EFFLUENT AND WASTE DISPOSAL

SEMIANNUAL REPORT

FOR

THIRD AND FOURTH QUARTERS 1992

Yankee Atomic Electric Company
Rowe, Massachusetts

R11\8

9303030244 930301
PDR ADOCK 05000029
R PDR

NOTES:

1. Yankee Nuclear Power Station's last day at any power level was October 1, 1991. The facility is permanently shut down for future decommissioning. Due to ceased operations, short-lived nuclides have been deleted from the gaseous and liquid effluent tables. Their activity concentrations in the fuel core inventory have decayed to zero values.

2. License Amendment No. 146 allowed the transfer of several technical specifications from the technical specification document into the Process Control Program and Off-Site Dose Calculation Manual. The transferred technical specifications are now designated as controls, and references to these technical specifications have been revised accordingly throughout the Effluent and Waste Disposal Semiannual Report. Further details are provided in Appendices F and G. Please note that the transfers did not involve any modifications to the technical specifications' contents or programmatic controls.

LIST OF TABLES AND APPENDICES

TABLE 1A

Yankee Atomic Electric Company, Rowe, Massachusetts
Effluent and Waste Disposal Semiannual Report
Third and Fourth Quarters 1992
Gaseous Effluents - Summation of All Releases

		Unit	Quarter 3	Quarter 4	Est. Total Error, %
A.	Fission and Activation Gases				
	1. Total Release	Ci	0.00E+00	0.00E+00	±2.50E+01
	2. Average Release Rate for Period	uCi/sec	0.00E+00	0.00E+00	
	3. Percent of Control Limit(1)(2)	%	0.00E+00	0.00E+00	
B.	Iodines				
	1. Total Iodine-131	Ci	<4.73E-07	0.00E+00	±2.50E+01
	2. Average Release Rate for Period	uCi/sec	<6.02E-08	0.00E+00	
	3. Percent of Control Limit(2)(3)	%	5.73E-02	3.20E-02	
C.	Particulates				
	1. Particulates with T-1/2 > 8 days	Ci	1.15E-06	4.00E-07	±3.00E+01
	2. Average Release Rate for Period	uCi/sec	1.46E-07	5.09E-08	
	3. Percent of Control Limit	%	(4)	(4)	
	4. Gross Alpha Radioactivity	Ci	<3.26E-08	<2.39E-08	
D.	Tritium				
	1. Total Release	Ci	5.91E-01	3.20E-01	±3.00E+01
	2. Average Release Rate for Period	uCi/sec	7.52E-02	4.07E-02	
	3. Percent of Control Limit	%	(4)	(4)	

(1) Control 3.4.a (previously Technical Specification 3.11.2.2.a) for gamma-air dose. Percent values for Control 3.4.b (previously Technical Specification 3.11.2.2.b) for beta-air dose are approximately the same.

(2) The percent of control limit is based on the conservative plant quarterly dose determinations. These values will be updated pending issuance of the supplemental report which will include the annual dose summary.

(3) Control 3.5.a (previously Technical Specification 3.11.2.3.a) for dose from Iodine-131, tritium and radionuclides in particulate form. Please note that Iodine-131 did not contribute to the percent of control limit value for this reporting period.

(4) Per Control 3.5, percent dose contribution from tritium and particulates is above in Part B.

TABLE 1B

Yankee Atomic Electric Company, Rowe, Massachusetts
Effluent and Waste Disposal Semiannual Report
Third and Fourth Quarters 1992
Gaseous Effluents - Elevated Releases

	Nuclides Released	Unit	Continuous Mode		Batch Mode*	
			Quarter 3	Quarter 4	Quarter 3	Quarter 4
1.	Fission Gases					
	Krypton-85	Ci	0.00E+00	0.00E+00		
	Xenon-131m	Ci	0.00E+00	0.00E+00		
	Total for Period	Ci	0.00E+00	0.00E+00		
2.	Iodines					
	Iodine-131	Ci	<4.73E-07	0.00E+00		
	Total for Period	Ci	<4.73E-07	0.00E+00		
3.	Particulates					
	Strontium-89	Ci	<4.16E-07	<4.16E-07		
	Strontium-90	Ci	<2.02E-07	<3.01E-07		
	Cesium-134	Ci	<2.64E-07	<2.12E-07		
	Cesium-137	Ci	2.23E-08	3.67E-08		
	Barium-Lanthanum-140	Ci	<8.76E-07	<7.03E-07		
	Zinc-65	Ci	<6.57E-07	<5.34E-07		
	Cobalt-58	Ci	<2.87E-07	<2.41E-07		
	Cobalt-60	Ci	1.13E-06	3.63E-07		
	Iron-59	Ci	<5.64E-07	<4.85E-07		
	Chromium-51	Ci	<1.79E-06	<1.42E-06		
	Zirconium-Niobium-95	Ci	<4.95E-07	<4.09E-07		
	Cerium-141	Ci	<2.36E-07	<1.73E-07		
	Cerium-144	Ci	<1.01E-06	<7.58E-07		
	Antimony-124	Ci	<2.56E-07	<2.08E-07		
	Manganese-54	Ci	<3.13E-07	<2.44E-07		
	Silver-110m	Ci	<2.55E-07	<2.05E-07		
	Molybdenum-99	Ci	<1.97E-06	<1.54E-06		
	Ruthenium-103	Ci	<2.27E-07	<1.80E-07		
	Total for Period	Ci	1.15E-06	4.00E-07		

* There were no batch mode gaseous releases during the third and fourth quarters of 1992.

TABLE 1C

Yankee Atomic Electric Company, Rowe, Massachusetts
Effluent and Waste Disposal Semiannual Report
Third and Fourth Quarters 1992
Gaseous Effluents - Ground Level Releases

There were no routine measured ground level continuous or batch mode gaseous releases during the third or fourth quarters of 1992.

TABLE 2A

Yankee Atomic Electric Company, Rowe, Massachusetts
Effluent and Waste Disposal Semiannual Report
Third and Fourth Quarters 1992
Liquid Effluents - Summation of All Releases

		Unit	Quarter 3	Quarter 4	Est. Total Error, %
A.	Fission and Activation Products				
	1. Total Release (not including tritium, gases, alpha)	Ci	3.99E-04	2.78E-04	±2.00E+01
	2. Average Diluted Concentration During Period	uCi/ml	2.63E-09	1.93E-09	
	3. Percent of Applicable Limit(1)	%	4.00E-03	3.03E-03	
B.	Tritium				
	1. Total Release	Ci	2.47E+00	5.02E-01	±1.00E+01
	2. Average Diluted Concentration During Period	uCi/ml	1.63E-05	3.49E-06	
	3. Percent of Applicable Limit(1)	%	5.42E-01	1.16E-01	
C.	Dissolved and Entrained Gases				
	1. Total Release	Ci	<5.06E-03	<8.72E-04	±2.00E+01
	2. Average Diluted Concentration During Period	uCi/ml	<3.33E-08	<6.06E-09	
	3. Percent of Applicable Limit(2)	%	<1.66E-02	<3.03E-03	
D.	Gross Alpha Radioactivity				
	1. Total Release	Ci	<1.97E-07	<1.55E-07	±3.50E+01
E.	Volume of Waste Released (prior to dilution)	liters	5.58E+05	3.23E+05	±3.00E+01
F.	Volume of Dilution Water Used During Period	liters	1.52E+08	1.44E+08	±1.50E+01

(1) Concentration limits specified in 10CFR20, Appendix B, Table II, Column 2 (Control 2.1, previously Technical Specification 3.11.1.1). The percent of applicable limit reported is based on the average diluted concentration during the period. At no time did any release exceed the concentration limit.

(2) Concentration limits for dissolved and entrained noble gases is 2.00E-04 uCi/ml (Control 2.1). The percent of applicable limit reported is based on the average diluted concentration during the period. At no time did any release exceed the concentration limit.

TABLE 2B

Yankee Atomic Electric Company, Rowe, Massachusetts
Effluent and Waste Disposal Semiannual Report
Third and Fourth Quarters 1992
Liquid Effluents - Routine Releases

Nuclides Released	Unit	Continuous Mode Quarter 3	Continuous Mode Quarter 4	Batch Mode Quarter 3	Batch Mode Quarter 4
Strontium-89	Ci	<1.14E-06	<1.84E-06	<2.23E-05	<8.14E-06
Strontium-90	Ci	<5.56E-07	<5.30E-07	<6.36E-06	<3.73E-06
Cesium-134	Ci	5.58E-07	3.75E-07	1.16E-07	2.87E-06
Cesium-137	Ci	1.77E-06	1.89E-06	8.35E-06	6.57E-05
Cobalt-58	Ci	<1.56E-07	<3.45E-07	<2.01E-05	<2.51E-06
Cobalt-60	Ci	<2.04E-07	6.92E-08	1.56E-04	9.46E-06
Iron-59	Ci	<3.27E-07	<7.55E-07	<3.63E-05	<4.92E-06
Zinc-65	Ci	<3.50E-07	<8.02E-07	<4.88E-05	<5.36E-06
Manganese-54	Ci	<1.81E-07	<4.23E-07	<2.02E-05	<2.57E-06
Chromium-51	Ci	<1.31E-06	<2.71E-06	<1.20E-04	<2.07E-05
Zirconium-Niobium-95	Ci	<2.73E-07	<5.87E-07	<3.54E-05	<4.68E-06
Molybdenum-99	Ci	<1.14E-06	<2.49E-06	<1.38E-04	<1.74E-05
Technetium-99m	Ci	<1.37E-07	<3.05E-07	<1.27E-05	<2.21E-06
Barium-Lanthanum-140	Ci	<5.74E-07	<1.21E-06	<5.57E-05	<9.82E-06
Cerium-141	Ci	<2.19E-07	<4.70E-07	<2.02E-05	<3.47E-06
Ruthenium-103	Ci	<1.70E-07	<3.39E-07	<1.44E-05	<2.68E-06
Cerium-144	Ci	<9.83E-07	<2.06E-06	<9.16E-05	<1.59E-05
Selenium-75	Ci	<1.79E-07	<3.70E-07	<1.63E-05	<2.91E-06
Silver-110m	Ci	<1.73E-07	<3.71E-07	<1.78E-05	<2.77E-06
Antimony-124	Ci	<1.93E-07	<4.23E-07	<1.61E-05	<3.00E-06
Carbon-14	Ci	-	-	2.33E-04	1.98E-04
Iron-55	Ci	<5.28E-06	<7.46E-06	<9.54E-05	<4.88E-05
Antimony-125	Ci	<4.77E-07	<9.94E-07	<4.50E-05	<8.90E-06
Unidentified	Ci	-	-	-	-
Total for Period (above)	Ci	2.33E-06	2.33E-06	3.97E-04	2.76E-04
Krypton-85	Ci	<4.95E-05	<1.07E-04	<5.01E-03	<7.65E-04

TABLE 3

Yankee Atomic Electric Company, Rowe, Massachusetts
Effluent and Waste Disposal Semiannual Report
Third and Fourth Quarters 1992
Solid Waste and Irradiated Fuel Shipments

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

1.	Type of Waste	Unit	6-Month Period	Est. Total Error, %
a.	Evaporator bottoms*, Class A waste, LSA containers, 55-gallon drums	m^3 Ci (Est.)	5.60E+01 3.57E+00	±3.00E+01
b.	Dry compressible waste, contaminated equipment, etc., Class A waste, LSA, 105-ft^3 boxes	m^3 Ci (Est.)	6.87E+01 2.15E+00	±5.00E+01
c.	Dewatered spent resin, Class C waste, poly HIC, 120.3 ft^3	m^3 Ci (Meas.)	3.40E+00 5.12E+01	±5.00E+01
d.	Irradiated components, control rods, Class C waste, 57.4-ft^3 steel liner	m^3 Ci (Est.)	3.25E+00 3.23E+04	±5.00E+01

2. Estimate of Major Nuclide Composition (by type of waste)**

a.	Iron-55	3.84E+01	b.	Iron-55	4.96E+01	c.	Cesium-137	3.89E+01	d.	Silver-110m	3.80E+01
	Tritium	3.75E+01		Cesium-137	2.41E+01		Iron-55	1.74E+01		Iron-55	2.57E+01
	Cesium-137	8.56E+00		Cesium-134	1.48E+01		Cesium-134	1.49E+01		Cobalt-60	2.14E+01
	Cobalt-60	5.26E+00		Cobalt-60	5.60E+00		Cobalt-60	1.07E+01		Cadmium-109	7.86E+00
	Cesium-134	5.04E+00		Nickel-63	3.91E+00		Manganese-54	9.49E+00		Antimony-125	2.02E+00
	Nickel-63	4.45E+00		Carbon-14	4.97E-01		Nickel-63	5.21E+00		Nickel-63	2.49E+00
	Carbon-14	7.19E-01					Strontium-90	1.71E+00		Manganese-54	1.51E+00
	Strontium-90	1.50E-02								Niobium-95	5.98E-01

3. Solid Waste Disposition

Number of Shipments	Mode of Transportation	Destination
10	Truck	Barnwell, South Carolina

B. Irradiated Fuel Shipments (Disposition): None

* Solidification agent is cement.

** Excluding radionuclides with half-lives less than 12.8 days.

APPENDIX A

Radioactive Liquid Effluent Monitoring Instrumentation

Requirement: Radioactive liquid effluent monitoring instrumentation channels are required to be operable in accordance with Control 5.1 (previously Technical Specification 3.3.3.6). With less than the minimum number of channels operable and reasonable efforts to return the instrument(s) to operable status within 30 days being unsuccessful, Control 5.1 requires an explanation for the delay in correcting the inoperability in the next Semiannual Effluent Release Report.

Responses: Response provided in the Effluent and Waste Disposal Semiannual Report for First and Second Quarters 1992 - On May 21, 1992, the steam generator effluent monitor was declared inoperable. The last release from this pathway occurred on March 4, 1992. It is not anticipated that this flow path be used in the near future. If required, the monitor will be re-established in its operable condition prior to any release from this pathway.

Response for this Reporting Period - On November 20, 1992, the steam generator effluent monitor was declared operable and remained in this status during the fourth quarter.

APPENDIX B

Radioactive Gaseous Effluent Monitoring Instrumentation

Requirement: Radioactive gaseous effluent monitoring instrumentation channels are required to be operable in accordance with Control 5.2 (previously Technical Specification 3.3.3.7). With less than the minimum number of channels operable and reasonable efforts to return the instrument(s) to operable status within 30 days being unsuccessful, Control 5.2 requires an explanation for the delay in correcting the inoperability in the next Semiannual Effluent Release Report.

Response: Since the requirements of Control 5.2 governing the operability of radioactive gaseous effluent monitoring instrumentation were met for this reporting period, no response is required.

APPENDIX C

Liquid Holdup Tanks

Requirement: Technical Specification 3.11.1.4 limits the quantity of radioactive material contained in any outside temporary tank. With the quantity of radioactive material in any outside temporary tank exceeding the limits of Technical Specification 3.11.1.4, a description of the events leading to this condition is required in the next Semiannual Effluent Release Report.

Response: The limits of Technical Specification 3.11.1.4 were not exceeded during this reporting period.

APPENDIX D

Radiological Environmental Monitoring Program

Requirement: The Radiological Environmental Monitoring Program is conducted in accordance with Control 4.1 (previously Technical Specification 3/4.12.1). With milk or fresh leafy vegetation samples no longer available from one or more of the required sample locations, Control 4.1.c (previously Technical Specification 3.12.1.c) requires the identification of the new location(s) for obtaining replacement sample(s) in the next Semiannual Effluent Release Report and inclusion of revised Off-Site Dose Calculation Manual figure(s) and table(s) reflecting the new location(s).

Response: No milk or fresh leafy vegetation samples were found to be unavailable during this reporting period.

APPENDIX E

Land Use Census

Requirement: A land use census is conducted in accordance with Control 4.2 (previously Technical Specification 3.12.2). With a land use census identifying a location(s) which yields at least a 20 percent greater dose or dose commitment than the values currently being calculated in Control 3.5 (previously Technical Specification 4.11.2.3). Control 4.2.a (previously Technical Specification 3.12.2.a) requires the identification of the new location(s) in the next Semiannual Effluent Release Report.

Response: The land use census for this reporting period did not identify any locations yielding at least a 20 percent greater dose or dose commitment than the values currently being calculated in Control 3.5.

Requirement: With a land use census identifying a location(s) which yields 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 4.1 (previously Technical Specification 3.12.1). Control 4.2.b (previously Technical Specification 3.12.2.b) requires that the new location(s) be added to the Radiological Environmental Monitoring Program if permission from the owner to collect samples can be obtained and sufficient sample volume is available. The identification of the new location(s) is required in the next Semiannual Effluent Release Report.

Response: No changes were made in the Radiological Environmental Monitoring Program as a result of the 1992 land use census.

APPENDIX F

Process Control Program (PCP)

Requirement: PCP Control 2.0 (previously Technical Specification 6.14.1) requires that licensee initiated changes to the PCP be submitted to the Commission in the Semiannual Effluent Release Report for the period in which the change(s) was made.

Response: License Amendment No. 146 allowed the following changes to be completed in the PCP document:

1. Title page revised to include signature table.

2. Revision Record, List of Affected Pages, and List of Controls pages added to document.

3. The following changes were implemented in accordance with NRC Generic Letter 89-01, which provided guidance on the relocation of selected Radiological Effluent Technical Specifications to the PCP document:

   a. Section 1.0, Definitions of Process Control Program and Solidification - Solidified added to include definitions pertinent to the relocated Technical Specifications;

   b. Technical Specification 3/4.11.3, now referred to as PCP Control 1.0, relocated to Section 1.0;

   c. Corresponding Technical Specification 3/4.11.3 Bases relocated to Section 1.0;

   d. Sections 4.2 and 4.3 created to contain reporting details for the Semiannual Effluent Release Reports for solid waste (PCP Control 2.0) and Major Changes to the Solid Radioactive Waste Treatment System (PCP Control 3.0), respectively.

4. All pages renumbered.

APPENDIX G

Off-Site Dose Calculation Manual (ODCM)

Requirement: Control 7.2 (previously Technical Specification 6.15.2) requires that licensee initiated changes to the ODCM be submitted to the Commission in the Semiannual Effluent Release Report for the period in which the change(s) was made effective.

Response: License Amendment No. 146 allowed the following changes to be implemented into the ODCM in accordance with NRC Generic Letter 89-01. NRC Generic Letter 89-01 provided guidance on the relocation of the Radiological Effluent Technical Specifications to the ODCM.

1. Addition of List of Controls page (succeeds Table of Contents);

2. Section 1.0, Introduction updated to reflect the change in scope of the ODCM;

3. Technical Specifications 3/4.0.1, 3/4.0.2, 3/4.0.3, and 3/4.0.4 listed in Section 1.2, Applicability of Controls and Surveillance Requirements (SR), and now referred to as Controls 1.1, 1.2, 1.3, and 1.4, respectively;

4. Table 1.6, Definition of Terms, modified to include definitions pertinent to the relocated Technical Specifications;

5. Tables 1.9, Operational Modes, and 1.10, Frequency Notations, added to Section 1.0;

6. Technical Specification 3/4.11.1.1, now referred to as Control 2.1, relocated to Section 2.0;

7. Technical Specifications 3/4.11.1.2, 3/4.11.4, 3/4.11.2.1, 3/4.11.2.2,and 3/4.11.2.3, now referred to as Controls 3.1, 3.2, 3.3, 3.4, and 3.5, respectively, relocated to Section 3.0;

8. Technical Specifications 3/4.12.1, 3/4.12.2, and 3/4.12.3, now referred to as Controls 4.1, 4.2, and 4.3, respectively, relocated to Section 4.0;

9. Technical Specification 3/4.3.3.6, now referred to as Control 5.1, relocated to Section 5.0;

10. Technical Specification 3/4.3.3.7, now referred to as Control 5.2, relocated to Section 5.0 (Existing requirements for explosive gas monitoring instrumentation retained in Technical Specification 3/4.3.3.7);

11. Technical Specifications 3/4.11.1.3 and 3/4.11.2.4, now referred to as Controls 6.1 and 6.2, respectively, relocated to Section 6.0;

12. Section 7.0 created to contain reporting details for the Annual Radiological Environmental Monitoring Operating (Control 7.1) and Semiannual Effluent Release Reports (Control 7.2), and Major Changes to the Liquid and Gaseous Radioactive Waste Treatment Systems (Control 7.3); and

13. Corresponding Technical Specification Bases relocated with Technical Specifications to become part of controls.

14. All pages renumbered.

All changes to the document were reviewed by the NRC as part of License Amendment 146.

In accordance with Technical Specification 6.15.c, the changes to the ODCM are submitted in the form of a complete, legible copy of the entire document concurrent with the Effluent and Waste Disposal Semiannual Report for the period of the report in which the changes were made. To review the revised ODCM, please refer to the following page.

YANKEE NUCLEAR POWER COMPANY

OFF-SITE DOSE CALCULATION MANUAL

YANKEE ATOMIC ELECTRIC COMPANY

NUCLEAR SERVICES DIVISION
580 MAIN STREET
BOLTON, MASSACHUSETTS 01740

	PREPARED BY/DATE	REVIEWED BY/DATE	PORC MEETING NO./DATE
REVISION 8	Christine L. Albright August 19, 1992	8-19-92	Meeting No. 92-72 August 19, 1992

R12\29

9209020311 21681

REVISION RECORD

Revision	Date	Description
0	12/01/82	Initial printing. Approved by PORC 11/29/82. Submitted for USNRC approval 12/03/82.
1	03/30/84	Change in environmental monitoring sampling locations based on 1983 land use census. Errors in Table 4.1 corrected. Maps revised.
2	07/30/85	Addition of Intercomparison Program description to Section 4.0. Reviewed by PORC 07/30/85.
3	03/19/86	Addition of a PVS I-131 inspection limit to demonstrate compliance with Technical Specification 3.11.2.1.b.
4	05/21/86	Change in milk sampling location. Samples no longer available at Station TM-11.
5	09/30/86	Change in food product sampling location based on 1986 land use census.
6	02/18/88	Change in liquid dose factors to reflect additional dose pathways. Change in gaseous dose factors to reflect five-year average meteorology. Change in gaseous dose rate factors to reflect a shielding factor of 1.0. Deletion of food product location TF-12 (samples no longer required after 10/31/86). Update of fence line location and several building names and locations in Figure 4-4.
7	05/21/90	Addition of Appendix A which documents the commitments for disposal of septage as provided in YNPS's Application For Approval to Routinely Dispose of Septage under 10CFR Part 20.302, and the NRC's acceptance as transmitted in their Safety Assessment, dated May 17, 1990.
8	08/19/92	a. The following changes were implemented in accordance with NRC Generic Letter 89-01, which provided guidance on the relocation of the Radiological Effluent Technical Specifications to the ODCM: 1. Addition of List of Controls Page (succeeds Table of Contents);

REVISION RECORD

Revision	Date	Description
8	08/19/92	2. Section 1.0, Introduction updated to reflect the change in scope of the ODCM;
		3. Technical Specifications 3/4.0.1, 3/4.0.2, 3/4.0.3, and 3/4.0.4 listed in Section 1.2, Applicability of Controls and Surveillance Requirements (SR), and now referred to as Controls 1.1, 1.2, 1.3, and 1.4, respectively;
		4. Table 1.6, Definition of Terms, modified to include definitions pertinent to the relocated Technical Specifications;
		5. Tables 1.9, OPERATIONAL MODES, and 1.10, FREQUENCY NOTATIONS, added to Section 1.0;
		6. Technical Specification 3/4.11.1.1, now referred to as Control 2.1, relocated to Section 2.0;
		7. Technical Specifications 3/4.11.1.2, 3/4.11.4, 3/4.11.2.1, 3/4.11.2.2, and 3/4.11.2.3, now referred to as Controls 3.1, 3.2, 3.3, 3.4, and 3.5, respectively, relocated to Section 3.0;
		8. Technical Specifications 3/4.12.1, 3/4.12.2, and 3/4.12.3, now referred to as Controls 4.1, 4.2, and 4.3, respectively, relocated to Section 4.0;
		9. Technical Specification 3/4.3.3.6, now referred to as Control 5.1, relocated to Section 5.0;
		10. Technical Specification 3/4.3.3.7, now referred to as Control 5.2, relocated to Section 5.0 (Existing requirements for explosive gas monitoring instrumentation retained in Technical Specification 3/4.3.3.7);

REVISION RECORD

Revision	Date	Description
B	08/19/92	11. Technical Specifications 3/4.11.1.3 and 3/4.11.2.4, now referred to as Controls 6.1 and 6.2, respectively, relocated to Section 6.0;
		12. Section 7.0 created to contain reporting details for the Annual Radiological Environmental Monitoring Operating (Control 7.1) and Semiannual Effluent Release Reports (Control 7.2), and Major Changes to the Liquid and GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEMS (Control 7.3); and
		13. Corresponding Technical Specification Bases relocated with Technical Specifications to become part of controls.
		b. All pages renumbered.

LIST OF AFFECTED PAGES

Changes, deletions, or additions in the most recent revision are indicated by a bar in the margin or by a dot near the page number if the entire page is affected.

Page	Revision
ii to xiv	8
1-1 to 1-21	8
2-1 to 2-10	8
3-1 to 3-55	8
4-1 to 4-16	8
5-1 to 5-25	8
6-1 to 6-10	8
7-1 to 7-5	8

DISCLAIMER OF RESPONSIBILITY

This document was prepared by Yankee Atomic Electric Company ("Yankee"). The use of information contained in this document by anyone other than Yankee, or the Organization for which the document was prepared under contract, is not authorized and, with respect to any unauthorized use, neither Yankee nor its officers, directors, agents, or employees assume any obligation, responsibility, or liability or make any warranty or representation as to the accuracy or completeness of the material contained in this document.

ABSTRACT

The Yankee Nuclear Power Station (YNPS) OFF-SITE DOSE CALCULATION MANUAL (ODCM) contains the methodology and parameters used in the calculation of off-site doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM also contains (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Specifications 6.9.5.a and 6.9.5.b. With initial approval by the U.S. Nuclear Regulatory Commission and the YNPS Plant Operation Review Committee (PORC) and approval of subsequent revisions by PORC (as required by the Technical Specifications) this manual is suitable to show compliance where referred to by the Technical Specifications and controls listed in this document.

TABLE OF CONTENTS

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

LIST OF CONTROLS

LIST OF TABLES

LIST OF TABLES
(Continued)

LIST OF FIGURES

1.0 INTRODUCTION

According to Definition of Terms (Table 1.6), the OFF-SITE DOSE CALCULATION MANUAL (ODCM) contains the methodology and parameters used in the calculation of off-site doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM also contains: (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Program required by Section 6.8 of the Technical Specification document and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Effluent Release Reports required by Controls 7.1 and 7.2, respectively. The ODCM forms the basis for plant procedures which document the off-site doses due to plant operation which are used to show compliance with the numerical guides for design controls of Section II, Appendix I, 10CFR Part 50.

The methods contained herein follow accepted NRC guidance, unless otherwise noted in the text. The basis for each method is sufficiently documented to allow regeneration of the methods by an experienced health physicist.

All changes to the ODCM shall be reviewed and approved by the Plant Operation Review Committee (PORC) in accordance with Technical Specification 6.15 prior to implementation. Changes made to the ODCM shall be submitted to the Commission for their information in the Semiannual Effluent Release Report for the period in which the change(s) was made effective.

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

This section summarizes the methods for the user. In addition, the applicability of controls and surveillance requirements are listed in this section. The first time user should read Chapters 2 through 5. The concentration and setpoint methods are documented in Table 1.1, as well as the Method I dose equations. Where more accurate dose calculations are needed, use the Method II for the appropriate dose as described in Sections 3.7 through 3.14 and 3.16. The dose factors used in the equations are in Tables 1.2, 1.7, and 1.8 and the regulatory limits are summarized in Table 1.3. The constants, variables, special definitions, OPERATIONAL MODES, and FREQUENCY

NOTATION used in the ODCM are in Tables 1.4, 1.5, 1.6, 1.9, and 1.10, respectively. Lastly, Figures 1-1 and 1-2 depict the Yankee plant site boundary line and liquid effluent discharge points, respectively.

1.2 Applicability of Controls and Surveillance Requirements (SR)

Control 1.1 The controls and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each control.

Control 1.2 Adherence to the requirements of the controls and/or associated ACTION within the specified time interval shall constitute compliance with the control. In the event that the control is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

Control 1.3 In the event that a control and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the control, the facility shall be placed in at least HOT STANDBY within 1 hour, and in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial discovery, or until the reactor is placed in a MODE in which the control is not applicable. Exceptions to these requirements shall be stated in the individual controls.

Control 1.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the conditions of the controls are met without reliance on provisions contained in the ACTION statements unless otherwise excepted. This provision shall not prevent passage through OPERATIONAL MODES as required to comply with ACTION statements.

SR 1.1 Surveillance requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual controls unless otherwise stated in an individual surveillance requirement.

SR 1.2 Each surveillance requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25 percent of the surveillance interval, and

b. A total maximum combined interval time for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

SR 1.3 Performance of a surveillance requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a control and associated ACTION statements unless otherwise required by the control.

SR 1.4 Entry into the OPERATIONAL MODE or other specified applicability condition shall not be made unless the surveillance requirement(s) associated with the control have been performed within the stated surveillance interval or as otherwise specified.

TABLE 1.1

Summary of Concentration and Setpoint Methods, and Method I Dose Equations for Normal Operations at the Yankee Plant

Equation No.	Maximum	Equation[a]
2-1	Unrestricted Area, Total Fraction of MPC in Liquids, Except Noble Gases	$F_1^{ENG} = \sum_i \frac{C_i^{ENG}}{MPC_i}$
2-2	Unrestricted Area, Concentration of Noble Gases in Liquids	$C^{NG} = \sum_i C_i^{NG}$
3-1	Total Body Dose Due to Liquids	D_{tb} (mrem) $= K \sum_i Q_i \, DFL_{itb}$
3-2	Maximum Organ Dose Due to Liquids	D_{organ} (mrem) $= K \sum_i Q_i \, DFL_{imo}$
3-3	Total Body Dose Rate Due to Noble Gases	$\dot{D}_{tb} \left(\frac{mrem}{yr}\right) = 7.83 \sum_i \dot{Q}_i \, DFB_i$
3-4	Skin Dose Rate Due to Noble Gases	$\dot{D}_{skin} \left(\frac{mrem}{yr}\right) = \sum_i \dot{Q}_i \, DF_i'$
3-5	Critical Organ Dose Rate Due to I-131, H-3, and Particulates with $T_{½} > 8$ Days	$\dot{D}_{co} \left(\frac{mrem}{yr}\right) = \sum_i \dot{Q}_i \, DFG_{ico}'$
3-6	Gamma Air Dose Due to Noble Gases	D_{air}^{γ} (mrad) $= 0.25 \sum_i Q_i \, DF_i^{\gamma}$
3-6.1	Gamma Air Dose Due to Ground Level Noble Gas Releases	D_{grd}^{γ} (mrad) $= (1.23 \times 10^{-4})$ (Q_{Xe-133} equivalent)
3-7	Beta Air Dose Due to Noble Gases	D_{air}^{β} (mrad) $= 0.76 \sum_i Q_i \, DF_i^{\beta}$
3-8	Critical Organ Dose Due to I-131, H-3, and Particulates with $T_{½} > 8$ Days	D_{co} (mrem) $= \sum_i Q_i \, DFG_{ico}$

TABLE 1.1
(Continued)

Summary of Concentration and Setpoint Methods, and Method I Dose Equations for Normal Operations at the Yankee Plant

Equation No.	Maximum	Equation(a)
3-9	Direct Dose	$D_d = (0.057 + \dot{E}_r)(T_e)(0.00087)$
5-1	Liquid Release Rate Reading	$R = \left(\frac{f_3}{f_1 + f_2}\right)(MPC_c)(S_\ell)$
5-3	Gaseous Release Rate Reading for Total Body Dose Limit	$R_{tb} = \frac{(S_g)\left(\sum_i f_i^{NG} s_i\right)(500)(60)}{(F)(7.83)\left(\sum_i f_i^{NG} DBF_i\right)}$
5-4	Gaseous Release Rate Reading for Skin Dose Limit	$R_{sk} = \frac{(S_g)\left(\sum_i f_i^{NG} s_i\right)(3000)(60)}{(F)\left(\sum_i f_i^{NG} DF_i'\right)}$

Note (a):

C_i = Concentration of radionuclide "i" in a mixture (µCi/ml).

$\dot{E}_r$ = Exposure rate at critical receptor from nonvapor container sources as measured or estimated for the period (µR/hr).

F = Primary vent stack flow rate (cc/min).

f_i^{NG} = Fraction of radionuclide "i" activity to total noble gas activity.

C_i^{ENG} = Concentration of radionuclide "i", except noble gases, at the point of discharge.

TABLE 1.1
(Continued)

Summary of Concentration and Setpoint Methods, and
Method I Dose Equations for Normal Operations at the Yankee Plant

C_i^{NG} = Concentration of radionuclide "i", except noble gases, at the point of discharge.

DF_i' = Skin dose factor for radionuclide "i".

DF_i^{γ} = Gamma dose factor to air for radionuclide "i".

DF_i^{β} = Beta dose factor to air for radionuclide "i".

DFB_i = Total body dose factor for radionuclide "i".

DFG_{ico} = Site-specific, critical organ dose factor for a gaseous release of radionuclide "i".

DFG_{ico}' = Site-specific, critical organ dose rate factor for a gaseous release of radionuclide "i".

DFL_{itb} = Site-specific, total body dose factor for a liquid release of radionuclide "i".

DFL_{imo} = Site-specific, maximum organ dose factor for a liquid release of radionuclide "i".

f_1 = Flow rate past the test tank monitor (gpm).

f_2 = Flow rate past the steam generator blowdown monitor (gpm).

f_3 = Flow rate at the point of discharge (gpm).

K = Deerfield River flow rate correction factor.

MPC_c = Composite MPC for the mix of radionuclides (μCi/ml).

$$= \frac{\sum_i C_i}{\sum_i \frac{C_i}{MPC_i}} \qquad \text{(Eq. 5-2)}$$

Q_i = Total release (Curies) for radionuclide "i".

$Q_{Xe\text{-}133\ equivalent}$ = Total release of noble gases expressed as Xe-133 equivalent.

TABLE 1.1
(Continued)

Summary of Concentration and Setpoint Methods, and
Method I Dose Equations for Normal Operations at the Yankee Plant

$\dot{Q}_i$ = Release rate (µCi/sec) for radionuclide "i".

S_g = Gaseous instrumentation response factor (cpm/(µCi/cc)).

S_ℓ = Liquid instrumentation response factor (cpm/(µCi/cc)).

s_i = Ratio of response from equal activities of radionuclide "i" to a reference radionuclide (i.e., Xe-133).

T_e = Length of exposure period (hours).

TABLE 1.2

Dose Factors Specific to the Yankee Plant for Noble Gas Releases

Radionuclide	Gamma Total Body Dose Factor $DFB_i \left(\frac{mrem\text{-}m^3}{pCi\text{-}yr}\right)$	Beta Skin Dose Factor $DFS_i \left(\frac{mrem\text{-}m^3}{pCi\text{-}yr}\right)$	Combined Skin Dose Factor $DF'_i \left(\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}\right)$	Beta Air Dose Factor $DF_i^{\beta} \left(\frac{mrad\text{-}m^3}{pCi\text{-}yr}\right)$	Gamma Air Dose Factor $DF_i^{\gamma} \left(\frac{mrad\text{-}m^3}{pCi\text{-}yr}\right)$
Ar-41	8.84×10^{-3}	2.69×10^{-3}	1.45×10^{-1}	3.28×10^{-3}	9.30×10^{-3}
Kr-83m	7.56×10^{-8}	-	1.68×10^{-4}	2.88×10^{-4}	1.93×10^{-5}
Kr-85m	1.17×10^{-3}	1.46×10^{-3}	4.56×10^{-2}	1.97×10^{-3}	1.23×10^{-3}
Kr-85	1.61×10^{-5}	1.34×10^{-3}	3.22×10^{-2}	1.95×10^{-3}	1.72×10^{-5}
Kr-87	5.92×10^{-3}	9.73×10^{-3}	2.86×10^{-1}	1.03×10^{-2}	6.17×10^{-3}
Kr-88	1.47×10^{-2}	2.37×10^{-3}	1.89×10^{-1}	2.93×10^{-3}	1.52×10^{-2}
Kr-89	1.66×10^{-2}	1.01×10^{-2}	3.92×10^{-1}	1.06×10^{-2}	1.73×10^{-2}
Kr-90	1.56×10^{-2}	7.29×10^{-3}	3.16×10^{-1}	7.83×10^{-3}	1.63×10^{-2}
Xe-131m	9.15×10^{-5}	4.76×10^{-4}	1.27×10^{-2}	1.11×10^{-3}	1.56×10^{-4}
Xe-133m	2.51×10^{-4}	9.94×10^{-4}	2.66×10^{-2}	1.48×10^{-3}	3.27×10^{-4}
Xe-133	2.94×10^{-4}	3.06×10^{-4}	1.04×10^{-2}	1.05×10^{-3}	3.53×10^{-4}
Xe-135m	3.12×10^{-3}	7.11×10^{-4}	4.62×10^{-2}	7.39×10^{-3}	3.36×10^{-3}
Xe-135	1.81×10^{-3}	1.86×10^{-3}	6.11×10^{-2}	2.46×10^{-3}	1.92×10^{-3}
Xe-137	1.42×10^{-3}	1.22×10^{-2}	3.05×10^{-1}	1.27×10^{-2}	1.51×10^{-3}
Xe-138	8.83×10^{-3}	4.13×10^{-3}	1.79×10^{-1}	4.75×10^{-3}	9.21×10^{-3}

TABLE 1.3

Summary of Radiological Effluent Controls and Implementing Equations

Control	Category	Method*	Limit
2.1 Off-Site Concentrations of Liquids	Total Fraction of MPC Excluding Noble Gases	Eq. 2-1	≤ 1.0
	Total Noble Gas Concentration	Eq. 2-2	$\leq 2.00 \times 10^{-4} \frac{\mu Ci}{cc}$
3.1 Dose Due to Liquid Effluents	Total Body Dose	Eq. 3-1	$\leq$ 1.5 mrem in a qtr. $\leq$ 3.0 mrem in a yr.
	Organ Dose	Eq. 3-2	$\leq$ 5.0 mrem in a qtr. $\leq$ 10.0 mrem in a yr.
3.2 Total Dose Due to Liquid and Gaseous Effluents	Total Body Dose Organ Dose Thyroid Dose	Eq. 3-1 Eq. 3-6 Eq. 3-9	$\leq$ 25.0 mrem in a yr.
		Eq. 3-2 Eq. 3-8 Eq. 3-9	$\leq$ 25.0 mrem in a yr.
		Eq. 3-2 Eq. 3-8 Eq. 3-9	$\leq$ 75.0 mrem in a yr.
3.3 Dose Rate Due to Gaseous Effluents	Total Body Dose Rate Due to Noble Gases	Eq. 3-3	$\leq 500.0 \frac{mrem}{yr}$
	Skin Dose Rate Due to Noble Gases	Eq. 3-4	$\leq 3000.0 \frac{mrem}{yr}$
	Organ Dose Rate Due to I-131, H-3, and Particulates with $T_{1/2} > 8$ Days	Eq. 3-5	$\leq 1500.0 \frac{mrem}{yr}$
3.4 Dose Due to Noble Gases in Gaseous Effluents	Gamma Air Dose Due to Noble Gases	Eq. 3-6	$\leq$ 5.0 mrad in a qtr. $\leq$ 10.0 mrad in a yr.
	Beta Air Dose Due to Noble Gases	Eq. 3-7	$\leq$ 10.0 mrad in a qtr. $\leq$ 20.0 mrad in a yr.

TABLE 1.3
(Continued)

Summary of Radiological Effluent Controls and Implementing Equations

Control	Category	Method*	Limit
3.5 Dose Due to Iodine-131, Tritium, and Particulates in Gaseous Effluents	Organ Dose Due to I-131, H-3, and Particulates with $T_{1/2}$ > 8 Days	Eq. 3-8	≤ 7.5 mrem in a qtr. ≤ 15.0 mrem in a yr.
5.1 Liquid Effluent Monitor Setpoint	Alarm/Trip Setpoint	Eq. 5-1	Control 2.1
5.2 Gaseous Effluent Monitor Setpoint	Alarm Setpoint for Total Body Dose Rate	Eq. 5-3	Control 3.3.a (Total Body)
	Alarm Setpoint for Skin Dose Rate	Eq. 5-4	Control 3.3.a (Skin)
6.1 Liquid Radioactive Waste Treatment	Total Body Dose	Eq. 3-1	≤ 0.06 mrem in a mo.
	Organ Dose	Eq. 3-2	≤ 0.2 mrem in a mo.
6.2 Gaseous Radioactive Waste Treatment	Gamma Air Dose Due to Noble Gases	Eq. 3-6	≤ 0.2 mrad in a mo.
	Beta Air Dose Due to Noble Gases	Eq. 3-7	≤ 0.4 mrad in a mo.
	Organ Dose Due to I-131, H-3, and Particulates with $T_{1/2}$ > 8 Days	Eq. 3-8	≤ 0.3 mrem in a mo.

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

TABLE 1.4

Summary of Constants

Constant	Definition	Units
0.00087	= Conversion factor.	$\frac{mrem}{\mu R}$
0.25	= $(3.17 \times 10^{+4})\ (\frac{pCi\text{-}yr}{Ci\text{-}sec})\ [X/Q]^{\gamma}\ (sec/m^3)$	
	= $(3.17 \times 10^{+4})\ (7.83 \times 10^{-6})$	$\frac{pCi\text{-}yr}{Ci\text{-}m^3}$
0.76	= $(3.17 \times 10^{+4})\ (\frac{pCi\text{-}yr}{Ci\text{-}sec})\ [X/Q]\ (sec/m^3)$	
	= $(3.17 \times 10^{+4})\ (2.39 \times 10^{-5})$	$\frac{pCi\text{-}yr}{Ci\text{-}m^3}$
1.11	= Average ratio of tissue to air energy absorption ratio coefficient.	
7.83	= $(10^{+6})\ (pCi/\mu Ci)\ (1.00)\ (7.83 \times 10^{-6})\ (sec/m^3)$	$\frac{pCi\text{-}sec}{\mu Ci\text{-}m^3}$
8.69	= $(1.11)\ (S_F)\ [X/Q]^{\gamma}\ (sec/m^3)\ (1.00 \times 10^{+6})\ (pCi/Ci)$	
	= $(1.11)\ (1.00)\ (7.83 \times 10^{-6})\ (1.00 \times 10^{+6})$	$\frac{pCi\text{-}sec}{Ci\text{-}m^3}$
23.90	= $(1.00 \times 10^{+6})\ (X/Q)$	
	= $(1.00 \times 10^{+6})\ (2.39 \times 10^{-5})$	
60.00	= Conversion factor.	$\frac{sec}{min}$
500.00	= Total body annual dose limit from ICRP-2.	mrem
3000.00	= Skin annual dose limit from ICRP-2.	mrem
$3.17 \times 10^{+4}$	= Number of picocuries per Curie divided by the number of seconds per year.	$\frac{pCi\text{-}sec}{Ci\text{-}yr}$
$1.00 \times 10^{+6}$	= Number of picocuries per microcurie.	$\frac{pCi}{Ci}$

TABLE 1.5

Summary of Variables

Variable	Definition	Units
C^{NG}	= Total activity of all dissolved and entrained noble gases from all station sources.	$\frac{\mu Ci}{cc}$
C_i^{ENG}	= Concentration of radionuclide "i", except noble gases, at the point of discharge.	$\frac{\mu Ci}{cc}$
C_i^{NG}	= Concentration of radionuclide "i", except noble gases, at the point of discharge.	$\frac{\mu Ci}{cc}$
C_i	= Concentration of radionuclide "i".	$\mu Ci/m^3$ or $\mu Ci/cc$
D_{air}^{β}	= Beta dose to air.	mrad
D_{air}^{γ}	= Gamma dose to air.	mrad
D_{grd}^{γ}	= Gamma dose to air from a ground level release.	mrad
D_{co}	= Dose to the critical organ.	mrem
D_d	= Direct dose.	mrem
D_{organ}	= Dose to the maximum organ.	mrem
D_{skin}	= Beta and gamma dose to the skin.	mrem
D_{tb}	= Dose to the total body.	mrem
DFB_i	= Total body gamma dose factor for radionuclide "i".	$\frac{mrem\text{-}m^3}{pCi\text{-}yr}$
DFS_i	= Beta skin dose factor for radionuclide "i".	$\frac{mrem\text{-}m^3}{pCi\text{-}yr}$
DF_i'	= Combined site-specific skin dose factor.	$\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}$
DF_i^{γ}	= Gamma air dose factor for radionuclide "i".	$\frac{mrad\text{-}m^3}{pCi\text{-}yr}$
DF_i^{β}	= Beta air dose factor for radionuclide "i".	$\frac{mrad\text{-}m^3}{pCi\text{-}yr}$

Summary of Variables

Variable	Definition	Units
DFG_{ico}	= Critical organ gaseous dose factor for radionuclide "i".	$\frac{mrem}{Ci}$
DFG'_{ico}	= Critical organ gaseous dose rate factor for radionuclide "i".	$\frac{mrem\text{-}sec}{\mu Ci\text{-}yr}$
DFL_{imo}	= Maximum organ liquid dose factor for radionuclide "i".	$\frac{mrem}{Ci}$
DFL_{itb}	= Total body liquid dose factor for radionuclide "i".	$\frac{mrem}{Ci}$
$\dot{D}_{co}$	= Critical organ dose rate due to iodines, tritium, and particulates.	$\frac{mrem}{yr}$
$\dot{D}_{skin}$	= Skin dose rate due to noble gases.	$\frac{mrem}{yr}$
$\dot{D}_{tb}$	= Total body dose rate due to noble gases.	$\frac{mrem}{yr}$
D/Q	= Deposition factor for dry deposition of elemental radioiodines and other particulates.	$\frac{sec}{m^2}$
$\dot{E}_r$	= Exposure rate at the critical receptor from non-vapor container sources as measured or estimated for the period.	$\frac{\mu R}{hr}$
$\dot{E}_{VC}$	= Limiting exposure rate at the critical receptor from the vapor container during normal operations.	$\frac{\mu R}{hr}$
F	= Primary vent stack flow rate.	$\frac{cc}{min}$
f_i^{NG}	= Fraction of radionuclide "i" activity to total noble gas activity.	
F_1	= Total fraction of MPC in liquid pathways.	
F_1^{ENG}	= Total fraction of MPC in liquid pathways excluding noble gases.	
f_1	= Flow rate past the test tank monitor.	gpm

TABLE 1.5
(Continued)

Summary of Variables

Variable	Definition	Units
f_2	= Flow rate past the steam generator monitor.	gpm
f_3	= Flow rate at the point of discharge.	gpm
MPC_c	= Composite MPC for the mix of radionuclides. See Equation 5-2.	$\frac{\mu Ci}{cc}$
MPC_i	= Maximum permissible concentration of radionuclide "i" (10CFR Part 20, Appendix B, Table 2, Column 2).	$\frac{\mu Ci}{cc}$
Q	= Total release of all noble gases.	Ci
Q_i	= Release for radionuclide "i".	Ci
$\dot{Q}$	= Total release rate of all noble gases.	$\frac{\mu Ci}{sec}$
$\dot{Q}_i$	= Release rate for radionuclide "i".	$\frac{\mu Ci}{sec}$
X/Q	= Average undepleted dispersion factor.	$\frac{sec}{m^3}$
$[X/Q]^D$	= Average depleted dispersion factor.	$\frac{sec}{m^3}$
$[X/Q]^\gamma$	= Effective average gamma dispersion factor.	$\frac{sec}{m^3}$
S_F	= Shielding factor.	
S_g	= Gaseous monitor response factor.	$\frac{cpm}{\mu Ci/cc}$
s_i	= Ratio of response from equal activities of radionuclide "i" to a reference radionuclide (such as Xe-133).	
S_ℓ	= Liquid monitor response factor.	$\frac{cpm}{\mu Ci/cc}$
T_e	= Exposure period.	hours
ΔD	= Conservative increment in annual average dose.	mrem

TABLE 1.6

Definition of Terms

The defined terms of this section appear in capitalized type and are applicable throughout this document.

ACTION

ACTION shall be those additional requirements specified as corollary statements to each principle control and shall be part of the controls.

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY, including alarm and/or trip functions.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

TABLE 1.6
(Continued)

Definition of Terms

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals defined in Table 1.10.

GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEM

A GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting Primary Coolant System offgases from the Primary System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category, are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the production of electricity.

OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The ODCM contains the methodology and parameters used in the calculation of off-site doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM also contains (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8 of the Technical Specification document and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Controls 7.1 and 7.2, respectively.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.9.

PURGE - PURGING

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

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 600 MWt.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee. Any area within the SITE BOUNDARY used for residential quarters or recreational purposes shall be considered to be beyond the SITE BOUNDARY for purposes of meeting gaseous effluent dose controls. (Realistic occupancy factors shall be applied at these locations for the purposes of dose calculations.)

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TABLE 1.6
(Continued)

Definition of Terms

VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust steam prior to release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

TABLE 1.7

Dose Factors Specific to the Yankee Plant for Liquid Releases

Radionuclide	Total Body Dose Factor $DFL_{itb}\left(\frac{mrem}{Ci}\right)$	Maximum Organ Dose Factor $DFL_{imo}\left(\frac{mrem}{Ci}\right)$
H-3	5.99×10^{-4}	5.99×10^{-4}
C-14	$1.64 \times 10^{+0}$	$8.18 \times 10^{+0}$
Cr-51	7.20×10^{-5}	1.07×10^{-2}
Mn-54	6.07×10^{-2}	5.47×10^{-1}
Fe-55	3.46×10^{-2}	2.11×10^{-1}
Fe-59	1.00×10^{-1}	4.53×10^{-1}
Co-58	4.76×10^{-2}	1.81×10^{-1}
Co-60	2.79×10^{-1}	9.04×10^{-1}
Zn-65	$1.65 \times 10^{+0}$	$2.71 \times 10^{+0}$
Sr-89	2.30×10^{-1}	$8.04 \times 10^{+0}$
Sr-90	$6.97 \times 10^{+1}$	$2.75 \times 10^{+2}$
Zr-95/Nb-95	1.40×10^{-3}	2.87×10^{-1}
Ru-103	2.48×10^{-3}	3.57×10^{-1}
Ag-110m	2.32×10^{-2}	$2.21 \times 10^{+0}$
Sb-124	2.62×10^{-2}	6.48×10^{-1}
I-131	8.57×10^{-3}	$4.96 \times 10^{+0}$
I-133	6.52×10^{-4}	3.18×10^{-1}
Cs-134	$1.79 \times 10^{+1}$	$2.40 \times 10^{+1}$
Cs-136	$2.28 \times 10^{+0}$	$3.20 \times 10^{+0}$
Cs-137	$1.07 \times 10^{+1}$	$2.07 \times 10^{+1}$
Ba-140/La-140	3.40×10^{-3}	5.80×10^{-2}
Ce-141	7.73×10^{-5}	1.06×10^{-1}
Ce-144	1.41×10^{-3}	$2.58 \times 10^{+0}$

TABLE 1.8

Dose and Dose Rate Factors Specific to the Yankee Plant for Iodines, Tritium, and Particulate Releases

Radionuclide	Critical Organ Dose Factor $DFG_{ico} \left(\frac{mrem}{Ci}\right)$	Critical Organ Dose Rate Factor $DFG'_{ico} \left(\frac{mrem-sec}{yr-\mu Ci}\right)$
H-3	7.21×10^{-3}	2.27×10^{-1}
C-14	$4.38 \times 10^{+0}$	$1.38 \times 10^{+2}$
Cr-51	3.44×10^{-2}	$1.19 \times 10^{+0}$
Mn-54	$3.78 \times 10^{+0}$	$1.49 \times 10^{+2}$
Fe-59	$3.83 \times 10^{+0}$	$1.27 \times 10^{+2}$
Co-58	$1.98 \times 10^{+0}$	$7.06 \times 10^{+1}$
Co-60	$4.08 \times 10^{+1}$	$1.81 \times 10^{+3}$
Zn-65	$1.99 \times 10^{+1}$	$6.43 \times 10^{+2}$
Sr-89	$6.10 \times 10^{+1}$	$1.92 \times 10^{+3}$
Sr-90	$2.36 \times 10^{+3}$	$7.44 \times 10^{+4}$
Zr-95/Nb-95	$3.77 \times 10^{+0}$	$1.24 \times 10^{+2}$
Ru-103	$1.02 \times 10^{+1}$	$3.22 \times 10^{+2}$
Ag-110m	$3.63 \times 10^{+1}$	$1.22 \times 10^{+3}$
Sb-124	$6.95 \times 10^{+0}$	$2.32 \times 10^{+2}$
I-131	$4.19 \times 10^{+2}$	$1.32 \times 10^{+4}$
I-133	$6.29 \times 10^{+0}$	$1.98 \times 10^{+2}$
Cs-134	$8.52 \times 10^{+1}$	$2.83 \times 10^{+3}$
Cs-136	$4.71 \times 10^{+0}$	$1.52 \times 10^{+2}$
Cs-137	$8.71 \times 10^{+1}$	$2.97 \times 10^{+3}$
Ba-140/La-140	$1.44 \times 10^{+0}$	$4.60 \times 10^{+1}$
Ce-141	9.75×10^{-1}	$3.10 \times 10^{+1}$
Ce-144	$2.10 \times 10^{+1}$	$6.65 \times 10^{+2}$

TABLE 1.9

OPERATIONAL MODES

MODE	Reactivity Condition, K_{eff}	% RATED THERMAL POWER*	Average Coolant Temperature
Power Operation	≥0.99	>2%	≥330°F
Startup	≥0.99	≤2%	≥330°F
Hot Standby	<0.99	0	≥330°F
Hot Shutdown	<0.96	0	330°F > T_{avg} > 200°F
Cold Shutdown	<0.96	0	≤200°F
Refueling**	≤0.95	0	≤140°F

* Excluding decay heat.

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

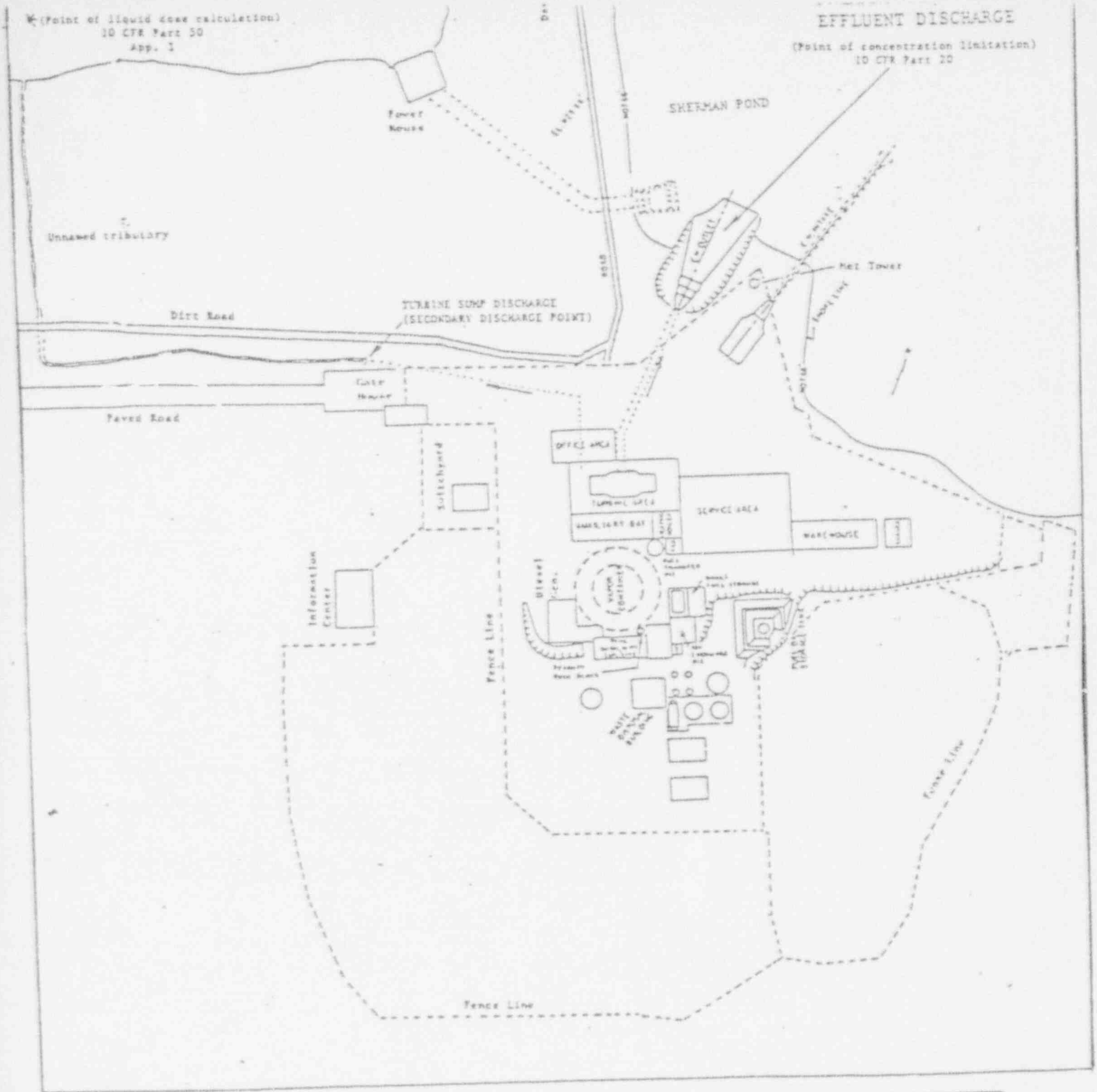
TABLE 1.10

FREQUENCY NOTATION

Notation	Frequency
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Prior to each release.
N.A.	Not applicable.

Yankee Atomic Electric Company	SITE BOU ARY LINE Figure -1

Revision 8



Yankee Atomic Electric Company	LIQUID EFFLUENT DISCHARGE POINTS Site Plot Plan
	Figure 1-2

2.0 RADIOACTIVE LIQUID EFFLUENTS

2.1 Off-Site Concentrations

Control 2.1 In accordance with Yankee Technical Specification 6.8.5.a., items 2 and 3, the concentration of radioactive material released from the site (see Figure 1-2) shall be limited to the concentrations specified in 10CFR Part 20, Appendix B, Table II, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2.00×10^{-4} µCi/ml total activity.

Applicability

At all times.

ACTION

With the concentration of radioactive material released from the site exceeding the above limits, without delay, take actions to restore the concentration to within the above limits.

Surveillance Requirements

SR 2.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 2.1.

SR 2.1.2 The results of radioactive analysis shall be used in accordance with the methods of the ODCM to assure that concentrations at the point of release are maintained within the limits of Control 2.1.

Bases

Control 2.1 is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site (at the point of discharge from the plant discharge structure into Sherman Pond) will be less than the concentration levels specified in 10CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design controls of Appendix I, 10CFR Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10CFR Part 20.106(e) to the population.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLD). Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report, ARH-SA-215 (June 1975). The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radionuclide, and its maximum permissible concentration (MPC) in air (submersion) was converted to an equivalent concentration in water using the International Commission on Radiological Protection (ICRP) Publication 2.

TABLE 2.1

Radioactive Liquid Waste Sampling and Analysis Program

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection LLD[a] (μCi/ml)
A. Batch Waste Release Tanks[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5.00×10^{-7}
			I-131	1.00×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1.00×10^{-5}
	P Each Batch	M Composite[c]	Tritium	1.00×10^{-5}
			Gross Alpha	1.00×10^{-7}
	P Each Batch	Q Composite[c]	Sr-89, Sr-90	5.00×10^{-8}
			Fe-55	1.00×10^{-6}
B. Plant Continuous Releases[e] (steam generator blowdown, secondary coolant and condensate leakage, Turbine Building sump)	Continuous[d]	W Composite[d]	Principal Gamma Emitters[f]	5.00×10^{-7}
			I-131	1.00×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1.00×10^{-5}
	Continuous[d]	M Composite[d]	Tritium	1.00×10^{-5}
			Gross Alpha	1.00×10^{-7}
	Continuous[d]	Q Composite[d]	Sr-89, Sr-90	5.00×10^{-8}
			Fe-55	1.00×10^{-6}

TABLE 2.1
(Continued)

Table Notation

a. The LLD is defined in Table Notation (a) of Table 4.3 of SR 4.1.

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 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.

d. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the average effluent release.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume or system that has an input flow during the continuous release.

f. The principal gamma emitters for which the LLD requirement applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these radionuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above radionuclides, also shall be identified and reported. Radionuclides that are below the LLD for the analyses should not be reported as being present at the LLD level.

2.2 Method to Calculate Off-Site Liquid Concentrations

The basis for plant procedures that the plant operator requires to meet Control 2.1, which limits the total fraction of MPC (F_1^{ENG}) in liquid pathways (excluding noble gases) at the point of discharge (see ODCM Figures 6-1 and 6-2) is discussed. F_1^{ENG} is limited to less than or equal to one, i.e.,

$$1 \geq \sum_i \frac{C_i}{MPC_i}$$

The total noble gas concentration, C^{NG}, is limited to 2.00×10^{-4} µCi/ml.

Evaluation of F_1^{ENG} and C^{NG} is required concurrent with the sampling and analysis program in Table 2.1 of Control 2.1. If this control is exceeded, and operator ACTION cannot be taken to reduce the concentration to within the required limit, then shutdown may be required; that is, Controls 1.3 and 1.4 apply.

Determine the total fraction of MPC in liquid pathways (excluding noble gases) as follows:

$$F_1^{ENG} = \sum_i \frac{C_i}{MPC_i} \qquad \text{(Eq.2-1)}$$

MPC_i = Maximum permissible concentration of radionuclide "i" except for dissolved and entrained noble gases (10CFR Part 20, Appendix B, Table 2, Column 2).

Determine the total noble gas concentration:

C^{NG} = Total activity of all dissolved and entrained noble gases from all station sources.

Where:

$$C_i = C_i^{TT} + C_i^{SGB} + C_i^{SCCL} + C_i^{Other} \qquad \text{(Eq.2-2)}$$

C_i^{TT} = Concentration at the point of discharge of radionuclide "i" except for dissolved and entrained noble gases from the test tank.

C_i^{SGB} = Concentration at the point of discharge of radionuclide "i" except for dissolved and entrained noble gases from steam generator blowdown.

C_i^{SCCL} = Concentration at the point of discharge of radionuclide "i" except for dissolved and entrained noble gases from secondary coolant and coolant leakage pathway.

C_i^{Other} = Concentration at the point of discharge of radionuclide "i" except for dissolved and entrained noble gases from all other significant sources.

These equations are always applicable.

2.3 Method to Determine Radionuclide Concentration for Each Liquid Effluent Pathway

2.3.1 Test Tank Pathway

C_i^{TT} is determined for each radionuclide above the analytical LLD from the activity in a proportional grab sample of the test tank and the predicted flow at the point of discharge.

Most periodic batch releases are from the two 7000-gallon capacity test tanks. Whenever a test tank is filled with liquid waste evaporator distillate, it is isolated for sampling and release. The volume of the tank contents is determined from the liquid level in the tank. A chemist extracts a sample for radionuclide analysis. Aliquots of the sample proportional to the volume of the tank contents are composited for appropriate radionuclide analyses. The composites contain suitable acids, alkalis, or carriers to assure the composite is representative of the sample. Several liters of samples are passed through a strong acid, strong base ion exchange resin. Soluble ionic and particulate radionuclides are concentrated with nearly 100 percent efficiency. The effluent is collected for tritium and dissolved noble gas analyses. Each test tank batch is analyzed for tritium, dissolved noble gases, and dissolved and suspended gamma emitters. Table 2 2 shows typical quantities of radionuclides released from the test tanks Most of the activity is tritium and Xenon-133. Gamma emitters are usually present near or below their LLD concentration.

2.3.2 Steam Generator Blowdown Pathway

C_i^{SGB} is determined for each radionuclide above the analytical LLD from the activity in a composite sample of steam generator blowdown flow and the predicted flow at the point of discharge.

About one part in five thousand of steam generator blowdown is collected for radionuclide analyses. The discharge valve on the blowdown tank cycles opens about one minute of every twenty. When the valve is open, a micro sampling pump collects water in the tank discharge line. Since the discharge flow rate is fixed by piping geometry and a constant head in the tank, the sampling flow rate is proportional to the discharge flow rate.

2.3.3 Secondary Coolant and Coolant Leakage Pathway

C_i^{SCCL} is determined for each radionuclide above the analytical LLD from the activity in a proportional composite sample from the discharge of the condensate pumps. The sampling method is similar to that outlined in Section 2.3.2.

2.3.4 Remaining Pathways

C_i^{Other} is determined for each of the remaining pathways as follows:

Miscellaneous batch releases of potentially contaminated water, i.e., draining a laid-up steam generator or rain water collected behind the radioactive waste tank dikes, are treated according to Section 2.3.1.

A proportional composite of the Turbine Building floor drains is continuously collected. Since the water in this pathway is practically all Sherman Pond water (>99 percent) used for cooling purposes, it is unlikely that it will ever be a significant effluent pathway.

TABLE 2.2

Typical Radionuclides Released From Test Tanks
(Reference Year 1979)

Radionuclide	Curies	Average Concentration In Tanks (Sampled)	Control LLD
Tritium	176	7×10^{-2}	1.00×10^{-5}
Xenon-133	0.2	1×10^{-4}	1.00×10^{-5}
Iodine-131	0.002	1×10^{-6}	1.00×10^{-6}
Cesium-134, -137	0.005	2×10^{-6}	5.00×10^{-7}
All Others (I-133, Xe-135, Co-60, Mn-54)	0.003	1.5×10^{-6} (total)	5.00×10^{-7}

2.4 Background Information

The results of the necessary analyses and calculations are conveniently summarized on the Radioactive Discharge Permit.

This permit shows the following information:

a. The radionuclides in the program library that were analyzed in the analysis;

b. The analytical LLD for each radionuclide for that particular analysis;

c. Whether the radionuclide was present (concentration above analytical LLD) or absent (concentration below analytical LLD);

d. Whether all analytical LLDs were less than the control LLD (i.e., as specified by Control 2.1, Table 2.1);

e. The activity of each radionuclide present in the tank contents;

f. The dose, in mrem, to a MEMBER OF THE PUBLIC associated with the release as calculated in the ODCM;

g. The calculated discharge concentration, C_i^{TT}, in µCi/ml, and the fraction of MPC;

h. The estimated (generally from the most recent data) unrestricted area discharge concentration from steam generator blowdown effluent expressed as µCi/ml and a fraction of MPC taken as 3.00×10^{-7} µCi/ml; and

i. A mathematical summation of all radionuclides identified above their analytical LLD.

$$F_1 = \frac{C_{H-3}}{3.00 \times 10^{-3}} + \frac{C_{\text{Noble Gases}}}{2.00 \times 10^{-4}} + \frac{C_{I-131}}{3.00 \times 10^{-7}} + \frac{C_{\text{SG Blowdown}}}{3.00 \times 10^{-7}} + \frac{C_{\text{Gamma}_i}}{MPC_i} + \ldots$$

If the summation is less than an administrative limit of 0.80, then the release is allowed.

3.0 DOSE/DOSE RATE CONTROLS AND CALCULATIONS

3.1 Dose Due to Radioactive Liquid Effluents

Control 3.1 In accordance with Yankee Technical Specification 6.8.5.a, items 4 and 5, the dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from the site (see Figure 1-2) to available uptake pathways shall be limited:

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

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, and if not applicable to 10CFR Part 50.73, prepare and submit to the Commission within 30 days, pursuant to Control 7.4, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be within the above limits.

b. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 3.1 Dose Calculations - Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

Bases

Control 3.1 is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10CFR Part 50. The control implements the guides set forth in Section II.A. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in liquid effluents will be kept as low as is reasonably achievable. The surveillance requirement implements 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. Existing pathways of liquid exposure to MEMBER(S) OF THE PUBLIC which form the basis for calculating liquid doses in the ODCM are described in detail in Yankee Atomic Electric Company's design report, "Supplemental Information for the Purpose of Evaluation of 10CFR Part 50, Appendix I", dated June 2, 1976 (with amendments). The point of exposure from existing pathways for dose calculational purposes is taken downstream of Sherman Dam in the Deerfield River. 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 10CFR 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," April 1977. Also, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in finished drinking water that are in excess of the requirements of 40CFR141. No drinking water supplies from the Deerfield River below the plant have been identified.

3.2 Total Dose Due to Radioactive Liquid and Gaseous Effluents

Control 3.2 In accordance with Yankee Technical Specification 6.8.5.a.10, the dose or dose commitment to any real MEMBER OF THE PUBLIC from all station sources is limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which is limited to less than or equal to 75 mrem) over a calendar year.

Applicability

At all times.

ACTION

a. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Controls 3.1.a, 3.1.b, 3.4.a, 3.4.b, 3.5.a, or 3.5.b, calculations should be made including direct radiation contributions from the reactor and from outside storage tanks to determine whether the above limits of Control 3.2 have been exceeded. If such is the case, and if not applicable to 10CFR Part 50.73, prepare and submit to the Commission within 30 days, pursuant to Control 7.4, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. The Special Report, as required by 10CFR Part 20.405(c), 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 the report. It also shall describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40CFR190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 3.2 Dose Calculations - Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with SR 3.1, 3.4, and 3.5 and in accordance with the ODCM.

Bases

Control 3.2 is provided to meet the dose limitations of 40CFR Part 190 that have been incorporated into 10CFR Part 20 by 46FR18525. The control requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40CFR Part 190 if the individual reactors remain within the reporting requirement level. 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 40CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to a MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40CFR Part 190 have not already been corrected), in accordance with the provisions of 40CFR Part 190.11 and 10CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10CFR Part 20, as addressed in liquid and gaseous effluent controls. 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 is part of the nuclear fuel cycle.

3.3 Dose Rate Due to Radioactive Gaseous Effluents

Control 3.3 In accordance with Yankee Technical Specification 6.8.5.a, items 3 and 7, the dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 1-1) shall be limited to the following:

a. For noble gases: less than or equal to 500 mrem/yr to the total body and less than or equal to 3,000 mrem/yr to the skin, and

b. For Iodine-131, 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, without delay, take actions to decrease the release rate to within the above limit(s).

Surveillance Requirements

SR 3.3.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

SR 3.3.2 The dose rate due to Iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 3.1.

Bases

Control 3.3 is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of MEMBER(S) OF THE PUBLIC either within or outside the SITE BOUNDARY to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10CFR , , 20.106(b). For MEMBER(S) OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits as determined by the procedures in the ODCM, restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to 500 mrem/year to the total body or to 3,000 mrem/year to the skin.

Control 3.3.b also restricts at all times comparable with the length of the sampling periods of Table 3.1, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to 1,500 mrem/year for the nearest cow to the plant.

TABLE 3.1

Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	LLD μCi/ml[a]
A. Waste Gas Surge Drum (one only)	P Grab Sample	P	Principal Gamma Emitters[b]	1.00×10^{-4}
B. Containment PURGE	P Grab Sample (Each PURGE)	P (Each PURGE)	Principal Gamma Emitters[b]	1.00×10^{-4}
			Tritium	1.00×10^{-6}
C. Plant Vent (Primary Vent Stack)[1]	M[c] Grab Sample	M[c]	Principal Gamma Emitters[b]	1.00×10^{-4}
			Tritium	1.00×10^{-6}
	Continuous[e]	W[d] Radioiodine Canister	I-131	1.00×10^{-12}
	Continuous[e]	W[d] Particulate	Principal Gamma Emitters[b] I-131	1.00×10^{-11}
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1.00×10^{-11}
	Continuous[e]	Q Composite Particulate Sample	Sr-89, Sr-90	1.00×10^{-11}
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1.00×10^{-5}

TABLE 3.1
(Continued)

Table Notation

NOTE 1 - The ventilation header channels air through the ventilation system to the plant vent stack. The following ventilation systems discharge directly into the ventilation header:

- Auxiliary Building
- Radioactive Waste Building
- Containment PURGE
- Fuel Storage Area
- Condenser Air Ejector
- Waste Gas Holdup System Discharge

The steam generator blowdown vent discharges directly into the plant vent stack.

a. The LLD is defined in Table Notation (a) of Table 4.1 of Control 4.1.

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

c. Sampling and analysis also shall be performed following a THERMAL POWER change of greater than 15 percent of RATED THERMAL POWER within one hour. A grab sample for noble gas analysis shall be taken within 8 hours and analyzed within 24 hours of the THERMAL POWER change. This requirement does not apply if: (1) the analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3, and (2) the noble gas activity monitor shows that effluent activity has not increased more than a factor of 3.

TABLE 3.1
(Continued)

Table Notation

d. Samples shall be changed at least once per 7 days, and analyses shall be completed within 48 hours after changing or after removal from samplers. Sampling also shall be performed at least once per 24 hours for at least 7 days following a THERMAL POWER change of greater than 15 percent of RATED THERMAL POWER within one hour. Samples collected for 24 hours will be analyzed within 48 hours of changing, and the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) the analysis shows that the DOSE EQUIVALENT I-131 concentration of the primary coolant has not increased more than a factor of 3, and (2) the noble gas activity monitor shows that the effluent activity has not increased more than a factor of 3.

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, 3.4, and 3.5.

3.4 Dose Due to Noble Gases Released in Radioactive Gaseous Effluents

Control 3.4 In accordance with Yankee Technical Specification 6.8.5.a, items 5 and 8, the air dose due to noble gases released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 1-1) shall be limited to the following:

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

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

Applicability

At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits and if not applicable to 10CFR Part 50.73, prepare and submit to the Commission within 30 days, pursuant to Control 7.4, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be within the above limits.

b. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 3.4 Dose Calculations - Cumulative dose contributions for current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once every 31 days.

Bases

Control 3.4 is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10CFR Part 50. The control implements the guides set forth in Section II.B. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The surveillance requirement implements 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 radioactive noble gases in gaseous 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 10CFR Part 50, Appendix I," Revision 1, October 1977, and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM provides for determining the air doses at the SITE BOUNDARY based upon the historical average atmospheric conditions.

3.5 Dose Due to Iodine-131, Tritium, and Radionuclides in Particulate Form With Half-Lives Greater than Eight Days

Control 3.5 In accordance with Yankee Technical Specification 6.8.5.a, items 5 and 9, the dose to a MEMBER OF THE PUBLIC from Iodine-131, 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 (see Figure 1-1) 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

a. With the calculated dose from the release of radioiodines, radioactive materials in particulate form, or radionuclides other than noble gases in gaseous effluents exceeding any of the above limits, and if not applicable to 10CFR Part 50.73, prepare and submit to the Commission within 30 days, pursuant to Control 7.4, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be within the above limits.

b. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 3.5 Dose Calculations - Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the ODCM at least once every 31 days.

Bases

Control 3.5 is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I, 10CFR Part 50. The control is the guide set forth in Section II.C. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The surveillance requirement implements 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 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 10CFR 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 Iodine-131, tritium, and radionuclides in particulate form with half-lives greater than eight days are dependent on the existing radionuclide pathways to man in areas at and beyond the 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 and meat animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3.6 Dose Calculation Concepts

Controls 3.1 through 3.5 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 refer to the fifty-year dose commitment from one year's worth of releases. "Dose in a quarter" similarly means a fifty-year dose commitment from one quarter's releases. The term "dose," with respect to external exposures, such as to noble gas clouds, 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 also are controlled such that the maximum "dose rates" at the SITE BOUNDARY at any time are limited to 500 mrem/yr to the whole body or 3000 mrem/yr to the skin. The annual dose limits are the doses associated with the concentrations of Appendix B, Table II, Column 1 of 10CFR Part 20.106(a). The use of the annual dose limits embodied in 10CFR Part 20 as plant "dose rate" values (to be applied at any time consistent with the capabilities of the monitoring instrumentation) provides reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of MEMBER(S) OF THE PUBLIC either within or outside the SITE BOUNDARY to annual average concentrations exceeding the federal regulations.

It also should be noted that a dose rate due to noble gases that exceeds for a short time period (less than one hour in duration) the equivalent 500 mrem/year dose rate limit stated in Control 3.3 does not necessarily by itself constitute a Licensee Event Report (LER) under 10CFR Part 50.73 unless it is determined that the air concentration of radioactive effluents in unrestricted areas also has exceeded two times the MPC when averaged over one hour (four-hour notification per 10CFR Part 50.72 and 30-day LER per 10CFR Part 50.73).

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

The dose, D, is the quantity calculated by the Chapter 3 dose equations. The D calculated by "Method I" equations is not necessarily the actual dose received by a real individual, but usually provides an upper bound for a given release because of the conservative margin built into the dose factors and the selection and definition of the critical receptors. The radioisotope specific dose factors in each "Method I" dose equation represent the greatest dose to any organ of any age group accounting for existing or potential pathways of exposure. The critical receptor assumed by "Method I" equations is typically a hypothetical individual whose behavior in terms of location and intake results in a dose which is expected to be higher than any real individual. Method II allows for a more exact dose calculation for real individuals if necessary by considering only existing pathways of exposure or actual concurrent meteorology with the recorded release.

$\dot{D}$ is the quantity calculated in the Chapter 3 dose rate equations. It is calculated using the plant's effluent monitoring system reading and an annual average or long-term atmospheric dispersion factor. If plant release rates were such that a $\dot{D}$ equal to the Control 3.3 value were continued for one year, the annual dose limits of 10CFR Part 20 would be reached. However, since maximum allowed release rates and the resulting dose rates in the range of the objective limits are very infrequent and are typically of short time duration, this approach of limiting dose rates equivalent to the annual dose limits then assures that 10CFR Part 20.106 limits on an annual average air concentration in unrestricted areas will be met.

Each of the methods to calculate dose or dose rate are presented in separate sections of Chapter 3, and are summarized in Table 1.1. Each method has two levels of complexity and conservative margin and are called Method I and Method II. Method I has the greatest margin and is the simplest; generally a linear equation. Method II is a more detailed analysis which allows for use of site-specific factors and variable parameters to be selected to best fit the actual release. Guidance is provided, but the appropriate margin and depth of analysis are determined in each instance at the time of analysis under Method II.

3.7 Method to Calculate the Total Body Dose from Liquid Releases

Control 3.1 limits the total body dose commitment to a MEMBER OF THE PUBLIC from radioactive material in liquid effluents to 1.5 mrem per quarter and 3 mrem per year. Control 6.1 requires liquid radioactive waste treatment when the total body dose estimate exceeds 0.06 mrem in any 31-day period. Control 3.2 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 31 days. If the liquid radioactive waste 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 from the plant.

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.

To evaluate total body dose for Control 6.1 add the total body dose from today's expected releases to the total body dose accumulated for the time period of interest.

3.7.1 Method I

The total body dose from a liquid release is:

$$\underset{(\text{mrem})}{D_{tb}} = K \sum_i Q_i \, DFL_{itb} \qquad \text{(Eq. 3-1)}$$

Where:

DFL_{itb} = Site-specific total body dose factor (mrem/Ci) for liquid release. See Table 1.7.

Q_i = Total activity (Curies) released to liquids of radionuclide "i" during the period of interest. For i = Fe-55, Sr-89, Sr-90, or H-3, use the best estimates (such as the most recent measurements).

K = $366/F_d$; where F_d is the average (typically monthly average) dilution flow of the Deerfield River below Sherman Dam (in ft^3/sec). If F_d cannot be obtained or F_d is greater than 366, K can be assumed to equal 1.0. The value, 366, is the ten-year minimum monthly average Deerfield River flow rate below Sherman Dam (in ft^3/sec).

Equation 3-1 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

a. Liquid releases to the circulating water pathway to Sherman Reservoir or to the west storm drain pathway to the Deerfield River,

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

3.7.2 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 is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.7.3 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 controls which limit off-site total body dose from liquids (Controls 3.1, 3.2, and 6.1) have been met for releases over the appropriate periods. These requirements are based on design objectives and standards in 10CFR Part 50 and 40CFR Part 190. Control 3.1 is based on the ALARA design controls in 10CFR Part 50, Appendix I, Subsection II A. Control 6.1 is an "appropriate fraction", determined by the NRC, of the ALARA design control. Control 3.2 is based on Environmental Standards for the Uranium Fuel Cycle in 40CFR Part 190 (hereafter called the Standard) which applies to direct radiation as well as liquid and gaseous effluents. Method I applies only to the liquid contribution.

Exceeding the design 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. This is unlike exceeding 10CFR Part 20 limits which could result in plant shutdown.

Method I was developed such that "the actual exposure of an individual ... is unlikely to be substantially underestimated (10CFR Part 50, Appendix I). The definition 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 and annual average conditions instead of any real individual. The 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.

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 in mrem/Ci) for a one Curie release of each radionuclide in liquid effluents was derived. The base case analysis uses the methods, data, and assumptions in Regulatory Guide 1.109 (Equations A-3, A-7, A-13, and A-16, Reference A). Tables 3.2 and 3.3 outline human consumption and environmental parameters used in the analysis. It is assumed that the critical receptor fishes below Sherman Dam and eats the fish caught from this location and consumes leafy vegetables and produce from a farm which is irrigated with water from the Deerfield River below Sherman Dam. It also is assumed that the critical receptor drinks milk and eats meat from cows who drink water from the Deerfield River below Sherman Dam and eat silage from the irrigated farm above. The model is conservative because no real individual is likely to have that critical combination of exposures. A real individual would have only one or two pathways of exposure. A plant discharge flow rate of 308 ft^3/sec was used with a mixing ratio of 0.84.

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

$$\Delta D_{itb} = (Q_i)\ (DFL_{itb})$$

where DFL_{itb} is the total body 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 because it is based on the following reduction of the base case. The dose factors, DFL_{itb}, used in Method I were chosen from the base case to be the highest of the four age groups for that radionuclide. In effect, each radionuclide is conservatively represented by its own critical age group.

TABLE 3.2

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

Variable			Potable Water	Aquatic Food	Shoreline Activity	Food Grown With Contaminated Water: Vegetables	Leafy Veg.	Meat	Cow Milk	Goat Milk*
MP	Mixing Ratio		-	0.84	0.84	0.84	0.84	0.84	0.84	0.84
TP	Transit Time	(hrs)	1[illegible]	24.00	0.00	0.00	0.00	480.00	48.00	48.00
YV	Agricultural Productivity	(kg/m^2)	-	-	-	2.00	2.00	2.00	2.00	2.00
P	Soil Surface Density	(kg/m^2)	-	-	-	240.00	240.00	240.00	240.00	240.00
IRR	Irrigation Rate	($l/m^2/hr$)	-	-	-	0.15	0.15	0.15	0.15	0.15
TE	Crop Exposure Time	(hrs)	-	-	-	1440.00	1440.00	1440.00	1440.00	1440.00
TH	Holdup Time	(hrs)	-	-	-	1440.00	24.00	2160.00	2160.00	2160.00
QAW	Water Uptake Rate for Animal	(l/d)	-	-	-	-	-	50.00	60.00	8.00
QF	Feed Uptake Rate for Animal	(kg/d)	-	-	-	-	-	50.00	50.00	6.00
Location of Critical Individual			None	Below Sherman Dam	Below Sherman Dam	Below Sherman Dam	Below Sherman Dam	Below Sherman Dam	Below Sherman Dam	None

* Pathway is not included in Method I. It is listed for informational purposes and the possible use in a Method II calculation.

TABLE 3.3

Age-Specific Usage Factors for Various Liquid Pathways at Yankee Rowe
(From Reference A, Table E-5. Zero where no pathway exists)

Age Group	Veg. (kg/yr)	Leafy Veg. (kg/yr)	Milk (l/yr)	Meat (kg/yr)	Fish (kg/yr)	Invert. (kg/yr)	Potable Water (l/yr)	Shoreline (hr/yr)
Adult	520.00	64.00	310.00	110.00	21.00	0.00	0.00	12.00
Teen	630.00	42.00	400.00	65.00	16.00	0.00	0.00	67.00
Child	520.00	26.00	330.00	41.00	6.90	0.00	0.00	14.00
Infant	0.00	0.00	330.00	0.00	0.00	0.00	0.00	0.00

3.8 Method to Calculate Maximum Organ Dose from Liquid Releases

Control 3.1 limits the maximum organ dose commitment to a MEMBER OF THE PUBLIC from radioactive material in liquid effluents to 5 mrem per quarter and 10 mrem per year. Control 6.1 requires liquid radioactive waste treatment when the maximum organ dose estimate exceeds 0.2 mrem in any 31-day period. Control 3.2 limits the maximum organ dose commitment to any real MEMBER OF THE PUBLIC from all station sources (including liquids) to 25 mrem in a year except for the thyroid, which is limited to 75 mrem in a year. Dose evaluation is required at least once per 31 days. If the Liquid Radioactive Waste 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 from the plant.

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.

To evaluate the maximum organ dose for Control 6.1, add the organ dose from the expected releases to the organ dose accumulated for the time period of interest.

3.8.1 Method I

The maximum organ dose from a liquid release is:

$$\underset{(mrem)}{D_{organ}} = K \sum_i Q_i \, DFL_{imo} \qquad \text{(Eq. 3-2)}$$

Where:

DFL_{imo} = Site-specific maximum organ dose factor (mrem/Ci) for a liquid release. See Table 1.7.

Q_i = Total activity (Curies) released to liquids of radionuclide "i" during the period of interest. For i = Fe-55, Sr-89, Sr-90, or

H-3, use the best estimates (such as the most recent measurements).

K = $366/F_d$; where F_d is the average (typically monthly average) dilution flow of the Deerfield River below Sherman Dam (in ft^3/sec). If F_d cannot be obtained or F_d is greater than 366, K can be assumed to equal 1.0. The value, 366, is the ten-year minimum monthly average Deerfield River flow rate below Sherman Dam (in ft^3/sec).

Equation 3-2 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

a. Liquid releases to the circulating water pathway to Sherman Reservoir or to the west storm drain pathway to the Deerfield River,

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

3.8.2 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, Revision 1 (Reference A), except where site-specific models, data, or assumptions are more applicable. The base case analysis is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.8.3 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 the maximum organ dose parallel the total body dose methods (see Section 3.7.3). Only the differences are presented here.

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

$$\Delta D_{imo} = (Q_i)\,(DFL_{imo})$$

where DFL_{imo} is the maximum organ dose factor for radionuclide "i", and Q_i is the activity of radionuclide "i" released in Curies.

The dose factors, DFL_{imo}, used in Method I were chosen from the base case to be the highest of the set of seven organs and four age groups for each radionuclide. This means that the maximum effect of each radionuclide is conservatively represented by its own critical age group and critical organ.

3.9 Method to Calculate the Total Body Dose Rate from Noble Gases

Control 3.3 limits the dose rate at any time to the total body from noble gas at any location at or beyond the SITE BOUNDARY equal to or less than 500 mrem/year. By limiting the maximum $\dot{D}_{tb}$ to a rate equivalent to no more than 500 mrem/year, assurance is provided that the total body dose accrued in any one year by any member of the general public will be less than 500 mrem in accordance with the annual dose limits of 10CFR Part 20 to unrestricted areas.

Use Method I first to calculate the total body dose rate from the peak release rate via the plant vent stack. Method I applies at all release rates.

Use Method II if Method I predicts a dose rate greater than the limit (i.e., use of actual meteorology over the period of interest) to determine if, in fact, Control 3.3 had actually been exceeded during a short time interval.

Compliance with the dose rate limits for noble gases is continuously demonstrated when effluent release rates are below the plant vent stack noble gas activity monitor alarm setpoint by virtue of the fact that the alarm setpoint is based on a value which corresponds to the off-site dose rate limit of Control 3.3 or a value below it.

Determinations of dose rates for compliance with Control 3.3 are performed when the effluent monitor alarm setpoint is exceeded, and the corrective action required by Control 3.3 is unsuccessful, or as required by the ACTION to Table 5.3 when the stack noble gas monitor is inoperable.

3.9.1 Method I

The total body dose rate due to noble gases can be determined as follows:

$$\dot{D}_{tb}\left(\frac{mrem}{yr}\right) = 7.83 \sum_i \dot{Q}_i \; DFB_i \qquad \text{(Eq. 3-3)}$$

Where:

$\dot{Q}_i$ = The release rate from the plant vent stack (μCi/sec) of radionuclide "i" in the case of noble gases. The release rate at the plant vent stack is based on the measured radionuclide distribution in the off-gas during plant operation and the recorded total gas effluent count rate from the stack noble gas activity monitor. The release rate at the stack also can be stated in the following equation:

$$\dot{Q} = (d_i)\,(M)\left(\frac{1}{S_g}\right)(F) \qquad \text{(Eq. 3-10)}$$

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

Where:

M = Plant vent stack monitor count rate (cpm).

S_g = Appropriate plant vent stack monitor detector counting efficiency (cpm/(μCi/cc)).

F = Plant vent stack flow rate (cc/sec).

d_i = Fraction of the release which is radionuclide "i". This fraction can be based on the last measured value of radionuclide "i" with respect to the total noble gas activity released at the plant vent stack. It also can be based on the fraction of radionuclide "i" in the primary coolant with respect to the total noble gas primary coolant activity.

DFB_i = Total body dose factor. See Table 1.2.

During periods (beyond the first two days) when the plant is shutdown and no radioactivity release rates can be measured at the plant vent stack, Xe-133 may be used as the referenced radionuclide to determine off-site dose rate and monitor setpoints. Alternately, a relative radionuclide "i" mix fraction (f_i) may be taken from Table 5.6 as a function of time after shutdown, and substituted in place of d_i in Equation 3-10 above to determine the relative fraction of each noble gas potentially available for release to the total. Just prior to plant startup, the monitor alarm setpoints should be based on Xe-138 as representing the most prevalent high dose factor noble gas expected to be present shortly after the plant returns to power. Monitor alarm setpoints which have been determined to be conservative under any plant conditions may be utilized at any time in lieu of the above assumptions.

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

a. Normal operations (not emergency event).

b. Noble gas releases via the plant vent stack to the atmosphere.

3.9.2 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 may be applied. Method II consists of the models, input data, and assumptions in Regulatory Guide 1.109, Revision 1 (Reference A), except where site-specific models, data, or assumptions are more applicable. The base case analysis is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.9.3 Basis for Method I

This section serves four purposes: (1) to document that Method I complies with appropriate NRC regulations, (2) to define the word "rate" as used in the control, (3) to provide background and training information to Method I users, and (4) to provide an introductory user's guide to Method II.

Method I may be used to show that the Control 3.3 limit for the total body dose rate from noble gases released to the atmosphere has been met for the peak noble gas release rate.

Control 3.3 ensures "... that the doses ... at and beyond the SITE BOUNDARY from gaseous effluents ... will be within the annual dose limits ... associated with the concentrations of 10CFR Part 20, Appendix B, Table II, Column 1." The maximum permissible concentrations for air in unrestricted areas, called MPC_a^{ura}, cannot be exceeded if this control is met. Hence, the requirements of 10CFR Part 20.106(d) are met. Because the plant has never approached even a small fraction of MPC_a^{ura} limits, Control 3.3 was given a greater conservative margin by the NRC. It additionally restricts release rate monitor readings to the level at which the plant could operate continuously and not exceed the annual dose limit. The annual total body dose limit is 500 mrem (from NBS Handbook 69, Reference G, Page 6), which is the basis for the MPC_a^{ura} limits.

Exceeding the annual average total body dose rate could result in plant shutdown, especially if the operators cannot take ACTION to reduce the peak release rate.

Method I was derived from Regulatory Guide 1.109 as follows:

$$D^T = (3.17 \times 10^{+4})\ (X/Q)\ (S_F) \sum_i Q_i\ DFB$$

The equation was derived by combining Equations B-4 and B-5 from Regulatory Guide 1.109, assuming $X/Q = X/Q^D$ for noble gases, and some simplification in the notation. Assuming that $D^T_{finite} = D^T\ [X/Q]^\gamma/[X/Q]$ and that $\dot{D}_{tb} = \dot{D}^T_{finite} \cdot \dot{Q}\ (\mu Ci/sec) \cdot 31.54/Q\ (Ci/yr)$, we get:

$$\dot{D}_{tb}\ (mrem/yr) = (1.00 \times 10^{+6})\ (S_F)\ [X/Q]^\gamma \sum_i \dot{Q}_i\ DFB_i$$

Substituting:

S_F = 1.00 (shielding factor).

$[X/Q]^{\gamma}$ = Long-term average gamma dilution factor.

= 7.83×10^{-6} (sec/m^3).

$\dot{Q}_i$ = Release rate of noble gas "i" (µCi/sec).

Gives:

$$\dot{D}_{tb} \text{ (mrem/yr)} = 7.83 \sum_i \dot{Q}_i \, DFB_i \qquad \text{(Eq. 3-3)}$$

Method II cannot provide much extra realism because $\dot{D}_{tb}$ is already based on several factors which make use of current plant parameters. However, should it be needed, the dose rate analysis for the critical receptor can be performed making use of current meteorology during the time interval of the recorded peak release rate in place of the default atmospheric dispersion factor used in Method I.

3.10 Method to Calculate the Skin Dose Rate from Noble Gases

Control 3.3 limits the dose rate to the skin from noble gases at any location at or beyond the SITE BOUNDARY to 3,000 mrem/year.

By limiting the maximum $\dot{D}_{sk}$ to a rate equivalent to no more than 3,000 mrem/year, assurance is provided that the skin dose accrued in any one year by any member of the general public is much less than 3,000 mrem.

Use Method I first to calculate the skin dose rate from the peak release rate via the plant vent stack. 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 had actually been exceeded during a short time interval.

Compliance with the dose rate limits for noble gases is continuously demonstrated when effluent release rates are below the plant vent stack noble gas activity monitor alarm setpoint by virtue of the fact that the alarm setpoint is based on a value which corresponds to the off-site dose rate limit or a value below it.

Determinations of dose rate for compliance with Control 3.3 are performed when the effluent monitor alarm setpoint is exceeded, and the corrective ACTION required by Control 3.3 is unsuccessful, or s required by the notations to Table 5.3 of Control 5.2 when the stack noble gas monitor is inoperable.

3.10.1 Method I

The skin dose rate due to noble gases is:

$$\dot{D}_{skin}\,(\text{mrem/yr}) = \sum_i \dot{Q}_i\, DF_i^{'} \qquad \text{(Eq. 3-4)}$$

Where:

$\dot{Q}_i$ = The release rate from the plant vent stack (μCi/sec) of radionuclide "i" in the case of noble gases. The release rate at the plant vent stack is based on the measured radionuclide distribution in the off-gas during plant operation and the recorded total gas effluent count rate from the stack noble gas activity monitor. The release rate at the stack also can be stated in the following equation:

$$\dot{Q} = (d_i)\,(M)\left(\frac{1}{S_g}\right)(F)$$

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

Where:

M = Plant vent stack monitor count rate (cpm).

S_g = Appropriate plant vent stack monitor detector counting efficiency (cpm/(μCi/cc)).

F = Plant vent stack flow rate (cc/sec).

d_i = Fraction of the release which is radionuclide "i". This fraction can be based on the last measured value of radionuclide "i" with respect to the total noble gas activity released at the plant vent stack. It also can be based on the fraction of radionuclide "i" in the primary coolant with respect to the total noble gas primary coolant activity.

DF_i' = Skin dose factor. See Table 1.2.

During periods (beyond the first two days) when the plant is shutdown and no radioactivity release rates can be measured at the plant vent stack, Xe-133 may be used as the referenced radionuclide to determine off-site dose rate and monitor setpoints. Alternately, a relative radionuclide "i" mix fraction (f_i) may be taken from Table 5.6 as a function of time after shutdown, and substituted in place of d_i in Equation 3-10 to determine the relative fraction of each noble gas potentially available for release to the total. Just prior to plant startup, the monitor alarm setpoints should be based on Xe-138 as representing the most prevalent high dose factor noble gas expected to be present shortly after the plant returns to power. Monitor alarm setpoints which have been determined to be conservative under any plant conditions may be utilized at any time in lieu of the above assumptions.

Equation 3-4 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

a. Normal operations (not emergency event).

b. Noble gas releases via the plant vent stack to the atmosphere.

3.10.2 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 may be applied. Method II consists of the models, input data, and assumptions in Regulatory Guide 1.109, Revision 1 (Reference A), except where site-specific models, data, or assumptions are more applicable. The base case analysis is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.10.3 Basis for Method I

This section serves four purposes: (1) to document that Method I complies with appropriate NRC regulations, (2) to define the word "rate" as used in the control, (3) to provide background and training information to Method I users, and (4) to provide an introductory user's guide to Method II. The methods to calculate the skin dose rate parallel the total body dose rate methods in Section 3.9.3. Only the differences are presented here.

Method I may be used to show that Control 3.3 which limits skin dose rate from noble gases released to the atmosphere has been met for the peak noble gas release rate.

The annual skin dose limit is 3,000 mrem (from NBS Handbook 69, Reference G, Pages 5 and 6, is 30 rem/10), which is the basis for the MPC_a^{ura}.

Method I was derived from Regulatory Guide 1.109 as follows:

$$D^S = (3.17 \times 10^{+4}) \left[(X/Q)(1.11)(S_F) \sum_i Q_i \, DF_i^{\gamma} + (X/Q) \sum_i Q_i \, DFS_i\right]$$

The equation was derived by combining Equations B-4, B-5, and B-7 from Regulatory Guide 1.109, assuming that $X/Q = X/Q^D$ for noble gases, and making some simplifications in notation. Assuming that $D^{\gamma}_{finite} = D^{\gamma}\,[X/Q]^{\gamma}/[X/Q]$ and that $\dot{D}_{skin} = D^S \cdot \dot{Q}\ (\mu Ci/sec) \cdot 31.54/Q\ (Ci/yr)$ yields:

$$\dot{D}_{skin}\,(mrem/yr) = (1.11)(S_F)(1.00 \times 10^{+6})\,[X/Q]^{\gamma} \sum_i \dot{Q}_i \, DF_i^{\gamma} + (1.00 \times 10^{+6})(X/Q) \sum_i \dot{Q}_i \, DFS_i$$

Where:

$[X/Q]^{\gamma} = 7.83 \times 10^{-6}\ sec/m^3$.

$X/Q = 2.39 \times 10^{-5}\ sec/m^3$.

S_F = 1.00 (shielding factor).

Substituting gives:

$$\dot{D}_{skin}\ (mrem/yr) = 8.69 \sum_i \dot{Q}_i\ DF_i^\gamma + \sum_i \dot{Q}_i\ DFS_i\ 23.9 \times 10^{+1}$$

$$= \sum_i \dot{Q}_i\ [8.69\ DF_i^\gamma + 23.9\ DFS_i]$$

Define:

$$DF_i^{'} = 8.69\ DF_i^\gamma + 23.9\ DFS_i$$

Then:

$$\dot{D}_{skin}\ (mrem/yr) = \sum_i \dot{Q}_i\ DF_i^{'} \qquad \text{(Eq. 3-4)}$$

3.11 Method to Calculate the Critical Organ Dose Rate from Iodine-131, Tritium, and Particulates with Half-Lives Greater Than Eight Days

Control 3.3 limits the dose rate to any organ from I-131, H-3, and radionuclides in particulate form with half-lives greater than eight days to 1,500 mrem/year to any organ.

The peak release rate averaging time in the case of iodines and particulates is commensurate with the time the iodine and particulate samplers are in service between changeouts (typically a week). By limiting the maximum $\dot{D}_{CO}$ to a rate equivalent to no more than 1,500 mrem/year, assurance is provided that the critical organ dose accrued in any one year by any member of the general public will be less than 1,500 mrem.

Use Method I first to calculate the critical organ dose rate from the peak release rate via the plant vent stack. 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 had actually been exceeded during the sampling period.

3.11.1 Method I

The critical organ dose rate can be determined as follows:

$$\dot{D}_{CO} = \sum_i \dot{Q}_i \, DFG'_{ico} \qquad \text{(Eq. 3-5)}$$

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

Where:

$\dot{Q}_i$ = Stack activity release rate determination of radionuclide "i"(iodine, tritium, and particulates with half-lives greater than eight days) in µCi/sec. For i = Sr-89, Sr-90, or H-3, use the best estimates (such as most recent measurements).

DFG'_{ico} = Site-specific critical organ dose rate factor $\left(\frac{\text{mrem-sec}}{\mu\text{Ci-yr}}\right)$ for a gaseous release. See Table 1.8.

Equation 3-5 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

a. Normal operations (not emergency event),

b. Tritium, iodine, and particulate releases via the plant vent stack to the atmosphere.

As an alternative to performing Method I calculations, compliance with the critical organ dose rate limit of 1,500 mrem/year in Control 3.3.b can be shown by two methods. They are a comparison of the measured I-131 release rate to determine if it is at or below an inspection limit of 0.0125 μCi/sec, or a concentration limit in the plant vent stack equivalent to:

$$C^{PVS}\ \text{I-131}\ (\mu Ci/cc) = (2.65 \times 10^{-5})/F\ (cfm)$$

Where:

F = average plant vent stack flow rate measured during the sampling interval.

This results from the fact that I-131 is the controlling radionuclide with respect to any critical organ dose, and the selected inspection limits represent approximately ten percent of the 1,500 mrem/year dose rate limit. Measured values greater than the inspection limits should be evaluated by Equation 3-5 or a Method II assessment.

3.11.2 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 may be applied. Method II consists of the models, input data, and assumptions in Regulatory Guide 1.109, Revision 1 (Reference A), except where site-specific models, data, or assumptions are more applicable. The base case analysis is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.11.3 Basis for Method I

This section serves four purposes: (1) to document that Method I complies with appropriate NRC regulations, (2) to define the word "rate" as used in the control, (3) to provide background and training information to Method I users, and (4) to provide an introductory user's guide to Method II. The methods to calculate the critical organ dose rate parallel the total body dose rate methods in Section 3.9.3. Only the differences are presented here.

Method I may be used to show that Control 3.3.b which limits organ dose rate from Iodine-131, tritium, and radionuclides in particulate form with half-lives greater than eight days released to the atmosphere has been met for the peak Iodine-131, tritium, and particulate release rates.

The equation for $\dot{D}_{co}$ is derived by modifying Equation 3-8 from Section 3.14 as follows:

$$D_{co} = \sum_i Q_i \, DFG_{ico} \qquad \text{(Eq. 3-8)}$$

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

Applying the conversion factor, 31.54 (Ci-sec/µCi-yr), and converting Q to $\dot{Q}$ in µCi/sec yields:

$$\dot{D}_{co} = 31.54 \sum_i \dot{Q}_i \, DFG_{ico}$$

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

Equation 3-5 is rewritten in the form:

$$\dot{D}_{co} = \sum_i \dot{Q}_i \, DFG'_{ico}$$

Where:

$$DFG'_{ico} = (DFG_{ico})\ (31.54)$$

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

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

3.12 Method to Calculate the Gamma Air Dose from Noble Gases

Control 3.4 limits the gamma dose to air from noble gases at any location at or beyond the SITE BOUNDARY to 5 mrad in any quarter and 10 mrad in any year. Dose evaluation is required at least once per 31 days.

Use Method I first to calculate the gamma air dose for the plant vent stack releases during the period. Method I applies at all dose levels.

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

3.12.1 Method I

The gamma air dose from plant vent stack releases is:

$$D^{\gamma}_{air}(\text{mrad}) = 0.25 \sum_i Q_i \, DF^{\gamma}_i \qquad \text{(Eq. 3-6)}$$

Where:

Q_i = Total activity (Curies) released to the atmosphere via the plant vent stack of each radionuclide "i" during the period of interest.

DF^{γ}_i = Gamma dose factor to air for radionuclide "i". See Table 1.2.

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

a. Normal operations (not emergency event).

b. Noble gas releases via the plant vent stack to the atmosphere.

3.12.1.1 Ground Level Releases

For ground level releases, the gamma air dose is:

$$D^{\gamma}_{grd} \text{ (mrad)} = (1.23 \times 10^{-4}) (Q_{Xe\text{-}133 \text{ equivalent}}) \qquad \text{(Eq. 3-6.1)}$$

Where:

$Q_{Xe\text{-}133 \text{ equivalent}}$ =
The Xe-133 equivalent of all the noble gases in the release (Curies) and is based on the dose conversion factors (DF^{γ}_{i}) as listed in Table 1.2 of the ODCM.

3.12.2 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 may be applied. Method II consists of the models, input data, and assumptions in Regulatory Guide 1.109, Revision 1 (Reference A), except where site-specific models, data, or assumptions are more applicable. The base case analysis is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.12.3 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 Control 3.4, which limits the off-site gamma air dose from gaseous effluents, has been met for releases over appropriate periods. Control 3.4 is based on 10CFR Part 50, Appendix I, Subsection B.1, which limits the estimated annual gamma air dose at unrestricted area locations. Exceeding the annual gamma air dose at unrestricted area locations does not immediately limit plant operation, but requires a report to the NRC within 30 days.

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

$$D^{\gamma}_{air}(mrad) = 3.17 \times 10^{+4} \left(\frac{pCi\text{-}yr}{Ci\text{-}sec}\right) [X/Q]^{\gamma} (sec/m^3) \sum_i Q_i \ (Ci) \ DF^{\gamma}_i \left(\frac{mrad\text{-}m^3}{pCi\text{-}yr}\right)$$

Where:

$[X/Q]^{\gamma}$ = Long-term average gamma dilution factor.

= 7.83×10^{-6} (sec/m^3).

Q_i = Number of Curies of noble gas "i" released.

Which leads to:

$$D^{\gamma}_{air} \ (mrad) = 0.25 \sum_i Q_i \ DF^{\gamma}_i \qquad \text{(Eq. 3-6)}$$

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

The gamma air dose from a ground level release is determined by using the same Regulatory Guide 1.109 equation to derive Equation 3-6. The only differences are:

$[X/Q]^{\gamma}$ = 1.10×10^{-5} sec/m^3, which is the long-term average ground level $[X/Q]^{\gamma}$ based on the time period from May 1977 through April 1982.

$Q_{Xe\text{-}133 \ equivalent}$ = The Xe-133 equivalent of all the noble gases in the equivalent release (Curies).

$DF^{\gamma}_i = DF^{\gamma}_{Xe\text{-}133} = 3.53 \times 10^{-4} \left(\frac{mrad\text{-}m^3}{pCi\text{-}yr}\right)$ obtained from Table 1.2 to account for the release being expressed in terms of the Xe-133 equivalent.

Substituting the above into the Regulatory Guide 1.109 general equation gives:

$$D^{\gamma}_{grd}\ (\text{mrad}) = (1.23 \times 10^{-4})\ (Q_{Xe\text{-}133\ equivalent}) \qquad \text{(Eq. 3-6.1)}$$

3.13 Method to Calculate the Beta Air Dose from Noble Gases

Control 3.4 limits the beta dose to air from noble gases at any location at or beyond the SITE BOUNDARY to 10 mrad in any quarter and 20 mrad in any year. Dose evaluation is required at least once per 31 days.

Use Method I first to calculate the beta air dose for the plant vent stack releases during the period. Method I applies at all dose levels.

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

3.13.1 Method I

The beta air dose from plant vent stack releases is:

$$D^{\beta}_{air}\ (\text{mrad}) = 0.76 \sum_i Q_i\ DF^{\beta}_i \qquad \text{(Eq. 3-7)}$$

Where:

DF^{β}_i = Beta dose factor to air for radionuclide "i". See Table 1.2.

Q_i = Total activity (Curies) released to the atmosphere via the plant vent stack of each radionuclide "i" during the period of interest.

Equation 3-7 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

a. Normal operations (not emergency event),

b. Noble gas releases via the plant vent stack to the atmosphere.

3.13.1.1 Ground Level Releases

For ground level releases, the beta air dose can be determined by using Equation 3-7. Equation 3-7 results in doses that are approximately ten percent more conservative than calculating releases using ground level methodology.

3.13.2 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 may be applied. Method II consists of the models, input data, and assumptions in Regulatory Guide 1.109, Revision 1 (Reference A), except where site-specific models, data, or assumptions are more applicable. The base case analysis is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.13.3 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 the beta air dose parallel the gamma air dose methods in Section 3.12.3. Only the differences are presented here.

Method I may be used to show that Control 3.4, which limits the off-site beta air dose from gaseous effluents, has been met for releases over appropriate periods. Control 3.4 is based on 10CFR Part 50, Appendix I, Subsection B.1, which limits the estimated annual beta air dose at unrestricted area locations. Exceeding the estimated annual beta air dose at unrestricted area locations does not immediately limit plant operation, but requires a report to the NRC within 30 days.

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

$$D_{air}^{\beta}\ (mrad) = (3.17 \times 10^{+4})\ [X/Q] \sum_i Q_i\ DF_i^{\beta}$$

Substituting:

$$X/Q = 2.39 \times 10^{-5}\ sec/m^3$$

We have:

$$D_{air}^{\beta}\ (mrad) = 0.76 \sum_i Q_i\ DF_i^{\beta} \qquad \text{(Eq. 3-7)}$$

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

Control 3.5 limits the critical organ dose to a MEMBER OF THE PUBLIC from radioactive tritium, Iodine-131, and particulates with half-lives greater than eight days in gaseous effluents to 7.5 mrem per quarter and 15 mrem per year. Control 3.2 limits the total body and organ dose to any real MEMBER OF THE PUBLIC from all station sources (including gaseous effluents) to 25 mrem in a year except for the thyroid, which is limited to 75 mrem in a year.

Use Method I first to calculate the critical organ dose from a vent stack release as it is simpler to execute and more conservative than Method II.

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), or if Method I cannot be applied.

3.14.1 Method I

The critical organ dose from a gaseous release is:

$$D_{co} \atop (mrem) = \sum_i Q_i \, DFG_{ico} \qquad \text{(Eq. 3-8)}$$

Where:

Q_i = Total activity (Curies) released to the atmosphere of radionuclide "i" during the period of interest. For i = Sr-89, Sr-90, or H-3, use the best estimates (such as the most recent measurements).

DFG_{ico} = Site-specific critical organ dose factor (mrem/Ci) for a gaseous release. See Table 1.8.

Equation 3-8 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

a. Normal operations (not emergency event).

b. Iodine, tritium, and particulate releases via the plant vent stack to the atmosphere.

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

3.14.2 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, Revision 1 (Reference A), except where site-specific models, data, or assumptions are more applicable. The base case analysis, documented below, is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.14.3 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 Controls 3.2 and 3.5, which limit off-site organ dose from gases, have been met for releases over the appropriate periods. These controls are based on requirements and standards in 10CFR and 40CFR. Control 3.5 is based on the ALARA requirements in 10CFR Part 50, Appendix I, Subsection II C. Control 3.2 is based on Environmental Standards for Uranium Fuel Cycle in 40CFR190 (hereafter called the standard) which applies to direct radiation as well as to liquid and gaseous effluents. These methods apply only to iodine, tritium, and particulates in gaseous effluents.

Exceeding the ALARA requirements 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" (10CFR Part 50, Appendix I). The use 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 Controls 3.2 and 3.5, annual average dilution factors are appropriate for batch and continuous releases. The 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, DFG_{ico} (mrem/Ci), for a one Curie release of each iodine, tritium, and particulate radionuclide to gaseous effluents was derived. Then Method I was determined using simplifying and further conservative assumptions. The base case analysis uses the methods, data, and assumptions in Regulatory Guide 1.109 (Equations C-2, C-4, and C-13 in Reference A). Tables 3.4 and 3.5 outline human consumption and environmental parameters used in the analysis. It is conservatively assumed that the critical receptor lives at the "maximum SITE BOUNDARY dilution factor location" as defined in Section 3.15.

For any gas release during any period, the dose from radionuclide "i" is:

$$D_{ico} = (DFG_{ico})\ (Q_i)$$

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 Met od II in the region of the objective limits because it is based on the following reduction of the base case. The dose factors, DFG_{ico}, used in Method I were chosen from the base case to be the highest of the four age groups for that radionuclide. In effect, each radionuclide is conservatively represented by its own critical age group and critical organ.

TABLE 3.4

Age-Specific Usage Factors
(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	8,000.00
Teen	630.00	42.00	400.00	65.00	8,000.00
Child	520.00	26.00	330.00	41.00	3,700.00
Infant	0.00	0.00	330.00	0.00	1,400.00

TABLE 3.5

Environmental Parameters for Gaseous Effluents at the Yankee Plant
(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)	2.00	2.00	0.70	2.00	0.70	2.00	0.70	2.00
P	Soil Surface Density	(kg/m^2)	240.00	240.00	240.00	240.00	240.00	240.00	240.00	240.00
T	Transport Time to User	(hrs)	-	-	[illegible]	48.00	48.00	48.00	480.00	480.00
TB	Soil Exposure Time	(hrs)	131400.00	131400.00	131400.00	131400.00	131400.00	131400.00	131400.00	131400.00
TF	Crop Exposure Time to Plume	(hrs)	1440.00	1440.00	720.00	1440.00	720.00	1440.00	720.00	1440.00
TH	Holdup After Harvest	(hrs)	1440.00	24.00	0.00	2160.00	0.00	2160.00	0.00	2160.00
QF	Animals Daily Feed	(kg/day)	-	-	50.00	50.00	6.00	6.00	50.00	50.00
FP	Fraction of Year on Pasture		-	-	0.50	-	0.50	-	0.50	-

*Pathway is not included in Method I. It is listed for informational purposes and the possible use in a Method II analysis.

TABLE 3.5
(Continued)

Environmental Parameters for Gaseous Effluents at the Yankee Plant
(Derived from Reference A)

Variable		Vegetables		Cow Milk		Goat Milk*		Meat	
		Stored	Leafy	Pasture	Stored	Pasture	Stored	Pasture	Stored
FS Fraction Pasture When on Pasture		-	-	1.00	-	1.00	-	1.00	-
FG Fraction of Stored Veg. Grown in Garden		0.76	-	-	-	-	-	-	-
FL Fraction of Leafy Veg. Grown in Garden		-	1.00	-	-	-	-	-	-
FI Fraction Elemental Iodine = 0.5		-	-	-	-	-	-	-	-
H Absolute Humidity = 5.6	(gm/m^3)	-	-	-	-	-	-	-	-

*Pathway is not included in Method I. It is listed for informational purposes and the possible use in a Method II analysis.

3.15 Critical Receptors and Long-Term Average Atmospheric Dispersion Factors for Important Exposure Pathways

The gaseous effluent dose equations (Method I) have been simplified by assuming an individual whose behavior and living habits inevitably lead to a higher dose than anyone else. The following pathways of exposure to gaseous effluents as listed in Regulatory Guide 1.109 (Reference A) have been considered. They are:

a. Direct exposure to contaminated air,

b. Direct exposure to contaminated ground,

c. Inhalation of air,

d. Ingestion of vegetables,

e. Ingestion of cow milk, and

f. Ingestion of meat.

Section 3.15.1 details the selection of important off-site locations and receptors; Section 3.15.2 describes the atmospheric model used to convert meteorological data into dispersion factors; and Section 3.15.3 contains the resulting descriptions of the critical receptors and their dispersion factors as a function of exposure pathway.

3.15.1 Critical Receptors

The most limiting SITE BOUNDARY location in which individuals are or are likely to be located was assumed to be the receptor for all the gaseous pathways considered. This provides a conservative estimate of the dose to an individual from existing and potential gaseous pathways for the Method I analysis.

This point is the SSE sector, 800 meters.

3.15.2 Yankee Atmospheric Dispersion Model

The annual average dispersion factors are computed for routine (long-term) releases using the Yankee Atomic Electric Company's (YAEC) AEOLUS (Reference B) computer code.

AEOLUS produces the following annual average dispersion factors for each location:

a. X/Q, nondepleted dispersion factors for evaluating ground level concentrations;

b. $[X/Q]^D$, depleted dispersion factors for evaluating ground level concentrations of iodines and particulates;

c. X/Q^{γ}, effective gamma dispersion factors for evaluating gamma dose rates from a sector-averaged finite cloud (multiple-energy, undepleted source); and

d. D/Q, deposition factors for dry deposition of elemental radioiodines and other particulates.

The AEOLUS diffusion model is described in the AEOLUS manual (Reference B). AEOLUS is based, in part, on the straight-line airflow model as discussed in Regulatory Guide 1.111 (Reference C).

One difference is that the gamma dose rate is calculated throughout this ODCM using the finite cloud model presented in Meteorology and Atomic Energy 1968 (Reference H, Section 7-5.2.5). That model is implemented through the definition (Reference B, Section 6) of an effective gamma dispersion factor, X/Q^{γ}, and the replacement of X/Q in infinite cloud dose equations by the X/Q^{γ}.

The other difference is that the relatively narrow valley in which the plant sits is considered by the model. Wind channelling is assumed to occur in the seven sectors which make up the valley. The seven sectors are SSE, S, SSW, SW, WSW, W, and WNW. If a receptor location is in one of the valley sectors, the contributions from the other six valley sectors are averaged into the particular valley receptor. This is done for distances greater than 500 meters from the primary vent stack where the valley effects are assumed to cause channelling.

3.15.3 Long-Term Average Dispersion Factors for Critical Receptors

Actual measured meteorological data for the five-year period, January 1981 through December 1985, was analyzed to determine the locations of the maximum off-site atmospheric dispersion factors. Each dose and dose rate calculation incorporates the maximum applicable off-site, long-term average atmospheric dispersion factor. The values used and their locations are summarized in Table 3.6.

TABLE 3.6

Yankee Nuclear Power Station Five-Year Average Atmospheric Dispersion Factors*

	Dose Rate to Individual			Dose to Air		Dose to Critical Organ
	Total Body	Skin	Critical Organ	Gamma	Beta	Thyroid
X/Q Depleted $\left(\frac{sec}{m^3}\right)$	-	-	2.19×10^{-6}	-	-	2.19×10^{-6}
X/Q Undepleted $\left(\frac{sec}{m^3}\right)$	-	2.39×10^{-6}	-	-	2.39×10^{-6}	-
D/Q $\left(\frac{1}{m^2}\right)$	-	-	5.02×10^{-8}	-	-	5.02×10^{-8}
X/Q^{γ} $\left(\frac{sec}{m^3}\right)$	7.83×10^{-6}	7.83×10^{-6}	-	7.83×10^{-6}	-	-

*SSE SITE BOUNDARY, 800 meters from the primary vent stack.

3.16 Method to Calculate Direct Dose from Plant Operation

Control 3.2 restricts the dose to the whole body and any organ of any real MEMBER OF THE PUBLIC from all station sources (including direct radiation from the reactor and outside storage tanks, which is called the direct dose) to the limit of 25 mrem in a year, except for the thyroid which is limited to 75 mrem in a year. A determination of the need to conduct a total dose evaluation is required at least every 31 days.

Use Method I first to calculate the direct dose contribution to the whole body and any organ as it is simpler to execute and more conservative than Method II.

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

3.16.1 Method I

The maximum contribution of direct dose to the whole body or to any organ is:

$$D_d = (0.057 + \dot{E}_r)\,(T_e)\,(0.00087) \qquad \text{(Eq. 3-9)}$$

Where:

0.00087 = Conversion factor (mrem/μR).

T_e = Length of exposure period in hours.

$\dot{E}_r$ = Exposure rate at critical receptor from non-vapor container sources as measured or estimated for the period.

Equation 3-9 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

a. Normal operations (not emergency event).

b. All significant remaining sources are considered in $\dot{E}_r$.

c. Any exposure period.

3.16.2 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 measurement and site-specific models, data, and assumptions. The base case analysis is a good example of the use of Method II. It is an acceptable starting point for a Method II analysis.

3.16.3 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 Control 3.2, which limits direct dose off-site, has been met for any exposure period. Control 3.2 is based on the standard (40CFR190) which applies to direct sources of radiation as well as liquid and gaseous effluents. Method I applies to the direct sources only.

Exceeding 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. This is unlike exceeding 10CFR Part 20 limits which could result in plant shutdown.

Method I is developed by reducing the base case (a Method II analysis) using conservative assumptions. The base case involves the choice of a critical receptor and the development of an exposure factor for the vapor container source, $\dot{E}_{VC}$ (μR/hour operation). The critical receptor is the nearest resident who lives 450m from the vapor container centerline in the NNW direction. An occupancy of 1.00 is assumed.

The exposure factor, $\dot{E}_{VC}$, is developed by extrapolating measurements made close to the plant out to the critical receptor. All significant sources of direct radiation on-site are shielded by buildings and tanks from the critical receptor, with the exception of the vapor container and one of the liquid waste storage tanks (TK-31).

The dose (mrem) to the critical receptor, D_d, over the exposure period (in hours), T_e, is related simplistically to the exposure rate from the vapor container in μR/hour, $\dot{E}_{VC}$, and the exposure rate from remaining sources, $\dot{E}_r$ by the following equation:

$$D_d = (\dot{E}_{VC} + \dot{E}_r)(T_e)(0.00087) \qquad \text{(Eq. 3-11)}$$

What remains is to conservatively derive:

$\dot{E}_{VC}$ and $\dot{E}_r$

The dose from the vapor container is due to fission and activation gases which build up in plant systems and the vapor container during operation. Those sources decay or are ventilated through the Ventilation Exhaust Filtration System at the beginning of refueling outages.

This is done to allow worker access to the vapor container to reload the core. The estimate of $\dot{E}_{VC}$ is based on the extrapolation of measurements made during plant operation at the restricted area fence, $\dot{E}_1$, compared to background measurements made during a refueling outage after containment PURGE at the same locations, $\dot{E}_b$. $\dot{E}_{VC}$ is expected to remain constant over the years, so it can be estimated here as a function of exposure period.

Although regular measurements of direct radiation are made near the critical receptor as part of the Environmental Surveillance Program, the majority of the measured doses are due to natural background and fallout, variations which entirely obscure plant contributions to dose. Because they are closer to the sources, the measurement of direct radiation using TLDs at the restricted area fence can be extrapolated with greater net sensitivity (about 10 mR/year). However, the most sensitive method proved to be exposure rate measurements made with a high pressure ionization chamber, which had a history of 1 μR/hour plus or minus 1 μR/hour 95 percent confidence interval for exposure rates near the background rate for the procedure used (20 replicate measurements and periodic instrument checks). This extrapolates to approximately a plus or minus one mrem/year 95 percent confidence interval at the critical receptor.

$\dot{E}_{VC}$ is estimated using the following equation:

$$\dot{E}_{VC} = \frac{d_o^2}{d^2} (\dot{E} - \dot{E}_b) \qquad \text{(Eq. 3-11.2)}$$

Where:

d = Distance to the critical receptor from the vapor container centerline.

d_o = Distance to exposure measurement from the vapor container centerline.

$\dot{E}$ = Exposure rate (μR/hr) measurement during plant operation.

$\dot{E}_b$ = Exposure rate (μR/hr) measurement during plant outage.

$\dot{E}_{VC}$ is derived from data collected in 1981 and presented in Table 3.7 The mean value of $\dot{E}_{VC}$ for measurements at each of the nine TLD locations at the restricted area fence is 0.057 μR/hour. The mean value is used because it is insensitive to miscellaneous on-site sources which contribute to the measurements, but not to the dose at the critical receptor.

$\dot{E}_r$ will have to be made from measurements or estimates made for the specific exposure period.

Substituting the derived value of $\dot{E}_{VC}$ into Equation 3-11.1 yields:

$$D_d = (0.057 + \dot{E}_r)\ (T_e)\ (0.00087) \qquad \text{(Eq. 3-9)}$$

TABLE 3.7

Estimate of Exposure Rate at Critical Receptor from Vapor Container Shine, $\dot{E}_{VC}$, Made in Spring 1981

Monitoring Station No.	d_o (km)	Direction	$\dot{E}$ (μR/hr)	$\dot{E}_b$ (μR/hr)	$\dot{E}_{VC}$ (μR/hr)
13	0.08	225°	20.4+	18.0*	0.076
14	0.11	300°	15.0	13.2*	0.108
15	0.08	345°	14.4	13.1*	0.041
16	0.13	30°	17.4	16.6*	0.067
17	0.14	70°	15.9	15.6**	0.029
18	0.14	115°	24.9	24.6**	0.029
19	0.16	140°	23.2	23.3**	0.013
20	0.16	160°	19.4	18.8**	0.076
21	0.11	205°	20.3	18.7**	0.096
				Total =	0.509

Average = 0.057 μR/hr

+ All measurements of $\dot{E}$ taken April 28, 1981. Average daily power level was 97.5 percent MWe-net, five days before shutdown at End-of-Cycle 13/14.

* Measurements of $\dot{E}_b$ taken May 13, 1981 during outage. Containment PURGED.

** Measurements of $\dot{E}_b$ taken June 2, 1981 during shutdown. Containment PURGED.

4.0 RADIOLOGICAL ENVIRONMENTAL MONITORING

4.1 Monitoring Program

Control 4.1 In accordance with Yankee Technical Specification 6.8.5.b.1, the Radiological Environmental Monitoring Program shall be conducted as specified in Table 4.1.

Applicability

At all times.

ACTION

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 4.1, prepare and submit to the Commission in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

b. 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 Table 4.1 exceeding the reporting levels of Table 4.2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the receipt of the laboratory analyses, pursuant to Control 7.4, a Special Report which includes an evaluation of any release conditions, environmental factors, or other aspects which caused the limits of Table 4.2 to be exceeded. When more than one of the radionuclides in Table 4.2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 4.2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal or greater than the calendar year limits of Controls 3.1, 3.3, and 3.4. This report is not required if the measured level of radioactivity was not the result of plant effluents, however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetation samples no longer available from one or more sample locations required by Table 4.1, identify new location(s), if available, for obtaining replacement samples and add them to the Radiological Environmental Monitoring Program within 30 days. The specific location(s) from which samples were no longer available may then be deleted from the monitoring program. Pursuant to Control 7.2, identify the cause of the samples no longer being available and identify the new location(s) for obtaining available replacement samples in the next Semiannual Radioactive Effluent Release Report and include revised ODCM figure(s) and table(s) reflecting the new location(s).

d. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 4.1 The radiological environmental monitoring samples shall be collected pursuant to Table 4.1 from the locations given in the ODCM and shall be analyzed pursuant to the requirements of Table 4.1 and the detection capabilities required by Table 4.3.

Bases

The Radiological Environmental Monitoring Program required by Control 4.1 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. The monitoring program implements Section IV.B.2 of Appendix I, 10CFR 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. Guidance for the monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. Program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. This does not preclude the calculation of an a posteriori LLD for a particular measurement based upon the actual parameters for the sample in question.

TABLE 4.1

Radiological Environmental Monitoring Program*

Exposure Pathway and/or Sample	Number of Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Radioiodine and Particulates	5	Continuous operation of sampler with sample collection as required by dust loading, but at least once per week.	Radioiodine canister. Analyze weekly for I-131. Particulate sampler: Gross beta radioactivity following filter change. Composite (by location) for gamma isotopic at least once per quarter.
2. DIRECT RADIATION			
	a. 22	Quarterly	Gamma dose, at least once per quarter.
	b. 16	Quarterly	Incident response TLDs in the outer (four to five mile) ring, de-dose only quarterly unless gaseous release control was exceeded during period.
3. WATERBORNE			
a. Surface	2	Composite sample collected over a period of one month.**	Gross beta and gamma isotopic analysis of each sample. Tritium analysis of composite sample at least once per quarter.
b. Ground	2	At least once per quarter.	Gamma isotopic and tritium analyses of each sample.

TABLE 4.1
(Continued)

Radiological Environmental Monitoring Program*

Exposure Pathway and/or Sample	Number of Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Sediment from Shoreline	1	At least once per six months.	Gamma isotopic analysis of each sample.
4. INGESTION			
a. Milk	3	At least once per two weeks when animals are on pasture***; at least once per month at other times.	Gamma isotopic and I-131 analysis of each sample.
b. Fish	2	Commercially and recreationally important species. Seasonal or semiannually, if not seasonal.	Gamma isotopic analysis on edible portions.
c. Food Products	3	At time of harvest. One sample of any of the following classes of food products: 1. Tuberous vegetable 2. Above ground vegetable 3. Fruit	Gamma isotopic analysis in edible portions.
	1	At time of harvest. One sample of broad leaf vegetation.	I-131 analysis.

* Specific sample locations for all media are specified in the ODCM and reported in the Annual Radiological Environmental Operating Report.

** Composite samples shall be obtained by collecting an aliquot at intervals not exceeding two hours.

*** Grazing season extends from June 1 to November 1.

TABLE 4.2

Reporting Levels for Radioactivity Concentrations in Environmental Samples

Analysis	Water* (pCi/l)	Airborne Particulates or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	$3 \times 10^{+4}$	-	-	-	-
Mn-54	$1 \times 10^{+3}$	-	$3 \times 10^{+4}$	-	-
Fe-59	$4 \times 10^{+2}$	-	$1 \times 10^{+4}$	-	-
Co-58	$1 \times 10^{+3}$	-	$3 \times 10^{+4}$	-	-
Co-60	$3 \times 10^{+2}$	-	$1 \times 10^{+4}$	-	-
Zn-65	$3 \times 10^{+2}$	-	$2 \times 10^{+4}$	-	-
Zr-Nb-95	$4 \times 10^{+2}$	-	-	-	-
I-131	$2 \times 10^{+0}$	9×10^{-1}	-	$3 \times 10^{+0}$	$1 \times 10^{+2}$
Cs-134	$3 \times 10^{+1}$	$1 \times 10^{+1}$	$1 \times 10^{+3}$	$6 \times 10^{+1}$	$1 \times 10^{+3}$
Cs-137	$5 \times 10^{+1}$	$2 \times 10^{+1}$	$2 \times 10^{+3}$	$7 \times 10^{+1}$	$2 \times 10^{+3}$
Ba-La-140	$2 \times 10^{+2}$	-	-	$3 \times 10^{+2}$	-

*Reporting levels for nondrinking water pathways.

TABLE 4.3

Detection Capabilities for Environmental Sample Analysis[a][d]

Analysis[e]	Water (pCi/l)	Airborne Particulates or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
Gross beta	$4 \times 10^{+0}$	1×10^{-2}	-	-	-	-
H-3	$2 \times 10^{+3}$	-	-	-	-	-
Mn-54	$1.5 \times 10^{+1}$	-	$1.3 \times 10^{+2}$	-	-	-
Fe-59	$3 \times 10^{+1}$	-	$2.6 \times 10^{+2}$	-	-	-
Co-58, -60	$1.5 \times 10^{+1}$	-	$1.3 \times 10^{+2}$	-	-	-
Zn-65	$3 \times 10^{+1}$	-	$2.6 \times 10^{+2}$	-	-	-
Zr-Nb-95	$1.5 \times 10^{+1}$[c]	-	-	-	-	-
I-131	$1 \times 10^{+0}$[b]	7×10^{-2}	-	$1 \times 10^{+0}$	$6 \times 10^{+1}$[g]	-
Cs-134	$1.5 \times 10^{+1}$	5×10^{-2}	$1.3 \times 10^{+2}$	$1.5 \times 10^{+1}$	$6 \times 10^{+1}$	$1.5 \times 10^{+2}$
Cs-137	$1.8 \times 10^{+1}$	6×10^{-2}	$1.5 \times 10^{+2}$	$1.8 \times 10^{+1}$	$8 \times 10^{+1}$	$1.8 \times 10^{+2}$
Ba-La-140	$1.5 \times 10^{+1}$[c][f]	-	-	$1.5 \times 10^{+1}$[c][f]	-	-

TABLE 4.3
(Continued)

Table 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 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{(4.66)\ (S_b)}{(E)\ (V)\ (2.22)\ (Y)\ [Exp(-\lambda\ \Delta t)]}$$

Where:

LLD = A priori lower limit of detection as defined above (microcuries or picocuries per unit mass or volume).

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

E = Counting efficiency (counts per disintegration).

V = Sample size (units of mass or volume).

2.22 = Number of disintegrations per minute per picocurie.

Y = Fractional radiochemical yield (when applicable).

λ = Radioactive decay constant for the particular radionuclide.

Δt = Elapsed time between sample collection and analysis.

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

TABLE 4.3
(Continued)

Table Notation

contributions of other radionuclides normally present in the samples (e.g., Potassium-40 in milk 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 radionuclides, or other uncontrollable circumstances may render these LLDs unavailable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. This does not preclude the calculation of an a posteriori 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.

b. LLD for drinking water.

c. Parent only.

d. 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.

e. This list does not mean that only these radionuclides are to be considered. Other peaks that are identifiable, together with those of the listed radionuclides, also shall be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Control 7.1.

f. The Ba-140 LLD and concentration can be determined by the analysis of its short-lived daughter product, La-140, subsequent to an

TABLE 4.3
(Continued)

Table Notation

eight-day period following collection. The calculation shall be predicted on the normal ingrowth equations for a parent-daughter situation and the assumption that any unsupported La-140 in the sample would have decayed to an insignificant amount (at least 3.6 percent of its original value). The ingrowth equations will assume that the supported La-140 activity at the time of collection is zero.

g. LLD for leafy vegetation.

4.2 Land Use Census

Control 4.2 In accordance with Yankee Technical Specification 6.8.5.b.2, a land use census shall be conducted to identify the location of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables 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 a location(s) which yields at least a 20 percent greater dose or dose commitment than the values currently being calculated in SR 3.4, identify the new location(s) in the next Semiannual Effluent Release Report.

b. With a land use census identifying a location(s) which yields 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 4.1, 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. Identify the new location(s) in the next Semiannual Effluent Release Report.

c. The provisions of Controls 1.3 and 1.4 are not applicable.

---

* In lieu of the garden census, broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q.

Surveillance Requirement

SR 4.2 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1 by either a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Control 7.1.

Bases

Control 4.2 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 the land use census. The census satisfies the requirements of Section IV.B.3 of Appendix I, 10CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used: (1) 20 percent of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage) and (2) a vegetation yield of 2 kg/square meter. In lieu of the garden census, broad leaf vegetation samples from the SITE BOUNDARY in the direction sector with the highest D/Q may be substituted. The use of the maximum off-site D/Q value predicted for gaseous effluents from the plant stack (the plant stack does not qualify for an elevated release as defined in Regulatory Guide 1.111, March 1976) will generate the maximum possible calculated dose, and thus, no real garden located at any other point could have a greater calculated dose or dose commitment.

The addition of new sampling locations to Control 4.1, based on the land use census, is limited to those locations which yield a calculated dose or dose commitment 20 percent greater than 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, contribute 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 radionuclides.

4.3 Intercomparison Program

Control 4.3 In accordance with Yankee Technical Specification 6.8.5.b.3, analyses shall be performed on referenced radioactive materials supplied as part of the Intercomparison Program which has been approved by the NRC.

Applicability

At all times.

ACTION

a. With analyses 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.

b. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 4.3 A summary of the results of analyses performed as part of the above required Intercomparison Program shall be included in the Annual Radiological Environmental Operating Report. The identification of which NRC-approved Intercomparison Program that is being participated in shall be stated in the ODCM.

Bases

The control for participation in the 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. The independent checks are completed 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, 10CFR Part 50.

4.4 Environmental Monitoring Locations

The radiological environmental monitoring stations are listed in Table 4.4. The locations of these stations with respect to the Yankee plant facility are shown on the maps in Figures 4-1 through 4-7.

All routine radiological analyses for environmental samples are performed at the Yankee Environmental Laboratory. The laboratory participates in the U.S. Environmental Protection Agency's Environmental Radioactivity Laboratory Intercomparison Studies Program for all the available species and matrices routinely analyzed.

TABLE 4.4

Radiological Environmental Monitoring Stations

Exposure Pathway and/or Sample	Sample Location and Designated Code+		Distance From the Plant (km)	Direction From Plant
1. AIRBORNE (Radioiodine and Particulates)				
	AP/CF-11	Observation Stand	0.50	NW
	AP/CF-12	Monroe Bridge	1.10	SW
	AP/CF-13	Rowe School	4.20	SE
	AP/CF-14	Harriman Power Station	3.20	N
	AP/CF-21	Williamstown, MA	22.20	W
2. WATERBORNE				
a. Surface	WR-11	Bear Swamp Lower Reservoir	6.30	Downriver
	WR-21	Harriman Reservoir	10.10	Upriver
b. Ground	WG-11	Plant Potable	On-Site Well	
	WG-12	Sherman Spring	0.20	NW
c. Sediment From Shoreline	SE-11	Number 4 Station	36.20	Downriver
	SE-21	Harriman Reservoir	10.10	Upriver
3. INGESTION				
a. Milk	TM-13	Whitingham, VT	8.40	ENE
	TM-12	Readsboro, VT	6.10	N
	TM-21	Williamstown, MA	21.00	WSW
b. Fish and Inverter-brates	FH-11	Sherman Pond	1.50	At Discharge Point
	FH-21	Harriman Reservoir	10.10	Upriver
c. Food Products	TF-11	Monroe Bridge	1.30	SW
	TF-13	Monroe, MA	1.90	WNW
	TF-21	Williamstown, MA	21.00	WSW
	TV-11	Monroe Bridge**	1.30	SW

Radiological Environmental Monitoring Stations*

Exposure Pathway and/or Sample	Sample Location and Designated Code+		Distance From the Plant (km)	Direction From Plant
4. DIRECT RADIATION				
	GM-1	Furlon House	0.80	SW
	GM-2	Observation Stand	0.50	NW
	GM-3	Rowe School	4.20	SE
	GM-4	Harriman Station	3.20	N
	GM-5	Monroe Bridge	1.10	SW
	GM-6	Readsboro Road Barrier	1.30	N
	GM-7	Whitingham Line	3.50	NE
	GM-8	Monroe Hill Barrier	1.80	S
	GM-9	Dunbar Brook	3.20	SW
	GM-10	Cross Road	3.50	E
	GM-11	Adams High Line	2.10	WNW
	GM-12	Readsboro, VT	5.50	NNW
	GM-13	Restricted Area Fence	0.08	WSW
	GM-14	Restricted Area Fence	0.11	WNW
	GM-15	Restricted Area Fence	0.08	NNW
	GM-16	Restricted Area Fence	0.13	NNE
	GM-17	Restricted Area Fence	0.14	ENE
	GM-18	Restricted Area Fence	0.14	ESE
	GM-19	Restricted Area Fence	0.16	SE
	GM-20	Restricted Area Fence	0.16	SSE
	GM-21	Restricted Area Fence	0.11	SSW
	GM-22	Heartwellville	12.60	NNW
	GM-23	Williamstown Substation	22.20	W
	GM-24	Harriman Dam	7.30	N
	GM-25	Whitingham, VT	7.70	NNE
	GM-26	Sadoga Road	7.60	NE
	GM-27	Number 9 Road	7.60	ENE
	GM-28	Number 9 Road	6.00	E
	GM-29	Route 8A	8.20	ESE
	GM-30	Route 8A	9.40	SE
	GM-31	Legate Hill Road	7.60	SSE
	GM-32	Rowe Road	7.90	S
	GM-33	Zoar Road	6.90	SSW
	GM-34	Fife Brook Road	6.40	SW
	GM-35	Whitcomb Summit	8.60	WSW
	GM-36	Tilda Road	6.60	W
	GM-37	Turner Hill Road	6.70	WNW
	GM-38	West Hill Road	6.60	NW
	GM-39	Route 100	6.80	NNW
	GM-40	Readsboro Road	0.50	W

* Sample locations are shown on Figures 4-1 through 4-7.
** TV-11 Station is for leafy vegetables.
\+ Station 1X's are indicator stations, and Station 2X's are control stations (excluding the direct radiation stations).

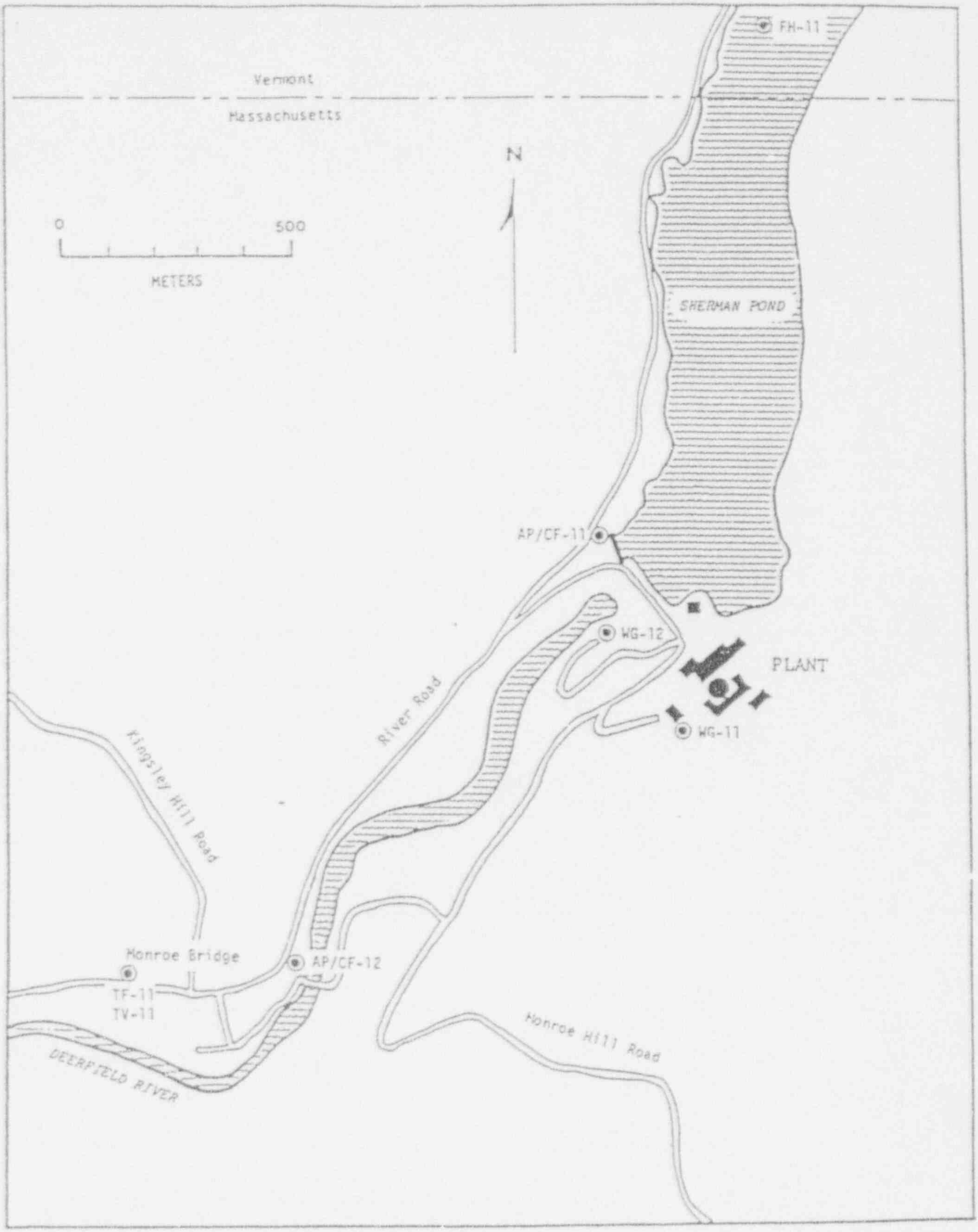


Figure 4-1 Yankee Plant Radiological Environmental Monitoring Locations Within 1 Mile (Airborne, Waterborne and Ingestion Pathways)

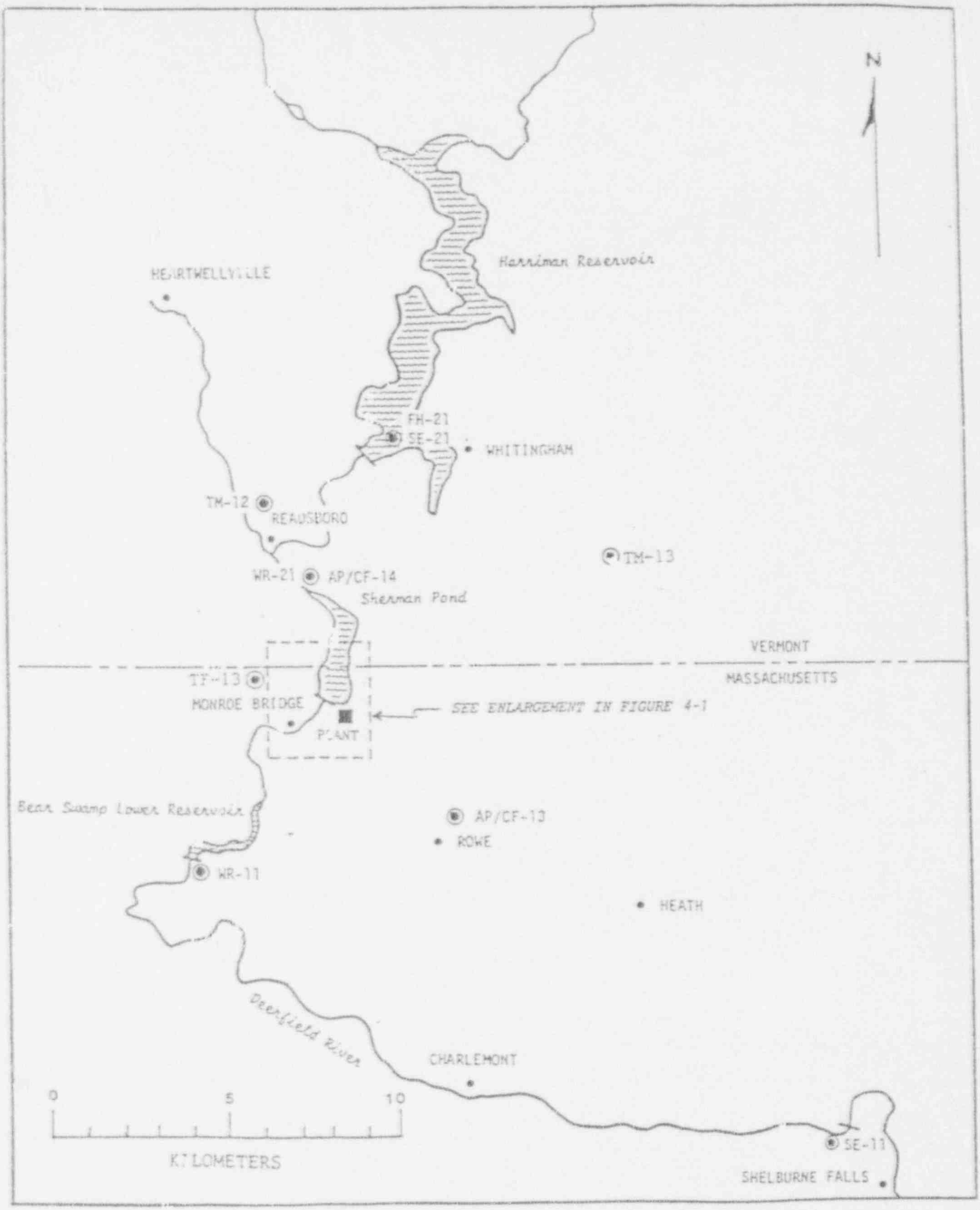


Figure 4-2 Yankee Plant Radiological Environmental Monitoring Locations Within 12 Miles (Airborne, Waterborne and Ingestion Pathways)

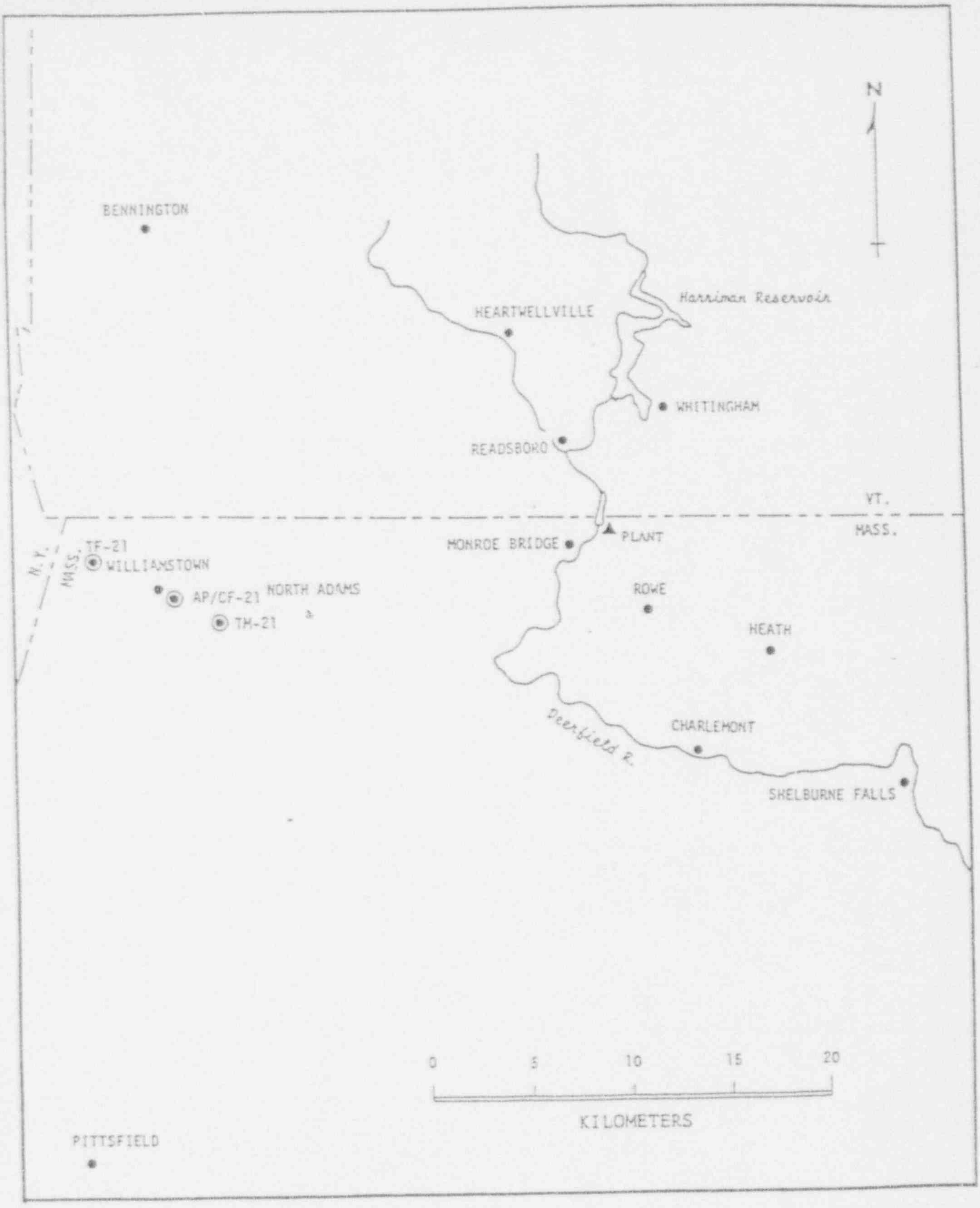


Figure 4-3 Yankee Plant Radiological Environmental Monitoring Locations Outside 12 Miles (Airborne, Waterborne and Ingestion Pathways)

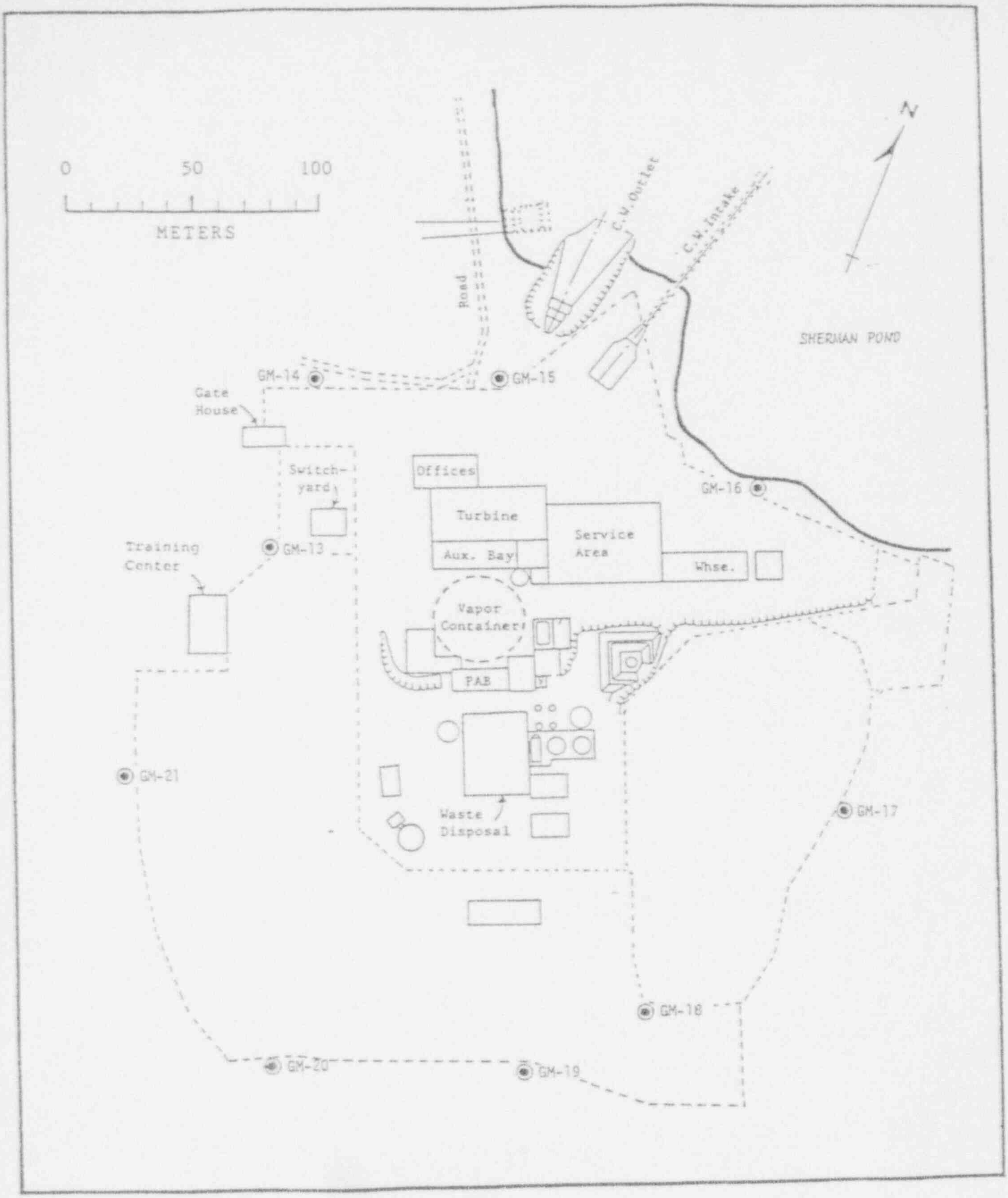


Figure 4-4 Yankee Plant Radiological Environmental Monitoring Locations at the Restricted Area Fence (Direct Radiation Pathway)

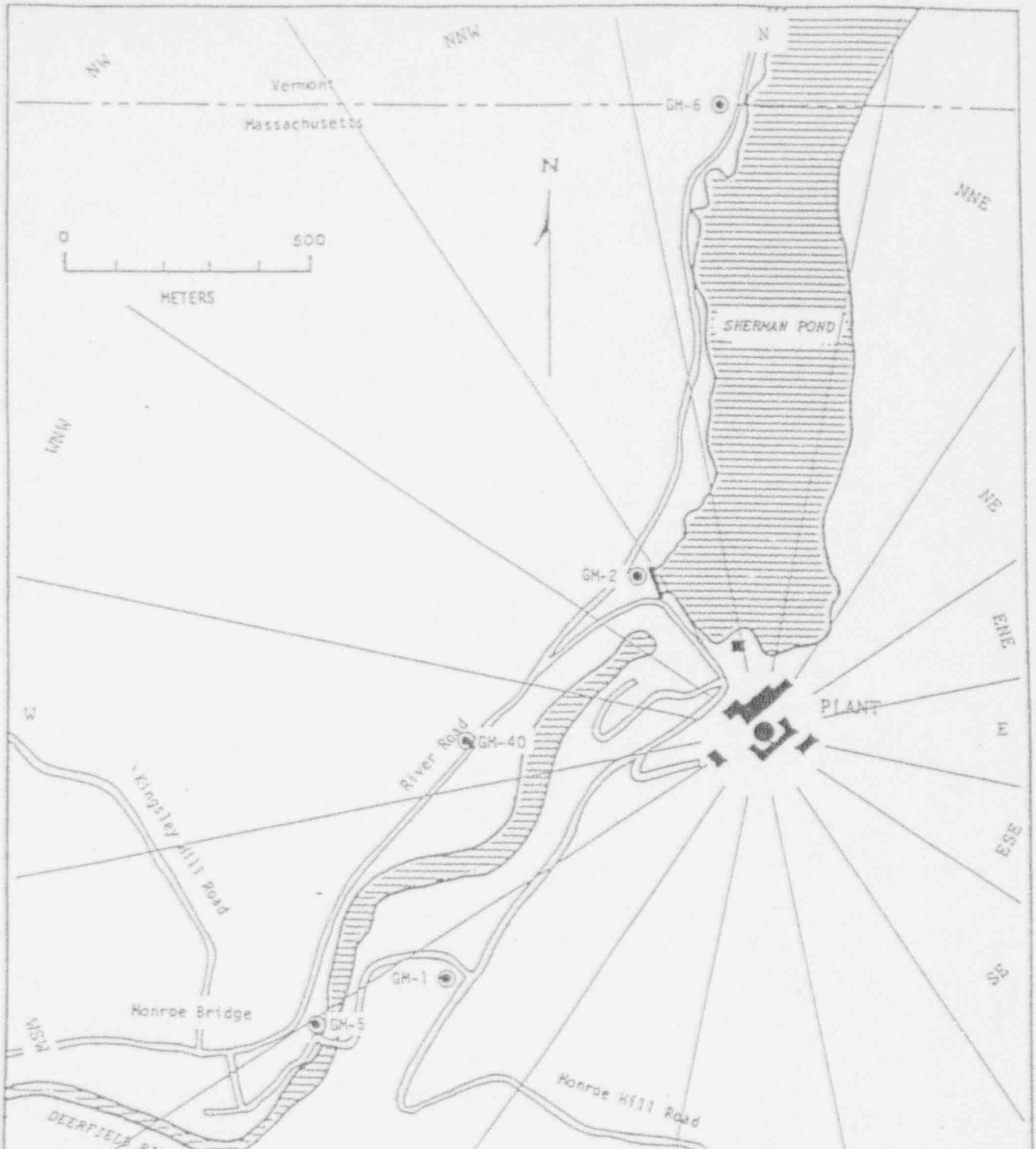


Figure 4-5 Yankee Plant Radiological Environmental Monitoring Locations Within 1 Mile (Direct Radiation Pathway)

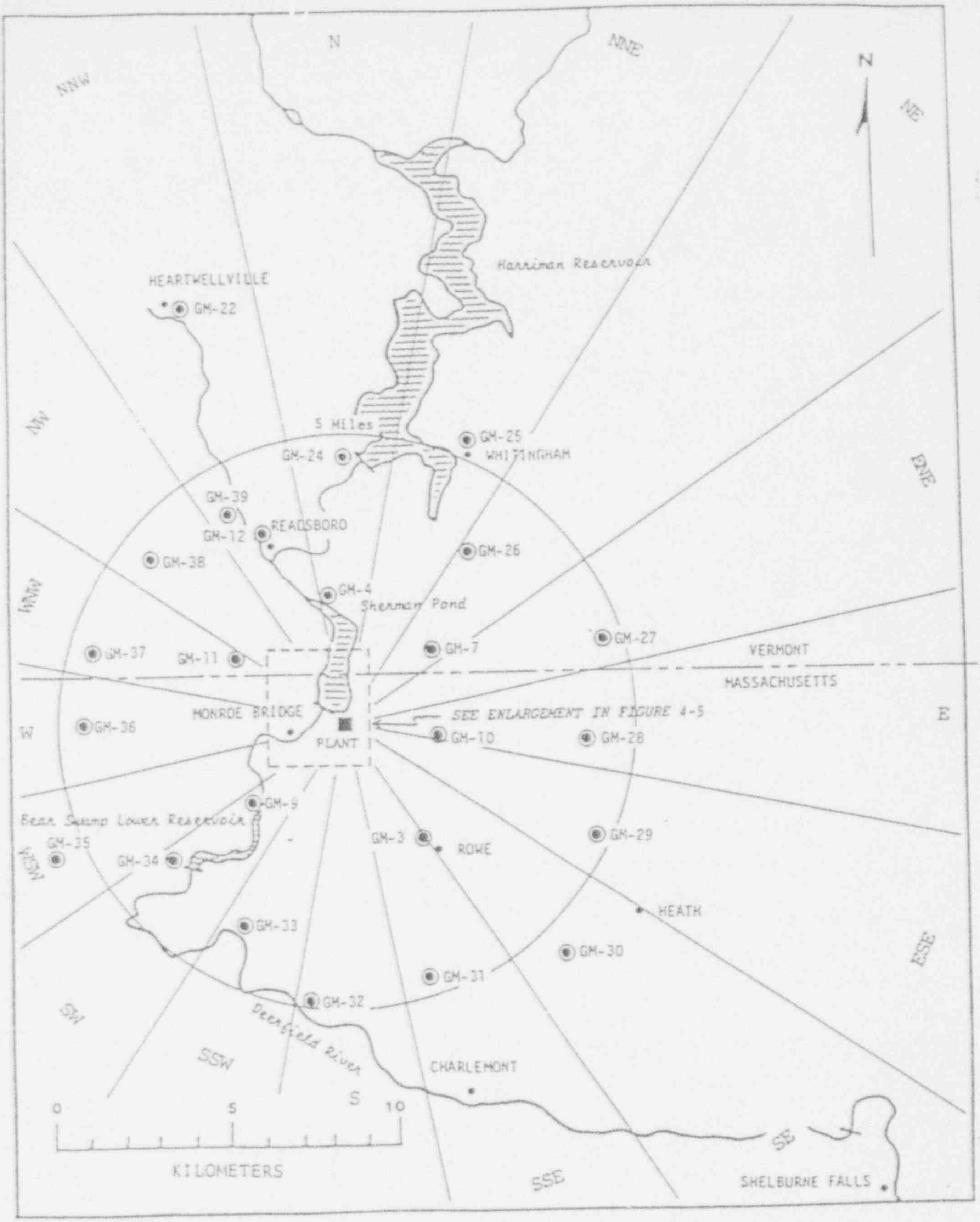


Figure 4-6 Yankee Plant Radiological Environmental Monitoring Locations Within 12 Miles (Direct Radiation Pathway)

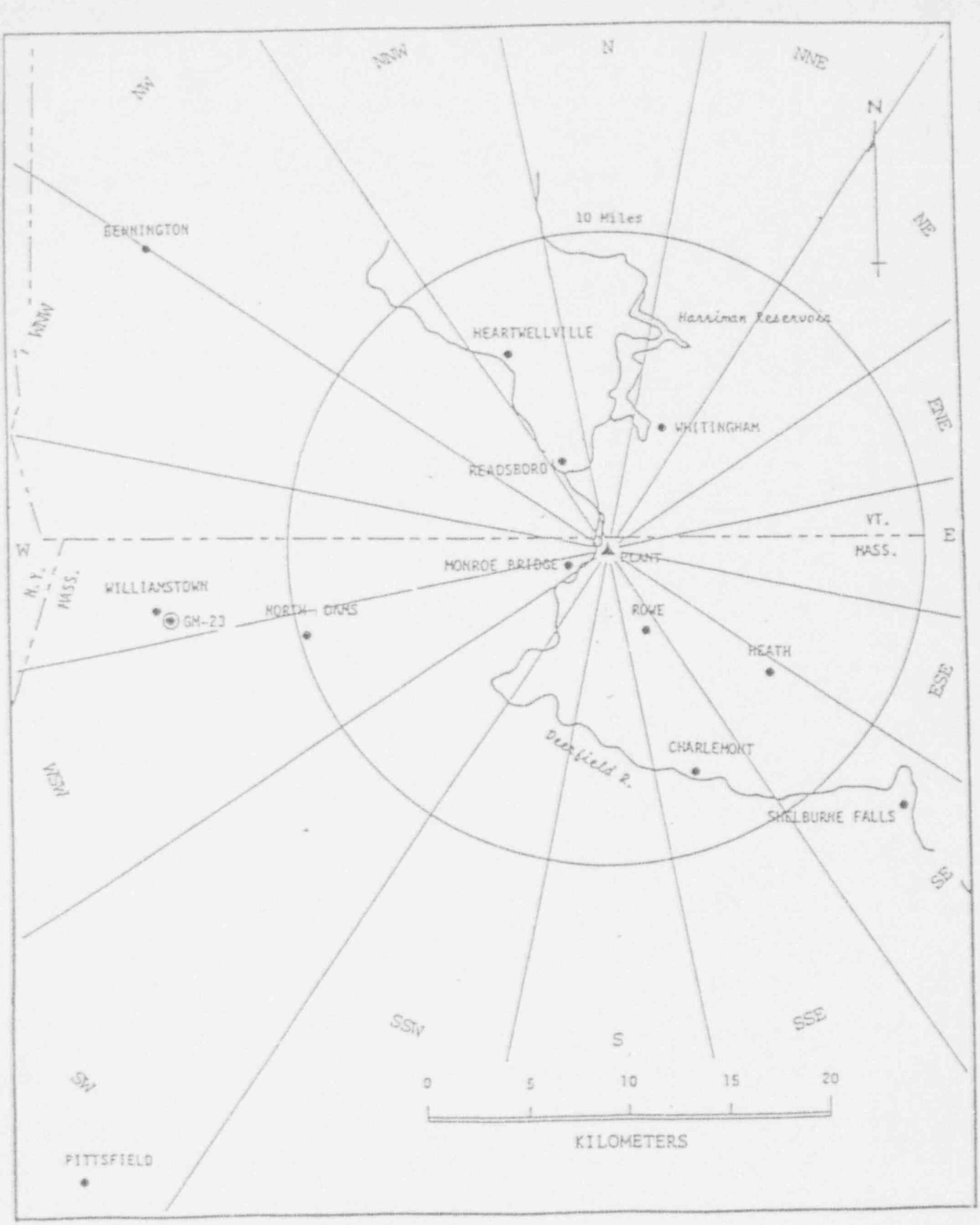


Figure 4-7 Yankee Plant Radiological Environmental Monitoring Locations Outside 12 Miles (Direct Radiation Pathway)

5.0 INSTRUMENTATION

5.1 Radioactive Liquid Effluents

Control 5.1 In accordance with Yankee Technical Specification 6.8.5.a.1, the radioactive liquid effluent monitoring instrumentation channels shown in Table 5.1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Control 2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

Applicability

As shown in Table 5.1.

ACTION

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Control 2.1 are met, without delay, take actions to suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint, so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 5.1. Exert reasonable efforts to return the instrument(s) to OPERABLE status within 30 days and if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report the reason for the delay in correcting the inoperability.

c. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 5.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 5.2.

Bases

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A, 10CFR Part 50.

Gross radioactivity monitors which provide for automatic isolation of liquid discharges on detection of radioactivity concentrations in excess of the limits of 10CFR Part 20 are included on the two principal radioactive effluent discharge pathways from the plant (liquid radioactive waste effluent line and steam generator blowdown effluent line). The automatic alarm/trip function provided by these monitors gives assurance as a final check that all conditions assumed, measured, or calculated that were used to determine effluent discharge rates have been appropriately made. This provides a degree of protection against calculational errors on discharge rate, operator errors in setting discharge flow, nonrepresentative samples used for isotopic content of discharge volume, or crud releases during discharge which could lead to the discharge concentration limits of Control 2.1 being exceeded.

Composite samples are provided on all continuous and secondary radioactive effluent pathways to give assurance that all potential radioactive liquid releases to the environment are accounted for.

TABLE 5.1

Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	Minimum Channels OPERABLE	Applicability	ACTION
1. Gross Radioactivity Monitors Providing Automatic Isolation			
a. Liquid Radwaste Effluent Line	(1)	At All Times	15
b. Steam Generator Blowdown Tank Effluent Line	(1)	At All Times	16
2. Continuous Composite Samplers			
a. Steam Generator Blowdown Tank Effluent Line	(1)	*	16
b. Secondary Coolant and Condensate Leakage	(1)	*	16
c. Turbine Building Sump	(1)	*	16
3. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line+	(1)	*	17
b. Circulating Water System Discharge+	(1)	*	17
c. Steam Generator Blowdown Tank Effluent	(1)	*	17

* Via this pathway during releases.

+ Pump curves may be utilized to estimate flow. In such cases, the ACTION statement is not required.

TABLE 5.1
(Continued)

ACTION Statements

ACTION 15 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases from the tank may continue, provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed in accordance with SR 2.1.1.

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving,

otherwise, suspend release of radioactive effluents via this pathway.

ACTION 16 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 1.00×10^{-7} microcuries/gram:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram I-131.

ACTION 17 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue, provided that the flow rate is estimated at least once per four hours during actual releases. Pump curves may be used to estimate flow.

TABLE 5.2

Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES in Which Surveillance is Required
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation					
a. Liquid Radwaste Effluent Line	D	P	$R^{(2)}$	$Q^{(1)}$	At All Times
b. Steam Generator Blowdown Tank Effluent Line	D	M	$R^{(2)}$	$Q^{(1)}$	At All Times
2. Continuous Composite Samplers and Sample Flow Measurement Device					
a. Steam Generator Blowdown Tank Effluent Line	D	NA	R	Q	**
b. Secondary Coolant and Condensate Leakage	D	NA	R	Q	**
c. Turbine Building Sump	D	NA	R	Q	**
3. Flow Rate Measurement Devices					
a. Liquid Radwaste Effluent Line	$D^{(3)}$	NA	R	Q	**
b. Circulating Water System Discharge*	$D^{(3)}$	NA	NA	NA	**
c. Steam Generator Blowdown Tank Effluent	NA	NA	$R^{(4)}$	NA	**

* Pump curves utilized for flow rate determination.

**Via this pathway during releases.

TABLE 5.2
(Continued)

Table Notation

(1) - The CHANNEL FUNCTIONAL TEST also shall demonstrate that automatic isolation of this pathway and Control Room alarm annunciation occurs if any of the following conditions, except as noted, exist:

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

b. Circuit failure.

c. Instrument indicates a downscale failure (automatic pathway isolation, and Control Room warning light indicator - no audible alarm annunciation).

(2) - The CHANNEL CALIBRATION shall include the use of a known radioactive source(s) positioned in a reproducible geometry with respect to the sensor whose effect on the system was established at the time of the primary calibration. Primary calibration is the determination of the electronic system accuracy when the detector is exposed in a known geometry to radiation from sources emitting beta and gamma radiation with fluences and energies in the ranges anticipated to be measured by the channel during normal operation. Sources should be traceable to the National Institute of Standards and Technology (NIST).

(3) - The CHANNEL CHECK shall consist of verifying indication of flow during periods of release except where pump curves are used to estimate flow. When pump curves are utilized as means of determining flow, no CHANNEL CHECK is required. The CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or batch releases are made.

(4) - The method of estimation of flow rate shall be verified and adjusted as necessary, by observing the response of blowdown tank level (high level, low level) sensors and the operation (opening and closing) of the tank's discharge valve.

5.2 Radioactive Gaseous Effluents

Control 5.2 In accordance with Yankee Technical Specification 6.8.5.a.1, the radioactive gaseous effluent monitoring instrumentation channels shown in Table 5.3 shall be OPERABLE with their alarm setpoints set to ensure that the limits of Control 3.3 are not exceeded. The alarm setpoints of these channels shall be determined in accordance with the ODCM.

Applicability

As shown in Table 5.3.

ACTION

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of Control 3.3 are met, without delay, take actions to suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint, so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 5.3. Exert reasonable efforts to return the instrument(s) to OPERABLE status within 30 days and if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report the reason for the delay in correcting the inoperability.

c. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 5.2 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 5.4.

Bases

The radioactive gaseous effluent instrumentation in the primary vent stack is provided to monitor, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm setpoints for these instruments are set conservatively to ensure that the limits of 10CFR Part 20 are not exceeded. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A, 10CFR Part 50.

The primary vent stack exhausts building ventilation air, as well as gaseous process streams, to the atmosphere and as such, cannot be isolated due to building ventilation requirements. The indications of the source of abnormally high radiation measurements at the primary vent stack are provided by a series of area monitors which are located throughout the plant in areas where potentially high radiation levels could be encountered. Each area radiation detector readout is displayed in the Main Control Room and is provided with both audible and visual alarms. The locations and operating ranges of the principal in-plant area monitors are as follows:

Detector Location	Range
Waste Disposal Building	1.00 mR/hr - 10^{+5} mR/hr
Primary Auxiliary Building Chem. Sample	1.00 mR/hr - 10^{+5} mR/hr
Primary Auxiliary Building Valve Room	1.00 mR/hr - 10^{+5} mR/hr
Spent Fuel Pit	1.00 mR/hr - 10^{+5} mR/hr
Primary Auxiliary Building Fan Room	1.00 mR/hr - 10^{+5} mR/hr
Primary Auxiliary Building Corridor	1.00 mR/hr - 10^{+5} mR/hr
Primary Auxiliary Building Charging Pump No. 1	1.00 mR/hr - 10^{+5} mR/hr
Primary Auxiliary Building Charging Pump No. 2	1.00 mR/hr - 10^{+5} mR/hr
Primary Auxiliary Building Charging Pump No. 3	1.00 mR/hr - 10^{+5} mR/hr
New Fuel Vault	1.00 mR/hr - 10^{+5} mR/hr
Diesel and SI Building	1.00 mR/hr - 10^{+5} mR/hr
Auxiliary Boiler Feed	1.00 mR/hr - 10^{+5} mR/hr

Detector Location	Range
Turbine Hall	1.00 mR/hr - 10^{+5} mR/hr
Vapor Container (Fuel Crane)	1.00 mR/hr - 10^{+5} mR/hr
Vapor Container (PAM No. 1)	10^{+0} R/hr - 10^{+7} R/hr
Vapor Container (PAM No. 2)	10^{+0} R/hr - 10^{+7} R/hr

The capability of the area monitors, such as in the charging pump cubicles which have detected leaking pumps before any significant increase in activity was detected at the primary vent stack, help identify the source of radioactivity measured at the primary vent stack, so that corrective actions can be taken to curtail the release. In addition to the area monitors, process monitors assist in identifying process streams with abnormal levels of radioactivity which include the steam generator blowdown (4), main steam lines (4), condenser air ejector, waste gas header loop seal, and the Main Coolant System leakage air particulate monitors (2) in the vapor container. In total, the process and area monitors provide the Control Room operators with substantial information to indicate and identify contaminated areas within the plant or subsystems with high activity levels which could lead to high activity measurements at the primary vent stack. Corrective actions can then be initiated by the Control Room operators to mitigate the release of radioactivity from the plant.

TABLE 5.3

Radioactive Gaseous Effluent Monitoring Instrumentation

Instrument	Minimum Channels OPERABLE	Applicability	Parameter	ACTION
1. Primary Vent Stack				
a. Noble Gas Activity Monitor	(1)	*	Radioactivity Rate Measurement	21
b. Iodine Sampler Cartridge	(1)	*	Verify Presence of Cartridge	19
c. Particulate Sampler Filter	(1)	*	Verify Presence of Filter	19
d. Effluent System Flow Rate Measuring Device	(1)	*	System Flow Rate Measurement	18
e. Sampler Flow Rate Measuring Device	(1)	*	Sampler Flow Rate Measurement	18

* At all times.

TABLE 5.3
(Continued)

ACTION Statements

ACTION 18 - With the number of channels OPERABLE less than the minimum channels OPERABLE requirement, effluent releases may continue, provided the flow rate is estimated at least once per 8 hours.

ACTION 19 - With the number of channels OPERABLE less than the minimum channels OPERABLE requirement, effluent releases via this pathway may continue, provided samples are continuously collected with auxiliary sampling equipment as required in Table 3.1.

ACTION 21 - With the number of channels OPERABLE less than the minimum channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are taken at least once per 8 hours, and these samples are analyzed for gross activity within 24 hours.

TABLE 5.4

Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES in Which Surveillance is Required
1. Primary Vent Stack					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Sampler Cartridge	W	NA	NA	NA	*
c. Particulate Sampler Filter	W	NA	NA	NA	*
d. System Effluent Flow Rate Measuring Device	D	NA	NA	NA	*
e. Sampler Flow Rate Measuring Device	D	NA	R	Q	*

* At all times, except when the line is valved out and locked.

TABLE 5.4
(Continued)

Table Notation

(1) The CHANNEL FUNCTIONAL TEST also shall demonstrate that Control Room alarm annunciation occurs if any of the following conditions exist:

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

b. Circuit failure,

c. Instrument indicates a downscale failure.

(2) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the NIST or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. Subsequent CHANNEL CALIBRATION sources that have been related to the initial calibration can be used at intervals of at least once per 18 months.

5.3 Liquid Effluent Instrumentation Setpoints

Control 5.1 requires that the radioactive liquid effluent instrumentation in Table 5.1 have alarm/trip setpoints in order to ensure that Control 2.1 is not exceeded. Control 2.1 limits the activity concentration in liquid effluents to the appropriate MPCs in 10CFR Part 20 and a total noble gas MPC.

Use the method below to determine the setpoints for the required instrumentation.

5.3.1 Method

The instrument response (cpm) for the limiting concentration at the point of discharge is the setpoint, denoted R, and is determined as follows:

$$R = \left(\frac{f_3}{f_1 + f_2}\right)(MPC_c)\ (S_\ell) \qquad \text{(Eq. 5-1)}$$

Where:

f_1 = Flow rate past the test tank monitor (gpm).

f_2 = Flow rate past the steam generator blowdown monitor (gpm).

f_3 = Flow rate at the point of discharge (gpm).

S_ℓ = Instrument response factor (cpm/(μCi/ml)).

MPC_c = Composite MPC for the mix of radionuclides (μCi/ml).

$$MPC_c = \sum_i C_i / \sum_i C_i/MPC_i = \sum_i f_i / \sum_i f_i/MPC_i \qquad \text{(Eq. 5-2)}$$

Where:

MPC_i = MPC for radionuclide "i" from 10CFR Part 20, Appendix B, Table 2, Column 2 (µCi/ml).

C_i = Concentration of radionuclide "i" in mixture (µCi/ml).

f_i = Fraction of radionuclide "i" in mixture.

Other setpoint methodologies also can be applied which are more restrictive than the approach used here.

The setpoint, R, may be set lower to accommodate pathways without on-line monitors (secondary coolant or condensate leakage). When MPC_c is not stable or when dilution flow is low, the setpoint may have to be evaluated for each release.

5.3.2 Liquid Effluent Setpoint Example

The effluent monitors for the test tank and steam generator blowdown release pathways are gamma sensitive monitors. They both have a typical sensitivity, S, of $7.50 \times 10^{+7}$ cpm per µCi/ml of gamma emitters which emit one photon per disintegration and a typical background of 10,000 cpm. Both monitors have adjustable alarm/setpoints. However, the setpoint adjust control is located inside the panel-mounted electronics cabinet and is not easily accessible.

The principal gamma emitting radionuclide in waste effluent streams is Xenon-133, averaging two orders of magnitude higher than any other specie. However, it is not the intent of effluent monitors to respond to dissolved noble gases because Xenon-133 concentrations have never approached the MPC.

However, Iodine-131, Cesium-134, and Cesium-137 are detected in every liquid effluent release in roughly equal quantities and are the principal gamma emitters because they can approach their MPCs. Therefore, for purposes of adjusting the alarm/setpoints of the effluent monitors to comply with Control 2.1, the composite MPC (MPC_c) of 6.00×10^{-7} µCi/ml will be used.

It is calculated based on the following data (to be conservative, iodine is weighted greater than the cesiums):

i	f_i	MPC_i
Cs-134	0.25	9.00×10^{-6}
Cs-137	0.25	2.00×10^{-5}
I-131	0.50	3.00×10^{-7}

$$MPC_C = \frac{\sum_i f_i}{\sum_i f_i/MPC_i} \qquad \text{(Eq. 5-2)}$$

$$= \frac{1}{(0.25/9.00 \times 10^{-6} + 0.25/2.00 \times 10^{-5} + 0.50/3.00 \times 10^{-7})}$$

$$= 6.00 \times 10^{-7}$$

The maximum liquid effluent flow rate, $f_1 + f_2$, is taken as 130 gpm, based on a maximum 30-gpm flow rate from the test tank effluent pathway and a maximum 100-gpm flow rate from the steam generator blowdown pathway. Both pathways will be assumed to operate continuously and simultaneously.

Dilution water flow, f_3, is taken as 140,000 gpm based on 138,000 gpm through the condenser and 2,000 gpm through the auxiliary cooling loop. Throttling of cooling water is not practiced.

In this example, the setpoint for both monitors when both effluent pathways are operating is:

$$R = \left(\frac{f_3}{f_1 + f_2}\right)(MPC_C)\ (S_\ell) \qquad \text{(Eq. 5-1)}$$

$$= \left(\frac{140{,}000 \text{ gpm}}{30 \text{ gpm} + 100 \text{ gpm}}\right)(6.00 \times 10^{-7}\ \mu\text{Ci/ml})\ (7.50 \times 10^{+7}\ \text{cpm}/(\mu\text{Ci/ml}))$$

$$= 48{,}500 \text{ cpm}$$

Note that both effluent monitors have their lower level discriminators set to reject the pulses originating from the 80-KeV gamma emissions from

Xenon-133, and their high count rate alarms are set at 48,500 cpm above background.

5.3.3 Basis

The liquid effluent monitor setpoint must ensure that Control 2.1 is not exceeded for the appropriate in-plant pathways. The monitor is placed upstream of the major source of dilution flow and responds to the concentration of radioactivity as follows:

$$\underset{(cpm)}{R} = (S_{\ell})\,(\sum_i f_i\, s_i)\,(C_{MON}) \qquad \text{(Eq. 5-5)}$$

Where variables are the same as those in Section 5.3.1 except:

C_{MON} = Total concentration (µCi/ml) seen by the monitor.

s_i = Ratio of response from equal activities of radionuclide "i" to a reference radionuclide.

Calibration of the radiation monitors have established that the gross gamma detector response, $S \sum_i f_i\, s_i$ was fairly independent of gamma energy as expected. Thus, the response is a function of radioactivity concentration and the gamma yield of the mixture. Since $\sum_i f_i\, s_i$ is approximately one:

$$R = (S_{\ell})\,(C_{MON}) \qquad \text{(Eq. 5-6)}$$

For simplicity, assume that both monitors look at the total flow for f_1 and f_2. We know that:

$$C = \left(\frac{f_1 + f_2}{f_3}\right)(C_{MON}) \qquad \text{(Eq. 5-7)}$$

Where:

C = Total concentration at the point of discharge.

Solve Equation 5-5 for C_{MON} and substitute into Equation 5-4 to get:

$$R = \left(\frac{f_3}{f_1 + f_2}\right)(C)\ (S_\ell) \qquad \text{(Eq. 5-8)}$$

We defined $C = \sum_i C_i$ and define MPC_c such that:

$$\frac{C}{MPC_c} = \sum_i \frac{C_i}{MPC_i} \qquad \text{(Eq. 5-9)}$$

The right side of the equation is the MPC limit in 10CFR Part 20. Solving for MPC_c, the composite MPC for the mixture, we get the definition of MPC_c:

$$MPC_c = \frac{\sum_i C_i}{\sum_i \frac{C_i}{MPC_i}} \qquad \text{(Eq. 5-2)}$$

Substituting MPC_c into Equation 5-6, we get the response of the monitor as MPC_c is reached at the point of discharge, which is the setpoint:

$$R = \left(\frac{f_3}{f_1 + f_2}\right)(MPC_c)\ (S_\ell) \qquad \text{(Eq. 5-1)}$$

5.4 Gaseous Effluent Instrumentation Setpoints

Control 5.2 requires that the radioactive gaseous effluent instrumentation in Table 5.3 have their alarm setpoints set to ensure that Control 3.3.a is not exceeded. Control 3.3.a limits the activity concentration in off-site gaseous effluents to well below the appropriate MPCs in 10CFR Part 20 by limiting total body, skin, and organ dose rate.

Use the method below to determine the setpoint for the noble gas activity monitor.

5.4.1 Method

The noble gas activity monitor response (cpm) at the limiting noble gas dose (either total body or skin off-site) is the setpoint, denoted R, and is determined as follows:

R is the lesser of:

$$R_{tb} = \frac{(S_g)\ (\sum_i f_i^{NG}\ s_i)\ (500)\ (60)}{(F)\ (7.83)\ (\sum_i f_i^{NG}\ DBF_i)} \qquad \text{(Eq. 5-3)}$$

And:

$$R_{sk} = \frac{(S_g)\ (\sum_i f_i^{NG}\ s_i)\ (3000)\ (60)}{(F)\ (\sum_i f_i^{NG}\ DF_i)} \qquad \text{(Eq. 5-4)}$$

Where:

s_i = Ratio of response from equal activities of radionuclide "i" to a reference radionuclide, i.e., Xe-133.

DF_i' = Skin dose factor. See Table 1.2.

DFB_i = Total body dose factor. See Table 1.2.

f_i^{NG} = Fraction of radionuclide "i" activity to total noble gas activity.

F = Primary vent stack flow rate (cc/min).

S_g = Instrument calibration factor (cpm/(μCi/cc)).

Other setpoint methodologies also can be applied which are more restrictive than the approach used here.

5.4.2 Gaseous Effluent Setpoint Example

The primary vent stack noble gas activity monitor is an off-line system consisting of a beta sensitive scintillation detector, electronics, an analog ratemeter readout, and a digital scaler which counts the detector output pulses. A strip chart recorder provides a permanent record of the ratemeter output. Calibration data is provided by the manufacturer which indicates the response, s_i, of the beta sensitive detector to various gaseous radionuclides. The calibration data was verified on installation and periodically thereafter. System characteristics are:

a. Typical sensitivity - 1 cpm = 3.00×10^{-8} μCi/cc of Xenon-133; that is, S = $3.30 \times 10^{+7}$ cpm/(μCi/cc)

b. Typical background - 10 to 20 cpm

Under normal plant stack flow, F, of $5.80 \times 10^{+8}$ cc/min (20,500 cfm x 28,300 cc/ft^3), one count on the scaler is equivalent to 17 microcuries of Xenon-133 noble gases released. Since the typical average primary vent stack concentrations of noble gases are only about 1.00×10^{-6} μCi/cc, direct grab sampling and isotopic analysis is not satisfactory. The isotopic distribution of noble gases dissolved in primary coolant is determined monthly and used as the distribution, f_i^{NG}, for gaseous effluent releases. The distribution,

f_i^{NG}, and the relative response, s_i, for each radionuclide in this example are presented in Table 5.5.

Applying Equations 5-3 and 5-4:

$$R_{tb} = \frac{(3.30 \times 10^{+7})\ (0.71)\ (500)\ (60)}{(5.80 \times 10^{+8})\ (7.83)\ (2.10 \times 10^{-3})} = 73{,}700 \text{ cpm}$$

$$R_{sk} = \frac{(3.30 \times 10^{+7})\ (0.71)\ (3000)\ (60)}{(5.80 \times 10^{+8})\ (4.30 \times 10^{-2})} = 169{,}000 \text{ cpm}$$

The setpoint, R, is the lesser of R_{tb} and R_{sk}; therefore, it is equal to 73,700 cpm. This is due to the noble gas mixture in this example, in which the total body dose rate is more restrictive.

5.4.3 Basis

The noble gas activity monitor setpoint must ensure that Control 3.3.a is not exceeded. Sections 3.9 and 3.10 show that Equations 3-3 and 3-4 are acceptable methods for complying with Control 3.3.a. The equation (i.e., dose - total body or skin) which is more limiting depends on the noble gas mixture. Therefore, each equation must be considered separately. The derivation of Equations 5-3 and 5-4 starts with the general equation for the response, R (cpm), of a radiation monitor:

$$R = (S_g)\ (\sum_i f_i^{NG}\ s_i)\ (C)$$
$$(\text{cpm})(\text{cpm}/(\mu\text{Ci/cc}))\ (1)\ (\mu\text{Ci/cc}) \qquad \text{(Eq. 5-5)}$$

Expanding for the concentration:

$$R = (S_g)\ (\sum_i f_i^{NG}\ s_i)\ (\dot{Q})\ (60/F)$$

$$(cpm)(cpm/(\mu Ci/cc))\ (1)\ (\mu Ci/sec)\ (sec/min)\ (cc/min) \qquad \text{(Eq. 5-10)}$$

The response of the monitor at the release rate which causes the total body dose rate limit to be reduced, R_{tb}, begins with Equation 3-3.

$$\dot{D}_{tb} = 7.83 \sum_i \dot{Q}_i\ DFB_i$$

Substituting $\dot{Q} = f_i^{NG}\ \dot{Q}$ gives: (Eq. 5-11)

$$\dot{D}_{tb} = 7.83\ \dot{Q} \sum_i f_i^{NG}\ DFB_i \qquad \text{(Eq. 5-12)}$$

Rearranging to solve for $\dot{Q}$:

$$\dot{Q} = \frac{\dot{D}_{tb}}{7.83 \sum_i f_i^{NG}\ DFB_i} \qquad \text{(Eq. 5-13)}$$

Substituting Equation 5-13 into Equation 5-10 and substituting the total body dose rate limit gives:

$$R_{tb} = \frac{S\ (cpm/(\mu Ci/cc))\ (\sum_i f_i^{NG}\ s_i)\ 500\ (mrem/yr)\ 60\ (sec/min)}{F\ (cc/min)\ 7.83\ (pCi\text{-}sec/\mu Ci\text{-}m^3) \sum_i f_i^{NG}\ DFB_i\ (mrem\text{-}m^3/pCi\text{-}yr)} \qquad \text{(Eq. 5-3)}$$

The response of the monitor at the release rate which causes the skin dose rate limit to be reduced, R_{sk}, begins with Equation 3-4:

$$\dot{D}_{sk} = \sum_i \dot{Q}_i\ DF_i'$$

Substituting $\dot{Q} = f_i^{NG}\, \dot{Q}_i$ gives: (Eq. 5-11)

$$\dot{D}_{sk} = \dot{Q} \sum_i f_i^{NG}\, DF_i' \qquad \text{(Eq. 5-14)}$$

Rearranging to solve for $\dot{Q}$:

$$\dot{Q} = \frac{\dot{D}_{sk}}{\sum_i f_i^{NG}\, DF_i'} \qquad \text{(Eq. 5-15)}$$

Substituting Equation 5-15 into Equation 5-10 and substituting the skin dose rate limit of 3,000 mrem/yr gives:

$$R_{sk} = \frac{S_g\ (\text{cpm}/(\mu\text{Ci/cc}))\ (\sum_i f_i^{NG}\, s_i)\ 3000\ (\text{mrem/yr})\ 60\ (\text{sec/min})}{F\ (\text{cc/min}) \sum_i f_i^{NG}\, DF_i'\ (\text{mrem-sec}/\mu\text{Ci-yr})}$$

TABLE 5.5

Sample Calculation of Gaseous Instrumentation Setpoint
(Based on 1981 Yankee Data)

Noble Gas Specie	Fraction of Total f_i^{NG}	Detector Response With Respect to Xe-133=1.0 s_i	Weighted Response $f_i^{NG} \times s_i$	Weighted Whole Body Dose Factor $f_i^{NG} \times DFB_i$	Weighed Skin Dose Factor $f_i^{NG} \times DF_i'$
Ar-41	0.008	1.20	0.010	7.10×10^{-5}	1.20×10^{-3}
Kr-85	0.000	1.15	0.000	~0.00	~0.00
Kr-85m	0.010	0.00	0.000	1.20×10^{-5}	4.60×10^{-4}
Kr-87	0.010	1.50	0.015	5.90×10^{-5}	2.90×10^{-3}
Kr-88	0.016	1.15	0.018	2.30×10^{-4}	3.00×10^{-3}
Xe-131m	0.020	0.00	0.000	1.80×10^{-6}	2.50×10^{-4}
Xe-133	0.380	1.00	0.380	1.10×10^{-4}	4.00×10^{-3}
Xe-133m	0.000	0.00	0.000	~0.00	~0.00
Xe-135	0.200	1.30	0.260	3.60×10^{-4}	1.20×10^{-2}
Xe-135m	0.340	0.00	0.000	1.10×10^{-3}	1.60×10^{-2}
Xe-138	0.020	1.50	0.030	1.80×10^{-4}	3.60×10^{-3}
Summation	1.000		0.710	2.10×10^{-3}	4.30×10^{-2}

TABLE 5.6

Relative Fractions of Core Inventory
Noble Gases After Shutdown

Time	Kr-85m	Kr-85	Kr-88	Xe-131m	Xe-133m	Xe-133	Xe-135m	Xe-135
t < 24 h	0.007	0.003	0.004	0.004	0.021	0.714	0.017	0.232
24 h ≤ t < 48 h	--	0.004	--	0.005	0.023	0.911	0.003	0.056
48 h ≤ t < 5 d	--	0.005	--	0.008	0.015	0.971	--	--
5 d ≤ t < 10 d	--	0.010	--	0.013	0.006	0.970	--	--
10 d ≤ t < 15 d	--	0.020	--	0.022	0.002	0.956	--	--
15 d ≤ t < 20 d	--	0.037	--	0.034	0.001	0.929	--	--
20 d ≤ t < 30 d	--	0.119	--	0.071	--	0.806	--	--
30 d ≤ t < 60 d	--	0.795	--	0.103	--	0.103	--	--
t ≤ 60 d	--	0.974	--	0.024	--	0.002	--	--

6.0 RADIOACTIVE WASTE TREATMENT SYSTEMS, EFFLUENT PATHWAYS, AND RADIATION MONITORS

6.1 Liquid Radioactive Waste Treatment

Control 6.1 In accordance with Yankee Technical Specification 6.8.5.a.6, the Liquid Radioactive Waste Treatment System shall be used to reduce the radioactive materials in the liquid waste prior to its discharge when the estimated doses due to the liquid effluent from the site (see Figure 1-2) when averaged over 31 days, would exceed 0.06 mrem to the total body or 0.20 mrem to any organ.

Applicability

At all times.

ACTION

a. With liquid waste being discharged without processing through appropriate treatment systems as defined in the ODCM and estimated doses in excess of the above limits, and if not applicable to 10CFR Part 50.73, prepare and submit to the Commission within 30 days pursuant to Control 7.4, a Special Report which includes the following information:

    1. Explanation of why liquid radioactive waste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reasons for the inoperability;

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

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

b. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 6.1 Doses due to liquid releases shall be estimated at least once per 31 days in accordance with the ODCM. No dose estimates are required if the Liquid Radioactive Waste Treatment System has been continually used to reduce the radioactive materials in liquid waste prior to its discharge or if no liquid discharges have taken place over the appropriate 31-day period.

Bases

The control that the appropriate portions of the Liquid Radioactive Waste Treatment System be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." Control 6.1 implements the requirements of 10CFR Part 50.36a, General Design Criterion 60 of Appendix A, 10CFR Part 50, and the design objective of Section II.D of Appendix I, 10CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radioactive Waste Treatment System were specified as a suitable fraction of the dose design controls set forth in Section II.A of Appendix I, 10CFR Part 50, for liquid effluents.

6.2 Gaseous Radioactive Waste Treatment

Control 6.2 In accordance with Yankee Technical Specification 6.8.5.a.6, the GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the estimated gaseous effluent air doses due to gaseous effluent releases from the site to areas at and beyond the SITE BOUNDARY (see Figure 1-1) would exceed 0.20 mrad for gamma radiation and 0.40 mrad for beta radiation over 31 days. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the estimated doses due to gaseous effluent releases from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed 0.30 mrem to any organ over 31 days.

Applicability

At all times.

ACTION

a. With gaseous waste being discharged without processing through appropriate treatment systems as defined in the ODCM and in excess of the above limits and if not applicable to 10CFR Part 50.73, prepare and submit to the Commission within 30 days, pursuant to Control 7.4, a Special Report that includes the following information:

1. Explanation of why gaseous radioactive waste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reasons for the inoperability;

2. Action(s) taken to restore any inoperable equipment to OPERABLE status, and

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

b. The provisions of Controls 1.3 and 1.4 are not applicable.

Surveillance Requirement

SR 6.2 Doses due to gaseous releases from the site shall be estimated at least once per 31 days in accordance with the ODCM.

Bases

The control that the appropriate portions of the GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." Control 6.2 implements the requirements of 10CFR Part 50.36a, General Design Criterion 60 of Appendix A, 10CFR Part 50, and the design controls of Appendix I, 10CFR Part 50. The ACTION levels governing the use of appropriate portions of the GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEM were specified as a suitable fraction of the guides set forth in Sections II.B and II.C of Appendix I, 10CFR Part 50, for gaseous effluents.

6.3 Liquid and Gaseous Effluent Streams, Radiation Monitors, and Radioactive Waste Treatment Systems

Figures 6-1 and 6-2 show the liquid effluent streams, radiation monitors, and the appropriate Liquid Radioactive Waste Treatment System. Figures 6-3 and 6-4 show the gaseous effluent streams, radiation monitors, and the appropriate GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEM.

6.4 In-Plant Liquid Effluent Pathways

Steam generator blowdown water is collected in a blowdown receiver tank and periodically released by automatic level controllers to the service water cooling loop. The tank fills at a nominal 5 gpm. The tank discharges about 100 gallons per cycle at a rate of about 110 gpm. The discharge rate is fixed because of a relatively constant head on the receiver tank and a fixed discharge geometry through a fully open/fully closed discharge valve. Blowdown discharges occur about one minute of every 20 during plant operation. This pathway is considered a continuous release during plant operation. A composite sampler operates whenever the blowdown receiver discharge valve is open. The sampler collects about one part in 5000 of the effluent. The total volume of liquid effluent per period (week) is determined by a cycle counter on the discharge valve (total cycles times volume per cycle). Steam generator blowdown may not be a continuous or routine expected effluent pathway when the plant is shut down. However, when the plant is shut down, liquids in a steam generator during wet layup could constitute a batch release pathway if discharged to service water. The liquid associated with a steam generator during wet layup should be sampled and analyzed in accordance with the requirements listed in Table 2.1, "Radioactive Liquid Waste Sampling and Analysis Program," of SR 2.1.1.

Whenever the plant is operating and secondary plant components are pressurized, then steam condensate/feedwater leaks from pump seals and other components are collected in drain headers and routed to the discharge of the main condenser. A single sampler collects a time integral composite sample of condensate water. This sample is assumed typical of water leakage in the condensate-feedwater train. The volume of leakage is determined by chemists each day by a water balance.

A number of auxiliary coolers and heat exchangers discharge service water (Sherman Pond water) into a common underfloor discharge header. The

Turbine Building floor drains and Auxiliary Boiler Room floor drains drain to this header. The water from this drain header discharges without further dilution into a tributary of the Deerfield River outside the controlled area. A composite sampler collects a sample of the water whenever there is discharge (water in the pipe).

Batch effluent tanks called "test tanks" collect the distillate from the liquid radioactive waste evaporator. Normally, liquid waste accumulates at about 1 gpm and is processed at about 4 gpm. When a 7000-gallon test tank is filled, it is sampled, analyzed, and released at a nominal 30 gpm.

The condenser cooling flow provides the major source of dilution and is assumed to be 138,000 gpm with two pumps operating and 69,000 gpm with one pump operating. Throttling of condenser cooling water is not practiced at the Yankee plant.

During shutdown periods, the 4,000-gpm service water provides dilution water flow. Flow rate is variable and estimated by pump curves. Typically, flow rates range from 1,500 gpm to 3,500 gpm.

The discharge rate from the steam generator blowdown tank is fixed by piping geometry and a relatively constant head on the tank. A flow meter estimates the discharge rate during periods of discharge. Verification is done periodically by measuring the time for the tank level to decrease during a normal release.

The discharge rate for the Turbine Building pathway is estimated to be 400 gpm. Approximately 1 gpm of this rate is secondary coolant (from pump leakage and sample stations). The remainder is service water from various secondary plant heat exchangers. All piping is buried and inaccessible, so flow is estimated from cooling water pump flows.

The discharge rate for the test tanks is controlled by the discharge line variorifices and limited to 30 gpm.

Calibrations of the radiation monitors have established that the gross gamma detector response was fairly independent of the gamma energy, as expected. Thus, the response is a function of the radioactivity concentration and the gamma yield of the mixture, but not the gamma energies of the mixture. The electronics of each monitor channel has an adjustable alarm setpoint. A

Control Room alarm will sound, and the effluent discharge valve will close whenever any one of the following occurs:

a. Ratemeter indication above the setpoint

b. Loss of detector high voltage

c. Loss of detector signal

d. Loss of power to channel

e. Instrument controls not set in operate position

6.5 In-Plant Gaseous Effluent Pathways

The primary vent stack noble gas effluent monitor is an off-line system consisting of a beta sensitive scintillation detector, electronics, an analog ratemeter readout, and a digital scaler which counts the detector output pulses. A strip chart recorder provides a permanent record of the ratemeter output. Calibration data is provided by the manufacturer which indicates the response of the beta sensitive detector to various gaseous radionuclides. The calibration data was verified on installation and periodically thereafter. System characteristics are:

a. Typical sensitivity 1 cpm ~ 3.00×10^{-8} µCi/cc Kr-85

b. Typical background 10-20 cpm

Under normal plant stack flow of 20,500 cfm, one count on the scaler is equivalent to 17 microcuries of Xenon-133 noble gases released. Since the typical average primary vent stack concentrations of noble gases are only about 1.00×10^{-6} µCi/cc, direct grab sampling and isotopic analysis is not satisfactory. The isotopic distribution of noble gases dissolved in primary coolant is determined monthly and used as a source term for gaseous effluent releases.

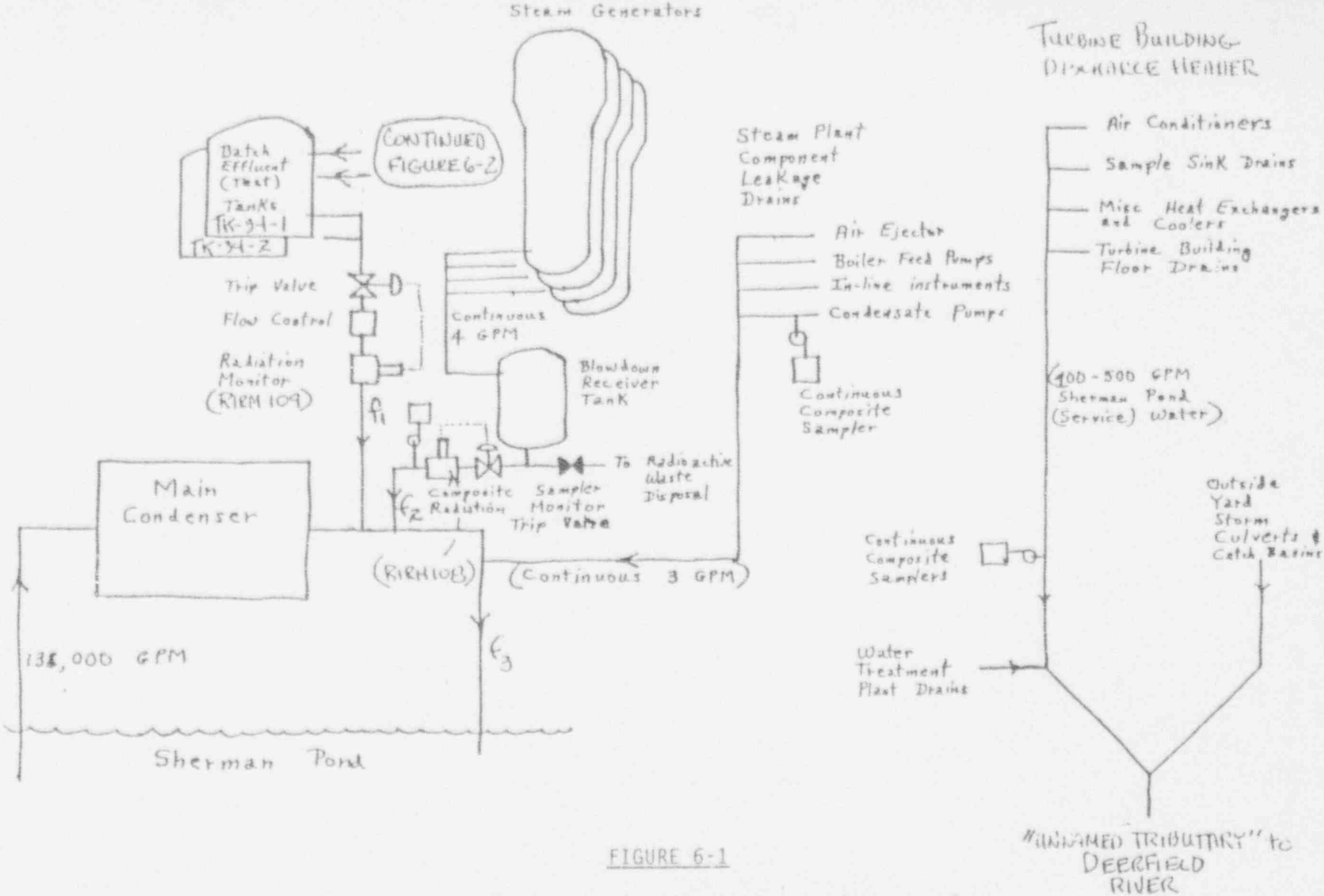


FIGURE 6-1

Liquid Effluent Streams, Radiation Monitors, and Radioactive Waste Treatment System at the Yankee Plant

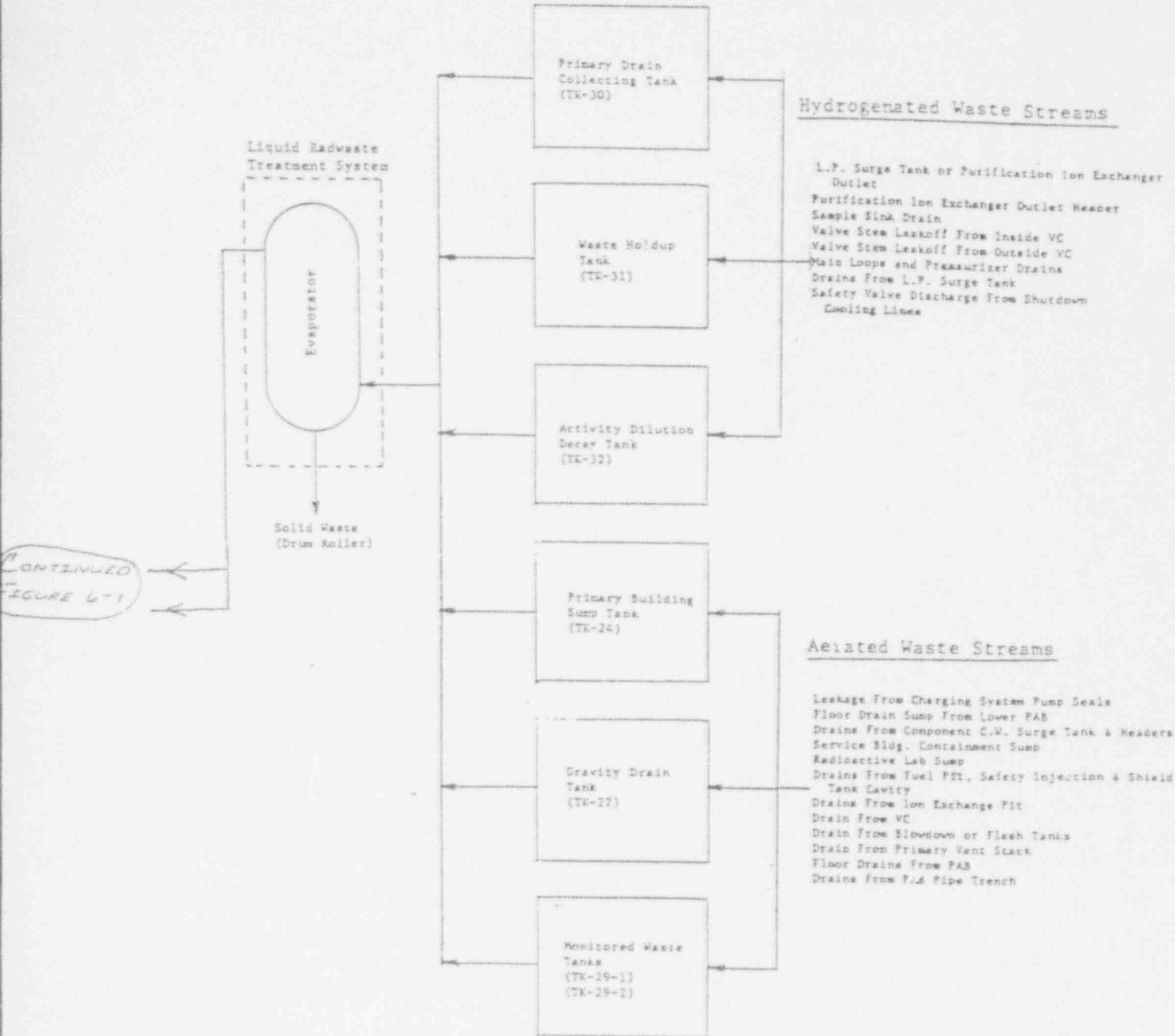


FIGURE 6-2

Liquid Effluent Streams, Radiation Monitors, and Radioactive Waste Treatment System at the Yankee Plant
(Continued)

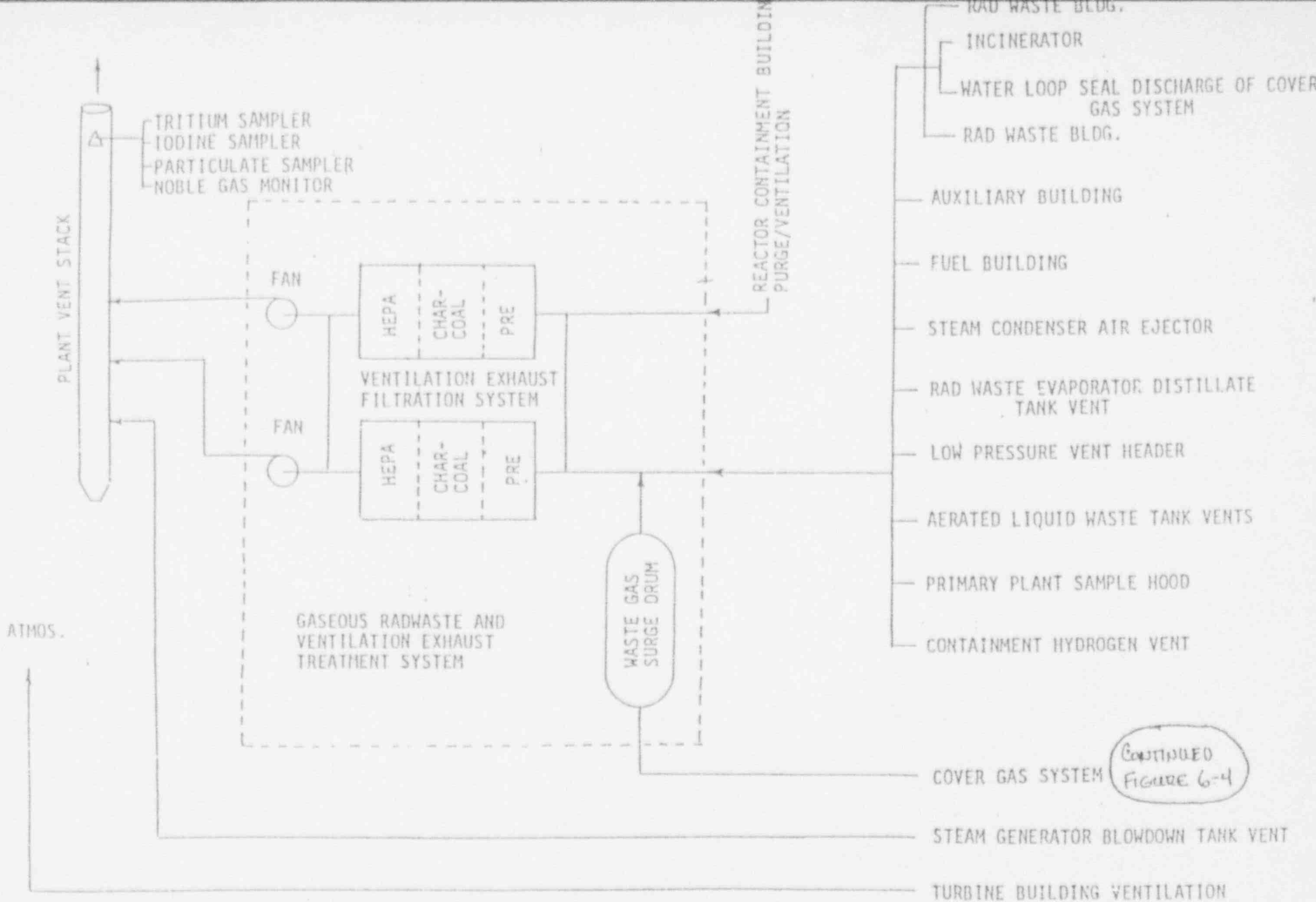


FIGURE 6-3

Gaseous Effluent Streams, Radiation Monitors, and Radioactive Waste Treatment System at the Yankee Plant

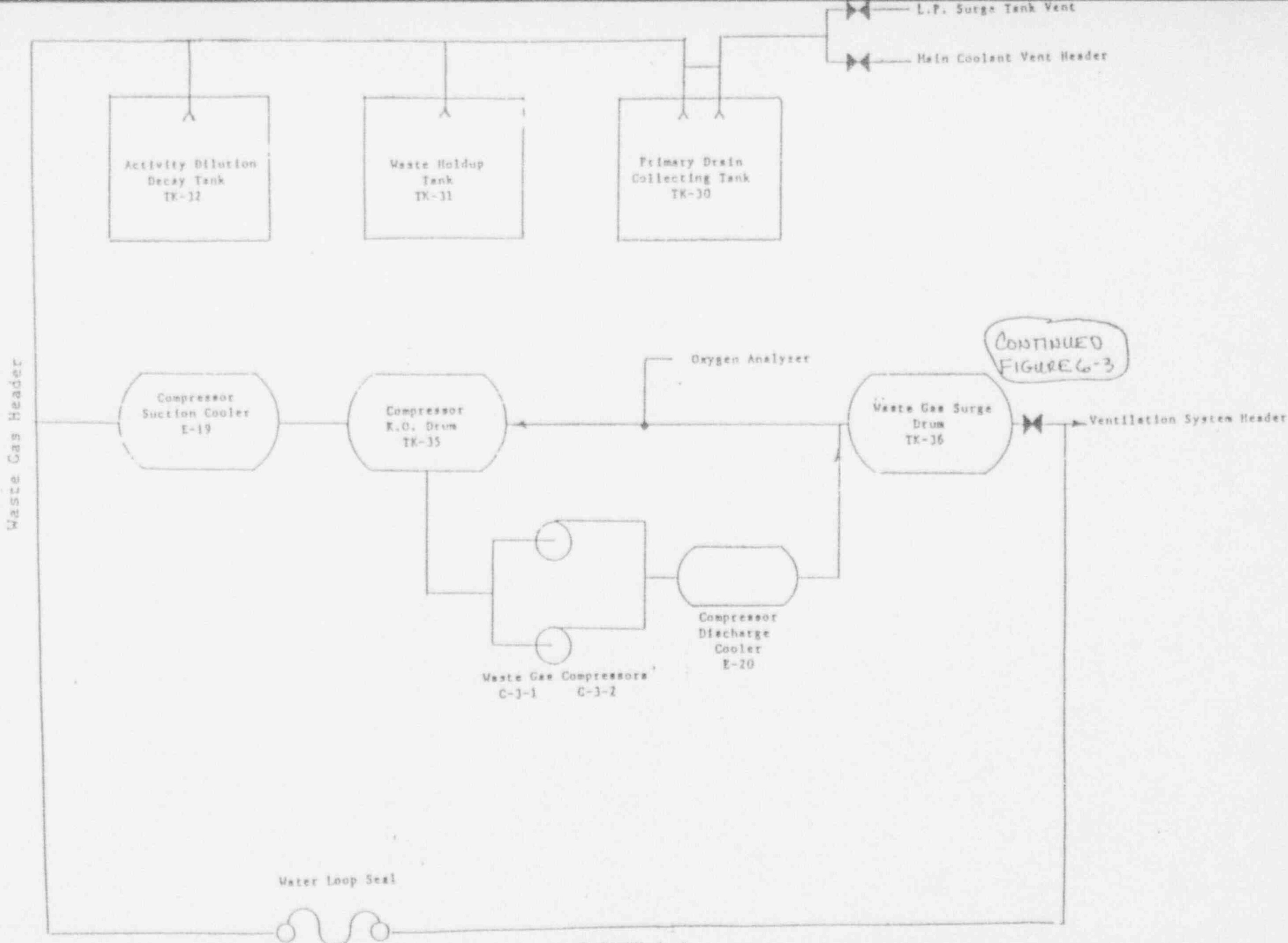


FIGURE 6-4

Gaseous Effluent Streams, Radiation Monitors, and
Radioactive Waste Treatment System at the Yankee Plant
(Continued)

7.0 REPORTING REQUIREMENTS

7.1 Annual Radiological Environmental Operating Report

Control 7.1

a. An Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

b. The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The report also shall include the results of the land use census required by Control 4.2.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of all radiological environmental samples taken during the report period pursuant to the table and figures in the ODCM. In the event that some results are not available to include in 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.

The report also shall include the following: a summary description of the Radiological Environmental Monitoring Program with a map of all sampling locations keyed to a table giving distances and directions from the reactor, the results of licensee participation in the Intercomparison Program required by Control 4.3, and a discussion of all analyses in which the LLD required by Table 4.3 was not achievable.

7.2 Semiannual Effluent Release Report

Control 7.2

a. Within 60 days after January 1 and July 1 of each year, a report shall be submitted covering the radioactive content of effluents released to unrestricted areas during the previous six months of operation.

b. The Semiannual Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents 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.

In addition, the Semiannual Effluent Release Report to be submitted 60 days after January 1 of each year also shall include an annual summary[a] 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[b]. This same report shall include an assessment of the radiation doses[a] due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. (The Semiannual Effluent Release Report submitted within 60 days of July 1 each year need not contain any dose estimates from the previous six months'

(a) In lieu of including in the Semiannual Effluent Release Rerort required to be submitted within 60 days after January 1 additional information that covers the summary of annual meteorological data and assessment of radiation doses, a supplemental report is permitted to be submitted within 90 days after January 1 containing this information.

(b) In lieu of submission with the first half-year Semiannual Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data in a file that shall be provided to the NRC upon request.

effluent releases.) This report also shall include an assessment of the radiation doses from radioactive effluents to MEMBER(S) OF THE PUBLIC due to the allowed recreational activities inside the SITE BOUNDARY (Figures 1-1 and 1-2) during the previous calendar year. All assumptions used in making these assessments (e.g., specific activity, exposure time, and location) shall be included in the report. For batch or discrete gas volume releases, the meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. For radioactive materials released in continuous effluent streams, quarterly average meteorological conditions concurrent with the quarterly release period shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the ODCM.

The Semiannual Effluent Release Report to be submitted 60 days after January 1 of each year also shall 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 40CFR190, "Environmental Radiation Protection Standards for Nuclear Power Operation," if Control 3.2 has been exceeded during the calendar year.

The Semiannual 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.

The Semiannual Effluent Release Report shall include any changes made during the reporting period to the ODCM, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Control 4.2.

7.3 <u>Major Changes to Liquid and Gaseous Radioactive Waste Treatment Systems</u>

<u>Control 7.3</u> Licensee initiated major changes to the liquid and gaseous radioactive waste systems:

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

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

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

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

   4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents 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 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 the PORC.

b. Shall become effective upon review and acceptance by the PORC.

7.4 Special Reports

Control 7.4 Special Reports shall be submitted pursuant to 10CFR50.4 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference controls:

a. Liquid Effluents, Controls 3.1 and 6.1.
b. Gaseous Effluents, Controls 3.4, 3.5, and 6.2.
c. Total Dose, Control 3.2.
d. Radiological Environmental Monitoring, Control 4.1.

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 10CFR Part 50, Appendix I," U.S. Nuclear Regulatory Commission, Revision 1, October 1977.

b. Hamawi, J. N., "AEOLUS - A Computer Code for Determining Hourly and Long-Term Atmospheric Dispersion of Power Plant Effluents and for Computing Statistical Distributions of Dose Intensity From Accidental Releases," Yankee Atomic Electric Company, Technical Report, YAEC-1120, January 1977.

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, March 1976.

d. NEP 1 and 2 Preliminary Safety Analysis Report, New England Power Company, Docket Nos. STN 50-568 and STN 50-569.

e. Yankee Atomic Technical Specifications.

f. Yankee Atomic Electric Company Supplemental Information for the Purposes of Evaluation of 10CFR Part 50, Appendix I, Amendment 2, October 1976 (Transmitted by J. L. French - YAEC to USNRC in letters dated June 2, 1976; August 31, 1976; and October 8, 1976).

g. 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.

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

i. TDR-122374, "Isotopic Standardization of Yankee Rowe Vent Stack Monitor."

APPENDIX A

DISPOSAL OF SEPTAGE

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

MAY 17 1990

Docket No. 50-029

Mr. George Papanic, Jr.
Senior Project Engineer - Licensing
Yankee Atomic Electric Company
580 Main Street
Bolton, Massachusetts 01740-1398

Dear Mr. Papanic:

SUBJECT: DISPOSAL OF SEPTAGE - YANKEE NUCLEAR POWER STATION

By letter dated April 11, 1990, you requested NRC approval for a proposed disposal of sewage sludge containing very low concentrations of radionuclide according to 10 CFR 20.302. We have completed our review of your request and our evaluation is enclosed. We have found that your proposed transfer of the sludge by a contracted vendor to a public owned treatment works is acceptable.

Sincerely,

Patrick Sears

Patrick Sears, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosed:
As stated

cc w/encl:
See next page

Mr. George Papanic, Jr.

cc:
Dr. Andrew C. Kadak, President
and Chief Operating Officer
Yankee Atomic Electric Company
580 Main Street
Bolton, Massachusetts 01740-1398

Thomas Dignan, Esquire
Ropes and Gray
225 Franklin Street
Boston, Massachusetts 02110

Mr. T. K. Henderson
Acting Plant Superintendent
Yankee Atomic Electric Company
Star Route
Rowe, Massachusetts 01367

Resident Inspector
Yankee Nuclear Power Station
U.S. Nuclear Regulatory Commission
Post Office Box 28
Monroe Bridge, Massachusetts 01350

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Robert M. Hallisey, Director
Radiation Control Program
Massachusetts Department of Public Health
150 Tremont Street, 7th Floor
Boston, Massachusetts 02111

Mr. George Sterzinger
Commissioner
Vermont Department of Public Service
120 State Street, 3rd Floor
Montpelier, Vermont 05602

Ms. Jane M. Grant
Senior Engineer - PLEX Licensing
Yankee Atomic Electric Company
580 Main Street
Bolton, Massachusetts 01740-1398

SAFETY ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-029

1.0 INTRODUCTION

By letter of April 11, 1990, the Yankee Atomic Electric Company (Yankee) submitted, pursuant to 10 CFR 20.302(a), a method for the routine disposal of septic tank waste containing very low levels of licensed material. Yankee proposed to periodically dispose of accumulated septic waste solids from the plant's sanitary system septic tank by transferring them to a public Sanitary Waste-Water Treatment Facility (SWTF) where they will be mixed with, processed with, and disposed as part of the sanitary waste generated from many sources. Yankee proposed to make such disposals every one to two years over a period of 30 years.

In the submittal, the licensee addressed specific information requested in accordance with 10 CFR 20.302(a), provided a detailed description of the licensed material, thoroughly analyzed and evaluated the information pertinent to the effects on the environment of the proposed disposal of the licensed material, and committed to follow specific procedures to minimize the risk of unexpected or hazardous exposures.

2.0 WASTE WATER STREAM DESCRIPTION

2.1 Physical and Chemical Properties

The waste involved consists of residual septage (the accumulated settled and suspended solids and scum) produced by the sanitary sewerage collection and treatment system at the Yankee plant. To safely dispose of the plant's sanitary waste stream, the Yankee plant supplements the onsite septic system supplemented with offsite treatment at a SWTF.

The onsite septic system consists of a 7,000-gallon buried septic tank and a subsurface soil-absorption leach field. In the overall system design, the septic tank collects sludge and scum and partially separates liquids from the incoming sanitary waste.

The septage is retained in the septic tank, and the remaining conditioned waste-water liquid flows into the underground leaching field for treatment. The leach field is the terminal point of the onsite portion of the plant sanitary waste treatment process.

In the offsite portion of this process, the septage is removed from the septic tank and transported to a SWTF.

2.2 Radiological Properties

The plant's sanitary system septic tank collects waste from the lavatories, showers, and janitorial facilities outside the Radiological Control Area (RCA). No radioactivity is intentionally discharged to the septic system. However, plant investigations into the source of low levels of licensed material found in septic tank waste have identified very small quantities of radioactive materials, which are below detection limits for radioactivity releases from the RCA. It is suspected that these materials are carried out of the control area on individuals and spread to floor areas outside the RCA. Floor wash water from these areas is poured through a filter bag to remove suspended solids and dirt before the water is released into a janitorial sink. Although the wash water is returned to the RCA for disposal, if it is known to contain radioactivity, very small quantities can be released to, and accumulate in the sceptic tank.

The following values are estimates of the maximum total activity presently in the septic tank based on measurements of radionuclide concentrations in the liquid and solid phases:

Nuclide	Total Activity (uCi)
Co-60	1.94
Mn-54	0.057
Cs-134	0.082
Cs-137	0.248
TOTAL	2.33

3.0 PROPOSED DISPOSAL METHOD

Yankee proposes to periodically dispose of accumulated septage from its septic tank by contracting with a septic tank pumper that is approved by the Board of Health, Rowe, Massachusetts and transfer the septage to a Massachusetts SWTF for treatment. This septic tank pumper will transfer the septage to an SWTF, where it is mixed and diluted with other raw sewage and introduced either into an anaerobic digester or an aeration pond for biological treatment. The resulting processed sludge from the SWTF is then mixed with sand and disposed of in a sanitary landfill, where it will be covered by clean soil daily. An alternate disposal means could result in the processed sludge being spread as a fertilizer, though generally for vegetation, such as sod, which is not consumed by humans. None of the region's SWTFs that receive sewage from local septic tank pumpers incinerate their sludge as a means of treatment.

This method of pumping the tank and transferring the septage to an SWTF is the same method normally applied to septic tank systems, regardless of the presence of licensed material.

3.1 Septic Tank Waste Procedural Requirements and Limits

The licensee will perform a gamma isotopic analysis on a representative sample of waste from the septic tank no more than 48 hours before a contracted septic tank pumper begins to pump the waste from the tank to transfer to a SWTF. The licensee will collect at least two septage samples from the plant's septic tank by taking a volumetric column sample that will allow the licensee to determine the ratio of the solid content to the total content of the tank. By determining the weight of the percentage of solid content of the collected sample and applying this value to the gamma isotopic analysis, the licensee will be able to estimate the total radioactivity of the contents of the tank.

To document the estimation of radiological effect of septage disposal, the licensee will perform these gamma isotopic analyses of the representative samples at the Technical Specification Environmental Lower Limit of Detection (LLD) requirements for liquids, as required in Technical Specification Table 4.12-1, "Detection Capabilities for Environmental Sample Analysis,"

The radionuclide concentrations and total radioactivity identified in the septage will be compared to the concentration and total curie limits established herein before disposal. The following limits apply to these analyses:

1. The concentration of radionuclides detected in the volume of septage to be pumped to a disposal truck shall be limited to a combined sum of fractional Maximum Permissible Concentrations in Water (MPC) (as listed in 10CFR Part 20, Appendix B, Table II, Column 2), summed over all nuclides present, of less than or equal to 1.0.

2. The total gamma activity that can be released during septage transfer to any SWTF or combination of such facilities in one year (12 consecutive months) is limited to not more than 20 microcuries (equivalent to a maximum whole-body dose of 1 mrem to any individual in the public).

3.2 Administrative Procedures

The licensee will maintain complete records of each disposal. In addition to copies of invoices with approved septic tank pumpers, these records will include the concentration of radionuclides in the septage, the total volume of septic waste disposed, the total activity in each batch, and the total accumulated activity of the septage pumped in any 12 consecutive months.

For periods in which disposal of septage occurs under this application, the licensee shall report, to the Nuclear Regulatory Commission (NRC) in the plant's Semiannual Effluent Release Report, the volume, liquid, and solid mass fractions, radionuclide concentrations in the liquid and solid fractions, and the total activity disposed.

4.0 EVALUATION OF ENVIRONMENTAL IMPACT

The proposed method for disposal of septage is the same as currently used by all facilities designed with septic tanks for the collection of septic waste. No new structures or facilities need be built or modified, nor any existing land uses changed. Septage from Yankee will be transnorted to an existing SWTF, where it will make up a small fraction of the total volume of sanitary waste treated each year. The normal method of septage handling and treatment would involve dilution of Yankee's septage with other waste-water at a public SWTF. The processed sludge from the SWTF is usually buried in a sanitary landfill, thus limiting the potential exposure pathways to man. Otherwise, the sludge is widely dispersed in fertilizer, thereby preventing any buildup of activity from successive annual pumpouts from the plant's septic tank. This method of disposal will not affect topography, geology, meteorology, hydrology, or nearby facilities.

5.0 RADIOLOGICAL IMPACTS

The licensee has evaluated the following potential exposure pathways to members of the general public: (1) inhalation of resuspended radionuclides, (2) ingestion of food grown on the disposal site, (3) external exposure to a truck driver or SWTF worker, and (4) external exposure caused by long-term buildup and external exposure from standing on the ground above the disposal site. The staff has reviewed the licensee's calculational methods and assumptions, and finds that they are consistent with regulatory Guide 1.109.[1] The staff finds the assessment methodology acceptable.

[1]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.

Doses calculated in this manner by the licensee for the maximum exposed member of the public were as follows (based on a total activity awaiting disposal of 2.3 μCi, more than 80% of which is Co-60):

Pathway	Maximally Exposed Individual/Whole Body (Child) (mrem/year)
Ground Irradiation	0.099
Inhalation	0.0001
Stored Vegetables	0.0214
Leafy Vegetables	0.0011
Milk Ingestion*	(0.0036)
TOTAL	0.12

The licensee then performed a similar calculation using a concervative upper bound activity of 20 μCi to be discharged in any one year. Based on this upper bound analysis, the dose to the maximally exposed individual member of the general public was estimated to be 1.1 mrem/year, as shown in the following table:

Pathway	Maximally Exposed Individual/Whole Body (mrem/year)
Ground Irradiation	0.980
inhalation	0.0004
Stored Vegetables	0.13
Leafy Vegetables	0.007
TOTAL	1.1

Based on this same total activity, the dose to truck drivers and SWTF workers was estimated to be 0.01 mrem/yr. These doses are within the design objectives of 10 CFR 50, Appendix I and well within the environmental standards for uranium fuel cycle activities as stated in 40 CFR 190.10(a) and are therefore acceptable.

6.0 SUMMARY AND CONCLUSIONS

The disposal of septage by transferring it to a public SWTF is in accordance with standard practices for treatment of the type of waste material generated by a septic tank and leach field sanitary waste system. Periodic pumping of the septic tank is necessary for the maintenance and continued operation of Yankee's sanitary waste system. Yankee requested approval for disposal of septic waste from the Yankee sanitary system to prevent failure of the sanitary system to adequately handle plant domestic waste.

An alternate means of disposal would involve the treatment of the septage as radwaste. Such a disposal would require that the licensee stabilize, solidify, and dispose of the material at a licensed burial ground, requiring excessive cost and valuable disposal ground.

The results of the radiological analysis indicate that the public health effects of the biological activity and pathogenic constituents of such sanitary waste far outweigh the concerns related to any radioactivity that is present. By setting release limits that restrict the exposure for an individual to a maximum value of 1 millirem per year, Yankee ensured that radiological risks from the proposed disposal method are insignificant.

The proposed release limits represent a small fraction of NRC limits permitted for disposal of similar waste by licensed facilities who have their sanitary systems connected directly to a public sanitary sewerage system. These proposed limits are also well within the plant's allowable release limits for the discharge of normal liquid waste to the environment. Any resulting dose to any individual in the public is less than exposures caused by natural background radiation.

Based on our review of the proposed disposal of septage, the staff makes the following conclusions: (1) the radionuclide concentrations in disposed septage will be a small percentage of permissible standards set forth in 10 CFR Part 20; (2) the radiation risk to workers involved in the disposal would be small compared to the routine occupational exposures at the Yankee Nuclear Power Station; (3) because the proposed action involves such very low levels of radioactivity, it will require no change in the decommissioning aspects of the facility and will require only insignificant changes in the handling or transport of radioactive material (septage); and (4) the licensee's procedures with commitments as documented in the submittal are acceptable, provided that the submittal is permanently incorporated into the licensee's Offsite Dose Calculation Manual (ODCM) as an Appendix, and future modifications will be reported to NRC in accordance with licensee commitments regarding ODCM changes.

Contributors: J. Minns
P. Sears

YANKEE ATOMIC ELECTRIC COMPANY

TWX 710-380-7619

YANKEE

580 Main Street, Bolton, Massachusetts 01740-1398

April 11, 1990

BYR #90-42

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

References: (A) License No. DPR-3 (Docker No. 50-29)

Subject: 10 CFR 20.302 Application

Dear Sirs:

Pursuant to 10 CFR 20.302, Yankee has prepared the attached application for the routine disposal of septage from Yankee Nuclear Power Station. This application utilizes guidance contained in NRC regulation 10 CFR 20.303 for the disposal of licensed material into a sanitary sewerage system.

We trust that you will find this submittal satisfactory, however, if you have any questions please contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

George Papanic Jr.

George Papanic, Jr.
Senior Project Engineer
Licensing

Enclosure

GP/emd

Revision 7 Date MAY 21 1990 Approved By:

YANKEE NUCLEAR POWER STATION

APPLICATION FOR APPROVAL

TO ROUTINELY DISPOSE OF

SEPTAGE UNDER 10CFR20.302

Revision - Date: MAY 21 1990 A-16 Approved By: [signature]

TABLE OF CONTENTS

Revision 7 - Date: MAY 21 1990 A-17 Approved By: Mark S Strum

LIST OF TABLES

LIST OF FIGURES

YANKEE NUCLEAR POWER STATION

Application for Approval
to Routinely Dispose of Septage Under 10CFR20.302

1.0 INTRODUCTION

Yankee Atomic Electric Company (YANKEE) requests approval, pursuant to 10CFR20.302(a), of a method proposed herein for the routine disposal (typically, once every one to two years) of septic tank waste containing very low levels of licensed material over an extended period of time of 30 years. Yankee proposes to periodically dispose of accumulated septic waste solids from the plant's sanitary system septic tank by transferring it to a public Sanitary Waste-Water Treatment Facility (SWTF) where it will be mixed with, processed, and disposed of, as part of sanitary waste generated from many sources. This is analogous to other Nuclear Regulatory Commission (NRC) licensed facilities who have their sanitary waste systems connected directly to a municipal sewer line. Part 20.303 of Title 10 to the Code of Federal Regulations already permits these NRC licensees to discharge licensed material into a sanitary sewerage system.

Routine maintenance of Yankee's septic system is necessary to ensure proper operation of the system. Periodic pumping of the septic tank to remove accumulated solids is necessary to prevent the carryover of solids into the subsurface leach field which would inhibit the soil absorption capabilities of the field.

This application addresses specific information requested in 10CFR20.302(a), and demonstrates that the periodic disposal of septage from Yankee's Sanitary Waste System over an extended periods of time (30 years), under both normal and unexpected conditions, will not result in significant impacts either to the environment or to individuals in the general public.

2.0 WASTE WATER STREAM DESCRIPTION

2.1 Physical/Chemical Properties

The waste involved in this application consists of residual septage (accumulated settled and suspended solids, and scum) associated with the sanitary sewerage collection and treatment system at the Yankee plant. The Yankee plant utilizes an on-site septic system supplemented with off-site treatment at a SWTF for the safe disposal of the plant's sanitary waste stream. Figure 1 is a schematic of the overall sanitary waste disposal process.

The on-site septic system consists of a 7,000 gallon buried septic tank and a subsurface soil absorption leach field. Sanitary sewage from the plant flows (estimated 2,600 gallons/day) into the septic tank. The septic tank function in the overall system design is for the collection of sludge and scum and partial separation of liquids from the incoming sanitary waste. Some of the solid particles settle to the bottom and form a layer of sludge, where greases and oils float to the surface creating a scum layer.

The septage is retained in the septic tank and the remaining conditioned waste-water liquid is permitted to flow into the underground leaching field for treatment. The leach field is the terminal point of the on-site portion of the plant sanitary waste treatment process. Some of the septage stored in the septic tank is reduced to liquid by bacterial action in the septic tank, but the rest of the septage remains essentially untreated. This material must be pumped out at regular intervals to prevent it from overflowing the tank and entering the leaching field (References 1, 2, 3, 4, 5, 6, 7, 8, 9, and 10) where it will clog the soil and eventually lead to septic system failure.

In general, septage pumped from septic tanks is discharged to a SWTF for treatment as part of the overall system design (Reference 10). The septage is then co-treated with other sanitary wastes at the SWTF. The septage pumped periodically from the Yankee plant has, in the past, been treated and disposed of in this fashion when no licensed material was determined to be present.

The removal of the septage from the septic tank and subsequent transportation to a SWTF constitutes the off-site portion of the Yankee plant overall sanitary waste disposal process.

2.2 Radiological Properties

The plant's sanitary system septic tank collects waste from the lavatories, showers, and janitorial facilities outside the Radiological Control Area (RCA). No radioactivity is intentionally discharged to the septic system. However, plant investigations into the source of low levels of licensed material found in septic tank waste have identified that very small quantities of radioactive materials, which are below detection limits for radioactivity releases from the RCA, appear to be carried out of the control area on individuals and accumulate in the septic tank. The suspected primary source of the radioactivity (i.e., floor wash water) is now either poured through a filter bag to remove suspended solids and dirt before the water is released into a janitorial sink, or the wash water is returned to the RCA for disposal.

An isotopic analysis, at environmental detection limits, of two composite volumetric sample columns of septage taken from the plant's septic tank identified the following plant-related radionuclides:

Nuclide	Activity Concentration (pCi/kg wet +/- 1 sigma)	
	West Manhole Sample Location	East Manhole Sample Location
Co-60	92.4 ± 3.9	13.2 ± 2.2
Cs-134	5.9 ± 1.3	--
Cs-137	9.2 ± 1.5	3.2 ± 1.0

After the initial analysis of the composite samples noted above, the samples were subsequently centrifuged into liquid and wet solids portions and reanalyzed. There was no activation or fission products identified in any of the liquid fraction samples indicating that the detected activity was in a form that had been carried out of solution with the solid fraction of the samples.

Analysis of the resulting solid fraction of the septage indicated the following radionuclide concentrations:

Nuclide	Activity Concentration (pCi/kg wet +/- 1 sigma)	
	West Manhole Sample Location	East Manhole Sample Location
Co-60	1,588 ± 42	528 ± 26
Mn-54	47 ± 13	--
Cs-134	67 ± 11	--
Cs-137	203 ± 17	100 ± 13

The original septic tank samples were volumetric samples representative of the distribution of solids and liquid from bottom to top of the tank. The ratio of the weight of the solid fraction sample to the weight of the solid fraction plus liquid fraction sample allows a determination of the percentage of total solids content of the septic tank. For the waste sample from the west manhole, the solid fraction of the composite sample was found to be 0.024, or 2.4 wt. %. For the east manhole, the solid fraction of the total sample was 0.046, or 4.6 wt.%. The principle radionuclide is Cobalt-60, which accounts for approximately 82% of all plant-related activity detected in the septage.

The total radioactivity content of the septic tank can be estimated by calculating the mass of solids present in the 7,000 gallon tank by taking the higher (conservative) solids fraction determined from the sample data. This is multiplied by the mass of septage in the tank and by the highest activity concentration determined in the solids. As a result, the estimated maximum total activity is:

Nuclide	Total Activity (µCi)
Co-60	1.94
Mn-54	0.057
Cs-134	0.082
Cs-137	0.248
TOTAL	2.33

3.0 PROPOSED DISPOSAL METHOD

Upon approval from the U.S. Nuclear Regulatory Commission (NRC), Yankee proposes to periodically dispose of accumulated septage from its septic tank by contracting with a town-approved (Board of Health, Rowe, Massachusetts) septic tank pumper for the removal and transfer by truck of the septage to a Massachusetts SWTF for treatment. At the SWTF, the septage would typically be mixed and diluted with other raw sewage and introduced either into an anaerobic digester or aeration pond for biological treatment. The resulting processed sludge from the SWTF is typically then mixed with sand in a ratio of 50/50 and disposed of in a sanitary landfill, where it would be covered by clean soil daily. An alternate disposal means could potentially result in the processed sludge being landspread as a fertilizer, though generally for nonhuman-consumed vegetation, such as sod. None of the regions SWTFs which would be used by local septic tank pumpers were identified as incinerating their sludge as a means of treatment.

This method of tank pumping and transfer to an SWTF is identical to that normally applied to septic tank systems, irrespective of the presence of licensed material. Once the septage is pumped into the contract vendor's transporting vehicle, the situation is analogous to the handling of licensed material under 10CFR20.303. Part 20.303 of Title 10 to the Code of Federal Regulations already permits these NRC licensees to discharge licensed material into a sanitary sewerage system if certain conditions are met. Due to the remoteness of the Yankee plant's location, it is impractical to directly connect sewer lines to a facility to handle sanitary waste. In this case, a tank truck acts as a sewer line in transferring septage to a SWTF. The quantity and form (soluble or dispersable) of any licensed material contained in our septage is not affected by the means employed to transfer it to a SWTF for processing. Therefore, it would be the same whether the plant was directly connected to a municipal sewerage system or trucked its septage on a periodic basis to a SWTF.

3.1 Septic Tank Waste Procedural Requirements and Limits

Gamma isotopic analysis of septic tank waste shall be made prior to transfer of the waste by a contracted septic tank pumper to a SWTF by obtaining a representative sample from the tank no more than 48 hours prior to initiating pump-out. At least two septage samples shall be collected from the plant's septic tank by taking a volumetric column sample which will allow for analysis of the solid's content of the tank. The weight percent of solid content of the collected sample will be determined and applied to the gamma isotopic analysis in order to estimate the total radioactivity content of the tank.

These gamma isotopic analyses of the representative samples will be performed at the Technical Specification Environmental Lower Limit of Detection (LLD) requirements for liquids (see Technical Specification Table 4.12-1, "Detection Capabilities for Environmental Sample Analysis") 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 limits to be applied are as follows:

1. The concentration of radionuclides detected in the volume of septage to be pumped to a disposal truck shall 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 1.0.

2. The total gamma activity which can be released via septage transfer to any SWTF in one year (12 consecutive months) is limited to not more than 20 microcuries (equivalent to a maximum whole body dose of 1 mrem to any individual in the public).

If the total activity limit is met, compliance with the self-imposed dose criteria will have been demonstrated since the radiological impact (Section 5) is based on evaluating the exposure to a maximally exposed hypothetical individual such that his annual whole body dose would be limited to approximately 1 mrem.

Both the concentration and total activity limits represent a small fraction of the allowable limits permitted under 10CFR20.303 to other NRC licensees who have their sanitary waste systems directly connected to a public sewerage system. If not for the biological nature of sanitary waste, the above release limits would also allow for the direct discharge of the waste under the plant's existing Technical Specification requirements for release of liquids to the environment.

3.2 Administrative Procedures

Complete records of each disposal will be maintained. In addition to copies of invoices with approved septic tank pumpers, 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 activity pumped in any 12 consecutive months.

For periods in which disposal of septage occurs under this application, the volume, total activity, and relative nuclide distribution, shall be reported to the NRC in the plant's Semiannual Effluent Release Report.

4.0 EVALUATION OF ENVIRONMENTAL IMPACT

The proposed method for disposal of septage is the same as currently used by all facilities designed with septic tanks for the collection of septic waste. No new structures or facilities need be built or modified, nor any existing land uses changed. Septage from Yankee will be trucked to an existing SWTF, where it will make up a small fraction of the total volume of sanitary waste treated each year. As a result, there will be no impact on topography, geology, meteorology, hydrology, or nearby facilities by the proposed method of disposal.

5.0 EVALUATION OF RADIOLOGICAL IMPACT

Radiological evaluations have been performed for the purpose of bounding the dose impact associated with the disposal of septage. The normal method of septage handling and treatment would provide for dilution of Yankee's septage with other waste-water at a public SWTF. The processed sludge would typically be buried in a sanitary landfill, thus limiting the potential exposure pathways to man, or widely dispersed if used as a fertilizer, thereby preventing any build-up of activity from successive annual pumpouts from the plant's septic tank. The dose assessments, however, did consider the maximum potential impact of long-term buildup of activity resulting from 30 years of placing septage waste in the same SWTF, with all the processed sludge assumed to be buried in one landfill disposal cell.

5.1 Septic Tank Sample Analysis Data

The analysis of the septic tank's measured radioactivity, and its distribution between liquid and solid fractions, provides the bases upon which a dose assessment of disposal of septage can be made. The composition of the septic tank waste determined from the sample analysis is:

	Composite Sample East End Manhole Location	Composite Sample West End Manhole Location
Wt. Liquid	3.502 kg	3.460 kg
Wt. Solid	0.087 kg	0.167 kg

Solid fraction of the composite sample as collected is equal to:

Solid fraction = Wt. solid/(Wt. solid + Wt. liquid)

The solid fraction for the East End sample was 0.0242, and 0.0460 for the West End. The activity in the solid fraction was basically found to contain all the detected radioactivity as noted below:

	East End Solids Sample (pCi/kg) Wet	West End Solids Sample (pCi/kg) Wet
Mn-54	-	47
Cs-134	-	67
Cs-137	100	203
Co-60	528	1,588

With the septic tank volume taken as approximately 7,000 gallons (26,500 liters), and assuming the maximum solid fraction (0.046) and maximum radionuclide concentration applies to the total tank's content, the total maximum radioactivity content is estimated to be:

Isotope	Half-Life	Qe (Ci)
Mn-54	312.2 day	5.73 E-08
Co-60	5.272 yr	1.94 E-06
Cs-134	2.065 yr	8.17 E-08
Cs-137	30.17 yr	2.48 E-07

5.2 Pathway Exposure Scenarios

Radiological evaluations were performed for both the expected activities associated with handling, processing, and disposal of septage waste at a SWTF, and a hypothetical event causing undiluted septage release. The bounding case was determined to be associated with a hypothetical event which lead to the spreading of undiluted septage from Yankee's septic tank directly on a garden area where food crops are grown. The contracts with town approved septic tank pumpers will direct that septage be disposed of at a SWTF in Massachusetts. It is not expected that any disposal will occur other than at an SWTF. It is, therefore, not considered credible that successive bounding case activities could occur which lead to a long-term buildup of activity on a single minimum size garden plot.

In addition, since incineration of septic waste is not a current practice in the local area, the potential exposures associated with incineration are not of current concern. However, the establishment of a conservative total whole body dose criteria for release of sanitary waste, via the above-noted garden scenario, assures that the potential resulting whole body dose due to incineration would not be expected to result in significant doses to any individual. This assessment is further detailed in Section 5.3.4.

The contributing pathways of exposure for the normal SWTF disposal process include:

1. External exposure to a truck driver.
2. External exposure to a SWTF worker.

3. External exposure to an individual standing on the SWTF landfill after 30 years of buildup and decay.

The following garden exposure pathways were addressed for the maximally exposed hypothetical individual:

1. Standing on the ground plane.
2. Inhalation of resuspended material.
3. Ingestion of leafy vegetables.
4. Ingestion of stored vegetables.
5. Ingestion of milk.
6. Liquid pathways.

It should be noted that the milk pathway is mutually exclusive to the other food production pathways since it would be impossible to support the grass-cow-milk-man exposure chain if the limited land area is utilized for the growing of food crops for direct human consumption. The two sets of ingestion pathways have been calculated so that the potential maximum impact can be assessed. Similarly, radionuclide movement into the ground water pathway would tend to reduce the impact of surface-related exposure paths and is, therefore, considered independently.

5.3 Dose Assessments

5.3.1 External Exposure to a Truck Driver/SWTF Worker

The external dose rate from a 3,500-gallon tank truck filled with septage containing the total measured activity in the septic tank (2.33 µCi) was calculated for the purpose of estimating exposures associated with shipping the waste to a SWTF. A three-dimensional point-kernel shielding code for the determination of direct radiation from gamma radiation emanating from a self-attenuating cylindrical source (DIDOS-IV, Reference 14) was utilized to calculate the external dose rate from the tank truck. The truck was modeled as a cylindrical radiation source with a radius equal to 1.22 meters and a length of 2.84 meters. A dose rate of 1.2E-04 mrem per hour for a point one meter from the end of the cylinder along the axis was calculated. No credit for shielding provided by the tank truck or cab was assumed. The dose to a truck driver making a 100-mile trip to a treatment facility at an average of 20 miles per hour plus a three-hour waiting period at the SWTF, is estimated to be 9.5E-04 mrem. It is concluded, based on the total activity limits proposed, that this pathway will not lead to significant exposure of any individual. It is also concluded that due to the sanitary properties of septage handling, a SWTF employee's direct exposure time is kept to a minimum. Using the dose rate estimated for the truck driver above, and conservatively assuming that it requires an employee at the SWTF a full eight-hour day to process each truckload of waste, and not taking any credit for dilution or increased distance from the waste, a waste processing facility employee's dose is also estimated to be 9.5E-04 mrem.

If the maximum activity content proposed to be disposed of each year were assumed as the source term (20 µCi), the dose to the truck driver/SWTF worker is estimated to be less than 1.0E-02 mrem using the same assumptions as noted above.

5.3.2 External Exposure Due to Long-Term Buildup

In order to assess the potential impact from the postulated buildup of activity resulting from 30 years of septage disposed at the maximum annual allowed activity content, it was conservatively assumed that the entire quantity of accumulated activity at the end of 30 years was buried in a common landfill disposal cell which was then available to the general public for uncontrolled access (8,760 hours per year).

For regional SWTFs, waste sludge is typically mixed with sand and placed in landfill disposal cells on a daily basis and covered by a layer of at least six inches of composited material before the end of each working day, as required by Massachusetts Department of Environmental Protection regulations (Reference 16). The landfill disposal cells range in size from about one acre up to about five acres. After a cell is full, a final layer of compacted material is required to be placed over the entire surface of the cell to a minimum depth of two feet (Reference 16).

Analytically, if Q_o is the amount of radioactivity per tank full of septage for a give nuclide, then the total accumulated radioactivity $Q_e(max)$ disposed of after 30 pumpouts is given by:

$$Q_e(max) = Q_o \ (1 + E + E^2 + E^3 + E^4 + ... + E^{29})$$

$$= Q_o \ (1 - E^{29})/(1 - E) \qquad (A)$$

where:

$E = \exp(-\lambda\Delta t)$

λ = is the decay constant for the selected nuclide (1/year), and

Δt = time interval between applications, assumed to be 1 year.

If the maximum total activity of 20 microcuries (with the same relative distribution as determined in the current septic tank analysis) were assumed to be released each year, then the accumulated activity at the end of 30 years is found in the following table:

Nuclide	Half Life	λ (1/year)	Q_o (uCi/batch)	$Q_e(max)$ uCi
Co-60	5.27 y	0.1315	16.65	132.14
Mn-54	312. d	0.8109	0.49	0.88
Co-134	2.07 y	0.3357	0.70	2.45
Co-137	30.2 y	0.023	2.15	46.04
		Total	20	182

If the 20 microcuries per year limit is assumed to be all Co-60, then the resulting accumulated total after 30 years would be 159 microcuries, and result in a higher calculated dose than that from the above mix.

Assuming a minimum landfill disposal cell to be one acre in area, and that the 30-year accumulated activity (159 uCi; Co-60) was disposed of in one year along with SWTF sludge that formed a minimum one foot layer which was placed immediately below the two-foot disposal cap of the cell, the resulting dose rate one meter above the ground surface was calculated to be 6.4E-07 mrem/hour. If it is also assumed that an individual remained on the landfill for a full year (8,760 hours) without taking any credit for shielding by a residential structure, the total whole body dose would be 5.6E-03 mrem, or about 56% of the truck driver's/SWTF workers calculated exposure.

Since the landfill cap (2' minimum) effectively isolates the vegetation zone of the top 15 cm plow layer, no garden pathways of exposure are included. However, it is noted that the 30-year accumulated activity concentration spread over a one acre landfill disposal cell would result in an area density of only 3.7E-03 microcuries per square foot. This is approximately a factor of 11 below the surface area density of the garden pathway scenario in Section 5.3.3 for the bounding case of placing 20 microcuries directly on a 500 ft^2 garden. Therefore, even if it is postulated that an individual were to dig a cellar hole for a new home on the landfill site after closure, the resulting dose impact would still be bounded by the garden scenario as described below.

It is, therefore, concluded that for normal handling, processing, and disposal of septage at a SWTF, the maximum annual dose is received by the truck driver or SWTF worker handling the annual batches of septage pumped for disposal, and not the result of accumulated activity buildup over extended time periods.

5.3.3 Garden Pathway Scenario

The radiological impact associated with an event which place undiluted septage directly on a garden was carried out using the dose assessment models in Regulatory Guide 1.109 (Reference 13), and in a manner consistent with the methodology employed by the plant's ODCM. Special consideration was given to the following:

1. The computation of an effective self-shielding factor to account for the effect provided by the soil after the waste is plowed or mixed in the top 15 cm surface layer.

2. The definition of an annual activity release rate, which following a year's time of continuous release, would yield the ground deposition expected to prevail after a tank pump-out and spreading on the 500 ft^2 garden.

3. The definition of an effective atmospheric dispersion factor to represent the resuspended radioactivity.

4. The proper representation of partial occupancy factors and usage data.

Landspreading, Resuspension, and Occupancy Factors

If it is assumed that the garden plot is limited to a surface area of 500 ft^2, then the land deposited radioactive material S_e (Ci/m^2) following landspreading will be equal to:

$$S_e = Q_e \ (Ci)/(500 \ ft^2 * 0.0929 \ m^2/ft^2) \tag{B}$$

The denominator of this equation is equivalent to the (D/Q) deposition factor normally employed in the airborne impact assessment of deposited radionuclides; that is:

$$(D/Q) = 1/(500 \ ft^2 * 0.0929 \ m^2/ft^2)$$

$$= 2.15E\text{-}02 \ (m^{-2}) \tag{C}$$

Following the application of undiluted septage on the garden, some of the radioactivity may become airborne as a result of resuspension effects. The model used to estimate the radionuclide concentration in air above the disposal plot was taken from WASH-1400, Appendix VI. According to that model, the relationship between the airborne concentration A_e (Ci/m^3) and the surface deposition is:

$$A_e = S_e \ (Ci/m^2) \times K \ (1/m) \tag{D}$$

where: K is the resuspension factor and is taken to be equal to 1.0E-06 (1/m) (Reference 11) which is believed conservative due to the limited surface area involved and the irrigation provided to a garden which minimizes airborne dust.

The 500 ft^2 garden size was selected based on the minimum surface area necessary to include a garden as part of the land-use census as required by Yankee's Technical Specification 3/4.12.2. This is the minimum area which could be expected to produce sufficient food to support the uptake assumption on food consumption noted below.

In addition, by limiting the garden surface area to 500 ft^2 (a circle with a 3.85 m radius) the concentration of radioactivity in the garden is maximized since the concentration for any given surface area is physically limited by the total activity available in the septage. For direct radiation estimates from standing on the ground plane, a commonly used assumption of an infinite plane source (which can be approximated by a circle with a radius of 15 m) would in fact undercalculate the surface dose rate from that of a 500 ft^2 garden by a factor of about 8 due to the dispersal of the fixed quantity of activity available to be spread. For use with the garden pathways of exposure, it is assumed that the septage is mixed in the top cultivated 15 cm of soil with no additional clean soil cover placed over it.

As for the occupancy factors for direct exposure to the ground deposition and for immersion in the resuspended radioactivity, 360 hours was used for the radiological impact analysis. The 360-hour interval is believed to be a reasonably conservative time frame a gardener would spend each year on a plot of land or garden during the growing season in the northeast (average two hours a day for six months).

Garden pathway data and usage factors as applicable to the area in the vicinity of the plant are shown below. These are the same factors as used in the plant's ODCM assessment of the off-site radiological impacts due to routine releases from the plant, with the following exceptions:

1. The soil exposure time was changed from 15 years to 1 year to account for the discrete application of septage on a garden plot.

2. The fraction of stored vegetables grown in the garden was conservatively increased from 0.76 to 1.0.

3. The crop exposure time was changed from 2,160 hours to 0 hours to reflect the condition that no radioactive material would be dispersed directly on crops for human or animal consumption, the deposition on crops of resuspended radioactivity being insignificantly small; that is, crop contamination is only through root uptake.

USAGE FACTORS

Individual	Vegetables (kg/yr)	Leafy Veg. (kg/yr)	Milk (liters/yr)	Inhalation* (m^3/yr)
Adult	520	64	310	329
Teen	630	42	400	329
Child	520	26	330	152
Infant	---	---	330	58

*Inhalation rates have been modified to reflect an annual occupancy factor of 360 hours.

VEGETABLE PATHWAY

	Stored Vegetables	Leafy Vegetables
Agricultural productivity (kg/m^2)	2.0	2.0
Soil surface density (kg/m^2)	240.0	240.0
Transport time to user (hours),	0.0	0.0
Soil exposure time (hours)	8,766.0	8,766.0
Crop exposure time to plume (hours)	.0	.0
Holdup after harvest (hours)	1,440.0	24.0
Fraction of stored vegetables grown in garden	1.0	
Fraction of leafy vegetables grown in garden		1.0

COW-MILK PATHWAY

	Pasture Feed	Stored Feed
Agricultural productivity (kg/m^2)	.7	2.0
Soil surface density (kg/m^2)	240.0	240.0
Transport time to user (hours)	48.0	48.0
Soil exposure time (hours)	8,766.0	8,766.0
Crop exposure time to plume (hours)	.0	.0
Holdup after harvest (hours)	.0	2,160.0
Animals daily feed (kg/day)	50.0	50.0
Fraction of year on pasture	.5	
Fraction pasture when on pasture	1.0	

As noted above, liquid exposure pathways are considered independent from those associated with garden exposures. Since the laboratory analysis data of septic tank waste shows that all the activity is associated with the suspended or settled solids fraction, and not dissolved in the liquid portion, transport of activity through groundwater would not be expected to lead to drinking water supplies being impacted by septage placed on farm lands. It is, therefore, not anticipated that the groundwater pathway could result in doses comparable to the direct surface exposure pathways. As confirmation of this, however, a methodology for groundwater analysis, as developed by Kennedy, et al. (1990) (Reference 12), was used as a check. This model assumes that the radionuclides on the ground are leached into the water table with a leach rate based on continuously saturated soil. Once into the water table, the radionuclides are immediately available for consumption. The volume of water used for dilution is limited to the quantity used by one person in one year (91,250 liters). No credit is taken by soil retardation of the nuclides, either during the leaching process or during groundwater movement. Consumption of water is assumed to be 2 liters/day. The resulting dose factors, by radionuclide, are listed in Table 3.4 of Reference 12.

Of the radionuclides detected in the septage, Co-60 is the dominant nuclide, and has the highest dose factors. The total effective dose equivalent from drinking water is 4.4E-6 mrem/yr for 1 µCi of disposed Co-60. The maximum organ dose is 1.9E-5 mrem/year per µCi, with the organ being the LLI wall. These results are several orders of magnitude below the direct surface exposure doses as detailed below. The groundwater pathway is, therefore, not significant.

Direct Ground Plane Exposure

To account for the gamma attenuation provided by the soil, it was necessary to carry out an appropriate shielding calculation. This was accom,lished through use of the DIDOS computer code which computed the radiation levels from a cylindrical volume source with a radius of 3.85 m and a height of 0.15 m, with the receptor located along the axis, 1 m above the source.

The source density was set equal to 1.6 g/cc, which is equivalent to the Regulatory Guide 1.109 value of 240 kg/m^2 for the effective surface density of soil within a 15 cm plow layer. If the total activity content of the septic tank, as listed earlier, were assumed to be uniformly distributed in the source disk, the volume source dose rate is equivalent to a dose rate of 2.8E-04 mrem/hr. The total dose from standing on the garden area for 360 hours each year is seen to be 0.099 mrem from the total activity content measure in the septic tank (2.33 μCi) being placed on the garden.

Garden Pathway Total Dose

The maximum individual ingestion/inhalation exposure assessments resulting from garden crops or pasture grass grown on a septage disposal plot were added to the direct ground plane doses discussed above. This results in a bounding estimate of dose to a hypothetical maximum exposed individual. The whole body and critical-organ radiation exposures after a tank pump-out and spreading on a garden at a concentration level equivalent to the measured concentrations in septic waste are as follows:

	Radiation Exposure	Individual/Organ
Maximum Exposed Individual	0.122 mrem/yr	Child/Whole Body
	0.157 mrem/yr	Child/Liver

The individual pathway contributions to the total dose are as follows:

Pathway-Dependent Critical Organ Doses

Pathway	Maximally Exposed Individual/Organ (Child/Liver) (mrem/year)	Maximally Exposed Individual/Whole Body (Child) (mrem/year)
Ground Irradiation	0.099	0.099
Inhalation	0.0003	0.0001
Stored Vegetables	0.055	0.0214
Leafy Vegetables	0.0028	0.0011
Milk Ingestion*	(0.019)	(0.0036)
TOTAL	0.157	0.122

Tables 1 through 4 detail the internal dose breakdown by radionuclide and pathway of exposure. As can be seen in the results, the whole body and maximum exposed organ dose are appropriately equivalent. This is due to the dominance of the external ground plane exposure pathway controlling the dose to both the organs and whole body.

5.3.4 Incineration Pathway Scenario

At the present time, there are no known facilities for the incineration of septage in the vicinity of the Yankee plant. For completeness, however, we have addressed the radiological impact expected from incineration. This will preclude the necessity of revising this application request if such a facility becomes available in the future.

The basis for the radiological assessment of incineration is a report by Murphy, et al. (1989) (Reference 15), in which they calculated individual and population dose impacts from low level waste disposal scenarios. This report used a radionuclide distribution that was based on extensive studies of

*As described above, the milk pathway is mutually exclusive to the vegetable ingestion pathway; and, therefore, not added into the total.

power reactor low level wastes. This distribution was similar to the measured distribution in the Yankee septage in that Co-60 and Cs-137 were the predominant gamma emitters.

The results of their analyses show that the transport worker receives the highest dose from the incineration scenario. The transport worker dose is approximately a factor of 5 higher than either the maximum incinerator worker or the maximum disposal site operator, and is several orders of magnitude higher than the maximum individual doses to the general public.

The dose to the transport worker has been discussed above (Section 5.3.1) for the off-site disposal of septage from Yankee. This transport worker dose will not change if the septage is incinerated, since it was conservatively assumed that the worker spends 8 hours traveling to the disposal site. Therefore, the dose to the individual landowner, from the garden scenario, will still be controlling for all disposal options, including incineration.

5.4 Maximum Releasable Activity

The above analysis for landspreading on a garden the measured activity levels detected in the septic tank indicates that over 80% of the total whole body dose received by the hypothetical individual is due to direct external exposure to the ground plane. Of this direct dose component, Co-60 accounts for about 96% of the exposure. In determining a practical means by which any future detectable levels of licensed material can be limited to ensure that the controlling hypothetical individual's annual dose is limited to approximately 1 mrem or less, the sum of all measured gamma emitting nuclides can be assessed as Co-60 to determine the quantity of gross activity that, if released in septage, would limit the dose to 1 mrem.

Repeating the above controlling analysis for the event which placed the septage shipment directly on a garden plot, and assuming that the activity available is all Co-60, the total activity which relates to the annual dose limit criteria of 1 mrem is determined to be approximately 20 microcuries. The breakdown by exposure pathway for this scenario, assuming an activity release of 20 microcuries in the form of Co-60 is as follows:

Pathway	Maximum Exposed Individual/Whole Body (mrem/year)
Ground Irradiation	0.980
Inhalation	0.0004
Stored Vegetables	0.13
Leafy Vegetables	0.0068
TOTAL	1.1

All other scenarios for the normal treatment and disposal of septage, including postulated accumulation and build-up of activity at a single SWTF for a 30-year period (at 20 microcuries/year), result in radiological impacts to individuals which are approximately a factor of 100 or more below the whole body dos . for the garden pathway.

The following summary compares the calculated whole body doses associated with normal handling of septage with the 1 mrem bounding event garden scenario. This demonstrates that by limiting the annual quantity of activity in septage to 20 microcuries, the expected dose impact for disposing of septage at a SWTF will in fact be well below a dose criterion of 1 mrem/year:

Scenario	Maximum Whole Body Annual Dose (mrem)
(a) Septic truck driver/SWTF worker. (20 uCi Co-60 per year)	1.0E-02
(b) SWTF landfill after closure. (30-year accumulation; 159 uCi Co-60)	5.6E-03

6.0 SUMMARY AND CONCLUSIONS

The disposal of septage by transferring it to a public SWTF is in accordance with standard practices for treatment of the type of waste material generated by a septic tank/leach field sanitary waste system. Periodic pumping of the septic tank is necessary for the maintenance and continued operation of Yankee's sanitary waste system. Approval for disposal of septic waste from the Yankee sanitary system is requested to prevent failure of the sanitary system to adequately handle plant domestic waste.

Alternate means of disposal of the septage would involve the treatment of it as radwaste, with the subsequent need to stabilize, solidify, and dispose of the material at a licensed burial ground at excessive cost and a loss in valuable disposal ground volume.

The radiological analysis results indicate that the public health effects due to the biological activity and infectious constituents of such sanitary waste far outweigh the concerns due to any radioactivity which is present. By setting release limits which restrict the exposure to a maximum hypothetical individual of 1 mrem per year, it is ensured that radiological risks from the proposed disposal method are of no significance.

The proposed release limits represent a small fraction of NRC limits permitted for disposal of similar waste by licensed facilities who have their sanitary systems connected directly to a public sanitary sewerage system. These proposed limits are also within the plant's current allowable release limits for discharge of normal liquid waste to the environment, with any resulting dose to any individual in the public being far less than committed exposures due to natural background radiation.

7.0 REFERENCES

1. "Design Manual - On-Site Waste-Water Treatment and Disposal Systems," U.S. Environmental Protection Agency, EPA-625/1-80-012, October 1980.

2. "Septage Management," U.S. Environmental Protection Agency, EPA-600/8-80-032, August 1980.

3. "Handbook - Septage Treatment and Disposal," U.S. Environmental Protection Agency, EPA-625/6-84-009, October 1984.

4. "Septic Tank Care, U.S. Department of Health," Education, and Welfare, U.S. Public Health Service, 1975.

5. "Manual of Septic Tank Practice," U.S. Public Health Service, Publication No. 526, 1957.

6. "Your Septic System," Prepared for the Massachusetts Department of Environmental Quality Engineering, Publication No. 10043-32-625-12-77-CR, January 1978.

7. "Septic System", Massachusetts Metropolitan Area Planning Council, 1981.

8. "Septic Systems," Massachusetts Division of Water Pollution Control, Publication No. 12551-24-300-9-81-CR, 1981.

9. Clark, J. W., W. Viessman, and M. J. Hammer, "Water Supply and Pollution Control," International Textbook Company, 1971.

10. Metcalf & Eddy, Inc., "Waste-Water Engineering: Treatment, Disposal, and Reuse," McGraw-Hill, 1979.

11. Cember, H., "Introduction to Health Physics," Page 321, Pergamon Press, 1969.

12. Kennedy, W. E., Peloquin, R. A., "Residual Radioactivity Contamination From Decommissioning," NUREG/CR-5512, January 1990 (Draft Report for Comment).

13. Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR, Part 50, Appendix I, USNRC, Revision 1, 1977.

14. J. N. Hamawi, "DIDOS-III - A Three-Dimensional Point-Kernel Shielding Code for Cylindrical Sources, ENTECH Engineering, Inc., Technical Report P100-R2, December 1982 (updated to Version DIDOS-IV, October 1989, Yankee Atomic Electric Company).

15. Murphy, E. S., Rogers, V. C., "Below Regulatory Concern Owners Group: Individual and Population Impacts From BRC Waste Treatment and Disposal," EPRI NP-5680, Interim Report, August 1989.

16. Massachusetts Department of Environmental Protection Regulations 310 CMR 19.15 (Disposal of Solid Waste in Sanitary Landfills).

TABLE 1

LANDSPREADING INGESTION PATHWAYS

(ADULT)

(2.33 UCI TOTAL ACTIVITY)

(MREM)

PATHWAY	BONE	LIVER	KIDNEY	LUNG	GI-LLI	THYROID	WHOLE BODY
INHALATION							
54 MN	0.00E+00	2.93E-06	7.28E-07	1.04E-04	5.72E-06	0.00E+00	4.66E-07
60 CO	0.00E+00	2.11E-05	0.00E+00	1.09E-02	5.21E-04	0.00E+00	2.71E-05
134 CS	3.17E-05	7.22E-05	2.44E-05	8.31E-06	8.85E-07	0.00E+00	6.19E-05
137 CS	1.07E-04	1.39E-04	4.98E-05	1.68E-05	1.88E-06	0.00E+00	9.58E-05
TOTAL FOR PATHWAY	1.39E-04	2.35E-04	[illegible].49E-05	1.11E-02	5.30E-04	0.00E+00	1.85E-04
STORED VEGETABLES							
54 MN	0.[illegible]E+00	3.10E-04	9.21E-05	0.00E+00	9.48E-04	0.00E+00	5.91E-05
60 CO	0.00E+00	1.78E-03	0.00E+00	0.00E+00	3.34E-02	0.00E+00	3.92E-03
134 CS	2.24E-03	5.33E-03	1.72E-03	5.72E-04	9.32E-05	0.00E+00	4.35E-03
137 CS	9.25E-03	1.27E-02	4.29E-03	1.43E-03	2.45E-04	0.00E+00	8.29E-03
TOTAL FOR PATHWAY	1.15E-02	2.01E-02	6.11E-03	2.00E-03	3.47E-02	0.00E+00	1.66E-02
LEAFY VEGETABLES							
54 MN	0.00E+00	4.34E-05	1.29E-05	0.00E+00	1.33E-04	0.00E+00	8.29E-06
60 CO	0.00E+00	2.24E-04	0.00E+00	0.00E+00	4.20E-03	0.00E+00	4.93E-04
134 CS	2.91E-04	6.92E-04	2.24E-04	7.44E-05	1.21E-05	0.00E+00	5.66E-04
137 CS	1.14E-03	1.56E-03	5.31E-04	1.76E-04	3.03E-05	0.00E+00	1.02E-03
TOTAL FOR PATHWAY	1.43E-03	2.52E-03	7.68E-04	2.51E-04	4.38E-03	0.00E+00	2.09E-03
COW MILK							
54 MN	0.00E+00	2.39E-06	7.10E-07	0.00E+00	7.31E-06	0.00E+00	4.55E-07
60 CO	0.00E+00	5.33E-05	0.00E+00	0.00E+00	1.00E-03	0.00E+00	1.18E-04
134 CS	8.11E-04	1.93E-03	6.25E-04	2.07E-04	3.38E-05	0.00E+00	1.58E-03
137 CS	3.31E-03	4.53E-03	1.54E-03	5.11E-04	8.77E-05	0.00E+00	2.97E-03
TOTAL FOR PATHWAY	4.12E-03	6.51E-03	2.16E-03	7.18E-04	1.13E-03	0.00E+00	4.66E-03

TABLE 2
LANDSPREADING INGESTION PATHWAYS
(TEEN)
(2.33 UCI TOTAL ACTIVITY)
(MREM)

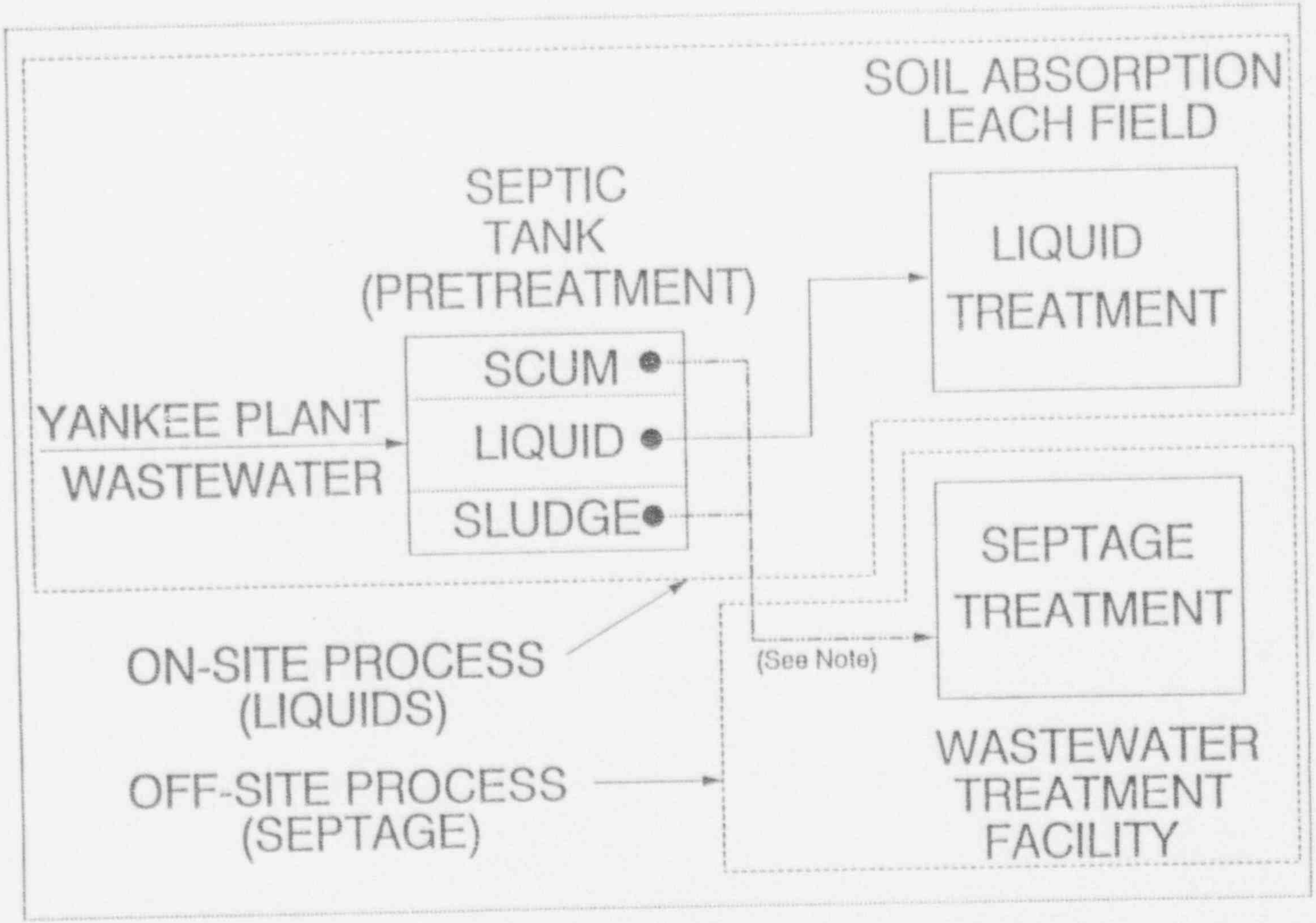
PATHWAY	BONE	LIVER	KIDNEY	LUNG	GI-LLI	THYROID	WHOLE BODY
INHALATION							
54 MN	0.00E+00	3.78E-06	9.41E-07	1.47E-04	4.94E-06	0.00E+00	6.21E-07
60 CO	0.00E+00	2.77E-05	0.00E+00	1.60E-02	4.75E-04	0.00E+00	3.63E-05
134 CS	4.28E-05	9.60E-05	3.19E-05	1.25E-05	8.31E-07	0.00E+00	4.67E-05
137 CS	1.50E-04	1.90E-04	6.80E-05	2.70E-05	1.90E-06	0.00E+00	6.96E-05
TOTAL FOR PATHWAY	1.93E-04	3.17E-04	1.01E-04	1.62E-02	4.82E-04	0.00E+00	1.53E-04
STORED VEGETABLES							
54 MN	0.00E+00	4.84E-04	1.44E-04	0.00E+00	9.93E-04	0.00E+00	9.60E-05
60 CO	0.00E+00	2.83E-03	0.00E+00	0.00E+00	3.69E-02	0.00E+00	6.37E-03
134 CS	3.65E-03	8.59E-03	2.73E-03	1.04E-03	1.07E-04	0.00E+00	3.98E-03
137 CS	1.57E-02	2.10E-02	7.13E-03	2.77E-03	2.78E-04	0.00E+00	7.30E-03
TOTAL FOR PATHWAY	1.94E-02	3.29E-02	1.00E-02	3.81E-03	3.83E-02	0.00E+00	1.78E-02
LEAFY VEGETABLES							
54 MN	0.00E+00	3.68E-05	1.10E-05	0.00E+00	7.55E-05	0.00E+00	7.30E-06
60 CO	0.00E+00	1.93E-04	0.00E+00	0.00E+00	2.51E-03	0.00E+00	4.34E-04
134 CS	2.57E-04	6.05E-04	1.92E-04	7.34E-05	7.52E-06	0.00E+00	2.81E-04
137 CS	1.05E-03	1.40E-03	4.77E-04	1.85E-04	1.99E-05	0.00E+00	4.88E-04
TOTAL FOR PATHWAY	1.31E-03	2.24E-03	6.80E-04	2.59E-04	2.61E-03	0.00E+00	1.21E-03
COW MILK							
54 MN	0.00E+00	3.98E-06	1.19E-06	0.00E+00	8.15E-06	0.00E+00	7.88E-07
60 CO	0.00E+00	9.03E-05	0.00E+00	0.00E+00	1.18E-03	0.00E+00	2.03E-04
134 CS	1.41E-03	3.31E-03	1.05E-03	4.02E-04	4.12E-05	0.00E+00	1.54E-03
137 CS	6.00E-03	7.99E-03	2.72E-03	1.06E-03	1.14E-04	0.00E+00	2.78E-03
TOTAL FOR PATHWAY	7.41E-03	1.14E-02	3.77E-03	1.46E-03	1.34E-03	0.00E+00	4.52E-03

TABLE 3
LANDSPREADING INGESTION PATHWAYS
(CHILD)
(2.33 UCI TOTAL ACTIVITY)
(MREM)

PATHWAY	BONE	LIVER	KIDNEY	LUNG	GI-LLI	THYROID	WHOLE BODY
INHALATION							
54 MN	0.00E+00	3.17E-06	7.41E-07	1.17E-04	1.69E-06	0.00E+00	7.03E-07
60 CO	0.00E+00	2.40E-05	0.00E+00	1.29E-02	1.76E-04	0.00E+00	4.15E-05
134 CS	5.54E-05	8.63E-05	2.81E-05	1.03E-05	3.27E-07	0.00E+00	1.91E-05
137 CS	2.03E-04	1.85E-04	6.32E-05	2.33E-05	8.10E-07	0.00E+00	2.87E-05
TOTAL FOR PATHWAY	2.58E-04	2.98E-04	9.20E-05	1.31E-02	1.79E-04	0.00E+00	9.00E-05
STORED VEGETABLES							
54 MN	0.00E+00	7.25E-04	2.03E-04	0.00E+00	6.08E-04	0.00E+00	1.93E-04
60 CO	0.00E+00	4.40E-03	0.00E+00	0.00E+00	2.44E-02	0.00E+00	1.30E-02
134 CS	8.42E-03	1.38E-02	4.28E-03	1.54E-03	7.45E-05	0.00E+00	2.91E-03
137 CS	3.80E-02	3.63E-02	1.18E-02	4.26E-03	2.27E-04	0.00E+00	5.36E-03
TOTAL FOR PATHWAY	4.64E-02	5.53E-02	1.63E-02	5.80E-03	2.53E-02	0.00E+00	2.14E-02
LEAFY VEGETABLES							
54 MN	0.00E+00	4.13E-05	1.16E-05	0.00E+00	3.47E-05	0.00E+00	1.10E-05
60 CO	0.00E+00	2.25E-04	0.00E+00	0.00E+00	1.24E-03	0.00E+00	6.62E-04
134 CS	4.45E-04	7.30E-04	2.26E-04	8.11E-05	3.93E-06	0.00E+00	1.54E-04
137 CS	1.90E-03	1.82E-03	5.94E-04	2.14E-04	1.14E-05	0.00E+00	2.69E-04
TOTAL FOR PATHWAY	2.35E-03	2.82E-03	8.32E-04	2.95E-04	1.29E-03	0.00E+00	1.10E-03
COW MILK							
54 MN	0.00E+00	5.95E-06	1.67E-06	0.00E+00	4.99E-06	0.00E+00	1.58E-06
60 CO	0.00E+00	1.40E-04	0.00E+00	0.00E+00	7.77E-04	0.00E+00	4.13E-04
134 CS	3.25E-03	5.33E-03	1.65E-03	5.93E-04	2.87E-05	0.00E+00	1.12E-03
137 CS	1.45E-02	1.38E-02	4.51E-03	1.62E-03	8.67E-05	0.00E+00	2.04E-03
TOTAL FOR PATHWAY	1.77E-02	1.93E-02	6.17E-03	2.22E-03	8.97E-04	0.00E+00	3.58E-03

TABLE 4
LANDSPREADING INGESTION PATHWAYS
(INFANT)
(2.33 UCI TOTAL ACTIVITY)
(MREM)

PATHWAY	BONE	LIVER	KIDNEY	LUNG	GI-LLI	THYROID	WHOLE BODY
INHALATION							
54 MN	0.00E+00	1.87E-06	3.69E-07	7.39E-05	5.22E-07	0.00E+00	3.69E-07
60 CO	0.00E+00	1.47E-05	0.00E+00	8.25E-03	5.84E-05	0.00E+00	2.16E-05
134 CS	3.37E-05	5.98E-05	1.62E-05	6.78E-06	1.14E-07	0.00E+00	6.34E-06
137 CS	1.23E-04	1.37E-04	3.85E-05	1.59E-05	2.99E-07	0.00E+00	1.02E-05
TOTAL FOR PATHWAY	1.57E-04	2.13E-04	5.51E-05	8.35E-03	5.94E-05	0.00E+00	3.84E-05
STORED VEGETABLES							
54 MN	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
60 CO	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
134 CS	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
137 CS	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PATHWAY	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
LEAFY VEGETABLES							
54 MN	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
60 CO	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
134 CS	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
137 CS	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PATHWAY	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
COW MILK							
54 MN	0.00E+00	1.11E-05	2.45E-06	0.00E+00	4.06E-06	0.00E+00	2.51E-06
60 CO	0.00E+00	2.86E-04	0.00E+00	0.00E+00	6.81E-04	0.00E+00	6.76E-04
134 CS	5.23E-03	9.76E-03	2.51E-03	1.03E-03	2.65E-05	0.00E+00	9.85E-04
137 CS	2.31E-02	2.70E-02	7.25E-03	2.94E-03	8.45E-05	0.00E+00	1.92E-03
TOTAL FOR PATHWAY	2.83E-02	3.71E-02	9.77E-03	3.97E-03	7.96E-04	0.00E+00	3.58E-03



Note: ------------ Septage Hauler/Tank Truck Pipe Line

YANKEE PLANT SANITARY WASTE DISPOSAL PROCESS

FIGURE 1

APPENDIX H

Radioactive Liquid, Gaseous, and Solid Waste Treatment Systems

Requirement: Control 7.3 (previously Technical Specification 6.16.1) requires that licensee initiated major changes to the radioactive waste systems (liquid, gaseous, and solid) be reported to the Commission in the Semiannual Effluent Release Report for the period in which the evaluation was reviewed by the Plant Operation Review Committee.

Response: There were no licensee initiated major changes to a radioactive waste system(s) during this reporting period.

APPENDIX I

Supplemental Information
Third and Fourth Quarters 1992

1. Control Limits - Dose and Dose Rate

Control and Category	Limit
a. Noble Gases	
Control 3.3 (previously TS 3.11.2.1), Total body dose rate	500 mrem/year
Control 3.3, Skin dose rate	3000 mrem/year
Control 3.4 (previously TS 3.11.2.2), Gamma air dose	5 mrad/quarter
Control 3.4, Gamma air dose	10 mrad/year
Control 3.4, Beta air dose	10 mrad/quarter
Control 3.4, Beta air dose	20 mrad/year
b. Iodine-131, Tritium, and Radionuclides in Particulate Form With Half-Lives Greater Than 8 Days	
Control 3.3, Organ dose rate	1500 mrem/year
Control 3.5 (previously TS 3.11.2.3), Organ dose	7.5 mrem/quarter
Control 3.5, Organ dose	15 mrem/year
c. Liquids	
Control 3.1 (previously TS 3.11.1.2), Total body dose	1.5 mrem/quarter
Control 3.1, Total body dose	3 mrem/year
Control 3.1, Organ dose	5 mrem/quarter
Control 3.1, Organ dose	10 mrem/year

2. Control Limits - Concentration

Control and Category	Limit
a. Liquids	
Control 2.1 (previously TS 3.11.1.1), Total sum of the fraction of MPC (10CFR20, Appendix B, Table II, Column 2), excluding noble gases less than:	1.0
Control 2.1, Total noble gas concentration	2.00E-04 µCi/cc

3. Measurements and Approximations of Total Radioactivity

a. Noble Gases, Krypton-85

Continuous discharges are determined by direct measurements. A primary vent stack gas sample is taken monthly and analyzed for Krypton-85. A review of the weekly primary vent stack noble gas integrator readings for any increase in values above the background level also is used as a reference. There are no longer any batch discharges. Errors associated with the above measurements are estimated to be ±25 percent.

b. Iodines

Iodines are continuously monitored by drawing a sample from the primary vent stack through a particulate filter and charcoal cartridge. The filter and charcoal cartridge are removed and analyzed weekly. The errors associated with these measurements are estimated to be ±25 percent.

c. Particulates

The particulate filter described in b. above is analyzed weekly. The errors associated with the determination of particulate effluents are estimated to be ±30 percent.

d. Liquid Effluents

A gamma isotopic analysis is performed on a representative sample using a Marinelli Beaker geometry for both a batch or continuous discharge. Composite samples for batch and continuous discharges are analyzed for strontium-89, strontium-90, iron-55, gross alpha activity, and carbon-14.

Tritium analysis is performed on composite samples for continuous discharges and on each batch discharge. The errors associated with these measurements are as follows: fission and activation products, ±20 percent; tritium, ±10 percent; dissolved fission gases, ±20 percent; alpha activity, ±35 percent.

4. Batch Releases

a. Liquids

The batch release data is presented in two categories to provide accurate average discharge rate values.

Third Quarter	Routine Batches	Outdoor Tank Moats
Number of batch releases:	17	2
Total time period for batch releases (minutes):	4693	6419
Maximum time period for a batch release (minutes):	335	3440
Average time period for batch releases (minutes):	276	3710
Minimum time period for a batch release (minutes):	180	2733
Average stream flow (Sherman Dam) during period (cfs):	290	
Average discharge rate (gpm):	23	2.7

Fourth Quarter	Routine Batches	Outdoor Tank Moats
Number of batch releases:	8	2
Total time period for batch releases (minutes):	2701	5763
Maximum time period for a batch release (minutes):	609	3030
Average time period for batch releases (minutes):	338	2882
Minimum time period for a batch release (minutes):	115	2733
Average stream flow (Sherman Dam) during period (cfs):	514	
Average discharge rate (gpm):	18	4.1

b. Gases

Third Quarter	
Number of batch releases:	0
Total time period for batch releases (minutes):	
Maximum time period for a batch release (minutes):	
Average time period for batch releases (minutes):	
Minimum time period for a batch release (minutes):	

Fourth Quarter	
Number of batch releases:	0
Tot:l time period for batch releases (minutes):	
Maximum time period for a batch release (minutes):	
Average time period for batch releases (minutes):	
Minimum time period for a batch release (minutes):	

5. Abnormal Releases

Control 7.2.b (previously Technical Specification 6.9.5.b) requires the reporting of any unplanned releases from the site to the site boundary of radioactive material in gaseous and liquid effluents made during the reporting period.

a. Liquid

There were no nonroutine liquid releases during the reporting period.

b. <u>Gases</u>

There were no nonroutine gaseous releases during the reporting period.