intruder Dose Conversion Factors

Isotope		Max Organ DCF Intruder (mrem/yr-μCi/acre)	Whole Body DCF Intruder (mrem/yr-μCi/acre)
Mn-54	Teen/Lung	1.02E-02	3.12E-03
Co-60	Teen/Lung	3.19E-02	9.09E-03
Zn-65	Child/Liver	1.89E-02	1.25E-02
Cs-134	Child/Liver	1.21E-02	9.36E-03
Cs-137	Child/Bone	6.98E-03	3.85E-03
Ce-141	Teen/Lung	1.21E-02	3.44E-04
Ce-144	Teen/Lung	5.00E-02	1.52E-03

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5.0 EVALUATIONS

In order to demonstrate compliance with the boundary dose conditions as stated in the ODCM, the critical organ and whole body dose from all pathways to the maximally exposed individual during Vermont Yankee Control Period and to the Inadvertent Intruder (for time periods following the end of the current operating plant license scheduled for 2013) were calculated for several scenarios (case studies) or combinations of disposal options. The dose calculations were performed using the site-specific dose factors for detected radionuclides as presented in Tables 2 and 3 (obtained from the VY ODCM, Appendix F, Tables 11 and 12). The objective is to demonstrate that the addition of the existing soil/sand materials currently in storage will not cause the radiological dose limits for materials spread on the South disposal field to be exceeded, even if it is assumed that all projected future annual disposals of septic waste, cooling tower silt and excavated soils/roadway sand containing the observed historical levels of plant related radionuclides are also placed on the same disposal plot.

5.1 Case Study I (Past Spreading Impacts)

The first case study (Case I) evaluated the spreading related dose impact associated with the past septic, Cooling Tower silt and soil spreading activity only. Table 4 shows the annual history and total amount of radioactivity in septic, silt and soil/sand waste materials by radionuclide that has been spread on the South field for the past 14 years (last spreading on 11/4/03). These radioactivity disposal values were taken from the past spreading records. Using (multiplying) the dose conversion factors listed on Tables 2 and 3 along with the total accumulated radioactivity content on the South disposal plot as of the last waste spreading in 2003 as shown on Table 18, the committed dose impact is found on Tables 21 and 22 for the Control Period dose as of the last spreading on November 4, 2003, as well as the Intruder Dose projected to 2013 from all materials currently spread on the South disposal plot. This assessment assumes no other material is spread on the disposal plot in the future, and therefore represents the existing dose commitments from all past spreadings. This establishes the dose margin in comparison to the Control Period and Intruder dose limits still available for the South disposal plot.

5.2 Case Study II (Stored Soil/Sand Inventory Impacts)

The second case study (Case II) looks at the spreading dose impact (Control Period and the Intruder impact at the end of plant license) first from the radioactivity associated with the existing soil in storage (18,653 ft³ as indicated on Table 1) between the Cooling Towers, and then in combination with all past spreadings in order to demonstrate that the single south disposal plot can accommodate the current backlog of soil now being stored if it were all spread on top of all past disposals.

Table 18 indicates the total accumulated septic, silt and soil activity per radionuclide remaining on the disposal plot as of the last spreading (11/04/03), as well as decayed to the projected reference date (6/1/04) for the spreading of all existing soil being held in storage, and to the projected end of the current plant license in 2013.

Tables 8, 9, 10 and 11 provide the estimate of radioactivity content in each of the storage piles of soil/sand being held between the Cooling Towers for the detected nuclides Cs-137, Co-60, Zn-65, and Mn-54, respectively. The activity concentrations are based on multiple grab samples collected

APPENDIX J (Continued)

from each of the six separate storage piles, and are taken from the grab sample laboratory radiological analyses for each of the storage piles provided in Appendixes A through F. The average Cs-137 concentration in each of the soil piles was determined by including in the average the minimum detectable concentration (MDC) of each radionuclide because most of the samples indicated a positive concentration for Cs-137. This slightly biases the assessment towards a conservative upper bound estimate of the potential activity in the pile since the use of the MDC does not represent the existence of positive measured activity in the sample, but is treated as such. For the other detected radionuclides (Co-60, Zn-65 and Mn-54), this biasing of the data was not applied since the occurrence of positive values only represented a small fraction of the total number of samples taken. The total activity determination for each pile is then calculated by taking the average measured concentration times the measured soil density times the estimated volume of the storage pile, correcting for decay time between the date of the sample analysis date to the estimated date of field disposal (6/1/04). This calculated total radionuclide activity value for each pile is assumed to be placed on the single 1.9 acre South disposal plot with the resulting surface concentration (uCi/acre) for the projected disposal calculated for each pile and totaled for all six piles currently in storage. The estimated average concentration for each of the four detected radionuclides is decay corrected from the date of the sample collection to June 1, 2004, as the reference date for estimating dose impacts from the proposed disposal of the accumulated material in storage. Table 12 summarizes the radioactivity associated with the soil/sand in storage decayed to June 1, 2004, and to the estimated end of the current plant license in 2013 for determining the Intruder dose impact at that time. Table 20 combines the radioactivity content of all soil/sand materials stored between the Cooling Towers with all remaining activity previously spread on the South disposal plot, decayed to 2013 for use in determining the Intruder dose at the end of plant license.

Table 23 applies the dose conversion factors from Table 2 for maximum organ and whole body doses with the projected activity on the South disposal plot from Tables 12, 18 and 20 to find the Control Period dose for all past material spreadings, stored material additions, and the sum total of past spreadings and proposed stored material additions. Table 24 illustrates the same dose impact combination of waste streams as applied to the site Intruder at the end of plant license. The combination of all past spreadings and the current stored soil material results in a maximum organ dose of only 17.7 % (0.177 mrem/yr) of the 1 mrem/yr Control Period dose limit. The maximum Intruder organ dose is estimated to be 0.456 mrem/yr or 9.1% of the 5 mrem/yr Intruder scenario dose limit. For comparison, Table 25 indicates that the maximum organ dose during the Control Period from only the inventory of 18,653 ft³ of soil/sand is estimated to be 0.0526 mrem/yr, or 5.3% of the 1 mrem/yr dose limit.

5.3 Case Study III (Projected Future Spreading Impacts)

The third case study (Case III) projects what the likely annual spreading additions of earthen material from all sources (septic waste, Cooling Tower silt, soil/sand mixes) would be based on historical records, combined with the existing 18,653 ft³ (528 m³) of material in storage, in order to determine the long term acceptability of the South disposal plot to continue to be used for all waste spreading applications.

Based on the historical spreading data listed in Table 4, Tables 5, 6 and 7 show the total accumulated septic waste, Cooling Tower silt and soil/sand annual average field spreading surface concentrations by radionuclide and waste source, respectively. This data breakdown is used in this disposal case study to predict future disposal rates to be applied to the South disposal plot. Table 13 provides a

APPENDIX J (Continued)

summary of the buildup of future spreadings over time after 2004 from all three waste streams (Septic waste, Cooling Tower silt, and soil/sand mixes), which could be projected to accumulate on the South disposal plot by 2013. The annual disposal quantity for each waste stream is based on the average annual disposal quantity observed for each stream since on-site disposal was originally approved in 1990. The total of all three streams (septic waste, Cooling Tower silt, and soil/sand) is taken to be representative of the future generation rate for each year from 2004 until 2013. The last column of Table 13 indicates the decay corrected accumulated activity on the South disposal plot from only additions to the South field from all future waste earthen materials. Tables 14, 15 and 16 provide additional detail of the projected annual and accumulated materials by waste stream (i.e., sand/soil mix, Cooling Tower silt and septic waste). The buildup equation used in Tables 13 through 18 accounts for both annual additions to the field as well as decay over this in-growth period is given by:

$$Act_i(t) = Act_i(a) * (1 - E^{(-\lambda_i t)}) / (1 - E)$$

Where:

$Act_i(t)$ = the total activity of radionuclide "i" (uCi) remaining at the end of the buildup period, t (years).

$Act_i(a)$ = the annual radioactivity addition of nuclide "i" to the disposal plot in uCi. The values for projected future additions are based on the annual average value observed for that nuclide for the specific disposal stream (i.e., septic waste, cooling tower slit, soil/sand).

E = $\exp(-\lambda_i \Delta t)$

λ_i = is the decay constant for the selected radionuclide "i" (1/year)

Δt = time interval between applications = 1 year.

In addition, Case Study II above evaluated the radiological impact from the disposal of the existing inventory of soil projected for mid-year, 2004 (including all past waste spreading operations), the total impact for 2004 should also include a projected disposal of both septic waste and Cooling Tower silt from one year's operation. Table 17 combines one year's generation of both septage and silt for assumed spreading in 2004, plus subsequent decay to 2013.

Table 26 combines the site-specific dose conversion factors from Tables 2 and 3 for the Control Period and Intruder scenario, respectively, with all previously spread radioactivity on the South disposal plot (Table 18) with both the proposed disposal of the existing soil stored inventory (Table 12) and projected annual additions of earth type waste materials based on the observed average annual generation rate (Tables 13 through 17). Table 19 provides a summary of accumulated radioactivity on the South field from the past spreading, materials in storage and future annual disposals on the same plot out to the assumed end of plant license, which corresponds to the Intruder dose scenario time frame. The resulting doses to the maximum organ and whole body of the maximum individual at the end of plant license (Table 26) reflects the maximum expected impact from all past and future disposals being placed on the South disposal field.

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6.0 RESULTS / CONCLUSIONS

The evaluation of the radiological impact of all past, accumulated storage inventory, and projected future waste spreading operations on the 1.9 acre South disposal plot have shown that the existing field is being operated within the previously approved dose limit criteria. The specific findings include:

1. For Case Study I (Past Spreading Impacts), Table 21 shows that after 14 years of spreading septic waste, Cooling Tower silt, and soil/sand mixes on the a single, 1.9 acre disposal plot the committed dose impact results in a maximum organ dose to a critical receptor (Control Period use) that accounts for only 13.7% of the 1 mrem/year limit. With respect to the Intruder dose limit of 5 mrem/year at the end of assumed active property control (i.e., end of plant license assumed for dose projection purposes), Table 22 indicates that all past spreadings through the end of 2003 account for only 7.2% of the maximum organ dose to the limiting receptor at the end of plant license. These finding illustrate that there is significant capacity remaining in the South disposal plot to continue to accept additional earthen materials that are suitable for land spreading without exceeding established dose limitations.
2. Table 23 shows that the impact from the projected spreading of the existing 18,653 ft³ of soil/sand material in storage is estimated to account for only 5.3% of the 1 mrem/year Control Period dose limit, or only 17.7% of the same limit when all past spreadings are combined with the materials currently in storage (as of the end of November 2003). The maximum Intruder organ dose from all past spreading and stored materials is calculated to be 0.456 mrem/yr (Table 24), or 9.1% of the 5 mrem/yr Intruder scenario dose limit. These results indicate that the existing inventory of waste materials in storage can be placed on the South disposal plot without exceeding dose impact limits previously established for a single disposal field, or using a significant proportion of the South disposal plot's capacity to receive additional materials for disposal in the future.
3. Assuming the same annual average generation rate of radioactivity in waste materials (septage/silt/soil) that has been observed over the last 14 years is added each year through 2013 to all past waste spreadings (including the stored soils inventory) already committed to the South disposal plot, Table 26 indicates that the projected dose at the end of the current plant licensing period (year 2013) yields a limiting maximum critical receptor dose (for either the control period or inadvertent intruder) equivalent to only 25.3% of the most restrictive annual dose limit (associated with the 1 mrem/year limit for the maximum organ during the Control Period). This finding demonstrates that the continued use of the South disposal plot, even with the addition of 18,653 ft³ of slightly contaminated soil currently in storage, will not exceed the approved limiting dose criteria established in the ODCM.
4. The approved dose impact methodology used to determine compliance with the on-site spreading dose limits are not driven by the volume of waste material disposed of, but by the total radioactivity content of the material that is spread over a fixed disposal plot area (1.9 acres for the South field). The dose modeling assumes that all radioactivity spread on the field remains in the top 15 cm surface layer of soil, even after subsequent additions are

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placed on the same field area. Existing limits on the concentration of radioactivity in waste media provides protection from small volumes of "hot" or high specific activity materials from being spread on the disposal field.

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Table 4

Record of Septic / Cooling Tower Silt / Construction Soil Radioactive Material Spreading
Each Year on the South Disposal Field

Year	Spreading Date	Material Type	Mn-54 (uCi/acre)	Co-60 (uCi/acre)	Zn-65 (uCi/acre)	Cs-134 (uCi/acre)	Cs-137 (uCi/acre)	Ce-141 (uCi/acre)
1990	10/31/90	Septage	0	3.89	0	0	0.26	0
	11/20/90	Septage	0.17	2.03	0.41	0	0.29	1.40E-08
1991		none	0	0	0	0	0	0
1992	10/19/92	Septage	0.11	1.73	0.52	0.05	0.32	0.006
1993	10/14/93	Septage	0.05	1.41	0.21	0	0.3	0
1994	06/14/94	Septage	0.08	0.43	0	0	0.09	0
1995	08/29/95	Septage	0	0.88	0	0	0	0
1996		none	0	0	0	0	0	0
1997	06/18/97	Septage	0.12	1	0	0	0.19	0
1998	07/30/98	Septage	0.14	0.72	0.09	0	0.12	0
	09/18/98	CT Silt	0	0	0	0	30.87	0
1999	07/15/99	Septage	0.11	1.47	0.2	0	0.25	0
2000	08/09/00	Septage*	0	0	0	0	0	0
	10/24/00	CT Silt	0.117	0.68	0	0	0	0
	10/24/00	Soil/Sand	0	0.602	0	0	3.698	0
2001	06-20-01	Septage	0	4.078	1.088	0	0.156	0.089
	09/25/01	Soil/Sand	0	0	0	0	1.4	0
2002	06/21/02	Septage	0.01	0.04	0	0.001	0.01	0
	11/11/02	Soil/Sand	0	0	0	0	1.37	0
2003	07/01/03	Septage	0	1.03	0	0	0	0
	10/25/03	Septage	0	0.12	0	0	0	0
	11/04/03	Soil/Sand	0	0	0	0	1.34	0
	11/04/03	CT Silt	0	0.256	0	0	0	0
Average Activity/yr (uCi/acre): (Over 14 year spreading history)			0.06	1.45	0.18	0.0036	2.90	0.01
Average activity (uCi/yr) disposed of on 1.9 acre field each year			0.123	2.76	0.342	0.007	5.52	0.013

* No radioactivity detected in septic waste samples.

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Table 5

Record of Septic Waste Only for Radioactive Material Spreading
Each Year on the South Disposal Field

Year	Spreading Date	Material Type	Mn-54 (uCi/acre)	Co-60 (uCi/acre)	Zn-65 (uCi/acre)	Cs-134 (uCi/acre)	Cs-137 (uCi/acre)	Ce-141 (uCi/acre)
1990	10/31/90	Septage	0	3.89	0	0	0.28	0
	11/20/90	Septage	0.17	2.03	0.41	0	0.29	1.40E-08
1991		none	0	0	0	0	0	0
1992	10/19/92	Septage	0.11	1.73	0.52	0.05	0.32	0.006
1993	10/14/93	Septage	0.05	1.41	0.21	0	0.3	0
1994	06/14/94	Septage	0.08	0.43	0	0	0.09	0
1995	08/29/95	Septage	0	0.88	0	0	0	0
1996		none	0	0	0	0	0	0
1997	06/18/97	Septage	0.12	1	0	0	0.19	0
1998	07/30/98	Septage	0.14	0.72	0.09	0	0.12	0
1999	07/15/99	Septage	0.11	1.47	0.2	0	0.25	0
2000	08/09/00	Septage*	0	0	0	0	0	0
2001	06-20-01	Septage	0	4.078	1.088	0	0.156	0.089
2002	06/21/02	Septage	0.01	0.04	0	0.001	0.01	0
2003	07/01/03	Septage	0	1.03	0	0	0	0
	10/25/03	Septage	0	0.12	0	0	0	0
Average Activity/yr (uCi/acre): (Over 14 year spreading history)			0.06	1.34	0.18	0.004	0.14	0.01
Average activity (uCi/yr) disposed of on 1.9 acre field each year			0.107	2.56	0.342	0.007	0.27	0.013

* No radioactivity detected in septic waste samples.

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Table 6

Record of Cooling Tower Silt Waste Only for Radioactive Material Spreading
Each Year on the South Disposal Field

Year	Spreading Date	Material Type	Mn-54 (uCi/acre)	Co-60 (uCi/acre)	Zn-65 (uCi/acre)	Cs-134 (uCi/acre)	Cs-137 (uCi/acre)	Ce-141 (uCi/acre)
1998	09/18/98	Silt	0	0	0	0	30.87	0
2000	10/24/00	Silt	0.117	0.68	0	0	0	0
2003	11/04/03	Silt	0	0.256	0	0	0	0
Average Activity/yr (uCi/acre) (31 year silt generation history)			0.004	0.030	0.000	0.000	0.996	0.000
Average activity (uCi/yr) disposed of on 1.9 acre field			0.007	0.057	0.000	0.000	1.892	0.000

Note: Cooling Tower yearly average is over 31 years of operation since the first disposal in 1998 included all accumulated material since plant startup

Table 7

Record of Construction Soil/Sand Waste Only for Radioactive Material Spreading
Each Year on the South Disposal Field

Year	Spreading Date	Material Type	Mn-54 (uCi/acre)	Co-60 (uCi/acre)	Zn-65 (uCi/acre)	Cs-134 (uCi/acre)	Cs-137 (uCi/acre)	Ce-141 (uCi/acre)
2000	10/24/00	Soil/Sand	0	0.602	0	0	3.698	0
2001	09/25/01	Soil/Sand	0	0	0	0	1.4	0
2002	11/11/02	Soil/Sand	0	0	0	0	1.37	0
2003	11/04/03	Soil/Sand	0	0	0	0	1.34	0
Average Activity/yr (uCi/acre):			0.00	0.15	0.00	0.00	1.95	0.00
Average activity (uCi/yr) disposed of on 1.9 acre field per year			0.00	0.29	0.00	0.00	3.71	0.00

Note: Soil/ road sand yearly average is over only the 4 years of operation since that is the period of material collection.

Cesium -137 in storage Piles after Last Spreading 2003*

Activity decayed to 06/01/04

Pile description	Estimated Volume	Date of Analysis	Decay time to 6/1/04	Aver. Cs-137 Conc.(w /LLD) No decay	Measured density	Total Cs-137 (decayed) (w /LLD)	Aver. Cs-137 Conc.(w/ LLD) decayed	Cs-137 applied to 1.9 acre field	% Cs-137 of total
	ft^3		years	uCi/gm	gm/cc	uCi	uCi/gm	uCi/acre	
2002-01: Security Fence Upgrade	4,679	05/06/02	2.08	7.12E-08	1.07	9.62E+00	6.79E-08	5.07E+00	31.48%
2002-02: Security Fence Upgrade	9,000	06/04/02	2.00	4.18E-08	1.11	1.13E+01	3.99E-08	5.94E+00	36.95%
Park Lot Sweep (inside protected area)	616	11/02/01	2.58	4.80E-08	1.56	1.23E+00	4.52E-08	6.48E-01	4.03%
2001 HWC Soils Excavation	931	10/22/01	2.58	5.41E-08	1.057	1.42E+00	5.10E-08	7.48E-01	4.65%
1996 Remnants(mixed contam. + non-cont)	970	04/13/95	8.13	9.22E-08	1.7	3.57E+00	7.65E-08	1.88E+00	11.89%
2002-03 Security Fence Chunks & soil mix	2,457	12/16/02	1.54	4.11E-08	1.24	3.42E+00	3.97E-08	1.80E+00	11.20%
totals =	18,853		Average =	5.81E-08	total =	3.06E+01	5.34E-08	1.61E+01	100.00%

* Note: Soil analysis data provided in Appendixes A through F.

Cobalt -60 in Storage Piles after Last Spreading in 2003*

Activity decayed to 06/01/04

Pile description	Estimated Volume	Date of Analysis	Decay time to 6/1/04	Aver.Co-60 Conc., no decay	Measured density	Total Co-60 (decayed)	Aver. Co-60 Conc., decayed	Co-60 applied to 1.9 acre field	% Co-60 of total
	ft^3		years	uCi/gm	gm/cc	uCi	uCi/gm	uCi/acre	
2002-01: Security Fence Upgrade	4,679	05/06/02	2.08	0.00E+00	1.07	0.00E+00	0.00E+00	0.00E+00	0.00%
2002-02: Security Fence Upgrade	9,000	06/04/02	2.00	0.00E+00	1.11	0.00E+00	0.00E+00	0.00E+00	0.00%
Park Lot Sweep (inside protected area)	816	11/02/01	2.58	8.72E-09	1.56	1.89E-01	8.22E-09	8.90E-02	33.01%
2001 HWC Soils Excavation	931	10/22/01	2.58	0.00E+00	1.057	0.00E+00	0.00E+00	0.00E+00	0.00%
1996 Remnants(mixed contam. + non-cont)	970	04/13/95	8.13	2.14E-08	1.7	3.43E-01	1.78E-08	1.81E-01	66.99%
2002-03 Security Fence Chunks & soil mix	2,457	12/16/02	1.54	0.00E+00	1.24	0.00E+00	0.00E+00	0.00E+00	0.00%
totals =	18,653		average =	5.02E-09	total =	5.12E-01	4.33E-09	2.70E-01	100.00%

* Note: Soil analysis data provided in Appendixes A through F.

Table 10

Pile description	Zinc -65 in Storage Piles after Last Spreading in 2003*					Activity decayed to 06/01/04			
	Estimated Volume	Date of Analysis	Decay time to 6/1/04	Aver. Zn-65 Conc., no decay	Measured density	Total Zn-65 (decayed)	Aver. Zn-65 Conc., decayed	Zn-65 applied to 1.9 acre field	% Zn-65 of total
	ft ³		years	uCi/gm	gm/cc	uCi	uCi/gm	uCi/acre	
2002-01: Security Fence Upgrade	4,679	05/06/02	2.08	0.00E+00	1.07	0.00E+00	0.00E+00	0.00E+00	0.00%
2002-02: Security Fence Upgrade	9,000	06/04/02	2.00	3.87E-09	1.11	1.05E+00	3.70E-09	5.50E-01	95.04%
Park Lot Sweep (inside protected area)	616	11/02/01	2.58	0.00E+00	1.56	0.00E+00	0.00E+00	0.00E+00	0.00%
2001 HWC Soils Excavation	931	10/22/01	2.58	2.08E-09	1.057	5.48E-02	1.96E-09	2.87E-02	4.96%
1996 Remnants(mixed contam. + non-cont)	970	04/13/95	8.13	0.00E+00	1.7	0.00E+00	0.00E+00	0.00E+00	0.00%
2002-03 Security Fence Chunks & soil mix	2,457	12/18/02	1.54	0.00E+00	1.24	0.00E+00	0.00E+00	0.00E+00	0.00%
totals =	18,653		Average =	9.92E-10	total =	1.10E+00	9.43E-10	5.79E-01	100.00%

* Note: Soil analysis data provided in Appendixes A through F.

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Table 11

Pile description	Mn-54 in Storage Piles after Last Spreading in 2003*					Activity decayed to 06/01/04			
	Estimated Volume	Date of Analysis	Decay time to 6/1/04	Aver. Mn-54 Conc., no decay	Measured density	Total Mn-54 (decayed)	Aver. Mn-54 Conc., decayed	Mn-54 applied to 1.9 acre field	% Mn-54 of total
	ft ³		years	uCi/gm	gm/cc	uCi	uCi/gm	uCi/acre	
2002-01: Security Fence Upgrade	4,679	05/06/02	2.08	2.09E-09	1.07	2.82E-01	1.99E-09	1.49E-01	69.78%
2002-02: Security Fence Upgrade	8,000	06/04/02	2.00	0.00E+00	1.11	0.00E+00	0.00E+00	0.00E+00	0.00%
Park Lot Sweep (inside protected area)	616	11/02/01	2.58	4.77E-09	1.56	1.22E-01	4.50E-09	6.44E-02	30.22%
2001 HWC Soils Excavation	931	10/22/01	2.58	0.00E+00	1.057	0.00E+00	0.00E+00	0.00E+00	0.00%
1996 Remnants (mixed contam. + non-cont)	970	04/13/95	8.13	0.00E+00	1.7	0.00E+00	0.00E+00	0.00E+00	0.00%
2002-03 Security Fence Chunks & soil mix	2,457	12/16/02	1.54	0.00E+00	1.24	0.00E+00	0.00E+00	0.00E+00	0.00%
totals =	18,653		Average=	1.14E-09	total =	4.05E-01	1.08E-09	2.13E-01	100.00%

* Note: Soil analysis data provided in Appendixes A through F.

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APPENDIX J (Continued)

Table 12

Soil/Sand in Storage Total Activity to be spread on 8/1/04 and decayed to 2013

Isotope	lamda 1/yr	As of 6/1/04 Qa (uCi/acre)*	Decay Time to 6/1/2013 (yrs)	As of 2013 Qa (uCi/acre)
Mn-54	0.8113	0.213	9	1.44E-04
Co-60	0.1315	0.270	9	8.27E-02
Zn-65	1.0382	0.579	9	5.07E-05
Cs-134	0.3356	0.000	9	0.00E+00
Cs137	0.0229	16.1	9	1.31E+01
Ce-141	7.7883	0.000	9	0.00E+00

* Note: Qa values from Table 8 (Cs-137), Table 9 (Co-60), Table 10 (Zn-65), and Table 11 (Mn-54)

Table 13

Projection of Additional Septic, silt, and soil/sand at Current Generation Rates to 2013

Isotope	lamda 1/yr	Annual Septic Addition Qa (uCi/acre)	Annual Silt Addition Qa (uCi/acre)	Annual Soil/Sand Addition Qa (uCi/acre)	Annual Total Additions Qa (uCi/acre)	Additional Accumulation at end of 9 years Qe (uCi/acre)
Mn-54	0.8113	0.056	0.0038	0.043	0.103	0.185
Co-60	0.1315	1.345	0.0302	0.174	1.549	8.183
Zn-65	1.0382	0.180	0.0000	0.116	0.296	0.458
Cs-134	0.3356	0.004	0.0000	0.000	0.004	0.012
Cs137	0.0229	0.142	0.9958	4.782	5.919	43.767
Ce-141	7.7883	0.007	0.0000	0.000	0.007	0.007

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APPENDIX J (Continued)

Table 14
Projection of Additional Sand/Soil Mix
At Historical Generation Rates To Year 2013 (9 years after 2004)

Isotope	lamda 1/yr	Projected Annual Soil/Sand Additions to Field Qa (uCi/acre)	Accumulated Sand/Soil at end of 9 years Qe (uCi/acre)
Mn-54	0.8113	4.26E-02	7.65E-02
Co-60	0.1315	1.74E-01	9.21E-01
Zn-65	1.0382	1.16E-01	1.79E-01
Cs-134	0.3356	0.00E+00	0.00E+00
Cs137	0.0229	4.78E+00	3.54E+01
Ce-141	7.7883	0.00E+00	0.00E+00

Table 15
Projection of Additional Cooling Tower (CT) Silt
At Historical Generation Rates To Year 2013 (9 years after 2004)

Isotope	lamda 1/yr	Projected Annual CT Silt Additions to Field Qa (uCi/acre)	Accumulated CT Silt at end of 9 years Qe (uCi/acre)
Mn-54	0.8113	3.77E-03	6.78E-03
Co-60	0.1315	3.02E-02	1.59E-01
Zn-65	1.0382	0.00E+00	0.00E+00
Cs-134	0.3356	0.00E+00	0.00E+00
Cs137	0.0229	9.96E-01	7.36E+00
Ce-141	7.7883	0.00E+00	0.00E+00

Table 16
Projection of Additional Septic Waste
At Historical Generation Rates To Year 2013 (9 years after 2004)

Isotope	lamda 1/yr	Projected Annual Septic Additions to Field Qa (uCi/acre)	Accumulated Septic Waste at end of 9 years Qe (uCi/acre)
Mn-54	0.8113	5.64E-02	1.01E-01
Co-60	0.1315	1.34E+00	7.10E+00
Zn-65	1.0382	1.80E-01	2.78E-01
Cs-134	0.3356	3.64E-03	1.19E-02
Cs137	0.0229	1.42E-01	1.05E+00
Ce-141	7.7883	6.79E-03	6.79E-03

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APPENDIX J (Continued)

Table 17

Projected 1 Yr Septage + Silt Spreading for 6/1/04 and decayed to 2013

Isotope	lamda 1/yr	As of 6/1/04 Qa (uCi/acre)	Decay Time to 6/1/2013	As of 2013 Qa (uCi/acre)
Mn-54	0.8113	0.060	9	4.06E-05
Co-60	0.1315	1.375	9	4.21E-01
Zn-65	1.0382	0.180	9	1.57E-05
Cs-134	0.3356	0.004	9	1.78E-04
Cs137	0.0229	1.138	9	9.26E-01
Ce-141	7.7883	0.007	9	2.45E-33

Table 18

Current Total Spreadings as of 11/4/03 and How much Remains at 6/1/04 and 2013

Isotope	lamda 1/yr	As of 11/4/03 Qa (uCi/acre)*	Decay Time to 6/1/04 (yrs)	As of 6/1/04 Qa (uCi/acre)	Decay Time to 2013 (yrs)	As of 2013 Qa (uCi/acre)
Mn-54	0.8113	0.241	0.5753	0.151	9	1.02E-04
Co-60	0.1315	16.33	0.5753	15.140	9	4.64
Zn-65	1.0382	1.47	0.5753	0.809	9	7.08E-05
Cs-134	0.3356	0.00063	0.5753	0.001	9	2.53E-05
Cs137	0.0229	38.18	0.5753	37.68	9	30.66
Ce-141	7.7883	8.80E-10	0.5753	1.0E-11	9	3.60E-42

* Data Taken from Plant Disposal
Records

APPENDIX J (Continued)

Table 19

Current Spreading Totals Plus Total Projected Future Spreadings to 2013

	Current Field Act. Decayed to 2013*	Stored Soil/sand Decayed to 2013**	1 Yr Septage+silt Decayed to 2013***	All Future Spreading Decayed to 2013****	Total all Activity Decayed to 2013
Isotope	(uCi/acre)	(uCi/acre)	(uCi/acre)	(uCi/acre)	(uCi/acre)
Mn-54	1.02E-04	1.44E-04	4.06E-05	0.185	1.85E-01
Co-60	4.636	8.27E-02	0.421	8.183	1.33E+01
Zn-65	7.08E-05	5.07E-05	1.57E-05	0.458	4.58E-01
Cs-134	2.53E-05	0.00E+00	1.78E-04	0.012	1.21E-02
Cs-137	30.66	13.10	0.926	43.767	8.85E+01
Ce-141	3.60E-42	0.00E+00	2.45E-33	0.007	6.79E-03

Notes: * Activity Concentration from Table 18.
 ** Activity Concentration from Table 12.
 *** Activity Concentration from Table 15, 16 and 17.
 **** Activity Concentration from Table 13 and 14.

Table 20

Radioactivity Content from Existing Materials (Past Spreadings & Stored Materials Only)

End Plant Operations for Intruder Dose: 6/1/2013 Last Application Date 6/1/2004
 End Date

Decay duration to end of Plant Operations: 9 years

Isotope	Total Activity on South Field plus storage piles (uCi/acre)* (6/1/04)	Total Activity decayed to year 2013 (uCi/acre)	Past Material Spread only up to 11/4/03 decayed to 2013 (uCi/acre)	Stored Material to be Spread decayed to 2013 (uCi/acre)	Total Past + Current Material decayed to 2013 (uCi/acre)
Mn-54	0.364	2.46E-04	1.02E-04	1.44E-04	2.46E-04
Co-60	15.41	4.72E+00	4.64E+00	8.27E-02	4.72E+00
Zn-65	1.388	1.21E-04	7.08E-05	5.07E-05	1.21E-04
Cs-134	0.001	2.53E-05	2.53E-05	0.00E+00	2.53E-05
Cs-137	53.78	4.38E+01	3.06E+01	1.31E+01	4.38E+01
Ce-141	1.0E-11	3.60E-42	3.60E-42	0.00E+00	3.60E-42

* Includes all material spread as of 11/4/03 decay corrected to the indicated date.

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Table 21
Past Spreading Control Period Doses
As of 11/04/03
(No Stored Material or Future Additions Included)

Isotope	Total Activity Remaining on South Field as of 11/04/03 uCi/acre	Max Organ Existing Material from Past Spreading Mrem/year	Whole Body Existing Material from Past Spreading Mrem/year
Mn-54	0.241	9.04E-05	4.65E-05
Co-60	16.33	1.17E-02	8.67E-03
Zn-65	1.47	2.41E-02	1.51E-02
Cs-134	0.00063	2.00E-06	8.06E-07
Cs-137	38.18	1.02E-01	2.68E-02
Ce-141	8.8E-10	1.36E-13	1.32E-14
	Total Dose = Dose Limit per field = % of Dose limit	1.37E-01 1 13.7%	5.07E-02 1 5.1%

Table 22
Past Spreading Intruder Period Doses
At End of Plant License in 2013
(No Stored Material or Future Additions Included after 11/4/03)

Isotope	Total Activity Remaining on South Field as of 11/04/03 uCi/acre	Total Activity Remaining on South Field decayed corrected to 2013 uCi/acre	Max Organ Existing Material from Past Spreading Mrem/year	Whole Body Existing Material from Past Spreading Mrem/year
Mn-54	0.241	1.02E-04	1.04E-06	3.18E-07
Co-60	16.33	4.636	1.48E-01	4.21E-02
Zn-65	1.47	7.08E-05	1.34E-06	8.85E-07
Cs-134	0.00063	2.53E-05	3.07E-07	2.37E-07
Cs-137	38.18	30.66	2.14E-01	1.18E-01
Ce-141	8.8E-10	3.80E-42	4.36E-44	1.24E-45
		Total Dose = Dose Limit per field = % of Dose limit	3.62E-01 5 7.2%	1.60E-01 5 3.2%

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Table 23
Control Period Dose: Past
Spreading & Current Stored Soil
Inventory (as of 6/1/04)

Isotope	Max Organ Existing Material from Past Spreading Mrem/year	Whole Body Existing Material from Past Spreading Mrem/year	Max Organ Stored Material to be Spread Mrem/year	Whole Body Stored Material to be Spread Mrem/year	Max Organ All Past Spreading Plus Stored Inventory Mrem/year	Whole Body All Past Spreading Plus Stored Inventory Mrem/year
Mn-54	5.67E-05	2.92E-05	7.99E-05	4.11E-05	1.37E-04	7.03E-05
Co-60	1.09E-02	8.04E-03	1.94E-04	1.43E-04	1.10E-02	8.18E-03
Zn-65	1.33E-02	8.33E-03	9.50E-03	5.96E-03	2.28E-02	1.43E-02
Cs-134	1.65E-06	6.65E-07	0.00E+00	0.00E+00	1.65E-06	6.65E-07
Cs-137	1.00E-01	2.65E-02	4.28E-02	1.13E-02	1.43E-01	3.78E-02
Ce-141	1.54E-15	1.50E-16	0.00E+00	0.00E+00	1.54E-15	1.50E-16
Total Dose = Dose Limit per field = % of Dose limit	1.24E-01 1 12.4%	4.29E-02 1 4.3%	5.26E-02 1 5.3%	1.75E-02 1 1.7%	1.77E-01 1 17.7%	6.03E-02 1 6.0%

Table 24
Past Spreading & Stored Soil Inventory (No Future Additions)
Intruder Dose at End of Plant License in 2013

Isotope	Max Organ Existing Material from Past Spreading Mrem/year	Whole Body Existing Material from Past Spreading Mrem/year	Max Organ Stored Material to be Spread Mrem/year	Whole Body Stored Material to be Spread Mrem/year	Max Organ All Past Spreading Plus Stored Inventory Mrem/year	Whole Body All Past Spreading Plus Stored Inventory Mrem/year
Mn-54	1.04E-06	3.18E-07	1.47E-06	4.48E-07	2.50E-06	7.66E-07
Co-60	1.48E-01	4.21E-02	2.64E-03	7.52E-04	1.51E-01	4.29E-02
Zn-65	1.34E-06	8.85E-07	9.58E-07	6.33E-07	2.30E-06	1.52E-06
Cs-134	3.07E-07	2.37E-07	0.00E+00	0.00E+00	3.07E-07	2.37E-07
Cs-137	2.14E-01	1.18E-01	9.14E-02	5.04E-02	3.05E-01	1.68E-01
Ce-141	4.36E-44	1.24E-45	0.00E+00	0.00E+00	4.36E-44	1.25E-45
Total Dose = Dose Limit per field = % of Dose limit	3.62E-01 5 7.2%	1.60E-01 5 3.2%	9.41E-02 5 1.9%	5.12E-02 5 1.0%	4.56E-01 5 9.1%	2.11E-01 5 4.2%

Table 25
Dose Impact from Spreading of Current Inventory of Stored Material Only*

Isotope	Total Stored Waste Activity As of 06/01/04** (uCi/acre)	Max Organ Control (06/01/04) mrem/year	Whole Body Control (06/01/04) mrem/year	Max Organ Intruder (2013) mrem/year	Whole Body Intruder (2013) mrem/yr	% Contribution Max organ by isotope Intruder Dose	% Contribution Max organ by isotope Control Period Dose
Mn-54	2.13E-01	7.99E-05	4.11E-05	1.47E-06	4.48E-07	0.00%	0.032%
Co-60	2.70E-01	1.94E-04	1.43E-04	2.64E-03	7.52E-04	0.25%	0.077%
Zn-65	5.79E-01	9.50E-03	5.96E-03	9.58E-07	6.33E-07	0.00%	3.760%
Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.0000%	0.0000%
Cs-137	1.61E+01	4.28E-02	1.13E-02	9.14E-02	5.04E-02	8.7%	17.0%
Ce-141	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.0000%	0.0000%
Ce-144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.0000%	0.0000%
	Total Dose = Dose Limit = % of Dose Limit	5.26E-02 1 5.3%	1.75E-02 1 1.7%	9.41E-02 5 1.9%	5.12E-02 5 1.0%	8.9%	20.8%

* Includes only Stored soil/sand materials collected as of end of 2003 (See Table 1)

** Total Activity values from Table 12.

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Table 26

Past, Stored Materials and Projected Future Disposal (all septage/silt and soils) Doses at End of Plant License*

Isotope	Total Waste Activity in 2013** (uCi/acre)	Max. Organ Control mrem/year	Whole Body Control mrem/year	Max. Organ Intruder mrem/year	Whole Body Intruder mrem/yr	% Contribution Max organ by isotope Intruder Dose	% Contribution Max organ by isotope Control Period Dose
Mn-54	1.85E-01	6.94E-05	3.57E-05	1.89E-03	5.77E-04	0.18%	0.027%
Co-60	1.33E+01	9.54E-03	7.06E-03	4.24E-01	1.21E-01	40.30%	3.776%
Zn-65	4.58E-01	7.51E-03	4.72E-03	8.66E-03	5.73E-03	0.82%	2.974%
Cs-134	1.21E-02	3.85E-05	1.55E-05	1.46E-04	1.13E-04	0.0139%	0.0152%
Cs-137	8.85E+01	2.35E-01	6.21E-02	6.18E-01	3.41E-01	58.7%	93.2%
Ce-141	6.79E-03	1.05E-06	1.02E-07	8.22E-05	2.34E-06	0.0078%	0.0004%
Ce-144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.0000%	0.0000%
	Total Dose =	2.53E-01	7.40E-02	1.05E+00	4.68E-01	100.0%	100.0%
	Dose Limit =	1	1	5	5		
	% of Dose Limit	25.3%	7.4%	21.1%	9.4%		

* Includes all past spreadings, soil stored Between Cooling Towers (as of 11/4/03), and all annual projected additions of septage/silt/soil.

** Total Activity values from last column of Table 19.

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APPENDIX J (Continued)

REFERENCES

- (1) Vermont Yankee Off site Dose Calculation Manual (ODCM), Revision 30, including the following appendixes:
 - (i) Appendix B, "Approval of Criteria for Disposal of Slightly Contaminated Septic Waste On-Site at Vermont Yankee" (Included NRC approval letter dated August 30, 1989, VY request for approval dated June 28, 1989 with Attachments I and II.)
 - (ii) Appendix F, "Approval Pursuant to 10CFR20.2002 For Onsite Disposal of Cooling Tower Silt" (Included NRC approval letter dated March 4, 1996, VY Request for Approval dated August 30, 1995.)
 - (iii) Appendix H, "Request to Amend Previous Approvals Granted Under 10CFR20.302(a) for Disposal of Contaminated Septic Waste and Cooling Tower Silt to Allow for Disposal of Contaminated Soil" dated June 23, 1999, with supplements dated January 4, 2000, and June 15, 2000.
- (2) USNRC Regulatory Guide 1.109, Rev.1; "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 40, Appendix I," dated October 1997.

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Appendix A

Security Fence Upgrade
Soil
Pile #2002-01

Sample location #	Soil Analysis Data		Density from samples data	
	Cs137 detected (uCi/gm)	Cs-137 LLD reported (uCi/gm)	Mn-54 detected (uCi/gm)	Volume (cc) wet weight (gm)
3B	5.76E-08		5.86E-08	1200 1266
3F	1.16E-07			1200 1062
4A	7.76E-08			1200 1393
4B	5.62E-08			1200 1512
4D	4.81E-08			1200 1483
4F	1.15E-07			1200 1137
5A	6.95E-08			1200 1390
5D	1.84E-07			1200 1194
5F	1.72E-07			1200 1051
1A		3.93E-08		1200 985
1B		5.97E-08		1200 1335
1C		2.92E-08		1200 1188
1D		7.39E-08		1200 1355
1E		3.65E-08		1200 1686
1F	4.18E-08			1200 1294
2A		6.83E-08		1200 1240
2B		8.53E-08		1200 1043
2C		4.88E-08		1200 1171
2D		4.16E-08		1200 1420
2E		3.43E-08		1200 1304
2F		4.75E-08		1200 1360
3A		8.35E-08		1200 1189
3C	9.59E-08			1200 1348
3E	5.65E-08			1200 1073
4C		4.44E-08		1200 1486
4E		9.37E-08		1200 1057
5B		7.23E-08		1200 1428
5C	4.53E-08			1200 1551

Wt.	4.06E-08	averages	1200	1285.75
Positive Ave. =		Density	1.07	gm/cc
Positive & LLD ave. =	7.12E-08	Density	1071.46	kg/m3

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Off-Site Dose Calculation Manual
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Appendix B

Security Fence Upgrade Soil
Pile # 2002-02

Soil Analysis data				Density from Sample Data	
Sample location #	Cs137 detected (uCi/gm)	Cs-137 LLD reported (uCi/gm)	Zn-65 detected (uCi/gm)	Volume (ml)	wet weight (gm)
A3	3.98E-08		7.73E-08	1200	1388
A4		6.93E-08		1200	1221
A5		4.53E-08		1200	1278
A6		4.69E-08		1200	1249
A7		4.04E-08		1200	1335
A8		7.29E-08		1200	1274
B1		4.06E-08		1200	1392
B2		3.02E-08		1200	1404
B3		4.08E-08		1200	1313
B4		3.34E-08		1200	1230
B5		3.38E-08		1200	1241
B6	2.13E-08			1200	1299
B7		4.24E-08		1200	1297
B8		3.73E-08		1200	1475
C3		3.90E-08		1200	1696
C4		4.25E-08		1200	1329
C5	4.06E-08			1200	1318
C6		4.14E-08		1200	1230
C7		4.60E-08		1200	1335
D7		3.22E-08		1200	1294
wtd positive ave.	5.09E-09		average	1200	1329.9
Positive & LLD ave. =		4.18E-08	density	1.11 gm/cc 1108.25 kg/m3	

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APPENDIX J (Continued)

Appendix C

Security Fence Upgrade
Soil
Pile # 2002-03

Soil Analysis Data			Density from samples data	
Sample location #	Cs137 detected (uCi/gm)	Cs-137 LLD reported (uCi/gm)	Volume (ml)	Wet weight (gm)
1	3.84E-08	3.00E-08	1000	1348
2		3.46E-08	1100	1424
3		2.58E-08	1100	1410
4		3.72E-08	1100	1410
5			1000	1034
6		4.58E-08	1200	1108
7		3.14E-08	1000	1526
8		4.29E-08	1000	1532
9		5.42E-08	1000	1086
10		5.00E-08	1000	1289
11		5.68E-08	1000	1185
12		4.62E-08	1000	1135
wt positive aver.= 3.2E-09			average density 1041.67 1.24 gm/cc	1290.58
Positive & LLD ave. = 4.11E-08			1238.96	kg/m3

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APPENDIX J (Continued)

Appendix D

2001 Protected Area Road Sweeping Pile

Soil Analysis Data					Density from Sample Data		
Sample location #	Cs137 detected (uCi/gm)	Cs-137 LLD reported (uCi/gm)	Co-60 detected (uCi/gm)	Co-60 LLD reported (uCi/gm)	Mn-54 detected (uCi/gm)	Volume (ml)	wet weight (gm)
1	6.55E-08			5.36E-08		1000	1435
2		3.43E-08		4.24E-08		1100	1753
3		2.91E-08		3.03E-08		1000	1550
4		4.83E-08		4.95E-08		1000	1615
5		6.37E-08		7.83E-08		1000	1657
6		3.67E-08		4.10E-08		1000	1609
7	1.96E-08		3.13E-08			1000	1565
8						1000	1650
9	3.11E-08			2.26E-08		1000	1439
10	6.09E-08		6.05E-08			1005	1762
11	4.07E-08		4.63E-08			1000	1421
12	4.75E-08			2.35E-08	1.23E-08	1000	1454
13	7.14E-08			4.30E-08		1000	1561
14	4.04E-08			3.22E-08		1000	1659
15		6.65E-08		6.47E-08	5.92E-08	1000	1503
16	4.20E-08			3.73E-08		1000	1607
17	4.69E-08			3.13E-08		1000	1536
18	5.05E-08			6.21E-08		1000	1594
19	5.45E-08			5.59E-08		1000	1474
20	6.25E-08		3.62E-08		2.38E-08	1000	1427
wt positive ave. =	3.17E-08		8.72E-09		4.77E-09	average 1005.2	1563.5
Positive & LLD ave. =		4.80E-08		4.43E-08		density 1.56 gm/cc 1555.4 kg/m3	

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APPENDIX J (Continued)

Appendix E

2001 HWC Soil Excavations

Soil Analysis Data			Density from Sample data		
Sample location #	Cs137 detected (uCi/gm)	Cs-137 LLD reported (uCi/gm)	Zn-65 detected (uCi/gm)	Volume (ml)	wet weight (gm)
1	4.09E-08			1000	1099
2		6.11E-08		1000	1043
3	6.04E-08			1000	1007
4	4.78E-08			1000	1008
5		5.30E-08		1000	1103
6		5.77E-08		1000	1025
7		5.94E-08		1000	956
8	4.75E-08			1000	957
9	5.05E-08			1000	976
10		6.16E-08		1000	1050
11		5.54E-08		1000	1043
12		6.50E-08		1000	1126
13	3.44E-08			1000	1141
14	5.67E-08			1000	997
15		5.26E-08		1000	1105
16		5.80E-08		1000	1190
17		5.58E-08	na	1000	1040
18		5.29E-08		1000	1141
19		5.85E-08		1000	1050
20	5.20E-08			1000	1073
wt positive ave.= 1.95E-08				average= 1000	1056.5
Positive & LLD ave.= 5.41E-08				density = 1.0565 1056.5	gm/cc kg/m3

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APPENDIX J (Continued)

Appendix F

1996 Soil Remnants Analysis - Security Fence Upgrade

The attached report ("Radioactivity Analyses for Soil Piles Stored Between Cooling Towers", REG-115/96, dated July 15, 1996) indicates that two soil piles totaling 4000 ft³ were collected in 1995 and stored between the Cooling Towers. Even though the larger of the two piles (3100 ft³) did not indicate any detectable plant related radioactivity, the two piles were eventually combined with portions disposed of by land spreading on the South disposal plot as annual disposal volume limits for soil/sand mixes permitted. The remnants of the piles currently contain about 970 ft³ of material. For the estimated concentration of Cs-137 in the combined piles, concentration values from Tables 1 and 3 were averaged in proportion to their volumes as shown:

Cs-137: 900 ft³ at an average concentration of 328 pCi/kg
 3100 ft³ with no detectable activity (average absolute value = 11 pCi/kg)

$$\begin{aligned} \text{Average concentration (weighted ave.)} &= \frac{900 \text{ ft}^3 * 328 \text{ pCi/kg} + 3100 \text{ ft}^3 * 11 \text{ pCi/kg}}{900 \text{ ft}^3 + 3100 \text{ ft}^3} \\ &= 82.33 \text{ pCi/kg (dry)} \end{aligned}$$

Therefore, on a wet weight basis applicable to the measured volume of the remaining pile of soil between the Cooling Towers, the weighted average concentration for Cs-137 is:

$$\begin{aligned} &= 82.33 \text{ pCi/kg} * 1 \text{ kg/1000 g} * 1 \text{ E-06 uCi/pCi} * 1.12 \text{ wet/dry volume ratio} \\ &= 9.22 \text{ E-08 uCi/gm (wet) for Cs-137} \end{aligned}$$

For Co-60;

$$\begin{aligned} \text{Average concentration (weighted ave.)} &= \frac{900 \text{ ft}^3 * 85 \text{ pCi/kg} + 3100 \text{ ft}^3 * 0.034 \text{ pCi/kg}}{900 \text{ ft}^3 + 3100 \text{ ft}^3} \\ &= 19.15 \text{ pCi/kg} \end{aligned}$$

On a wet weight basis applicable to the measured volume of the remaining pile of soil between the Cooling Towers, the weighted average concentration for Co-60 is:

$$\begin{aligned} &= 19.5 \text{ pCi/kg} * 1 \text{ kg/1000 g} * 1 \text{ E-06 uCi/pCi} * 1.12 \text{ wet/dry volume ratio} \\ &= 2.14 \text{ E-08 uCi/gm (wet) for Co-60.} \end{aligned}$$

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APPENDIX J (Continued)

MEMORANDUM

YANKEE ATOMIC - BOLTON

To	G. D. Weyman	Date	July 15, 1996
From	M. S. Strum	Group #	REG-115/96
Subject	Radioactivity Analyses for Soil Piles Stored Between the Cooling Towers	W.O.#	
		I.M.S.#	
		File #	vyzoilac.doc

REFERENCES

(1) Environmental Laboratory Analysis Reports Sample Numbers G22686 through G22735, soil - Fence and Repaving. Reference Date 4/13/95.

BACKGROUND

Site area construction activities have generated two piles of soil from the protected area that were placed between the plant's cooling towers pending radiological assessment and final disposal disposition. One pile was estimated at 3100 cubic feet and was initially marked as having come from security fence excavation. The second pile, estimated at about 900 cubic feet, was placed directly south of the first pile and east of the 14,000 cubic feet of cooling tower silt also stored between the towers. This second pile was initially designated as coming from repaving activities. (Note that these designations may have been reversed with the repaving activity generating the 3100 cubic feet of material and the security fence excavation generating the 900 cubic feet of material.)

DISCUSSION

As we discussed last week, I'm forwarding copies of the laboratory analyses (Reference 1) for two piles of dirt currently located immediately east of the cooling tower silt pile between the plant's cooling towers.

The 3100 cubic foot pile was sampled by collecting 30 composite grab samples (G22686 through G22715) taken at equal distances long its 82 foot length (the pile is about 15.5 feet wide and 4 feet high at its peak). Each composite sample is consist of 3 grab aliquots taken on the left, top, and right side of the pile at each reference distance starting from the pile's north end. The comment field on each analysis report indicates a sample location relative to the north end of the pile. As an example, sample G22691 has a comment of 6-15.0, indicating the 6th sample taken at a distance of 15.0 feet from the north end of the pile.

Tables 1 and 2 summarize the results of the gamma isotopic analyses for Cs-137 and Co-60. None of the 30 composite samples indicated any positive Cesium or Cobalt, or any other plant related radionuclide. As a consequence, this pile appears to be free from any radioactivity contamination.

The 900 cubic foot pile was sampled in the same manner as above, with a total of 20 composite samples collected (G22716 through G22735). Table 3 shows that both Cs-137 and Co-60 were detected in all or most of the grab samples, indicating that positive plant related radioactivity exists in the soil, and that 10CFR20.2002 approval for disposal will be needed.

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APPENDIX J (Continued)

REG-115/96, July 15, 1996
Page 2

Please call at your convenience to discuss the next steps necessary to handle the 900 cubic foot pile.



Mark S. Strum
Lead Radiological Engineer
Environmental Engineering Depart.

Attachments

c

R. Marcello
P. Littlefield
M. Marian

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Appendix J Original
Off-Site Dose Calculation Manual
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APPENDIX J (Continued)

Table 1
Vermont Yankee Soil Analysis
E-Lab soil data for 3100 ft³ soil marked as
"Fence" line construction material
Cs-137

Sample I.D.	Conc.	(pCi/kg dry) + - sigma	3 sigma	MDC
1-1.4	9	21	63	78
2-4.1	33	22	66	74
3-6.8	33	26	78	89
4-9.6	-9	23	69	91
5-12.3	20	21	63	73
6-16.0	16	19	57	69
7-17.8	-41	26	78	110
8-20.5	-6	21	63	86
9-23.2	-6	33	99	120
10-26.0	17	23	69	80
11-28.7	11	16	48	60
12-31.4	-14	21	63	82
13-34.2	-12	21	63	83
14-36.9	22	18	54	59
15-39.6	-12	22	66	87
16-42.4	15	18	54	64
17-45.1	12	28	84	99
18-47.8	17	21	63	77
19-50.6	-33	20	60	84
20-53.3	-42	23	69	94
21-56.0	26	36	108	130
22-58.8	26	27	81	93
23-61.5	30	25	75	86
24-64.2	18	25	75	89
25-67.0	28	26	78	89
26-69.7	46	23	69	73
27-72.4	33	23	69	76
28-75.2	48	23	69	70
29-77.9	37	26	78	84
30-80.6	9	29	87	100
Average:	11	23	70	86
Max. value:	48	36	108	130
Min. value:	-42	16	48	59

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APPENDIX J (Continued)

Table 2
Vermont Yankee Soil Analysis
E-Lab soil data for 3100 ft³ soil marked as
"Fence" line construction material

Sample I.D.	Conc.	Co-60 (pCi/kg dry)		MDC
		1 - sigma	3 sigma	
1-1.4	10	18	54	70
2-4.1	-12	23	69	100
3-8.8	-11	21	63	97
4-9.6	39	30	90	100
5-12.3	1	21	63	86
6-15.0	32	18	54	57
7-17.8	-50	25	75	120
8-20.5	16	15	45	56
9-23.2	43	28	84	91
10-26.0	-17	23	69	97
11-28.7	-13	21	63	93
12-31.4	-4	20	60	84
13-34.2	-2	17	51	75
14-36.9	4	25	75	98
15-39.6	-15	24	72	100
16-42.4	-39	28	84	120
17-45.1	-17	23	69	100
18-47.8	-14	29	87	130
19-50.6	-9	18	54	82
20-53.3	37	22	66	72
21-56.0	17	27	81	140
22-58.8	5	30	90	120
23-61.5	7	28	84	110
24-64.2	-10	19	57	90
25-67.0	27	27	81	98
26-69.7	1	29	87	120
27-72.4	16	21	63	79
28-75.2	6	25	75	98
29-77.9	-8	32	96	130
30-80.6	-27	23	69	110
Average:	0	24	71	97
Max. value:	43	32	96	140
Min. value:	-50	15	45	56

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APPENDIX J (Continued)

Table 3
Vermont Yankee Soil Analysis

E-Lab Analysis of soil from 900 ft3 pile initial marked "repaving dirt"

sample I.D.	Cs-137 (pCi/kg dry)				Co-60 (pCi/kg dry)			
	Conc.	+ - sigma	3 sigma	Positive Act. > 3 sigma	Conc.	+ - sigma	3 sigma	Positive Act. > 3 sigma
1-1.1	234	53	159	positive	49	38	114	ND
2-3.3	522	57	171	positive	143	29	87	positive
3-5.5	337	43	129	positive	37	21	63	ND
4-7.7	291	29	87	positive	111	17	51	positive
5-9.9	348	51	153	positive	47	29	87	ND
6-12.1	135	26	78	positive	73	23	69	positive
7-14.3	107	24	72	positive	82	15	45	positive
8-16.5	222	44	132	positive	140	28	84	positive
9-18.7	180	37	111	positive	92	21	63	positive
10-20.9	269	51	153	positive	118	31	93	positive
11-2.0	810	51	153	positive	114	21	63	positive
12-4.2	378	38	114	positive	108	21	63	positive
13-6.4	810	66	198	positive	124	27	81	positive
14-8.6	378	24	72	positive	62	13	39	positive
15-10.8	331	22	66	positive	87	12	36	positive
16-13	253	33	99	positive	5	22	66	ND
17-15.2	150	12	36	positive	57	9	27	positive
18-17.4	247	30	90	positive	105	17	51	positive
19-19.6	326	55	165	positive	54	40	120	ND
20-21.8	235	32	96	positive	100	23	69	positive
Average:	328	39			85	23		
Max. value:	810	66			143	40		
Min. value:	107	12			5	9		

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APPENDIX J (Continued)

Appendix G

Record for Last Spreadings (2003) on South Disposal Field

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Total Recorded Spreading Data for 2003

Serial #	Spreading Date	Media Type	Mn-54 (uCi/acre)	Co-60 (uCi/acre)	Zn-65 (uCi/acre)	Cs-134 (uCi/acre)	Cs-137 (uCi/acre)	Ce-141 (uCi/acre)	Ce-144 (uCi/acre)
2003-01	7-1-3	Septic	--	1.03	--	--	--	--	--
2003-02	10-25-03	Septic	--	0.12	--	--	--	--	--
2003-03	11-4-03	Sand/soil	--	--	--	--	1.34	--	--
2003-04	11-4-03	CT Silt	--	0.256	--	--	--	--	--
		Total	--	1.41	--	--	1.34	--	--

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APPENDIX J (Continued)



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

7/25/05 D

July 19, 2005

NW 05-090

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: SAFETY EVALUATION OF REQUEST TO AMEND PREVIOUS APPROVALS
GRANTED PURSUANT TO 10 CFR 20.2002 - VERMONT YANKEE NUCLEAR
POWER STATION (TAC NO. MC5104)

Dear Mr. Kansler:

By letter dated October 4, 2004, as supplemented on January 17, 2005, Entergy Nuclear Operations, Inc. (Entergy) submitted a request to the Nuclear Regulatory Commission (NRC) to modify previous approvals granted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 20.2002 (previously 10 CFR 20.302(a)), for on-site disposal of slightly contaminated material at Vermont Yankee Nuclear Power Station. Specifically, Entergy requested an increase of the current approved annual volume limit of 28.3 cubic meters of soil/sand to a new annual volume limit of 150 cubic meters of soil/sand. In addition, Entergy has requested a one-time approval for on-site disposal of the current backlog inventory of approximately 528 cubic meters of soil/sand.

The NRC staff has completed its review of the request and has determined that the proposed changes are acceptable as documented in the enclosed Safety Evaluation.

Pursuant to the provisions of 10 CFR Part 51, the NRC has published an Environmental Assessment and Finding of No Significant Impact in the Federal Register on July 19, 2005 (70 FR 41440).

Sincerely,

A handwritten signature in dark ink, appearing to read "R B Ennis".

Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: As stated

cc w/encl: See next page

Handwritten initials, possibly "J-51", in dark ink.

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APPENDIX J (Continued)

Vermont Yankee Nuclear Power Station

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APPENDIX J (Continued)

Vermont Yankee Nuclear Power Station

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APPENDIX J (Continued)



**UNITED STATES
NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ENTERGY NUCLEAR OPERATIONS, INC.

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated October 4, 2004, as supplemented on January 17, 2005, Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request to the Nuclear Regulatory Commission (NRC) to modify previous approvals granted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 20.2002 (previously 10 CFR 20.302(a)), for on-site disposal of slightly contaminated material at Vermont Yankee Nuclear Power Station (VYNPS). Specifically, Entergy requested an increase of the current approved annual volume limit of 28.3 cubic meters of soil/sand to a new annual volume limit of 150 cubic meters of soil/sand. In addition, Entergy has requested a one-time approval for on-site disposal of the current backlog inventory of approximately 528 cubic meters of soil/sand.

2.0 REGULATORY EVALUATION

As described in 10 CFR 20.2002, "Method for obtaining approval of proposed disposal procedures," licensees are required to obtain NRC approval of proposed procedures, not otherwise authorized in the regulations, to dispose of licensed material generated in the licensee's activities.

Previous NRC approval for VYNPS on-site disposal of various slightly contaminated waste materials is documented in letters dated August 30, 1989, March 4, 1996, June 18, 1997, June 15, 2000, and June 26, 2001. Based on these previous approvals, the licensee is currently authorized to dispose, in designated on-site areas, the following materials: (1) septic waste; (2) cooling tower silt; (3) soil/sand generated from the annual winter spreading on roads and walkways; and (4) soil resulting from on-site construction-related activities. Disposal of septic waste and cooling tower silt material is not limited by an annual volume, but by a total dose impact related to the radiological content of the material and the concentration of radioactivity contained within it. The combination of the soil/sand generated from the annual winter spreading on roads and walkways and the soil resulting from on-site construction-related activities is currently subject to an annual volume limit of 28.3 cubic meters. The licensee's application dated October 4, 2004, proposed to increase this annual volume limit for the same materials (i.e., soil/sand) to 150 cubic meters. In addition, the application proposed a one-time disposal of the current backlog inventory of approximately 528 cubic meters of soil/sand.

The current restrictions on the annual volume of slightly contaminated soil/sand that can be disposed on-site coupled with several plant facility projects in recent years, has resulted in the

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APPENDIX J (Continued)

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accumulation of a backlog of low-level contaminated earthen material that is awaiting disposal by land spreading on previously-approved on-site disposal areas. The current approved annual volume limit of 28.3 cubic meters of soil/sand for disposal was based on licensee estimates of soil and sand collected from road and walkway sweepings inside the Protected Area following each year's winter cleanup (i.e., current annual limit does not account for future site excavation and construction activities).

3.0 TECHNICAL EVALUATION

The licensee proposes to dispose of the current backlog inventory of soil/sand and the future soil/sand material using a land spreading technique consistent with the current commitments for on-site disposal of slightly contaminated material previously approved by the NRC. The licensee will continue to use designated areas of its property approved for this waste material. Determination of the radiological dose impact of the new material has been made based on the same dose assessment models and pathway assumptions used in the previously-approved applications.

The licensee will procedurally control and maintain records of all disposals. The following information will be recorded:

1. the radionuclide concentrations detected in the material;
2. the total volume of material disposed;
3. the total radioactivity in the disposal operation as well as the total radioactivity accumulated on each disposal plot at the time of spreading;
4. the plot of land on which the material was applied; and
5. dose calculations or maximum allowable accumulated activity determinations required to demonstrate that the dose values have not been exceeded.

The licensee's application states that the existing NRC-approved bounding dose conditions for the proposed on-site disposals will continue to be applied without change. The bounding dose conditions for the on-site disposals are as follows:

1. the annual dose to the whole body or any organ of a hypothetical maximally exposed individual will be less than 1.0 millirem (mrem) (during the period the licensee has active control over the disposal sites, i.e., during the current operating license period);
2. annual doses to the whole body and any organ of an inadvertent intruder from the probable pathways of exposure will be less than 5 mrem (following the period the licensee has active control over the disposal sites); and
3. disposal operations will be at an approved on-site location.

To ensure that the addition of new waste material will not exceed the bounding dose conditions for each new spreading operation, the licensee's total radioactivity and dose calculation will include all past disposals of septic waste, cooling tower silt, soil and soil/sand material on the

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designated disposal plots. In addition, concentration limits will be applied to the disposed material to restrict the placement of small volumes of material that may have relatively high radioactivity concentrations.

VYNPS is currently authorized to dispose of licensed material, pursuant to 10 CFR 20.2002, in two designated locations both within the site boundary security fence. The South field (approximately 1.9 acres in size) is centered approximately 1500 feet south of the reactor building. The North field (approximately 10 acres in size) is centered approximately 2000 feet northwest of the reactor building. The South field has been the only field utilized for disposal of licensed material to date. The licensee's application states that it is anticipated that future disposal operations will also use the South field since sufficient margin in comparison to the approved dose limit criteria still exists for anticipated waste disposal of the existing backlog of material now in storage, plus all expected future disposals of material, assuming the same observed generation rates.

The licensee performed an evaluation of the radiological impact of all past, accumulated storage inventory, and projected future waste spreading operations on the South field. The licensee assessed the dose that may be received by the maximally exposed individual during the period of plant control over the property, and to an inadvertent intruder after plant access control ends using the same pathway modeling, assumptions, and dose calculation methods that were previously approved by the NRC for the waste material. The dose models are based on the guidance in Regulatory Guide 1.109, Revision 1.

Table 26 in the attachment to the licensee's October 4, 2004, application provides the calculated maximum organ and whole-body doses, at the end of the current plant license period, based on the combination of all past disposal of waste materials, disposal of the current backlog inventory of waste material and projected annual disposal of waste materials. The results are summarized as follows:

	Individual Organ Dose (mrem/year)	Individual Whole Body Dose (mrem/year)	Intruder Organ Dose (mrem/year)	Intruder Whole Body Dose (mrem/year)
Total Dose	0.253	0.074	1.05	0.468
Dose Limit	1	1	5	5
% of Dose Limit	25.3%	7.4%	21.1%	9.4%

Based on the above calculated dose rates, the NRC staff concludes that the proposed increase of the current approved annual volume limit of 28.3 cubic meters of soil/sand to a new annual volume limit of 150 cubic meters of soil/sand and a one-time approval for on-site disposal of the current backlog inventory of approximately 528 cubic meters of soil/sand would result in dose rates within the bounding dose conditions for on-site disposal previously approved by the NRC.

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APPENDIX J (Continued)

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4.0 CONCLUSION

The NRC staff finds the licensee's request for disposal of a new annual volume limit of 150 cubic meters of soil/sand and a one-time approval for on-site disposal of the current backlog inventory of approximately 528 cubic meters of soil/sand, pursuant to 10 CFR 20.2002, in the same manner, location, and within the bounding dose conditions as previously approved by the NRC, to be acceptable.

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