

YANKEE NUCLEAR POWER STATION
OFF-SITE DOSE CALCULATION MANUAL

YANKEE ATOMIC ELECTRIC COMPANY

	PREPARED BY/DATE	REVIEWED BY/DATE	PORC MEETING NO./DATE
REVISION 8	C. L. Albright August 19, 1992	Mark Strum August 19, 1992	Meeting No. 92-72 August 19, 1992
REVISION 9	Edward R. Cumming May 12, 1993	Mark Strum May 18, 1993	Meeting No. 93-22 May 18, 1993
REVISION 10	R. Brad Harvey June 22, 1993	Mark Strum June 22, 1993	Meeting No. 93-28 June 22, 1993
REVISION 11	March Strum October 22, 1996	G. M. Babineau November 12, 1996	Meeting No. 96-63 October 31, 1996
REVISION 12	R. B. Harvey February 7, 1997	Mark Strum February 7, 1997	Meeting No. 97-3 January 9, 1997
REVISION 13	M. S. Strum June 17, 1999	John S. Gedulis June 17, 1999	Meeting No. 99-19 June 17, 1999
REVISION 14	M. S. Strum April 13, 2000	G. M. Babineau April 13, 2000	Meeting No. 00-15 April 13, 2000
REVISION 15	M. S. Strum November 19, 2001	G. M. Babineau December 17, 2001	Meeting No. 01-69 November 19, 2001
REVISION 16	<i>Mark Strum</i> Aug. 6, 2003	<i>G. M. Babineau</i> 8/12/03	Meeting No. 03-65 August 14, 2003

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	<p>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:</p> <ol style="list-style-type: none"> 1. Addition of List of Controls Page (succeeds Table of Contents);

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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);

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8	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.
9	05/18/93	Replacement of milk sampling location TM-12 with TM-14 in Table 4.4 and Figure 4-2.
10	06/22/93	Technical Specification 3/4.3.3.3 (now referred to as Control 5.5) and its Bases relocated to Section 5.0; Technical Specification 3/4.3.3.3 nominal sensor elevations revised to reflect actual measurement heights; Technical Specification 3/4.3.3.3 Bases revised to eliminate reference to protective action recommendations.
11	10/31/96	Surveillance and analyses schedules for both the in-plant Gaseous and Liquid Effluent Monitoring Programs and the off-site REMP have been reduced. These reductions reflect changes in plant configuration due to plant dismantlement and decommissioning activities, and the elimination of radioactive source terms due to the cessation of the fission process with the shutdown of power operations.
12	02/07/97	The requirement to submit an annual summary of hourly meteorological data with the Semi-annual Radioactive Effluent Release Report due 60 days after January 1 of each year was eliminated.

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Revision	Date	Description
13	6/17/99	Elimination of the Turbine Building composite sampler and analysis requirements due to change in plant configuration. Elimination of milk sampling location due to changes identified in the annual land use census. Clarification of actions necessary to perform maintenance and testing on an in--line rad-monitor, plus expanded flexibility of operating Aux. Service Water for Spent Fuel Pit cooling while the effluent rad-monitor is out of service. Correction to block diagram (Figure 6-1) to reflect current liquid waste processing system configuration. Change in the reporting requirement for the Radioactive Effluent Release Report from semiannual to annual based on Technical Specification Amendment No. 151. Editorial changes to improve readability and correct or eliminate unnecessary text.
14	4/13/00	Clarifications to liquid effluent monitoring ACTION STATEMENTS 1 and 4 (Table 5.1) are included to provide clearer guidance on the existing intent of actions needed if the liquid radiation effluent monitor is out of service.
15	11/19/01	<ol style="list-style-type: none">1. Eliminates the requirement to maintain the 200 foot On-Site Meteorological Monitoring System (Section 5.5).2. References changed to reflect the relocation of Technical Specification Section 6.7 into the Yankee Decommissioning Assurance Program, Appendix D, Administrative Controls.3. Change the auxiliary service water flow rate to 120 gpm to reflect current operating performance and deletes the use of pump and valve curves to estimate flow rate.

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Revision	Date	Description
16		<ol style="list-style-type: none">1. Eliminates detection requirements for radionuclides which, by natural decay, are no longer a significant effluent or of environmental concern.2. Eliminates milk sampling requirements from REMP.3. Adds new liquid waste treatment and discharge conditions for decommissioning of structures including the SFP.4. Updates site maps and monitoring programs to recognize the new ISFSI operations.5. Eliminates Control limits and measurement requirements for dissolved and entrained noble gases in liquid waste.6. Updates surveillance requirements for liquid effluent releases.

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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 Appendix D, Section B.4, of the Yankee Decommissioning Quality Assurance Program (YDQAP) and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Technical Specifications 6.8.2.a and 6.8.2.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 the Decommissioning Manager, this manual is suitable to show compliance where referred to by the Yankee Decommissioning Quality Assurance Program⁽¹⁾ and controls listed in this document.

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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 Appendix D, Section B.4, of the Yankee Decommissioning Quality Assurance Program (YDQAP) and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive 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 accepted by an Independent Safety Reviewer and approved by the Decommissioning Manager in accordance with Technical Specification 6.13 prior to implementation. Changes made to the ODCM shall be submitted to the Commission for their information in the Annual Radioactive 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, and FREQUENCY NOTATION used in the

ODCM are in Tables 1.4, 1.5, 1.6, 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 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.

SR 1.1 Surveillance requirements shall be applicable during the conditions specified for individual controls.

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.

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	$D_{tb} \text{ (mrem)} = K \sum_i Q_i DFL_{tb}$
3-2	Maximum Organ Dose Due to Liquids	$D_{organ} \text{ (mrem)} = K \sum_i Q_i DFL_{mo}$
3-3	Total Body Dose Rate Due to Noble Gases	$D_{tb} \left(\frac{\text{mrem}}{\text{yr}} \right) = 7.83 Q_{Kr-85} DFB_{Kr-85}$
3-4	Skin Dose Rate Due to Noble Gases	$D_{skin} \left(\frac{\text{mrem}}{\text{yr}} \right) = Q_{Kr-85} DF'_{Kr-85}$
3-5	Critical Organ Dose Rate Due to H-3 and Particulates with $T_{1/2} > 8$ Days	$D_{co} \left(\frac{\text{mrem}}{\text{yr}} \right) = \sum_i Q_i DFG'_{Kr-85}$
3-6	Gamma Air Dose Due to Noble Gases	$D_{air}^{\gamma} \text{ (mrad)} = 0.25 Q_{Kr-85} DF_{Kr-85}^{\gamma}$
3-6.1	Gamma Air Dose Due to Ground Level Noble Gas Releases	$D_{air}^{\gamma} \text{ (mrad)} = (6.0 \times 10^{-6}) (Q_{Kr-85})$
3-7	Beta Air Dose Due to Noble Gases	$D_{air}^{\beta} \text{ (mrad)} = 0.76 Q_{Kr-85} DF_{Kr-85}^{\beta}$
3-8	Critical Organ Dose Due to H-3 and Particulates with $T_{1/2} > 8$ Days	$D_{co} \text{ (mrem)} = \sum_i Q_i DFG_{co}$

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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)
5-1	Liquid Release Rate Reading	$R = \left(\frac{f_2}{f_1} \right) (MPC_c) (S \ell)$
5-3	Gaseous Release Rate Reading for Total Body Dose Limit	$R_b = \frac{(500) (60) (S_{Kr-85})}{(F) (7.83) DFB_{Kr-85}}$
5-4	Gaseous Release Rate Reading for Skin Dose Limit	$R_{sk} = \frac{(3000) (60) (S_{Kr-85})}{(F) DF'_{Kr-85}}$

Note (a):

- C_i = Concentration of radionuclide "i" in a mixture ($\mu\text{Ci/ml}$).
- F = Primary vent stack flow rate (cc/min).
- C_i^{ENG} = Concentration of radionuclide "i", except noble gases, at the point of discharge.
- 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".

TABLE 1.1
(Continued)

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

- 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_{tb} = Site-specific, total body dose factor for a liquid release of radionuclide "i".
 - DFL_{mo} = Site-specific, maximum organ dose factor for a liquid release of radionuclide "i".
 - f₁ = Flow rate past the test tank monitor (gpm).
 - f₂ = 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).
- $$= \frac{\sum_i C_i}{\sum_i \frac{C_i}{MPC_i}} \tag{Eq. 5-2}$$
- Q_i = Total release (Curies) for radionuclide "i".
 - Q_i = Release rate (μCi/sec) for radionuclide "i".
 - S_{Kr-85} = Gaseous instrumentation response factor for Kr-85 (cpm/(μCi/cc)).
 - S_l = Liquid instrumentation response factor (cpm/(μCi/cc)).

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TABLE 1.2

Dose Factors Specific to the Yankee Plant for
Noble Gas Releases

Radionuclide	Gamma Total Body Dose Factor $DF_{T\gamma} \left(\frac{\text{mrem} \cdot \text{m}^3}{\text{pCi} \cdot \text{yr}} \right)$	Beta Skin Dose Factor $DF_{S\beta} \left(\frac{\text{mrem} \cdot \text{m}^3}{\text{pCi} \cdot \text{yr}} \right)$	Combined Skin Dose Factor $DF_{S\gamma} \left(\frac{\text{mrem} \cdot \text{sec}}{\mu\text{Ci} \cdot \text{yr}} \right)$	Beta Air Dose Factor $DF_{I\beta} \left(\frac{\text{mrad} \cdot \text{m}^3}{\text{pCi} \cdot \text{yr}} \right)$	Gamma Air Dose Factor $DF_{I\gamma} \left(\frac{\text{mrad} \cdot \text{m}^3}{\text{pCi} \cdot \text{yr}} \right)$
Kr-85	1.61×10^{-6}	1.34×10^{-3}	3.22×10^{-2}	1.95×10^{-3}	1.72×10^{-4}

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	Eq. 2-1	≤ 1.0
3.1	Dose Due to Liquid Effluents	Total Body Dose	Eq. 3-1	≤ 1.5 mrem in a qtr. ≤ 3.0 mrem in a yr.
		Organ Dose	Eq. 3-2	≤ 5.0 mrem in a qtr. ≤ 10.0 mrem in a yr.
3.2	Total Dose Due to Liquid and Gaseous Effluents	Total Body Dose	Eq. 3-1 Eq. 3-6 Eq. 3-9	≤ 25.0 mrem in a yr.
		Organ Dose	Eq. 3-2 Eq. 3-8 Eq. 3-9	≤ 25.0 mrem in a yr.
		Thyroid Dose	Eq. 3-2 Eq. 3-8 Eq. 3-9	≤ 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	$\leq 500.0 \frac{\text{mrem}}{\text{yr}}$
		Skin Dose Rate Due to Noble Gases	Eq. 3-4	$\leq 3000.0 \frac{\text{mrem}}{\text{yr}}$
		Organ Dose Rate Due to H-3 and Particulates with $T_{1/2} > 8$ Days	Eq. 3-5	$\leq 1500.0 \frac{\text{mrem}}{\text{yr}}$
3.4	Dose Due to Noble Gases in Gaseous Effluents	Gamma Air Dose Due to Noble Gases	Eq. 3-6	≤ 5.0 mrad in a qtr. ≤ 10.0 mrad in a yr.
		Beta Air Dose Due to Noble Gases	Eq. 3-7	≤ 10.0 mrad in a qtr. ≤ 20.0 mrad in a yr.
3.5	Dose Due to Tritium and Particulates in Gaseous Effluents	Organ Dose Due to 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.

TABLE 1.3
(Continued)

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Summary of Radiological Effluent Controls
and Implementing Equations

	Control	Category	Method	Limit
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	≤ 0.4 mrad in a mo.
		Beta Air Dose Due to Noble Gases	Eq. 3-7	≤ 0.4 mrad in a mo.
		Organ Dose Due to H-3 and Particulates with $T_{1/2} > 8$ Days	Eq. 3-8	≤ 0.3 mrem in a mo.

TABLE 1.4

Summary of Constants

Constant	Definition	Units
0.25	$= (3.17 \times 10^{14}) \left(\frac{\text{pCi-yr}}{\text{Ci-sec}} \right) [X/Q]^r (\text{sec/m}^3)$ $= (3.17 \times 10^{14}) (7.83 \times 10^{-6})$	$\left(\frac{\text{pCi-yr}}{\text{Ci-m}^3} \right)$
0.76	$= (3.17 \times 10^{14}) \left(\frac{\text{pCi-yr}}{\text{Ci-sec}} \right) [X/Q] (\text{sec/m}^3)$ $= (3.17 \times 10^{14}) (2.39 \times 10^{-5})$	$\left(\frac{\text{pCi-yr}}{\text{Ci-m}^3} \right)$
1.11	= Average ratio of tissue to air energy absorption ratio coefficient.	
7.83	$= (10^{16}) (\text{pCi}/\mu\text{Ci}) (7.83 \times 10^{-6}) (\text{sec}/\text{m}^3)$	$\left(\frac{\text{pCi-sec}}{\mu\text{Ci-m}^3} \right)$
8.69	$= (1.11) (S_f) [X/Q]^r (\text{sec}/\text{m}^3) (1.00 \times 10^{16}) (\text{pCi}/\mu\text{Ci})$ $= (1.11) (1.00) (7.83 \times 10^{-6}) (1.00 \times 10^{16})$	$\left(\frac{\text{pCi-sec}}{\mu\text{Ci-m}^3} \right)$
23.90	$= (1.00 \times 10^{16}) (X/Q)$ $= (1.00 \times 10^{16}) (2.39 \times 10^{-5})$	$\left(\frac{\text{pCi-sec}}{\mu\text{Ci-m}^3} \right)$
60.00	= Conversion factor.	$\frac{\text{sec}}{\text{min}}$
500.00	= Total body annual dose limit from ICRP-2.	mrem
3000.00	= Skin annual dose limit from ICRP-2.	mrem
3.17×10^{14}	= Number of picocuries per Curie divided by the number of seconds per year.	$\left(\frac{\text{pCi-yr}}{\text{Ci-sec}} \right)$

TABLE 1.5

Summary of Variables

Variable	Definition	Units
$1.00 \times 10^{+6}$	= Number of picocuries per microcurie.	$\frac{\text{pCi}}{\mu\text{Ci}}$
31.54	= $(1.0 \times 10^{-6}) \left(\frac{\text{Ci}}{\mu\text{Ci}} \right) (3.154 \times 10^{+7}) \left(\frac{\text{sec}}{\text{yr}} \right)$	$\frac{\text{Ci} - \text{sec}}{\mu\text{Ci} - \text{yr}}$
C^{NG}	= Total concentration of all dissolved and entrained noble gases from all station sources.	$\frac{\mu\text{Ci}}{\text{cc}}$
C_i^{NG}	= Concentration of radionuclide "i", except noble gases, at the point of discharge.	$\frac{\mu\text{Ci}}{\text{cc}}$
C_i	= Concentration of radionuclide "i".	$\frac{\mu\text{Ci}}{\text{m}^3}$ or $\frac{\mu\text{Ci}}{\text{cc}}$
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_{organ}	= Dose to the maximum organ.	mrem
D_{skin}	= Beta and gamma dose to the skin.	mrem
D_{b}	= Dose to the total body.	mrem
$DF_{\text{Kr-85}}$	= Total body gamma dose factor for Kr-85.	$\frac{\text{mrem} - \text{m}^3}{\text{pCi} - \text{yr}}$
$DFS_{\text{Kr-85}}$	= Beta skin dose factor for Kr-85.	$\frac{\text{mrem} - \text{m}^3}{\text{pCi} - \text{yr}}$
$DF'_{\text{Kr-85}}$	= Combined site-specific skin dose factor for Kr-85.	$\frac{\text{mrem} - \text{sec}}{\mu\text{Ci} - \text{yr}}$

TABLE 1.5
(Continued)

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Summary of Variables

Variable	Definition	Units
DF_{Kr-85}^{γ}	= Gamma air dose factor for Kr-85.	$\frac{mrad - m^3}{pCi - yr}$
DF_{Kr-85}^{β}	= Beta air dose factor for Kr-85.	$\frac{mrad - m^3}{pCi - yr}$
DFG_{ico}	= Critical organ gaseous dose factor for radionuclide "i".	$\frac{mrem}{Ci}$
DFG_{ico}	= Critical organ gaseous dose rate factor for radionuclide "i".	$\frac{mrem - sec}{\mu Ci - yr}$
DFL_{lmo}	= Maximum organ liquid dose factor for radionuclide "i".	$\frac{mrem}{Ci}$
DFL_{tb}	= Total body liquid dose factor for radionuclide "i".	$\frac{mrem}{Ci}$
D_{∞}	= Critical organ dose rate due to tritium and particulates.	$\frac{mrem}{yr}$
D_{skin}	= Skin dose rate due to noble gases.	$\frac{mrem}{yr}$
D_b	= Total body dose rate due to noble gases.	$\frac{mrem}{yr}$
D/Q	= Deposition factor for dry deposition of particulates.	$\frac{sec}{m^2}$
F	= Primary vent stack flow rate.	$\frac{cc}{sec}$

TABLE 1.5
(Continued)

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Summary of Variables

Variable	Definition	Units
F_1	= Total fraction of MPC in liquid pathways.	
f_1	= Flow rate past the test tank monitor.	gpm
f_2	= Flow rate at the point of discharge.	gpm
MPC_c	= Composite MPC for the mix of radionuclides. See Equation 5-2.	$\frac{\mu\text{Ci}}{\text{cc}}$
MPC_i	= Maximum permissible concentration of radionuclide "i" (10CFR Part 20, Appendix B, Table 2, Column 2, see Appendix B of the ODCM).	$\frac{\mu\text{Ci}}{\text{cc}}$
Q_i	= Release for radionuclide "i".	Ci
Q	= Total release rate of noble gas (Kr-85).	$\frac{\mu\text{Ci}}{\text{sec}}$
Q_i	= Release rate for radionuclide "i".	$\frac{\mu\text{Ci}}{\text{sec}}$
X/Q	= Average undepleted dispersion factor.	$\frac{\text{sec}}{\text{m}^3}$
$[X/Q]^D$	= Average depleted dispersion factor.	$\frac{\text{sec}}{\text{m}^3}$
$[X/Q]^E$	= Effective average gamma dispersion factor.	$\frac{\text{sec}}{\text{m}^3}$
S_F	= Shielding factor.	

TABLE 1.5
(Continued)

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Summary of Variables

Variable	Definition	Units
S_{Kr-85}	= Gaseous monitor response factor for Kr-85.	$\frac{\text{cpm}}{\mu\text{Ci/cc}}$
S_l	= Liquid monitor response factor.	$\frac{\text{cpm}}{\mu\text{Ci/cc}}$

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.

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals defined in Table 1.9.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC (for purposes of 10CFR50, Appendix I) 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

TABLE 1.6
(Continued)

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Definition of Terms

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 site operations or decommissioning of the plant.

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 Appendix D, Section B.4 of the YDQAP and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual 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).

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.

TABLE 1.6
(Continued)

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Definition of Terms

VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce radioactive material in particulate form in gaseous effluents by passing ventilation or vent exhaust gases through HEPA filters for the purpose of removing particulates from the gaseous exhaust stream prior to release to the environment. Such a system is not considered to have any effect on noble gas effluents.

TABLE 1.7

Dose Factors Specific to the Yankee Plant for
Liquid Releases

Radionuclide	Total Body Dose Factor $DFL_{tb} \left(\frac{\text{mrem}}{\text{Ci}} \right)$	Maximum Organ Dose Factor $DFL_{mo} \left(\frac{\text{mrem}}{\text{Ci}} \right)$
H-3	5.99×10^{-4}	5.99×10^{-4}
C-14	$1.64 \times 10^{+0}$	$8.18 \times 10^{+0}$
Fe-55	3.46×10^{-2}	2.11×10^{-1}
Co-60	2.79×10^{-1}	9.04×10^{-1}
Sr-90	$6.97 \times 10^{+1}$	$2.75 \times 10^{+2}$
Ag-110m	2.32×10^{-2}	$2.21 \times 10^{+0}$
Cs-134	$1.79 \times 10^{+1}$	$2.40 \times 10^{+1}$
Cs-137	$1.07 \times 10^{+1}$	$2.07 \times 10^{+1}$
Ag108m/Ag108	5.70×10^{-1}	$1.81 \times 10^{+0}$

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TABLE 1.8

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

Radionuclide	Critical Organ Dose Factor $DFG_{100} \left(\frac{\text{mrem}}{\text{Ci}} \right)$	Critical Organ Dose Rate Factor $DFG'_{100} \left(\frac{\text{mrem-sec}}{\mu\text{Ci-yr}} \right)$
H-3	7.21×10^{-3}	2.27×10^{-1}
C-14	$4.38 \times 10^{+0}$	$1.38 \times 10^{+2}$
Co-60	$4.08 \times 10^{+1}$	$1.81 \times 10^{+3}$
Sr-90	$2.36 \times 10^{+3}$	$7.44 \times 10^{+4}$
Ag-110m	$3.63 \times 10^{+1}$	$1.22 \times 10^{+3}$
Cs-134	$8.52 \times 10^{+1}$	$2.83 \times 10^{+3}$
Cs-137	$8.71 \times 10^{+1}$	$2.97 \times 10^{+3}$
Ag108m/Ag108	$5.36 \times 10^{+2}$	$2.42 \times 10^{+4}$

TABLE 1.9

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.
P	Prior to each release.
N.A.	Not applicable.

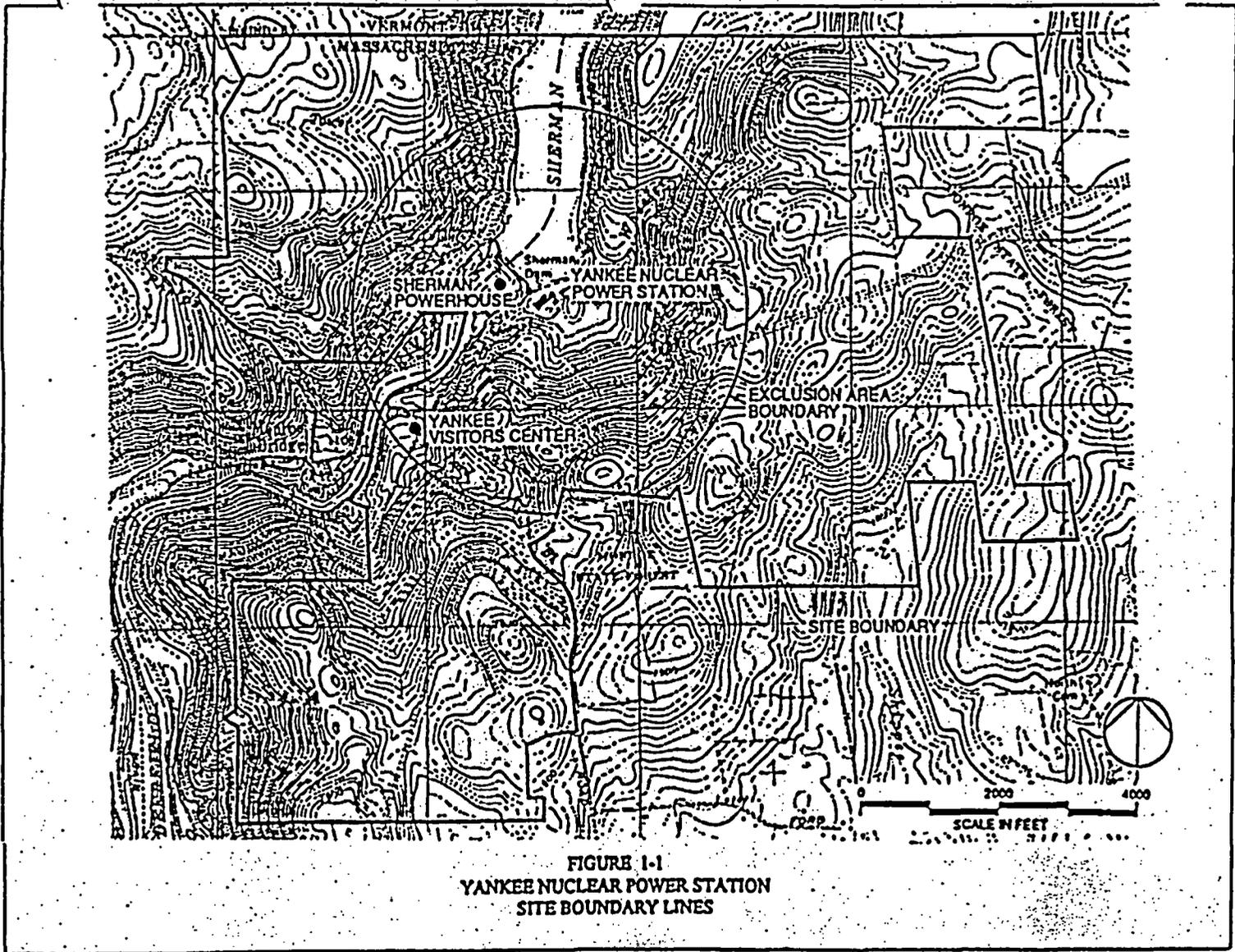


FIGURE 1-1
YANKEE NUCLEAR POWER STATION
SITE BOUNDARY LINES

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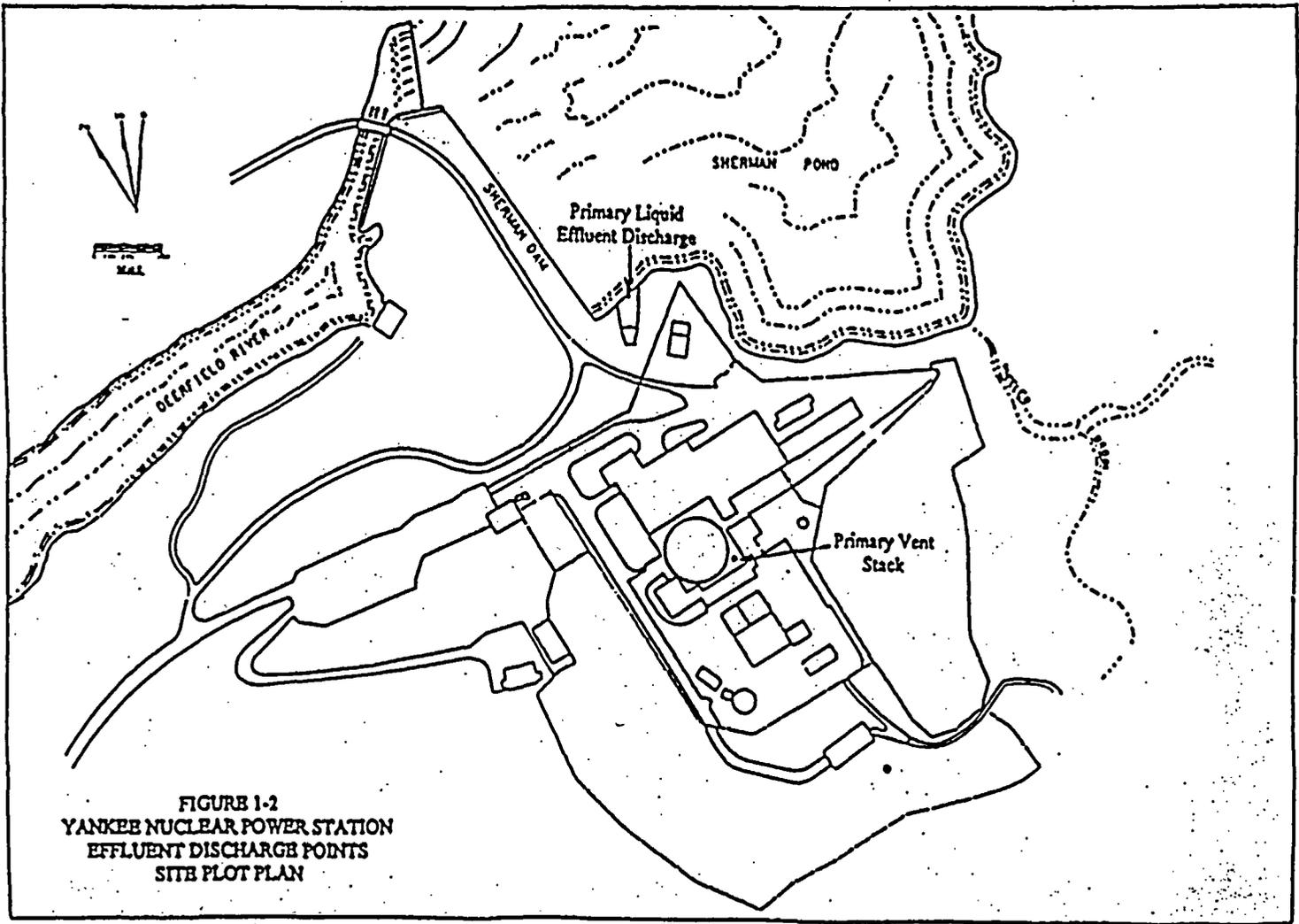


FIGURE 1-2
YANKEE NUCLEAR POWER STATION
EFFLUENT DISCHARGE POINTS
SITE PLOT PLAN

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2.0 RADIOACTIVE LIQUID EFFLUENTS

2.1 Off-Site Concentrations

Control 2.1 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), the concentration of radioactive material released to Unrestricted Areas (see Figure 1-2) shall be limited to the concentrations specified in 10CFR Part 20, Appendix B, Table II, Column 2, for all radionuclides.

Applicability

At all times.

ACTION

With the concentration of radioactive material released from the site to Unrestricted Areas 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 at any time concentration of radioactive materials released in liquid waste effluents from the site above background (unrestricted areas for liquids is 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 (Appendix B of the ODCM contains a listing of these values as taken from the regulations). These requirements provide operational flexibility, compatible with considerations

of health and safety, which may temporarily result in releases higher than the absolute value of the concentration numbers in Appendix B, but still within the annual average limitation of the revised (January 1, 1993 effective date) 10CFR, Part 20, regulation. Compliance with the design objective doses of Section II.A of Appendix I to 10CFR, Part 50, assure that doses are maintained ALARA, and that annual concentration limits of Appendix B to 10CFR20.1001-20.2401 will not be exceeded.

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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) (Including SFP drain down via batch effluent test tanks in batch mode)	P Each Batch	P Each Batch	Principal Gamma Emitters ^(f)	5.00 x 10 ⁻⁷
	P Each Batch	M Composite ^(c)	Tritium	1.00 x 10 ⁻⁵
			Gross Alpha	1.00 x 10 ⁻⁷
	P Each Batch	Q Composite ^(c)	Sr-90	5.00 x 10 ⁻⁸
Fe-55			1.00 x 10 ⁻⁶	
B. Plant Continuous Releases ^(e) Turbine Building Sump	Deleted (Pathway Abandoned)			
C. Plant Continuous Releases ^(e) SFP drain down skid for direct release or addition to TK-39 ^(g)	Continuous ^(d)	M/2 Composite ^(c)	Principal Gamma Emitters ^(f)	5.00 x 10 ⁻⁷
	Continuous ^(d)	M Composite ^(c)	Tritium	1.00 x 10 ⁻⁵
			Gross Alpha	1.00 x 10 ⁻⁷
	Continuous ^(d)	Q Composite ^(c)	Sr-90	5.00 x 10 ⁻⁸
Fe-55			1.00 x 10 ⁻⁶	
D. Construction Dewatering ^(b)	P Each Batch	P Each Batch	Principal Gamma Emitters ^(f)	5.00 x 10 ⁻⁷
			Tritium	1.00 x 10 ⁻⁵
	P Each Batch	M Composite ^(c)	Gross Alpha	1.00 x 10 ⁻⁷
			P Each Batch	Q Composite ^(c)
Fe-55	1.00 x 10 ⁻⁶			

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TABLE 2.1
(Continued)

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. For construction dewatering sources, a sample aliquot for every 1000 gallons (or less) of water transferred to temporary holding tanks or basins not capable of recirculation or internal mixing shall be collected and composited to satisfy representative sampling requirements. Alternately, if three separate samples taken from the dewatering source indicate that the gamma emitter/tritium radioactivity does not vary by more than a factor of two, then the dewatering operation may be considered as a continuous discharge where a composite sampler on the discharge line will collect a representative sample for analysis following the release.
- 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. If there is no effluent discharge during the period, no composite sample of collected waste is required.
- 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 periods when flow exist through the system.
- f. The principal gamma emitters for which the LLD requirement applies exclusively are the following radionuclides: Co-60, Cs-134, and Cs-137. 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.
- g. If SFP drain down process flow is halted to allow for SFP hydrolazing, a liquid grab sample from the SFP shall be taken and analyzed for gamma isotopic activity in accordance with notation f. above before discharge flow is re-established when the release path is in the continuous mode to either the Auxiliary Service Water (ASW) for immediate release to the environment, or to TK-39 for future release.

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_i) in liquid pathways at the point of discharge (see ODCM Figure 6-1) is discussed. (F_i) is limited to less than or equal to one, i.e.,

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

Evaluation of (F_i) is required concurrent with the sampling and analysis program in Table 2.1 of Control 2.1.

Determine the total fraction of MPC in liquid pathways as follows:

$$(F_i) = \sum_i \frac{C_i}{MPC_i} \leq 1 \quad \text{(Eq.2-1)}$$

Where:

MPC_i = Maximum permissible concentration of radionuclide "i" (10CFR Part 20, Appendix B, Table 2, Column 2. See Appendix B of ODCM for listing).

$$C_i = C_i^{TT} + C_i^{SWS} + C_i^{Other}$$

C_i^{TT} = Concentration at the point of discharge of radionuclide "i" from the test tank.

C_i^{SWS} = Concentration at the point of discharge of radionuclide "i" from the Auxiliary Service Water System.

C_i^{Other} = Concentration at the point of discharge of radionuclide "i" from any other significant sources which may be created during plant decommissioning activities.

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 5000-gallon capacity test tanks. When test tanks are filled with liquid waste, they are isolated for sampling and release. The volume of the tank's contents are determined from the liquid level in each tank. A chemist extracts a sample for radionuclide analysis. Aliquots of the sample proportional to the volume of the tank's contents are composited for appropriate radionuclide analyses. The composites contain suitable acids, alkalis, or carriers to assure the composite is representative of the sample. Composite samples are analyzed at a minimum for tritium and gross alpha activity. At a minimum, each test tank is analyzed for principal gamma emitters.

2.3.2 Auxiliary Service Water System Pathway

C_i^{SWS} is determined for each radionuclide above the analytical LLD from the activity in composite samples from the effluent lines of the Auxiliary Service Water System downstream of any potential leakage source.

2.3.3 Remaining Pathways

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

- a. Miscellaneous batch releases of potentially contaminated water, i.e., rain water collected in the containment liner of the inservice radioactive waste 20K tank, are analyzed to environmental detection levels and treated like a Test Tank according to Section 2.3.1 if plant related radioactivity is detected.
- b. Construction Dewatering: The dismantling of building and related structures, including foundation excavations, may fill with either ground water or storm water. The water-filled excavations, in many cases, must be dewatered to complete the dismantling activity. Construction dewatering may also include water generated during the process of digging new ground water monitoring wells, or other dismantlement / demolition related water and waste water sources. This discharge will be directed to either the existing storm drain systems or process treatment flow path with final release to Sherman Pond or the Deerfield River just below the Sherman Dam. With respect to effluent control, the dewatering will be sampled and analyzed to determine the radionuclide content as detailed in Table 2.1. Releases to the environment without treatment will occur only if the projected impact is less than ODCM Control 6.1 dose limits. If higher activity water is found, it will be treated as appropriate (see Figure 6.1) prior to release to reduce the radionuclide inventory (other than tritium).
- c. Spent Fuel Pool Draining: After all spent fuel and other contaminated materials are transferred to the Independent Spent Fuel Storage Installation (ISFSI), the Spent Fuel Pool (SFP) will be drained before dismantlement of this facility. Discharge of the SFP water in either batch or continuous mode will be treated by demineralization and filtration before release to Outfall 001. Figure 6.1 indicates the flow paths and in-line radiation monitoring prior to release.

3.0 DOSE/DOSE RATE CONTROLS AND CALCULATIONS

3.1 Dose Due to Radioactive Liquid Effluents

Control 3.1 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), 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.

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 Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), 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 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.

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, 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.

3.3 Dose Rate Due to Radioactive Gaseous Effluents

Control 3.3 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), 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 (Kr-85): 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 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 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

The specified limits as determined by the methodology 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. This instantaneous dose rate limit allows for operational flexibility when off normal occurrences may temporarily increase gaseous effluent release rates from the plant, while still providing controls to ensure that licensee meets the dose objectives of Appendix I to 10CFR50.

Control 3.3 also restricts, at all times, comparable with the length of the sampling periods of Table 3.1 the corresponding maximum organ dose rate above background to 1500 mrem/year for the highest impacted receptor 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 $\mu\text{Ci}/\text{ml}^{(a)}$
A. Plant Vent (Primary Vent Stack)	M Grab Sample	M	Principal Gamma Emitters ^(b)	1×10^{-4}
			Tritium	1×10^{-6}
	Continuous ^(d)	W ^(c) Particulate	Principal Gamma Emitters ^(b) , Gross Alpha	1×10^{-11}
	Continuous ^(d)	Q Composite Particulate Sample	Sr-90	1×10^{-11}
	Continuous ^(d)	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-5}

TABLE 3.1
(Continued)

Table Notation

- 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-85 for gaseous emissions and Co-60, Cs-134, and Cs-137 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. 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.
- d. 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 Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), 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.

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

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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 Tritium and Radionuclides In Particulate Form With Half-Lives Greater than Eight Days 61

Control 3.5 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), the dose to a MEMBER OF THE PUBLIC from tritium, and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the site to areas at and beyond the SITE BOUNDARY (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 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.

Surveillance Requirement

SR 3.5 Dose Calculations - Cumulative dose contributions for the current calendar quarter and current calendar year for 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.

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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 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.

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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.

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 radiolotope 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 with the recorded release.

3.7 Method to Calculate the Total Body Dose from Liquid Releases

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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.

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:

$$D_{tb} = K \sum_i Q_i DFL_{tb} \quad (\text{Eq. 3-1})$$

Where:

DFL_{tb} = 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-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).

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

- a. Liquid releases to the auxiliary service water pathway to Sherman Pond or Deerfield River just below the Sherman Power House Dam. (Outfalls 001, 003 and 004 as specified in Reference K.)
- 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

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 which applies to direct radiation as well as liquid and gaseous effluents. Method I applies only to the liquid contribution.

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.

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	Aquatic Food	Shoreline Activity	Food Grown with Contaminated Water				
			Vegetables	Leafy Veg.	Meat	Cow Milk	
MP	Mixing Ratio ⁽¹⁾	0.84	0.84	0.84	0.84	0.84	0.84
TP	Transit Time (hrs)	24.00	0.00	0.00	0.00	480.00	48.00
YV	Agricultural Productivity (kg/m ²)	-	-	2.00	2.00	2.00	2.00
P	Soil Surface Density (kg/m ²)	-	-	240.00	240.00	240.00	240.00
IRR	Irrigation Rate (l/m ² /hr)	-	-	0.15	0.15	0.15	0.15
TE	Crop Exposure Time (hrs)	-	-	1440.00	1440.00	1440.00	1440.00
TH	Holdup Time (hrs)	-	-	1440.00	24.00	2160.00	2160.00
QAW	Water Uptake Rate for Animal (l/d)	-	-	-	-	50.00	60.00
QF	Feed Uptake Rate (kg/d)	-	-	-	-	50.00	50.00
	Location of Critical Individual	Below Sherman Dam	Below Sherman Dam	Below Sherman Dam	Below Sherman Dam	Below Sherman Dam	Below Sherman Dam

⁽¹⁾ Listed mixing ratios apply to Method I dose factors. Method II analyses can apply calculated mixing ratios based on river flow and plant discharge dilution flow which exist over the period of actual release.

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/hr)	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.

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:

$$D_{organ} = K \sum_i Q_i DFL_{lmo} \tag{Eq. 3-2}$$

(mrem)

Where:

DFL_{lmo} = 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-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 ft³/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 ft³/sec).

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

- a. Liquid releases to Sherman Pond or Deerfield River just below the Sherman Power House Dam, as specified by Reference K.
- 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

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.

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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.

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:

$$D_b \left(\frac{\text{mrem}}{\text{yr}} \right) = 7.83 Q_{\text{Kr-85}} \text{DFB}_{\text{Kr-85}} \quad (\text{Eq. 3-3})$$

Where:

$Q_{\text{Kr-85}}$ = The release rate from the plant vent stack ($\mu\text{Ci}/\text{sec}$) of Kr-85. The release rate at the stack also can be stated in the following equation:

$$Q = (M) \left(\frac{1}{S_{\text{Kr-85}}} \right) (F) \quad (\text{Eq. 3-10})$$
$$\left(\frac{\mu\text{Ci}}{\text{sec}} \right) = (\text{cpm}) \left(\frac{\mu\text{Ci}/\text{cc}}{\text{cpm}} \right) \left(\frac{\text{cc}}{\text{sec}} \right)$$

Where:

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

S_{Kr-85} = Gaseous monitor response factor for Kr-85 (cpm/(μCi/cc)).

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

DFB_{Kr-85} = Total body dose factor for Kr-85. See Table 1.2.

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

- a. Normal conditions.
- b. Kr-85 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

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.

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

$D_{finite}^T = D^T [X/Q]^T/[X/Q]$ and that $D_{tb} = D_{finite}^T \cdot Q (\mu\text{Ci/sec}) \cdot 31.54/Q (\text{Ci/yr})$, we get:

$$D_{tb} (\text{mrem/yr}) = (1.0 \times 10^{-6} (S_F) [X/Q]^T Q_{\text{Kr-85}} \text{DFB}_{\text{Kr-85}}$$

Substituting:

$$S_F = 1.00 \text{ (shielding factor).}$$

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

$$= 7.83 \times 10^{-6} (\text{sec/m}^3).$$

$$Q_{\text{Kr-85}} = \text{Release rate of Kr-85 } (\mu\text{Ci/sec}).$$

Gives:

$$D_{tb} (\text{mrem/yr}) = 7.83 Q_{\text{Kr-85}} \text{DFB}_{\text{Kr-85}}$$

(Eq. 3-3)

3.10 Method to Calculate the Skin Dose Rate from Noble Gases

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

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 as 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 (Kr-85) is:

D_{skin} (mrem/yr) = Q_{Kr-85} DF_{Kr-85} (Eq. 3-4)

Where:

Q_{Kr-85} = The release rate from the plant vent stack (μCi/sec) of Kr-85. The release rate at the stack also can be stated in the following equation:

Q = (M) (1 / S_{Kr-85}) (F)
(μCi / sec) = (cpm) (μCi/cc / cpm) (cc / sec)

Where:

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

S_{Kr-85} = Gaseous monitor response factor for Kr-85 (cpm/(μCi/cc)).

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

DF_{Kr-85} = Combined skin dose factor for Kr-85. See Table 1.2.

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

- a. Normal conditions (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

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 was derived from Regulatory Guide 1.109 as follows:

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

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The equation was derived by combining Equations B-4, B-5, and B-7 from Regulatory Guide 1.109, assuming that $X/Q = X/Q^0$ for noble gases, and making some simplifications in notation. Assuming that $DF_{\text{finite}}^Y = D^Y [X/Q]^Y / [X/Q]$ and that $D_{\text{skin}} = D^3 \cdot Q$ ($\mu\text{Ci}/\text{sec}$) $\cdot 31.54/Q$ (Ci/yr) yields, and that Kr-85 is the only noble gas left in plant inventory:

$$D_{\text{skin}} (\text{mrem}/\text{yr}) = (1.11) (S_F) (1.00 \times 10^{+6}) [X/Q]^Y Q_{\text{Kr-85}} DF_{\text{Kr-85}}^Y + (1.00 \times 10^{+6}) (X/Q) Q_{\text{Kr-85}} DFS_{\text{Kr-85}}$$

Where:

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

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

$$S_F = 1.00 \text{ (shielding factor)}$$

Substituting gives:

$$D_{\text{skin}} (\text{mrem}/\text{yr}) = 8.69 Q_{\text{Kr-85}} DF_{\text{Kr-85}}^Y + Q_{\text{Kr-85}} DFS_{\text{Kr-85}} 23.9 \times 10^{+1}$$

$$= Q_{\text{Kr-85}} [8.69 DF_{\text{Kr-85}}^Y + 23.9 DFS_{\text{Kr-85}}]$$

Define:

$$DF_{\text{Kr-85}}^Y = 8.69 (DF^Y) + 23.9 (DFS_{\text{Kr-85}})$$

$$= 8.69 (1.72 \times 10^{-5}) + 23.9 (1.34 \times 10^{-3})$$

$$= 3.22 \times 10^{-2} \left(\frac{\text{mrem} - \text{sec}}{\mu\text{Ci} - \text{yr}} \right)$$

Then:

$$D_{\text{skin}} (\text{mrem}/\text{yr}) = Q_{\text{Kr-85}} DF_{\text{Kr-85}}^Y = Q_{\text{Kr-85}} 3.22 \times 10^{-2} \tag{Eq. 3-4}$$

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

Control 3.3 limits the dose rate at any time at location at or beyond the Site Boundary from 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 particulates is commensurate with the time the particulate samplers are in service between changeouts.

3.11.1 Method I

The critical organ dose rate can be determined as follows:

$$D_{\infty} = \sum_i Q_i DFG'_{i\infty} \tag{Eq. 3-5}$$

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

Where:

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

$DFG'_{i\infty}$ = Site-specific critical organ dose rate factor $\left(\frac{\text{mrem} - \text{sec}}{\mu\text{Ci} - \text{yr}}\right)$ 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 conditions (not emergency event),
- b. Tritium and particulate releases via the plant vent stack to the atmosphere.

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

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 tritium and radionuclides in particulate form with half-lives greater than eight days released to the atmosphere has been met for the peak tritium and particulate release rates.

The equation for D_{co} is derived by modifying Equation 3-8 from Section 3.14 as follows:

$$D_{co} = \sum_i Q_i DFG_{ico} \tag{Eq. 3-8}$$

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

Applying the conversion factor, 31.54 (Ci-sec/ μ Ci-yr), and converting Q to Q in μ Ci/sec yields:

$$D_{co} = 31.54 \sum_i Q_i DFG_{ico}$$

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

Equation 3-5 is rewritten in the form:

$$D_{co} = \sum_i Q_i DFG'_{ico}$$

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) may be performed with 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 (Kr-85)

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.

3.12.1 Method I

The gamma air dose from plant vent stack releases is:

$$D_{air}^{\gamma} \text{ (mrad)} = 0.25 Q_{Kr-85} DF_{Kr-85}^{\gamma} \tag{Eq. 3-6}$$

Where:

Q_{Kr-85} = Total Kr-85 (Curies) released to the atmosphere via the plant vent stack during the period of interest.

DF_{Kr-85}^{γ} = Gamma air dose factor for Kr-85. See Table 1.2.

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

- a. Normal conditions (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:

$$D_{gnd}^{\gamma} \text{ (mrad)} = (6.0 \times 10^{-6}) (Q_{Kr-85}) \tag{Eq. 3-6.1}$$

Where:

Q_{Kr-85} = The ground level release (in curies) of Kr-85.

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

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.

With Kr-85 being the only noble gas potentially available for release, the dose can be taken from Equations B-4 and B-5 of Regulatory Guide 1.109 with the added assumption that $D_{finite}^Y = D [X/Q]^Y/[X/Q]$:

$$D_{air}^Y \text{ (mrad)} = 3.17 \times 10^{-4} \left(\frac{\text{pCi-yr}}{\text{Ci-sec}} \right) [X/Q]^Y \text{ (sec/m}^3\text{)} Q_{Kr-85} \text{ (Ci)} DF_{Kr-85}^Y \left(\frac{\text{mrad-m}^3}{\text{pCi-yr}} \right)$$

Where:

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

= 7.83×10^{-6} (sec/m³).

Q_{Kr-85} = Number of Curies of noble gas (Kr-85) released.

Which leads to:

$$D_{air}^Y \text{ (mrad)} = 0.25 Q_{Kr-85} DF_{Kr-85}^Y \tag{Eq. 3-6}$$

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:

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

$D_{Kr-85}^T =$ the gamma air dose factor for Kr-85 (see Table 1.2)

$$= \text{to } 1.72 \times 10^{-5} \left(\frac{\text{mrad} - \text{m}^3}{\text{pCi} - \text{yr}} \right)$$

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

$$\begin{aligned}
D_{\text{grd}}^T \text{ (mrad)} &= 3.17 \times 10^{-4} \left[\frac{\text{pCi} - \text{yr}}{\text{Ci} - \text{sec}} \right] [X/Q]^T \text{ (sec/m}^3) DF^T \left[\frac{\text{mrad} - \text{m}^3}{\text{pCi} - \text{yr}} \right] Q \text{ (Ci)} \\
&= 3.17 \times 10^{-4} \times 1.10 \times 10^{-5} \times 1.72 \times 10^{-5} \times Q_{Kr-85} && \text{(Eq. 3-6.1)} \\
&= 6.0 \times 10^{-8} Q_{Kr-85} \text{ (Ci)}
\end{aligned}$$

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.

3.13.1 Method I

The beta air dose from plant vent stack releases is:

$$D_{\text{air}}^{\beta} \text{ (mrad)} = 0.76 Q_{\text{Kr-85}} DF_{\text{Kr-85}}^{\beta} \quad (\text{Eq. 3-7})$$

Where:

$DF_{\text{Kr-85}}^{\beta}$ = Beta air dose factor for Kr-85. See Table 1.2.

$Q_{\text{Kr-85}}$ = Total Kr-85 (Curies) released to the atmosphere via the plant vent stack 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 conditions (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 for stack releases. 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

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.

With Kr-85 being the only noble gas potentially available for release, the dose can be taken from Equations B-4 and B-5 of Regulatory Guide 1.109:

$$D_{air}^{\beta} \text{ (mrad)} = (3.17 \times 10^{-4}) [X/Q] Q_{Kr-85} DF_{Kr-85}^{\beta}$$

Substituting:

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

We have:

$$D_{air}^{\beta} \text{ (mrad)} = 0.76 Q_{Kr-85} DF_{Kr-85}^{\beta} \tag{Eq. 3-7}$$

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

Control 3.5 limits the critical organ dose to a MEMBER OF THE PUBLIC from radioactive tritium 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.

3.14.1 Method I

The critical organ dose from a gaseous release is:

D_{co} (mrem) = Σ Q_i DFG_{ico} (Eq. 3-8)

Where:

Q_i = Total activity (Curies) released to the atmosphere of radionuclide "i" during the period of interest. For i = 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 conditions (not emergency event),
b. 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

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. 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 which applies to direct radiation as well as to liquid and gaseous effluents. These methods apply only to tritium and particulates in gaseous effluents.

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 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 stack gas releases 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.

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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)	-	-	48.00	48.00	48.00	48.00	480.00	480.00
TB Soil Exposure Time ⁽¹⁾ (hrs)	131400.0 0	131400.0 0	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 ⁽²⁾	-	-	0.50	-	0.50	-	0.50	-
FS Fraction Pasture When on Pasture ⁽³⁾	-	-	1.00	-	1.00	-	1.00	-
FG Fraction of Stored Veg. Grown in Garden	0.76	-	-	-	-	-	-	-

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
FL Fraction of Leafy Veg. Grown in Garden		1.00	-	-	-	-	-	-
FI Fraction Elemental Iodine = 0.5	-	-	-	-	-	-	-	-
H Absolute ⁽⁴⁾ 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.

Notes:

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

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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 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 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} .

Another 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⁽¹⁾

	Dose Rate to Individual			Dose to Air		Dose to Critical Organ
	Total Body	Skin	Critical Organ	Gamma	Beta	Thyroid
X/Q Depleted $\left(\frac{\text{sec}}{\text{m}^3}\right)$	-	-	2.19×10^{-5}	-	-	2.19×10^{-5}
X/Q Undepleted $\left(\frac{\text{sec}}{\text{m}^3}\right)$	-	2.39×10^{-5}	-	-	2.39×10^{-5}	-
D/Q $\left(\frac{1}{\text{m}^2}\right)$	-	-	5.02×10^{-8}	-	-	5.02×10^{-8}
X/Q ^y $\left(\frac{\text{sec}}{\text{m}^3}\right)$	7.83×10^{-6}	7.83×10^{-6}	-	7.83×10^{-6}	-	-

⁽¹⁾SSE SITE BOUNDARY, 800 meters from the primary vent stack.

hb

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 at and beyond the Site Boundary from all station sources (including direct radiation) to the limit of 25 mrem in a year, except for the thyroid which is limited to 75 mrem in a year.

Estimates of direct exposure above background in areas at and beyond the site boundary (or in residential areas inside the site boundary) can be determined from measurements made by environmental TLDs that are part of the Environmental Monitoring Program (see Table 4.4). Alternatively, direct dose calculations from identified fixed sources on site can be used to estimate the off-site direct dose contribution where TLD information may not be applicable.

4.0 RADIOLOGICAL ENVIRONMENTAL MONITORING

4.1 Monitoring Program

Control 4.1 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.b), 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:

$$\frac{\text{concentration (1)}}{\text{reportinglevel (1)}} + \frac{\text{concentration (2)}}{\text{reportinglevel (2)}} + \dots \geq 1.0$$

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.

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. Particulates	5	Continuous operation of sampler with sample collections as required by dust loading, but at least once per two weeks.	Gross beta radioactivity following filter change. Composite (by location) for gamma isotopic at least once per quarter.
2. DIRECT RADIATION	24***	Quarterly	Gamma dose, at least once per quarter.
3. WATERBORNE a. Surface	2	Composite sample** collected over a period of one month at downstream location; monthly grab sample at upstream control location.	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.
c. Sediment from Shoreline	1****	At least once per six months.	Gamma isotopic analysis 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
4. INGESTION a. Fish	2	Commercially and recreationally important species. Seasonal or semiannually, if not seasonal.	Gamma isotopic analysis on edible portions.
b. 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 on edible portions.

- * 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.
- *** Does not include Restricted Area Fence locations or those TLDs associated with the ISFSI pad monitoring.
- **** One sample from downstream area with existing or potential recreational value.

TABLE 4.2

Reporting Levels for Radioactivity Concentrations in Environmental Samples

Analysis	Water (pCi/l)	Airborne Particulates (pCi/m ³)	Fish (pCi/kg, wet)	Food Products (pCi/kg, wet)
H-3	$3 \times 10^{+4}$	-	-	-
Mn-54	$1 \times 10^{+3}$	-	$3 \times 10^{+4}$	-
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	$4 \times 10^{+2}$	-	-	-
Cs-134	$3 \times 10^{+1}$	$1 \times 10^{+1}$	$1 \times 10^{+3}$	$1 \times 10^{+3}$
Cs-137	$5 \times 10^{+1}$	$2 \times 10^{+1}$	$2 \times 10^{+3}$	$2 \times 10^{+3}$

Reporting levels for nondrinking water pathways.

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TABLE 4.3

Detection Capabilities for Environmental Sample Analysis^{(a) (c)}

Analysis ^(d)	Water (pCi/l)	Airborne Particulates (pCi/m ³)	Fish (pCi/kg. wet)	Food Products (pCi/kg. wet)	Sediment (pCi/kg. dry)
Gross beta	4×10^0	1×10^{-2}	-	-	-
H-3	2×10^3	-	-	-	-
Co-60	$1.5 \times 10^{+1}$	-	$1.3 \times 10^{+2}$	-	-
Cs-134	$1.5 \times 10^{+1}$	5×10^{-2}	$1.3 \times 10^{+2}$	$6 \times 10^{+1}$	$1.5 \times 10^{+2}$
Cs-137	$1.8 \times 10^{+1}$	6×10^{-2}	$1.5 \times 10^{+2}$	$8 \times 10^{+1}$	$1.8 \times 10^{+2}$

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):

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

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 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. Parent only.
- c. If the measured concentration minus the 5 sigma counting statistics is found to exceed the specified LLD, the sample does not have to be analyzed to meet the specified LLD.
- d. This list does not mean that only these 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.

4.2 Land Use Census

Control 4.2 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.b), 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 3.5, identify the new location(s) in the next Annual Radioactive 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 Annual Radioactive Effluent Release Report.

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.

* 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.

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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 Decommissioning Quality Assurance Program (Appendix D, Section B.4.b), analyses shall be performed on referenced radioactive materials supplied as part of the quality assurance Laboratory Intercomparison Program.

Applicability

At all times.

ACTION

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.

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.

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.

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 (Particulates)	AP-11 Observation Stand AP-12 Monroe Bridge AP-13 Rowe School AP-14 Harriman Power Station AP-21 Williamstown, MA	0.5 1.1 4.2 3.2 22.2	NW SW SE N W
2. WATERBORNE a. Surface b. Ground c. Sediment From Shoreline	WR-11 Bear Swamp Lower Reservoir WR-21 Harriman Reservoir WG-11 Plant Potable WG-12 Sherman Spring SE-11 Number 4 Station SE-21 Harriman Reservoir	6.3 10.1 On-Site Well 0.2 36.2 10.1	Downriver Upriver NW Downriver Upriver
3. INGESTION a. Fish and Invertebrates b. Food Products	FH-11 Sherman Pond FH-21 Harriman Reservoir TF-11 Monroe Bridge TF-13 Monroe, MA TF-21 Williamstown, MA	1.5 10.1 1.9 1.9 21.0	At Discharge Point Upriver WSW WNW WSW

TABLE 4.4
(Continued)

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 (Plant)	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-25 Whitingham, VT	7.70	NNE
	GM-27 Number 9 Road	7.60	ENE
	GM-29 Route 8A	8.20	ESE
	GM-31 Legate Hill Road	7.60	SSE
	GM-32 Rowe Road	7.90	S
	GM-33 Zoar Road	6.90	SSW
	GM-35 Whitcomb Summit	8.60	WSW
	GM-36 Tilda Road	6.60	W
	GM-38 West Hill Road	6.60	NW
	GM-40 Readsboro Road	0.50	W

TABLE 4.4
(Continued)

Exposure Pathway and/or Sample	Sample Location and Designated Code			Approximate Distance from Center of ISFSI Pad (meters)	Direction from Center of ISFSI Pad
5. DIRECT RADIATION (Plant)	IF-1	ISFSI Security Fence	***	20	MNW
	IF-2	Observation Stand	***	560	NW
	IF-3	ISFSI Security Fence	***	20	N
	IF-4	ISFSI Security Fence	***	34	NE
	IF-5	ISFSI Security Fence	***	28	E
	IF-6	ISFSI Security Fence	***	15	SE
	IF-7	ISFSI Security Fence	***	23	S
	IF-8	ISFSI Security Fence	***	38	SW
	IF-9	Restricted Area Fence (Plant)	***	50	SE
	IF-10	Restricted Area Fence (Plant)	***	55	SSE
	IF-11	Restricted Area Fence (Plant)	***	135	SW
	IF-12	Restricted Area Fence (Warehouse)	***	225	N
	IF-18	C.W. Intake	***	240	NNW
	IF-19	Restricted Area Fence (Adm. Bldg.)	***	170	W
IF-20	Restricted Area Fence (Gatehouse)	***	235	WNW	
IF-40	Readsboro Road	***	700	N	

* Sample locations are shown on Figures 4-1 through 4-7.

** Station 1X's are indicator stations, and Station 2X's are control stations (excluding the direct radiation stations).

***Not included as part of the Radiological Environmental Monitoring Program.

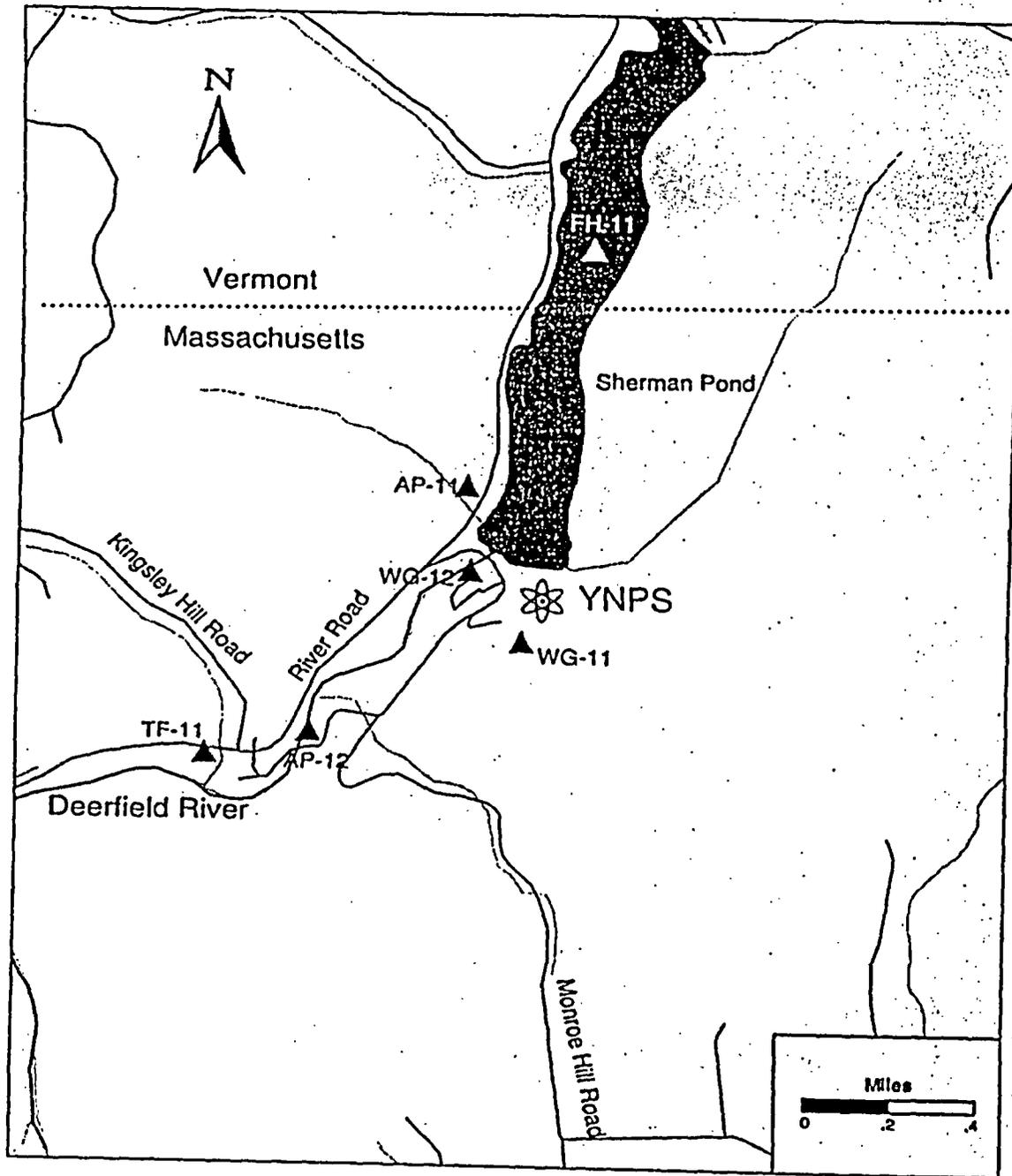


Figure 4-1 Yankee Plant Radiological Environmental Monitoring Locations within 1 Mile (Airborne, Waterborne and Ingestion Pathways)

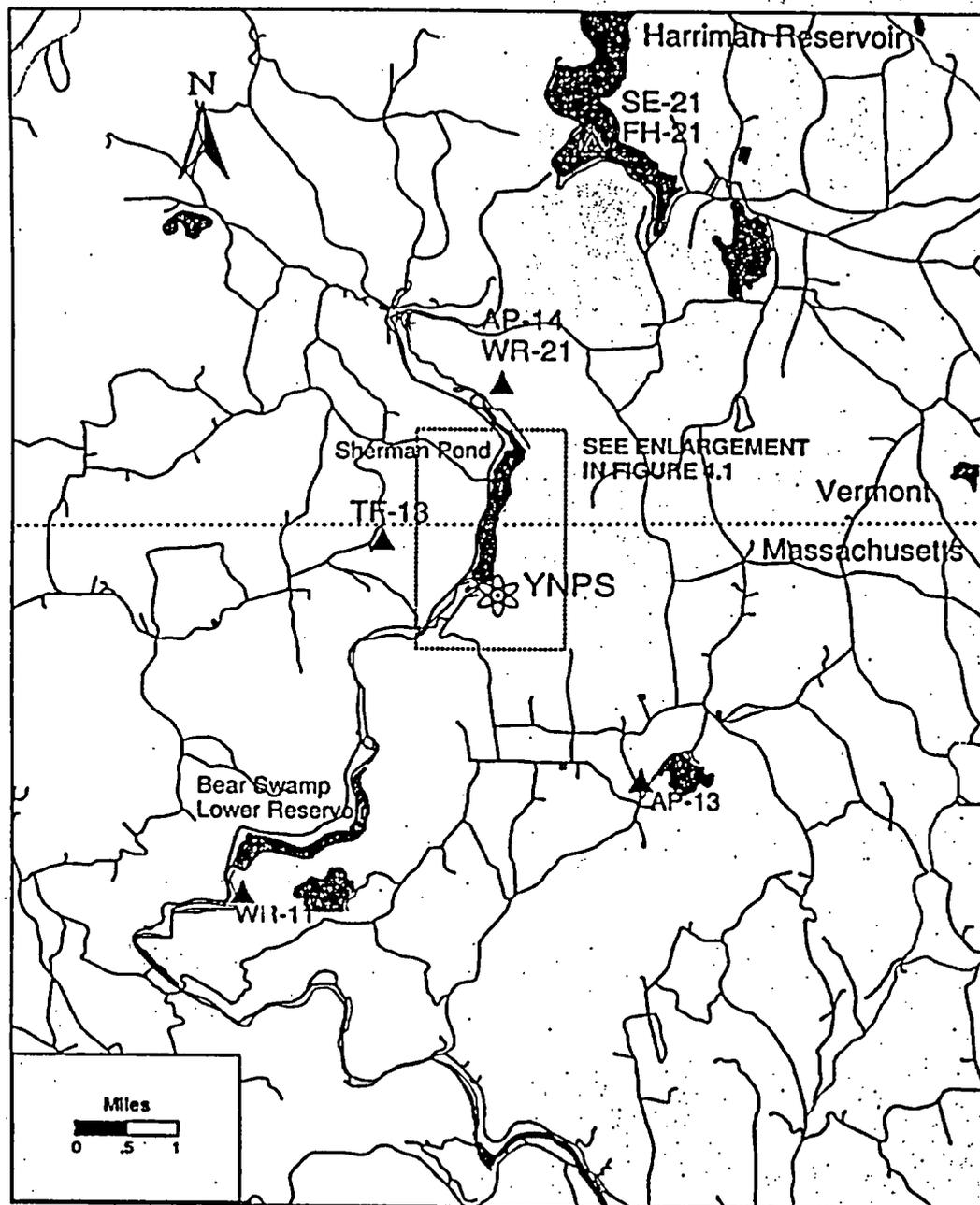


Figure 4-2 Yankee Plant Radiological Environmental Monitoring Locations Within 12 Miles (Airborne, Waterborne and Ingestion Pathways)

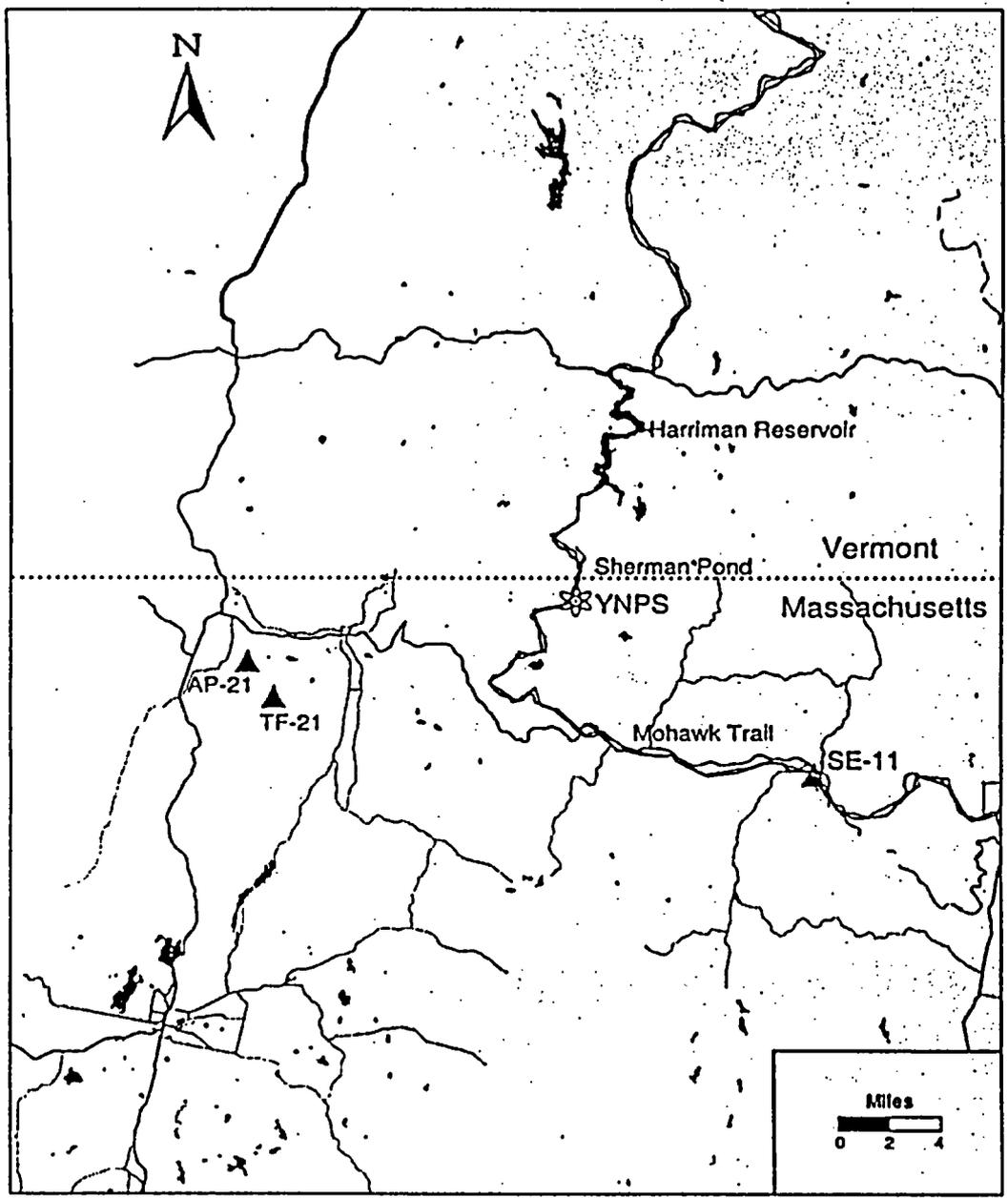


Figure 4-3 Yankee Plant Radiological Environmental Monitoring Locations Outside 12 miles (Airborne, Waterborne and Ingestion Pathways)

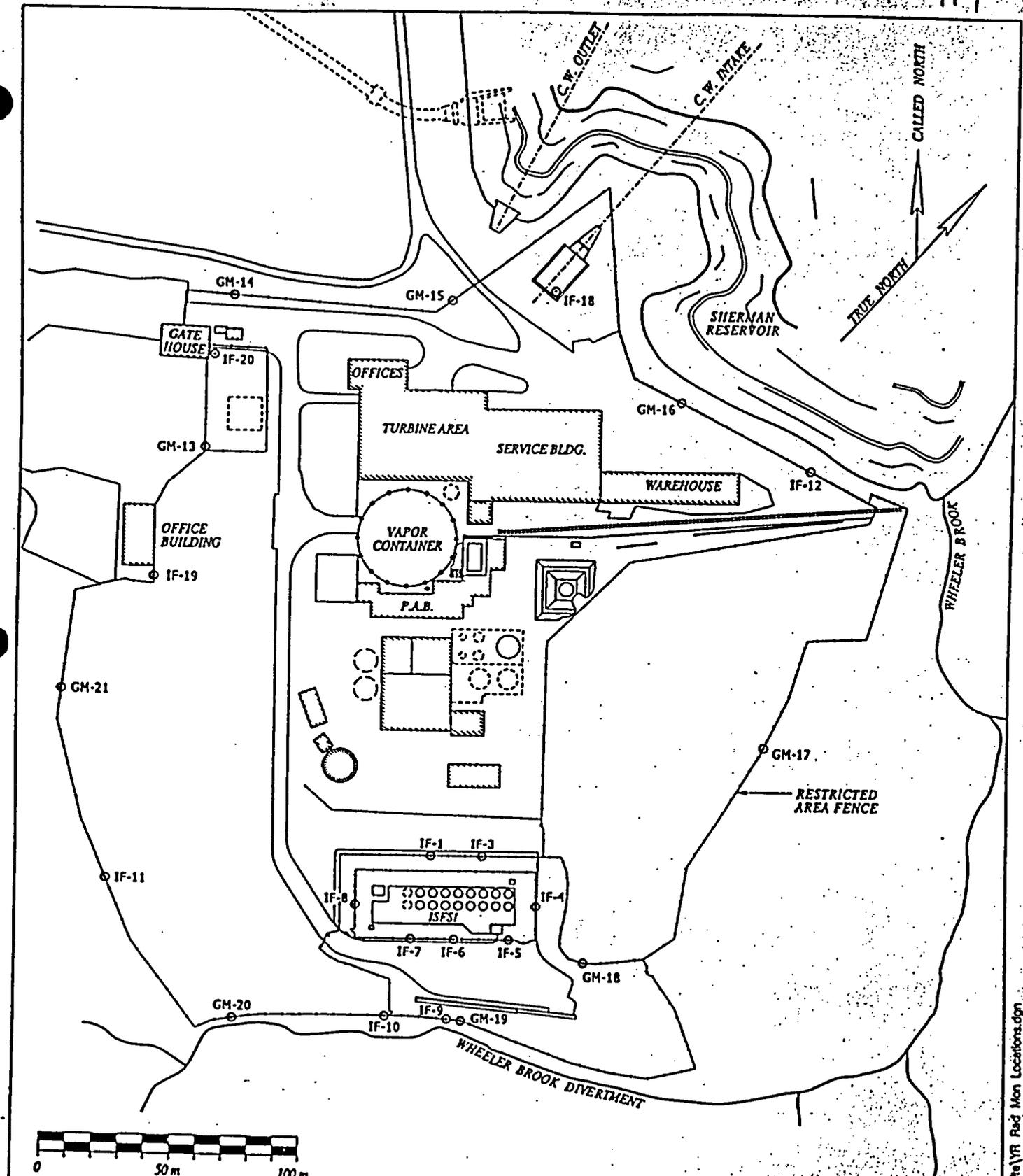


FIGURE 4-4
 Yankee Plant Radiological Environmental Monitoring Locations at the
 Restricted Area Fence (Direct Radiation Pathway)

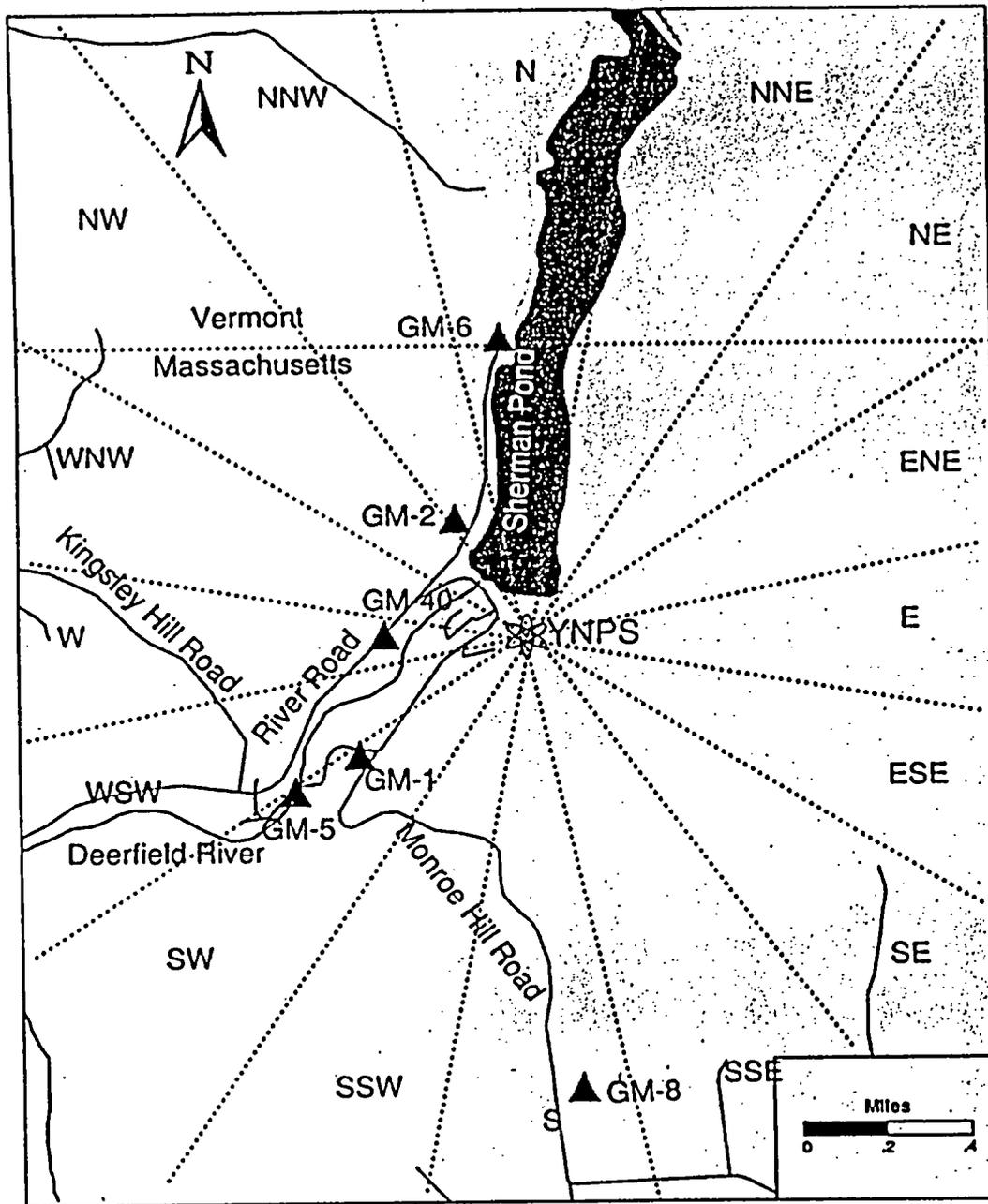


Figure 4-5 Yankee Plant Radiological Environmental Monitoring Locations within 1 mile (Direct Radiation Pathway)

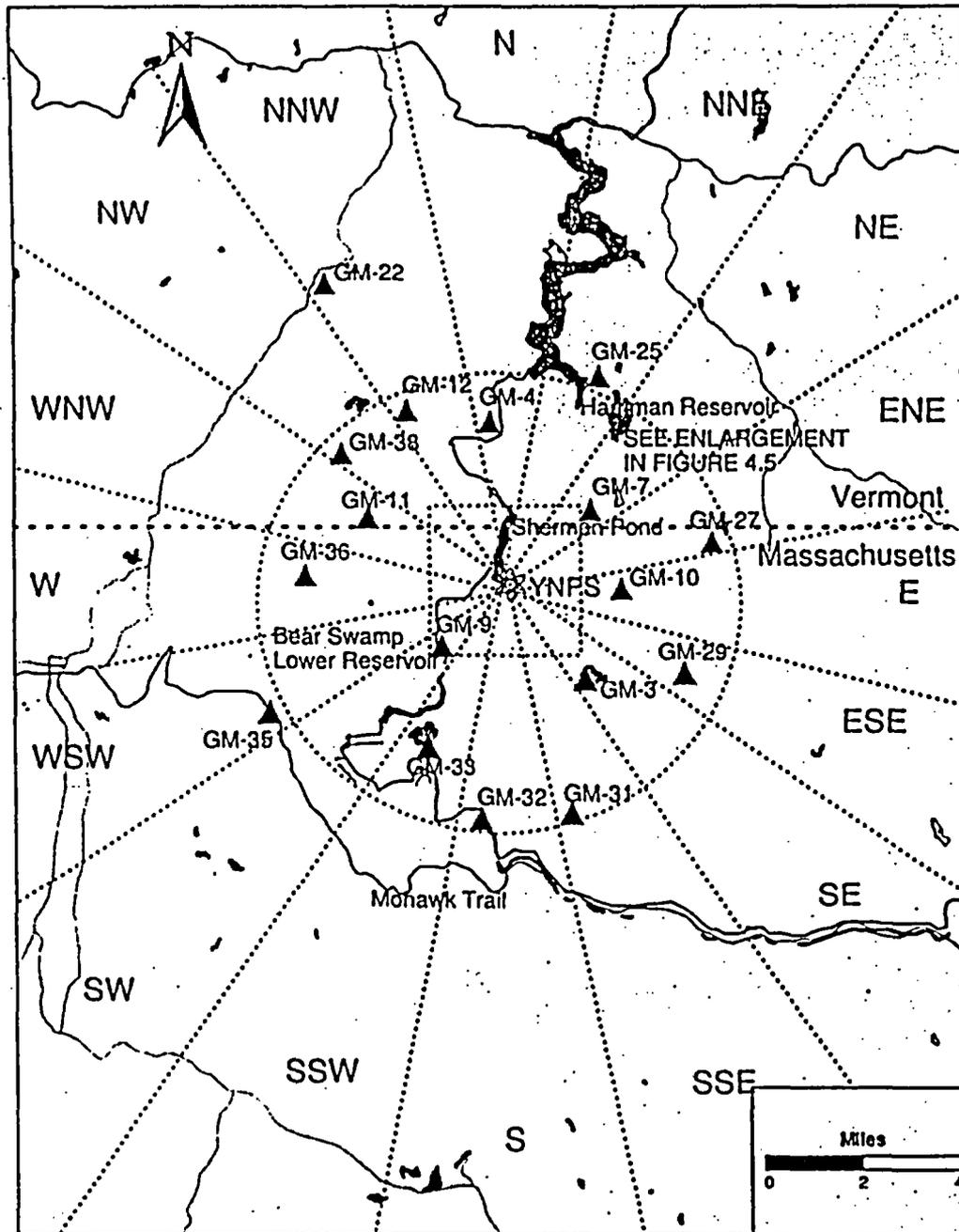


Figure 4-6 Yankee Plant Radiological Environmental Monitoring Locations within 12 Miles (Direct Radiation Pathway)

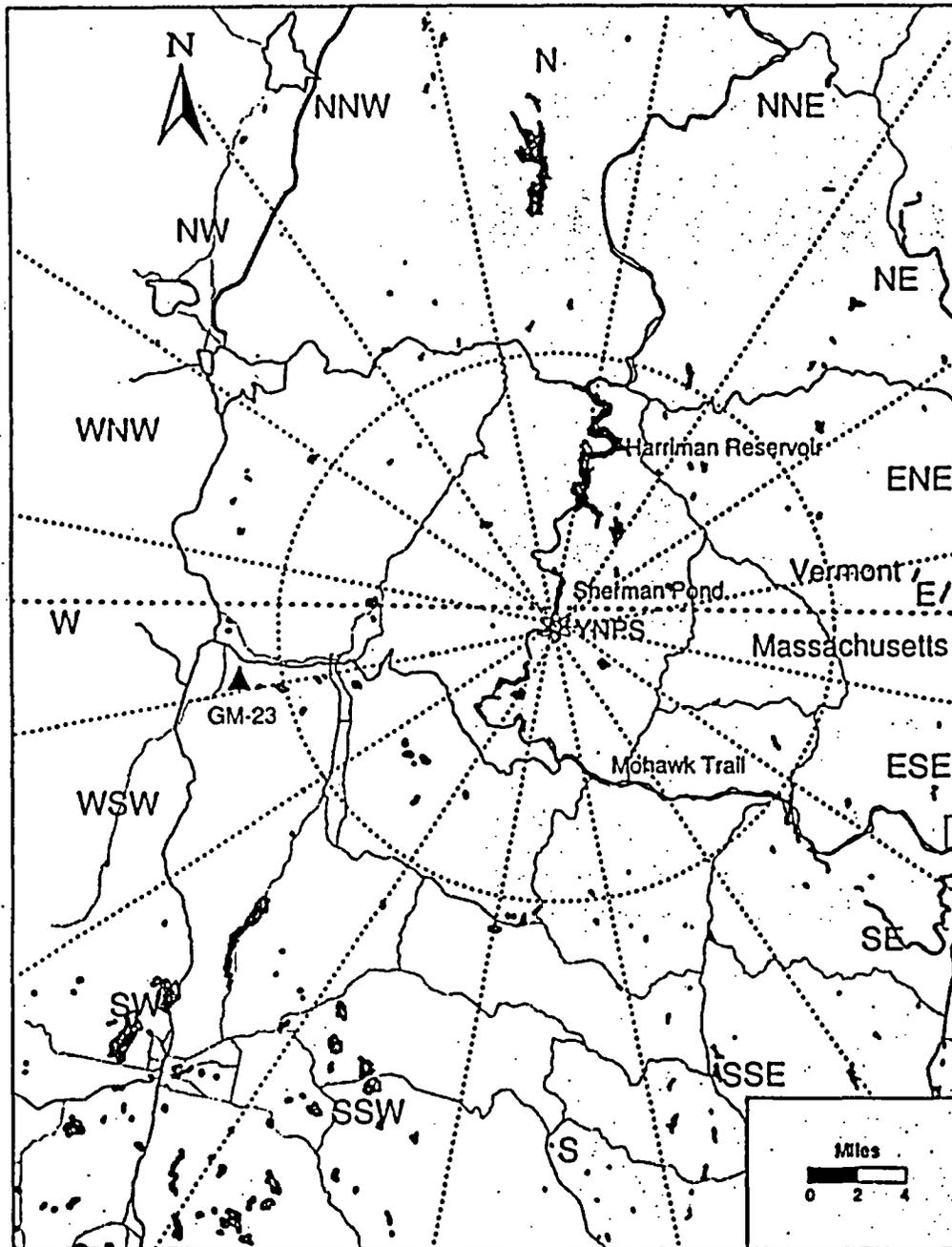


Figure 4-7 Yankee Radiological Environmental Monitoring Locations Outside 12 Miles (Direct Radiation Pathways)

5.0 INSTRUMENTATION

5.1 Radioactive Liquid Effluents

Control 5.1 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), 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 Annual Radioactive Effluent Release Report the reason for the delay in correcting the inoperability.

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.

A gross radioactivity monitor which provides for automatic isolation of liquid discharges on detection of radioactivity concentrations in excess of the values of 10CFR Part 20 (see Appendix B of ODCM), is included on the liquid radioactive waste effluent discharge line from the plant's test tanks. The automatic alarm/trip function provided by this monitor 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 for continuous potential radioactive effluent pathways (i.e., ASW discharge) to give assurance that potential radioactive liquid releases to the environment are accounted for (See Figure 6-1).

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TABLE 5.1

Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	Minimum Channels OPERABLE	Applicability	ACTION
1. Gross Radioactivity Monitor Providing Automatic Isolation a. Liquid Radwaste Effluent Line ¹	(1)	At All Times ⁴	1
2. Deleted			
3. Continuous Composite Samplers a. Auxiliary Service Water Effluent Line ² b. SFP Dewatering Skid Effluent Line ³	(1) (1)	* *	5 5
4. Flow Rate Measurement Devices a. Liquid Radwaste Effluent Line b. Auxiliary Service Water System Effluent c. SFP Dewatering Skid Effluent Line	(1) (1) (1)	** ** **	3 3 3

¹ A common radioactivity monitor (ASW-RM-001) provides indication for both the Liquid Radwaste Effluent Line and the Auxiliary Service Water Effluent. Actuation of the radioactivity monitor alarm will isolate test tank releases (if in progress).

² The ASW Effluent Line composite sampler is intended to provide water samples during the discharge of water from SFP drain down only. ASW composite sampler not required during discharge of evaporator processed test tanks.

TABLE 5.1
(Continued)

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Table Notation

- ³ The SFP Dewatering composite sampler is required only during SFP drain down processing operations.
- ⁴ Monitor preventive maintenance, testing and calibration are permitted under the ACTION requirements.
- * Via this pathway during SFP discharge (3.a) or SFP drain down (3.b.) only.
- ** Via this pathway during releases.

TABLE 5.1
(Continued)

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ACTION Statements

ACTION 1 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, all radioactive waste effluent releases from the Test Tanks, TK-39 or direct SFP drain down shall be curtailed. Effluent releases from the Test Tanks or TK-39 may continue, provided that prior to initiating the release:

- a. At least two independent samples of the Test Tank's contents, or two separate aliquots from the SFP dewatering skid composite sampler servicing TK-39, 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,

ACTION 2 - Deleted

ACTION 3 - 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.

ACTION 4 - Deleted

TABLE 5.1
(Continued)

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ACTION Statements

ACTION 5 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, operation of the potential effluent release pathway may continue, provided that the associated effluent monitor is verified to be operating and grab samples are taken at least once per 24 hours and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 1.0×10^7 microcuries/ml at least weekly on a composite sample.

ACTION 6 - Deleted

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 Monitor Providing Alarm and Automatic Isolation a. Liquid Radwaste Effluent Line ^(a)	D	P	R ⁽²⁾	Q ⁽¹⁾	At All Times
2. Deleted					
3. Continuous Composite Samplers a. Auxiliary Service Water Effluent Line b. SFP Dewatering Skid Effluent Line	D D	NA NA	NA NA	Q Q	*** ***
4. Flow Rate Measurement Devices a. Liquid Radwaste Effluent Line	D ⁽³⁾	NA	R	NA	**

TABLE 5.2
(Continued)

Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES in Which Surveillance is Required
4. Flow Rate Measurement Devices (continued)					
b. Auxiliary Service Water Discharge	D ⁽³⁾	NA	R	NA	**
c. SFP Dewatering Skid Effluent Line	D ⁽³⁾	NA	R	NA	**

** Via this pathway during waste effluent releases.

*** Via this pathway during SFP discharge (3.a) or SFP drain down (3.b) only.

(*) A common radioactivity monitor (ASW-RM-001) provides indication for both the Liquid Radwaste Effluent Line and Auxiliary Service Water Effluent Line. Actuation of the radioactivity monitor alarm will isolate test tank releases (if in progress).

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TABLE 5.2
(Continued)

Table Notation

- (1) - The CHANNEL FUNCTIONAL TEST also shall demonstrate that automatic isolation of the liquid radwaste effluent line, 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 alarm indication.

- (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. The CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or batch releases are made.

5.2 Radioactive Gaseous Effluents

Control 5.2 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), 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 Annual Radioactive Effluent Release Report the reason for the delay in correcting the inoperability.

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 Control 3.3 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.

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	9
b. Particulate Sampler Filter	(1)	*	Verify Presence of Filter	8
c. Effluent System Flow Rate Measuring Device	(1) ^a	*	System Rate Measurement	7
d. Sampler Flow Rate Measuring Device	(2) ^b	*	Sampler Flow Rate Measurement	7

* At all times.

^a One channel per pathway (VC and PAB ventilation flow paths) required.

^b One flow device channel is for the noble gas channel, and a separate channel is for the fixed particulate filter.

TABLE 5.3
(Continued)

ACTION Statements

- ACTION 7 - 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 12 hours.

- ACTION 8 - 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 9 - 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 12 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. Particulate Sampler Filter	W	NA	NA	NA	•
c. System Effluent Flow Rate Measuring Device	D	NA	NA	NA	•
d. Sampler Flow Rate Measuring Device	D	NA	R	Q	•

* At all times.

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 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, as listed in Appendix B of the ODCM, and a total noble gas MPC.

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

5.3.1 Method

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

$$R = \left(\frac{f_2}{f_1} \right) (MPC_c) (S \ell) \quad (\text{Eq. 5-1})$$

Where:

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

f_2 = Flow rate at the point of discharge (gpm).

$S \ell$ = Instrument response factor (cpm/($\mu\text{Ci/ml}$)).

MPC_c = Composite MPC for the mix of radionuclides ($\mu\text{Ci/ml}$).

$$MPC_c = \frac{\sum_i C_i}{\sum_i C_i / MPC_i} = \frac{\sum_i f_i}{\sum_i f_i / MPC_i} = 1 / \left(\sum_i f_i / MPC_i \right) \quad (\text{Eq. 5-2})$$

Where:

MPC_i = MPC for radionuclide "i" from 10CFR Part 20, Appendix B, Table II, Column 2 ($\mu\text{Ci/ml}$). See ODCM Appendix B.

C_i = Concentration of radionuclide "i" in mixture ($\mu\text{Ci/ml}$).

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 administratively set lower to accommodate pathways which normally are nonradioactive (Auxiliary Service Water). The auxiliary Service Water is a normally clean system. The same radiation effluent monitor provides detection of the presence of an off normal condition that may have unexpectedly introduced radioactive contamination to this clean system. The alarm setpoint when only ASW cooling flow is in operation is set at two to three times background to give as early an alarm as practicable. SFP cooling flow is secured from the SFP heat exchanger when test tank discharges are made. This requirement allows the common radiation monitor to see either the expected clean ASW pathway flow when SFP cooling operations are on going (potential source), or the expected radioactivity in the test tank effluent flow when this discharge pathway is in operation.

5.3.2 Liquid Effluent Setpoint Example

The effluent monitor for the test tank release pathway is gamma sensitive monitor. It has a typical sensitivity, S, of $2.8\text{E}+8$ cpm per $\mu\text{Ci/ml}$ of gamma emitters which emit one photon per disintegration and a typical background of about 330 cpm.

The composite MPC and setpoint can be calculated based on the following example data:

i	f_i	MPC_i ($\mu\text{Ci/ml}$)
Cs-134	0.02	9×10^{-6}
Cs-137	0.18	2×10^{-5}
Co-60	0.80	3×10^{-5}

$$MPC_c = \frac{\sum_i f_i}{\sum_i f_i / MPC_i} \tag{Eq. 5-2}$$

$$= \frac{1}{(0.02/9 \times 10^{-6} + 0.18/2 \times 10^{-5} + 0.80/3 \times 10^{-5})}$$

$$MPC_c = 2.6 \times 10^{-3} \text{ } (\mu\text{Ci/ml})$$

For this example, normal liquid effluent flow rate, (f_1), is assumed to be 2.8 gpm. Dilution water flow, f_2 , is assumed to be 90 gpm (equivalent to total flow of both contaminated and clean water). The setpoint for the monitor when the test tank effluent pathway is operating is then calculated for these example conditions to be:

$$R = \left(\frac{f_2}{f_1} \right) (MPC_c) (St) \tag{Eq. 5-1}$$

$$= \left(\frac{90 \text{ gpm}}{2.8 \text{ gpm}} \right) (2.6 \times 10^{-3} \text{ } \mu\text{Ci/ml}) (2.8E+8 \text{ cpm}/(\mu\text{Ci/ml}))$$

$$= 234,000 \text{ cpm}$$

This setpoint value may be administratively set lower than the maximum count rate for conservatism.

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 responds to the concentration of radioactivity as follows:

$$R_{(cpm)} = (St) \left(\sum_i f_i s_i \right) (C_{MON}) \tag{Eq. 5-5}$$

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

C_{MON} = Total concentration ($\mu\text{Ci/ml}$) 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:

$$R = (S \ell) (C_{MON}) \quad (\text{Eq. 5-6})$$

For simplicity, assume that the monitor looks at a flow for f_1 . We know that:

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

Where:

C = Total concentration at the point of discharge.

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

$$R = \left(\frac{f_2}{f_1} \right) (C) (S \ell) \quad (\text{Eq. 5-8})$$

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

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

The right side of the equation is the sum of the ratios of the MPC limits in 10CFR Part 20 (Appendix B of the ODCM). Solving for MPC_c , the composite MPC for the mixture, we get the definition of MPC_c :

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

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:

$$R = \left(\frac{f_2}{f_1} \right) (MPC_c) (S\ell) \quad (\text{Eq. 5-1})$$

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.

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 for Kr-85 as the only noble gas left in the spent fuel following permanent shutdown in October 1991.

R is the lesser of:

$$R_{tb} = \frac{(500) (60) (S_{Kr-85})}{(F) (7.83) DFB_{Kr-85}} \tag{Eq. 5-3}$$

And:

$$R_{sk} = \frac{(3000) (60) (S_{Kr-85})}{(F) DF'_{Kr-85}} \tag{Eq. 5-4}$$

Where:

$$7.83 = 10^6 \text{ (pCi/}\mu\text{Ci)} \cdot 7.83 \times 10^{-6} \text{ (sec/m}^3\text{)}$$

DF'_{Kr-85} = Combined Skin dose factor for Kr-85. See Table 1.2.

DFB_{Kr-85} = Total body dose factor for Kr-85. See Table 1.2.

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

S_{Kr-85} = Gaseous monitor response factor for Kr-85 (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. System characteristics are:

- a. Typical sensitivity - 1 cpm \approx 3×10^{-8} μCi/cc of Kr-85; that is, $S \approx 3.3 \times 10^{+7}$ cpm/(μCi/cc)
- b. Typical background - 100 to 200 cpm

Under normal plant stack flow, F, of $5.80 \times 10^{+8}$ cc/min (20,500 cfm x 28,300 cc/ft³).

Applying Equations 5-3 and 5-4:

$$R_{db} = \frac{(500)(60)(3.3 \times 10^{+7})}{(5.80 \times 10^{+8})(7.83)(1.61 \times 10^{-5})} = 1.35 \times 10^{+7} \text{ cpm}$$

$$R_{sk} = \frac{(3000)(60)(3.3 \times 10^{+7})}{(5.80 \times 10^{+8})(3.22 \times 10^{-2})} = 318,000 \text{ cpm}$$

The setpoint, R, is the lesser of R_{db} and R_{sk}; therefore, it is equal to 318,000 cpm. Since Kr-85 is the only noble gas radionuclide left for potential release, the skin dose rate limit calculated in the example is the maximum setpoint permitted.

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 derivation of Equations 5-3 and 5-4 starts with the general equation for the response, R (cpm), of a radiation monitor and the basic assumption that only one noble gas radionuclide is expected to be available for detection (Kr-85) for the long-term shutdown mode:

$$R = (S_{Kr-85}) (C) \tag{Eq. 5-5}$$

$$(cpm) = (cpm/(\mu Ci/cc)) (1) (\mu Ci/cc)$$

Expanding for the concentration:

$$R = (S_{Kr-85}) (Q) (60/F) \tag{Eq. 5-10}$$

$$(cpm) = (cpm/(\mu Ci/cc)) (1) (\mu Ci/sec) (sec/min)/(cc/min)$$

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.

$$D_{tb} = 7.83 Q_{Kr-85} DFB_{Kr-85}$$

where 7.83 is equal to the limiting off-site gamma X/Q from Table 3-6 times a unit conversion factor of 10^{+6} pCi/ μ Ci.

Rearranging to solve for Q:

$$Q = \frac{D_{tb}}{7.83 DFB_{Kr-85}} \tag{Eq. 5-13}$$

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

$$R_{tb} = \frac{S_{Kr-85} \text{ (cpm/(\mu Ci/cc)) } 500 \text{ (mrem/yr) } 60 \text{ (sec/min)}}{F \text{ (cc/min) } 7.83 \text{ (pci-sec/\mu Ci-m}^3\text{) } DFB_{Kr-85} \text{ (mrem-m}^3\text{/pCi-yr)}} \quad \text{(Eq. 5-3)}$$

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:

$$D_{sk} = QDF'_{Kr-85}$$

Rearranging to solve for Q:

$$Q = \frac{D_{sk}}{DF'_{Kr-85}} \quad \text{(Eq. 5-15)}$$

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

$$R_{sk} = \frac{S_{Kr-85} \text{ (cpm/(\mu Ci/cc)) } 3000 \text{ (mrem/yr) } 60 \text{ (sec/min)}}{F \text{ (cc/min) } DF'_{Kr-85} \text{ (mrem-sec/\mu Ci-yr)}}$$

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

6.1 Liquid Radioactive Waste Treatment

Control 6.1 In accordance with Yankee Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), 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.

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 Decommissioning Quality Assurance Program (Appendix D, Section B.4.a), 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.

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 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.

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

Figure 6-1 shows the liquid effluent streams, radiation monitors, and the appropriate Liquid Radioactive Waste Treatment System. Figure 6-2 shows the gaseous effluent streams, radiation monitors, and the appropriate Ventilation Exhaust Treatment System.

6.4 In-Plant Liquid Effluent Pathways

Batch effluent tanks called "test tanks" collect the distillate from the atmospheric liquid radioactive waste evaporator. Normally, liquid waste accumulates at about 0.3 gpm and is processed at about 1 gpm. When the test tanks are full, they are sampled, analyzed, and released at a nominal 5 gpm.

Auxiliary service water provides dilution water flow. Typically, flow rates range up to 90 gpm for normal auxiliary service water use.

Calibrations of the radiation monitor has 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 the monitor channel has an adjustable alarm setpoint.

Spent Fuel Pool drain down occurs after all spent fuel and other contaminated materials are transferred to the Independent Spent Fuel Storage Installatin (ISFSI). Discharge of the SPF water in either batch or continuous mode will be treated by demineralization and filtration before release to Outfall 001. Figure 6.1 indicates the flow paths and in-line radiation monitoring prior to release. The primary flow path for SFP dewatering is to direct the flow (continuous mode) after waste treatment through a composite sampler (for representative sampling of the discharge stream) upstream of ASW dilution flow and finally past the ASW in-line radiation monitor before release to Sherman Pond via Outfall 001.

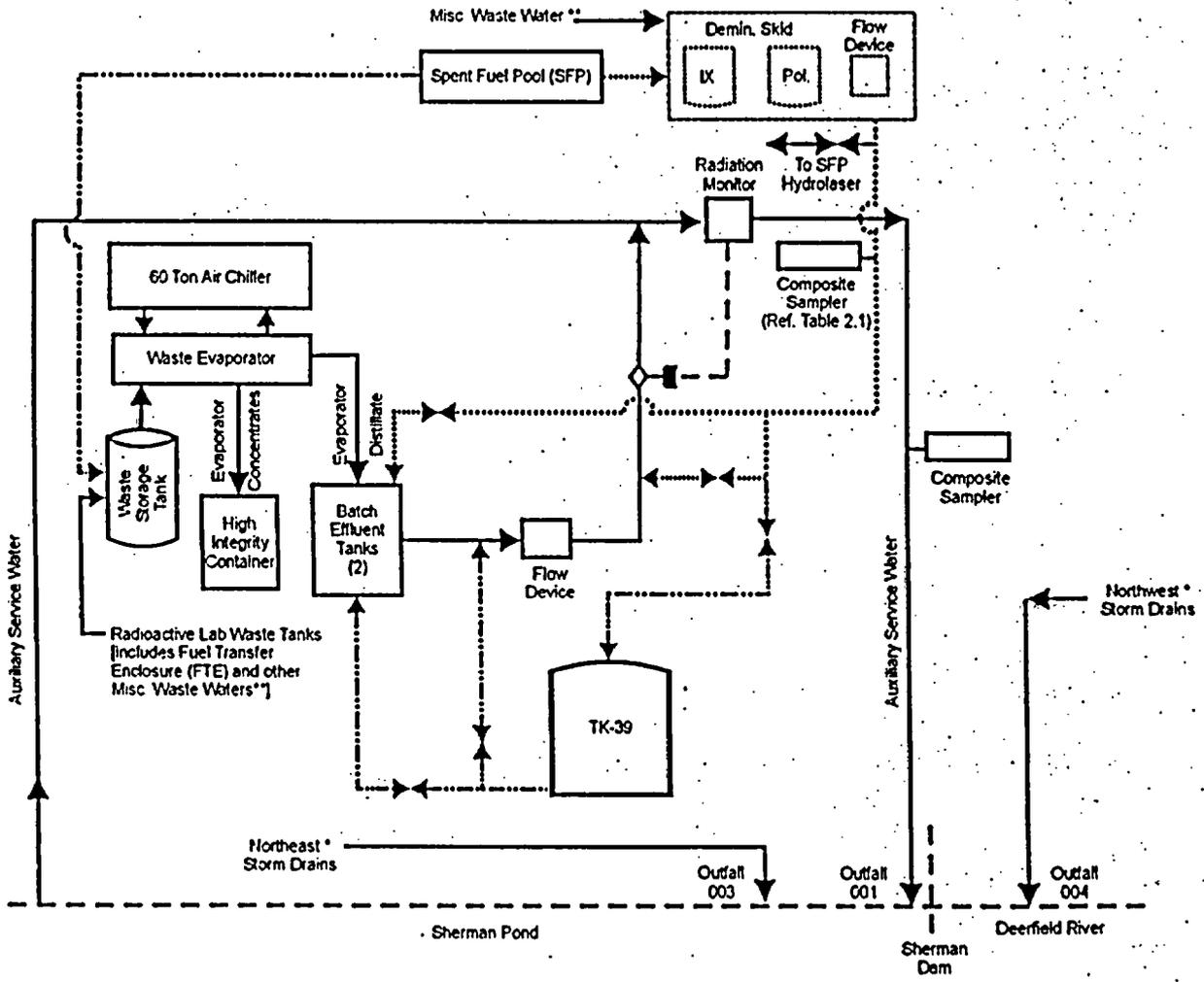
6.5 In-Plant Gaseous Effluent Pathways

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. 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 ~ 3×10^{-8} μ Ci/cc Kr-85
- b. Typical background 100-200 cpm

FIGURE 6-1

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



----- Contingency Path, SFP Dewatering
 Primary Path: SFP Dewatering

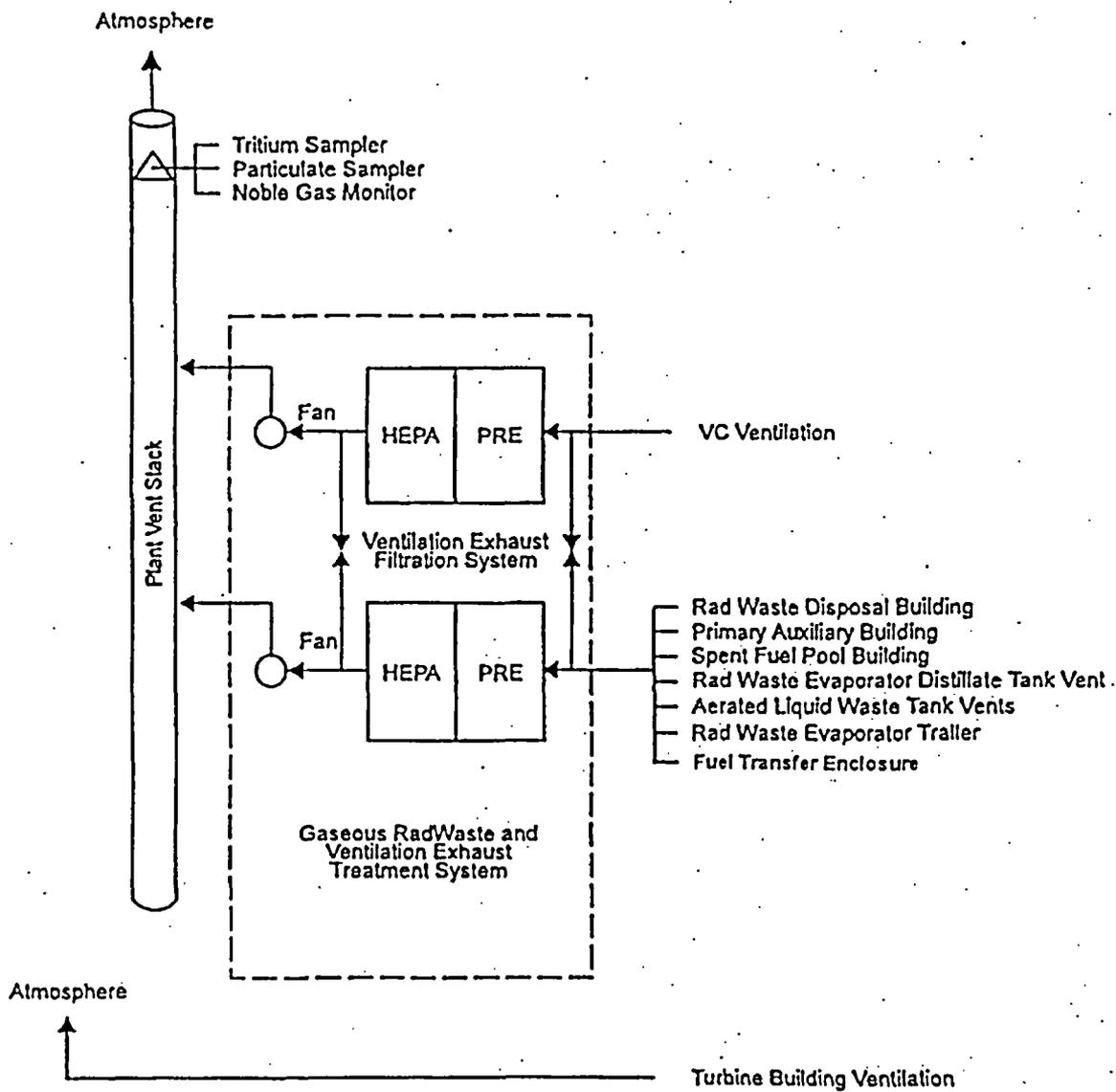
* For construction dewatering, process treatment, if needed, will be provided before release to the environment. Treatment for radioactivity reduction could include any combination of evaporation, ion exchange and/or filtration by either installed or temporary process equipment as appropriate for the radionuclide content and water quality of the source

** Miscellaneous waste water includes, but is not limited to, sources such as construction dewatering that are not derived from enclosed plant systems.

CDSTRUM/FIG6_1

FIGURE 6-2

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



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 to the NRC in accordance with Technical Specification 6.8.2.a 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 Annual Radioactive Effluent Release Report

Control 7.2

- a. Before May 1 of each year, a report shall be submitted to the NRC in accordance with the requirements of Technical Specification 6.8.2.b covering the radioactive content of effluents released to unrestricted areas during the previous calendar year.
- b. The Annual Radioactive 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 Annual Radioactive Effluent Release Report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. 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. Historical average meteorological conditions shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the ODCM.

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

The Annual Radioactive 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 Major Changes to Liquid and Gaseous Radioactive Waste Treatment Systems

Control 7.3 Licensee initiated major changes to the liquid and gaseous radioactive waste systems:

- a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:
 - 1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10CFR Part 50.59,
 - 2. Sufficient detailed information to support the reason for the change without benefit of additional or supplemental information,
 - 3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems,
 - 4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents 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.

8.0 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 59-569.
- 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."
- j. Yankee Decommissioning Quality Assurance Program (YDQAP), Yankee Atomic Electric Company.
- k. Issuance of NPDES Permit No. MA0004367, Letter to J. A. Kay from R. Janson, U.S. EPA, dated July 29, 2003

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,

A handwritten signature in cursive script that reads "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

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Approved By:

A handwritten signature in cursive script that reads "Mark Paterson".

Mr. George Papanic, Jr.

Yankee Rowe

cc:

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Approved By: 

SAFETY ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-029

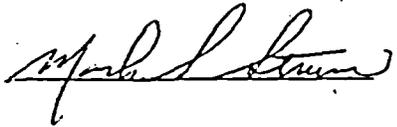
1.0 INTRODUCTION

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.

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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.

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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 septic 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:

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

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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

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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.

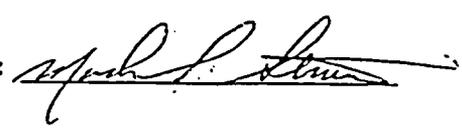
For periods in which disposal of septage occurs under this application, the licensee shall report, to the Nuclear Regulatory Commission (NRC) in the plant's Semiannual Effluent Release Report, the volume, liquid, and solid mass fractions, radionuclide concentrations in the liquid and solid fractions,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

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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.

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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):

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

The licensee then performed a similar calculation using a conservative 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:

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<u>Pathway</u>	<u>Maximally Exposed Individual/Whole Body (mrem/year)</u>
Ground Irradiation	0.980
inhalation	0.0004
Stored Vegetables	0.13
Leafy Vegetables	<u>0.007</u>
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.

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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

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Approved By: 

YANKEE ATOMIC ELECTRIC COMPANY

Telephone (508) 779-6711
TWX 710-380-7619



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

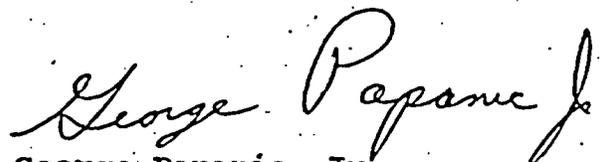
Dear Sirs:

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.
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 7 - Date: MAY 21 1990 A-16 Approved By: *Mich S. Stram*

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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.

-1-

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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



3.1 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:

	<u>Composite Sample East End Manhole Location</u>	<u>Composite Sample West End Manhole Location</u>
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:

$$\text{Solid fraction} = \text{Wt. solid} / (\text{Wt. solid} + \text{Wt. liquid})$$

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:

	<u>East End Solids Sample (pCi/kg) Wet</u>	<u>West End Solids Sample (pCi/kg) Wet</u>
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:

<u>Isotope</u>	<u>Half-Life</u>	<u>Qe (Ci)</u>
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 $1.2\text{E}-04$ 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 ($20 \mu\text{Ci}$), 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_0 is the amount of radioactivity per tank full of septage for a give nuclide, then the total accumulated radioactivity $Q_e(\text{max})$ disposed of after 30 pumpouts is given by:

$$Q_e(\text{max}) = Q_0 (1 + E + E^2 + E^3 + E^4 + \dots + E^{29})$$

$$= Q_0 (1 - E^{29}) / (1 - E) \quad (\text{A})$$

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:

<u>Nuclide</u>	<u>Half Life</u>	<u>λ (1/year)</u>	<u>Q_0 (uCi/batch)</u>	<u>$Q_e(\text{max})$ uCi</u>
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	<u>2.15</u>	<u>46.04</u>
		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 ft² 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 ft², then the land deposited radioactive material S_e (Ci/m²) following landspreading will be equal to:

$$S_e = Q_e \text{ (Ci)} / (500 \text{ ft}^2 * 0.0929 \text{ m}^2/\text{ft}^2) \quad \text{(B)}$$

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:

$$\begin{aligned} (D/Q) &= 1 / (500 \text{ ft}^2 * 0.0929 \text{ m}^2/\text{ft}^2) \\ &= 2.15\text{E}-02 \text{ (m}^{-2}\text{)} \end{aligned} \quad \text{(C)}$$

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/m³) and the surface deposition is:

$$A_e = S_e \text{ (Ci/m}^2\text{)} * K \text{ (1/m)} \quad \text{(D)}$$

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 ft² 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 ft² (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 ft² 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

<u>Individual</u>	<u>Vegetables (kg/yr)</u>	<u>Leafy Veg. (kg/yr)</u>	<u>Milk (liters/yr)</u>	<u>Inhalation* (m³/yr)</u>
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

	<u>Stored Vegetables</u>	<u>Leafy Vegetables</u>
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

	<u>Pasture Feed</u>	<u>Stored Feed</u>
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 accomplished 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/m² 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:

	<u>Radiation Exposure</u>	<u>Individual/Organ</u>
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

<u>Pathway</u>	<u>Maximally Exposed Individual/Organ (Child/Liver) (mrem/year)</u>	<u>Maximally Exposed Individual/Whole Body (Child) (mrem/year)</u>
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*	<u>(0.019)</u>	<u>(0.0036)</u>
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:

<u>Pathway</u>	<u>Maximum Exposed Individual/Whole Body (mrem/year)</u>
Ground Irradiation	0.980
Inhalation	0.0004
Stored Vegetables	0.13
Leafy Vegetables	<u>0.0068</u>
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 dose 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:

<u>Scenario</u>	<u>Maximum Whole Body Annual Dose (mrem)</u>
(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.

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	7.49E-05	1.11E-02	5.30E-04	0.00E+00	1.85E-04
STORED VEGETABLES							
54 MN	0.00E+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)

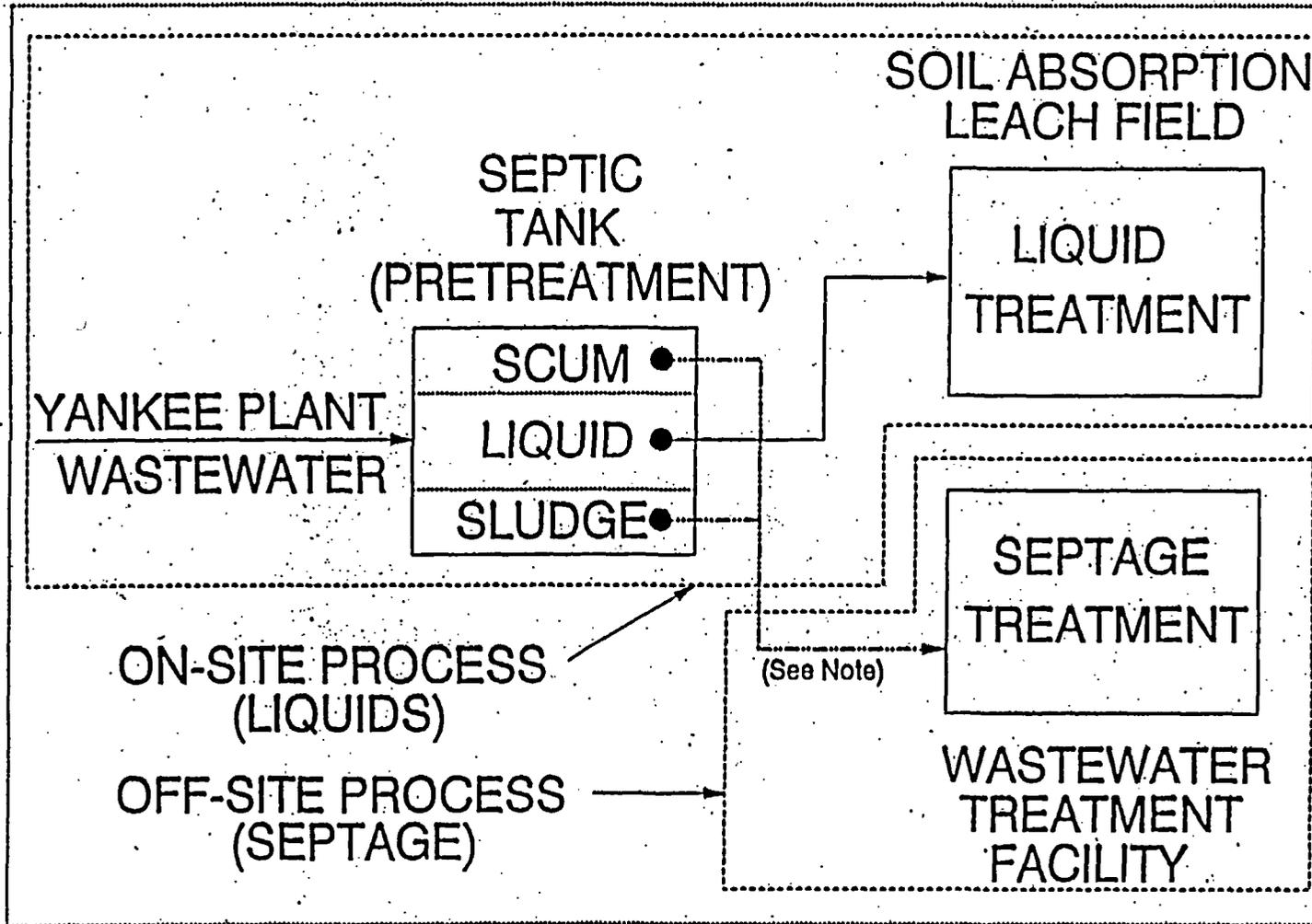
PATHWAY	BONE	LIVER	KIDNEY	LUNG	GI-LLI	THYROID	WHOLE BODY
INHALATION							
54 MN	0.00E+00	3.78E-06	9.41E-07	1.47E-04	4.94E-06	0.00E+00	6.21E-07
60 CO	0.00E+00	2.77E-05	0.00E+00	1.60E-02	4.75E-04	0.00E+00	3.63E-05
134 CS	4.28E-05	9.60E-05	3.19E-05	1.25E-05	8.31E-07	0.00E+00	4.67E-05
137 CS	1.50E-04	1.90E-04	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	2.98E-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						
60 CO	0.00E+00						
134 CS	0.00E+00						
137 CS	0.00E+00						
TOTAL FOR PATHWAY	0.00E+00						
LEAFY VEGETABLES							
54 MN	0.00E+00						
60 CO	0.00E+00						
134 CS	0.00E+00						
137 CS	0.00E+00						
TOTAL FOR PATHWAY	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 B

Concentrations in Air and Water Above Natural Background
(10CFR20.1-20.602, Appendix B)

Revision 11

R12\120

B-1

Appendix B

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

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2— Water ($\mu\text{Ci/ml}$)	Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2— Water ($\mu\text{Ci/ml}$)
Actinium (89)	Ac 227	S	2×10^{-12}	6×10^{-3}	8×10^{-14}	2×10^{-6}
		I	3×10^{-11}	9×10^{-3}	9×10^{-13}	3×10^{-4}
	Ac 228	S	8×10^{-9}	3×10^{-3}	3×10^{-9}	9×10^{-5}
		I	2×10^{-8}	3×10^{-3}	6×10^{-10}	9×10^{-5}
Americium (95)	Am 241	S	6×10^{-12}	1×10^{-4}	2×10^{-12}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-11}	3×10^{-5}
	Am 242m	S	6×10^{-12}	1×10^{-4}	2×10^{-12}	4×10^{-6}
		I	3×10^{-10}	3×10^{-3}	9×10^{-12}	9×10^{-5}
	Am 242	S	4×10^{-4}	4×10^{-3}	1×10^{-3}	1×10^{-4}
		I	5×10^{-4}	4×10^{-3}	2×10^{-3}	1×10^{-4}
	Am 243	S	6×10^{-12}	1×10^{-4}	2×10^{-12}	4×10^{-6}
	I	1×10^{-10}	8×10^{-4}	4×10^{-11}	3×10^{-5}	
	Am 244	S	4×10^{-6}	1×10^{-1}	1×10^{-1}	5×10^{-3}
		I	2×10^{-5}	1×10^{-1}	8×10^{-2}	5×10^{-3}
Antimony	Sb 122	S	2×10^{-7}	8×10^{-4}	6×10^{-9}	3×10^{-5}
		I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-5}
	Sb 124	S	2×10^{-7}	7×10^{-4}	5×10^{-9}	2×10^{-5}
		I	2×10^{-7}	7×10^{-4}	7×10^{-10}	2×10^{-5}
	Sb 125	S	5×10^{-7}	3×10^{-3}	2×10^{-6}	1×10^{-4}
		I	3×10^{-6}	3×10^{-3}	9×10^{-9}	1×10^{-4}
Argon (18)	A 37	Sub ²	6×10^{-3}		1×10^{-4}	
	A 41	Sub	2×10^{-6}		4×10^{-6}	
Arsenic (33)	As 73	S	2×10^{-6}	1×10^{-2}	7×10^{-6}	5×10^{-4}
		I	4×10^{-7}	1×10^{-2}	1×10^{-6}	5×10^{-4}
	As 74	S	3×10^{-7}	2×10^{-2}	1×10^{-6}	5×10^{-5}
		I	1×10^{-7}	2×10^{-2}	4×10^{-9}	5×10^{-5}
	As 76	S	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
		I	1×10^{-7}	6×10^{-4}	3×10^{-9}	2×10^{-5}
	As 77	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
		I	4×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-5}
Astatine (85)	At 211	S	7×10^{-9}	5×10^{-3}	2×10^{-10}	2×10^{-6}
		I	3×10^{-8}	2×10^{-3}	1×10^{-9}	7×10^{-5}
Barium (56)	Ba 131	S	1×10^{-6}	5×10^{-3}	4×10^{-6}	2×10^{-4}
		I	4×10^{-7}	5×10^{-3}	1×10^{-6}	2×10^{-4}
	Ba 140	S	1×10^{-7}	8×10^{-4}	4×10^{-9}	3×10^{-5}
		I	4×10^{-6}	7×10^{-4}	1×10^{-9}	2×10^{-5}
Berkelium (97)	Bk 249	S	9×10^{-10}	2×10^{-3}	3×10^{-11}	6×10^{-4}
		I	1×10^{-7}	2×10^{-3}	4×10^{-9}	6×10^{-4}
	Bk 250	S	1×10^{-7}	6×10^{-3}	5×10^{-9}	2×10^{-4}
		I	1×10^{-6}	6×10^{-3}	4×10^{-6}	2×10^{-4}
Beryllium (4)	Be 7	S	6×10^{-6}	5×10^{-1}	2×10^{-7}	2×10^{-3}
		I	1×10^{-6}	5×10^{-1}	4×10^{-6}	2×10^{-3}
Bismuth (83)	Bi 206	S	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
		I	1×10^{-7}	1×10^{-3}	5×10^{-9}	4×10^{-5}
	Bi 207	S	2×10^{-7}	2×10^{-3}	6×10^{-9}	6×10^{-5}
		I	1×10^{-6}	2×10^{-3}	5×10^{-10}	6×10^{-5}
	Bi 210	S	6×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-5}

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

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope		Table I		Table II	
			Col. 1—Air ($\mu\text{Ci}/\text{ml}$)	Col. 2— Water ($\mu\text{Ci}/\text{ml}$)	Col. 1—Air ($\mu\text{Ci}/\text{ml}$)	Col. 2— Water ($\mu\text{Ci}/\text{ml}$)
Bismuth (83)	Bi 212	I	6×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-3}
		S	1×10^{-7}	1×10^{-2}	3×10^{-9}	4×10^{-4}
Bromine (35)	Br 82	I	2×10^{-7}	1×10^{-2}	7×10^{-9}	4×10^{-4}
		S	1×10^{-6}	8×10^{-3}	4×10^{-6}	3×10^{-4}
Cadmium (48)	Cd 109	I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-3}
		S	5×10^{-6}	5×10^{-3}	2×10^{-9}	2×10^{-4}
	Cd 115m	I	7×10^{-6}	5×10^{-3}	3×10^{-9}	2×10^{-4}
		S	4×10^{-6}	7×10^{-4}	1×10^{-9}	3×10^{-5}
Calcium (20)	Cd 115	I	4×10^{-6}	7×10^{-4}	1×10^{-9}	3×10^{-5}
		S	2×10^{-7}	1×10^{-3}	8×10^{-9}	3×10^{-5}
	Ca 45	I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-5}
		S	3×10^{-6}	3×10^{-4}	1×10^{-9}	9×10^{-6}
Californium (98)	Ca 47	I	1×10^{-7}	5×10^{-3}	4×10^{-9}	2×10^{-4}
		S	2×10^{-7}	1×10^{-3}	6×10^{-9}	5×10^{-5}
	Cf 249	I	2×10^{-12}	1×10^{-3}	6×10^{-9}	3×10^{-5}
		S	2×10^{-12}	1×10^{-4}	5×10^{-11}	4×10^{-6}
Cesium (55)	Cf 250	I	1×10^{-10}	7×10^{-4}	3×10^{-11}	2×10^{-5}
		S	5×10^{-12}	4×10^{-4}	2×10^{-11}	1×10^{-5}
	Cf 251	I	1×10^{-10}	7×10^{-4}	3×10^{-11}	3×10^{-5}
		S	2×10^{-11}	1×10^{-4}	6×10^{-11}	4×10^{-6}
	Cf 252	I	1×10^{-10}	8×10^{-4}	3×10^{-11}	3×10^{-5}
		S	6×10^{-11}	2×10^{-4}	2×10^{-11}	7×10^{-6}
	Cf 253	I	3×10^{-11}	2×10^{-4}	1×10^{-11}	7×10^{-6}
		S	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
Carbon (6)	Cf 254	I	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
		S	5×10^{-12}	4×10^{-4}	2×10^{-11}	1×10^{-5}
Cerium (58)	C 14	I	4×10^{-6}	4×10^{-4}	2×10^{-11}	1×10^{-5}
	(CO ₂)	Sub	5×10^{-8}	2×10^{-2}	1×10^{-7}	8×10^{-4}
Cesium (55)	Ce 141	I	4×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
		S	2×10^{-7}	3×10^{-3}	5×10^{-9}	9×10^{-5}
	Ce 143	I	3×10^{-7}	1×10^{-3}	9×10^{-9}	4×10^{-5}
		S	2×10^{-7}	1×10^{-3}	7×10^{-9}	4×10^{-5}
Chlorine (17)	Ce 144	I	1×10^{-6}	3×10^{-4}	3×10^{-10}	1×10^{-5}
		S	6×10^{-9}	3×10^{-4}	2×10^{-10}	1×10^{-5}
	Cs 131	I	1×10^{-5}	7×10^{-2}	4×10^{-7}	2×10^{-3}
		S	3×10^{-6}	3×10^{-2}	1×10^{-7}	9×10^{-4}
	Cs 134m	I	4×10^{-5}	2×10^{-2}	1×10^{-6}	6×10^{-3}
		S	6×10^{-6}	3×10^{-2}	2×10^{-7}	1×10^{-3}
	Cs 134	I	4×10^{-5}	3×10^{-2}	1×10^{-6}	9×10^{-4}
		S	1×10^{-6}	1×10^{-3}	4×10^{-10}	4×10^{-5}
	Cs 135	I	5×10^{-7}	3×10^{-3}	2×10^{-9}	1×10^{-4}
		S	9×10^{-8}	7×10^{-3}	3×10^{-9}	2×10^{-4}
Chromium (24)	Cs 136	I	4×10^{-7}	2×10^{-3}	1×10^{-6}	9×10^{-5}
		S	2×10^{-7}	2×10^{-3}	6×10^{-9}	6×10^{-5}
	Cs 137	I	6×10^{-6}	4×10^{-4}	2×10^{-9}	2×10^{-5}
		S	1×10^{-6}	1×10^{-3}	5×10^{-10}	4×10^{-5}
Cobalt (27)	Cl 36	I	4×10^{-7}	2×10^{-3}	1×10^{-6}	8×10^{-5}
		S	2×10^{-6}	2×10^{-3}	8×10^{-10}	6×10^{-5}
Copper (29)	Cl 38	I	3×10^{-6}	1×10^{-3}	9×10^{-6}	4×10^{-4}
		S	2×10^{-6}	1×10^{-2}	7×10^{-6}	4×10^{-4}
	Cr 51	I	1×10^{-5}	5×10^{-2}	4×10^{-7}	2×10^{-3}
		S	2×10^{-5}	5×10^{-2}	8×10^{-6}	2×10^{-3}
Curium (96)	Co 57	I	3×10^{-6}	2×10^{-2}	1×10^{-7}	5×10^{-4}
		S	2×10^{-7}	1×10^{-2}	6×10^{-9}	4×10^{-4}
	Co 58m	I	2×10^{-5}	8×10^{-2}	6×10^{-7}	3×10^{-3}
		S	9×10^{-6}	6×10^{-2}	3×10^{-7}	2×10^{-3}
	Co 58	I	8×10^{-7}	4×10^{-3}	3×10^{-6}	1×10^{-4}
		S	5×10^{-6}	3×10^{-3}	2×10^{-9}	9×10^{-5}
Copper (29)	Co 60	I	3×10^{-7}	1×10^{-3}	1×10^{-6}	5×10^{-5}
		S	9×10^{-8}	1×10^{-3}	3×10^{-10}	3×10^{-5}
Copper (29)	Cu 64	I	2×10^{-6}	1×10^{-3}	7×10^{-6}	3×10^{-4}
		S	1×10^{-6}	6×10^{-3}	4×10^{-6}	2×10^{-4}
Curium (96)	Cm 242	I	1×10^{-10}	7×10^{-4}	4×10^{-12}	2×10^{-5}
		S				

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

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ^a		Table I		Table II	
			Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2—Water ($\mu\text{Ci/ml}$)	Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2—Water ($\mu\text{Ci/ml}$)
Curium (96)	Cm 243	I	2×10^{-10}	7×10^{-4}	6×10^{-12}	2×10^{-5}
		S	6×10^{-12}	1×10^{-4}	2×10^{-12}	5×10^{-6}
	Cm 244	I	1×10^{-10}	7×10^{-4}	3×10^{-12}	2×10^{-5}
		S	9×10^{-12}	2×10^{-4}	3×10^{-12}	7×10^{-6}
	Cm 245	I	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
		S	5×10^{-12}	1×10^{-4}	2×10^{-12}	4×10^{-6}
	Cm 246	I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
		S	5×10^{-12}	1×10^{-4}	2×10^{-12}	4×10^{-6}
	Cm 247	I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
		S	5×10^{-12}	1×10^{-4}	2×10^{-12}	4×10^{-6}
	Cm 248	I	1×10^{-10}	6×10^{-4}	4×10^{-12}	2×10^{-5}
		S	6×10^{-12}	1×10^{-3}	2×10^{-12}	4×10^{-6}
Cm 249	I	1×10^{-11}	4×10^{-3}	4×10^{-12}	1×10^{-4}	
	S	1×10^{-9}	6×10^{-2}	4×10^{-7}	2×10^{-3}	
Dysprosium (66)	Dy 165	I	1×10^{-9}	6×10^{-2}	4×10^{-7}	2×10^{-3}
		S	3×10^{-6}	1×10^{-2}	9×10^{-6}	4×10^{-4}
	Dy 166	I	2×10^{-6}	1×10^{-2}	7×10^{-6}	4×10^{-4}
		S	2×10^{-7}	1×10^{-3}	8×10^{-6}	4×10^{-5}
Einsteinium (99)	Es 253	I	2×10^{-7}	1×10^{-3}	7×10^{-6}	4×10^{-5}
		S	8×10^{-10}	7×10^{-4}	3×10^{-11}	2×10^{-5}
	Es 254m	I	6×10^{-10}	7×10^{-4}	2×10^{-11}	2×10^{-5}
		S	5×10^{-9}	5×10^{-4}	2×10^{-10}	2×10^{-5}
	Es 254	I	6×10^{-9}	5×10^{-4}	2×10^{-10}	2×10^{-5}
S		2×10^{-11}	4×10^{-4}	6×10^{-12}	1×10^{-5}	
Erbium (68)	Er 169	I	1×10^{-10}	4×10^{-4}	4×10^{-12}	1×10^{-5}
		S	5×10^{-10}	8×10^{-4}	2×10^{-11}	3×10^{-5}
	Er 171	I	4×10^{-10}	8×10^{-4}	1×10^{-11}	3×10^{-5}
		S	6×10^{-7}	3×10^{-3}	2×10^{-6}	9×10^{-5}
		Er 171	I	4×10^{-7}	3×10^{-3}	1×10^{-6}
S			7×10^{-7}	3×10^{-3}	2×10^{-6}	1×10^{-4}
Europium (63)	Eu 152 ($T/2=9.2$ hrs)	I	6×10^{-7}	3×10^{-3}	2×10^{-6}	1×10^{-4}
		S	4×10^{-7}	2×10^{-3}	1×10^{-6}	6×10^{-5}
	Eu 152 ($T/2=13$ yrs)	I	3×10^{-7}	2×10^{-3}	1×10^{-6}	6×10^{-5}
		S	1×10^{-6}	2×10^{-3}	4×10^{-10}	8×10^{-5}
	Eu 154	I	2×10^{-6}	2×10^{-3}	6×10^{-10}	8×10^{-5}
Eu 154	S	4×10^{-9}	6×10^{-4}	1×10^{-10}	2×10^{-5}	
Fermium (100)	Fm 254	I	7×10^{-9}	6×10^{-4}	2×10^{-10}	2×10^{-5}
		S	9×10^{-8}	6×10^{-3}	3×10^{-9}	2×10^{-4}
	Fm 255	I	7×10^{-8}	6×10^{-3}	3×10^{-9}	2×10^{-4}
		S	6×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
	Fm 256	I	7×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
Fm 256	S	2×10^{-8}	1×10^{-3}	6×10^{-10}	3×10^{-5}	
Fluorine (9)	F 18	I	1×10^{-8}	1×10^{-3}	4×10^{-10}	3×10^{-5}
		S	2×10^{-9}	3×10^{-3}	6×10^{-11}	9×10^{-7}
Gadolinium (64)	Gd 153	I	5×10^{-6}	2×10^{-3}	2×10^{-7}	8×10^{-4}
		S	3×10^{-6}	1×10^{-3}	9×10^{-6}	5×10^{-4}
	Gd 159	I	2×10^{-7}	6×10^{-3}	8×10^{-9}	2×10^{-4}
Gallium (31)	Ga 72	I	9×10^{-6}	6×10^{-3}	3×10^{-9}	2×10^{-4}
		S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
Germanium (32)	Ge 71	I	4×10^{-7}	2×10^{-3}	1×10^{-6}	6×10^{-5}
		S	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-5}
Gold (79)	Au 196	I	1×10^{-6}	5×10^{-3}	6×10^{-9}	4×10^{-5}
		S	6×10^{-6}	5×10^{-2}	4×10^{-7}	2×10^{-3}
	Au 198	I	6×10^{-6}	5×10^{-2}	2×10^{-7}	2×10^{-3}
Hafnium (72)	Au 199	I	1×10^{-6}	5×10^{-3}	4×10^{-8}	2×10^{-4}
		S	8×10^{-7}	4×10^{-3}	3×10^{-8}	2×10^{-4}
	Hf 181	I	3×10^{-7}	2×10^{-3}	1×10^{-8}	7×10^{-5}
Holmium (67)	Ho 166	I	2×10^{-7}	1×10^{-3}	8×10^{-9}	5×10^{-5}
		S	7×10^{-8}	2×10^{-3}	3×10^{-9}	7×10^{-5}

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

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope *		Table I		Table II	
			Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2— Water ($\mu\text{Ci/ml}$)	Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2— Water ($\mu\text{Ci/ml}$)
Hydrogen (1)	H3	I	2×10^{-3}	9×10^{-4}	6×10^{-3}	3×10^{-3}
		S	5×10^{-4}	1×10^{-4}	2×10^{-3}	3×10^{-3}
Indium (49)	In 113m	I	5×10^{-6}	1×10^{-6}	2×10^{-5}	3×10^{-5}
		Sub	2×10^{-3}		4×10^{-3}	
		S	8×10^{-6}	4×10^{-6}	3×10^{-5}	1×10^{-5}
		I	7×10^{-6}	4×10^{-6}	2×10^{-5}	1×10^{-5}
		S	1×10^{-5}	5×10^{-6}	4×10^{-5}	2×10^{-5}
Indium (49)	In 114m	I	2×10^{-6}	5×10^{-6}	7×10^{-6}	2×10^{-5}
		S	2×10^{-6}	1×10^{-6}	8×10^{-6}	4×10^{-5}
		I	2×10^{-6}	1×10^{-6}	6×10^{-6}	4×10^{-5}
		S	2×10^{-5}	3×10^{-5}	9×10^{-5}	9×10^{-5}
		I	3×10^{-6}	3×10^{-6}	1×10^{-5}	9×10^{-5}
Iodine (53)	I 125	S	5×10^{-6}	4×10^{-6}	8×10^{-6}	2×10^{-5}
		I	2×10^{-5}	6×10^{-5}	6×10^{-5}	2×10^{-4}
		S	8×10^{-6}	5×10^{-6}	9×10^{-6}	3×10^{-5}
		I	3×10^{-5}	3×10^{-5}	1×10^{-4}	9×10^{-5}
		S	2×10^{-5}	1×10^{-5}	2×10^{-5}	6×10^{-5}
		I	7×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-4}
		S	9×10^{-6}	6×10^{-6}	1×10^{-5}	3×10^{-5}
		I	3×10^{-5}	2×10^{-5}	1×10^{-4}	6×10^{-5}
		S	2×10^{-5}	2×10^{-5}	3×10^{-5}	8×10^{-5}
		I	9×10^{-6}	5×10^{-6}	3×10^{-5}	2×10^{-4}
		S	3×10^{-6}	2×10^{-6}	4×10^{-6}	1×10^{-5}
		I	2×10^{-5}	1×10^{-5}	7×10^{-5}	4×10^{-5}
		S	5×10^{-6}	4×10^{-6}	6×10^{-6}	2×10^{-5}
		I	3×10^{-6}	2×10^{-6}	1×10^{-5}	6×10^{-5}
		S	1×10^{-5}	7×10^{-6}	1×10^{-5}	4×10^{-5}
Iridium (77)	Ir 190	I	4×10^{-5}	2×10^{-5}	1×10^{-4}	7×10^{-5}
		S	1×10^{-6}	6×10^{-6}	4×10^{-6}	2×10^{-5}
		I	4×10^{-5}	5×10^{-5}	1×10^{-4}	2×10^{-4}
		S	1×10^{-7}	1×10^{-7}	4×10^{-6}	4×10^{-5}
Iridium (77)	Ir 192	I	3×10^{-6}	1×10^{-6}	9×10^{-6}	4×10^{-5}
		S	2×10^{-7}	1×10^{-7}	8×10^{-6}	3×10^{-5}
		I	2×10^{-7}	9×10^{-8}	5×10^{-6}	3×10^{-5}
		S	1×10^{-6}	7×10^{-7}	3×10^{-6}	8×10^{-6}
Iron (26)	Fe 55	I	1×10^{-6}	7×10^{-7}	3×10^{-6}	2×10^{-5}
		S	1×10^{-7}	2×10^{-7}	5×10^{-6}	6×10^{-5}
Iron (26)	Fe 59	I	5×10^{-6}	2×10^{-6}	2×10^{-5}	5×10^{-5}
		S	6×10^{-6}	2×10^{-6}	1×10^{-5}	5×10^{-5}
Krypton (36)	Kr 85m	Sub	1×10^{-6}		3×10^{-6}	
		Sub	1×10^{-6}		2×10^{-6}	
		Sub	1×10^{-6}		2×10^{-6}	
		Sub	1×10^{-6}		2×10^{-6}	
Lanthanum (57)	La 140	I	2×10^{-7}	7×10^{-8}	5×10^{-7}	2×10^{-6}
		S	1×10^{-7}	7×10^{-8}	4×10^{-7}	2×10^{-6}
Lead (82)	Pb 203	I	3×10^{-6}	1×10^{-6}	9×10^{-6}	4×10^{-5}
		S	2×10^{-6}	1×10^{-6}	6×10^{-6}	4×10^{-5}
		I	1×10^{-10}	4×10^{-6}	4×10^{-12}	1×10^{-7}
		S	2×10^{-10}	5×10^{-6}	8×10^{-12}	2×10^{-7}
		I	2×10^{-6}	6×10^{-6}	6×10^{-6}	2×10^{-5}
Lutetium (71)	Lu 177	S	6×10^{-7}	3×10^{-7}	2×10^{-6}	1×10^{-4}
		I	5×10^{-7}	3×10^{-7}	2×10^{-6}	1×10^{-4}
Manganese (25)	Mn 52	S	2×10^{-7}	1×10^{-7}	7×10^{-7}	3×10^{-5}
		I	1×10^{-7}	9×10^{-8}	5×10^{-7}	3×10^{-5}
		S	4×10^{-7}	4×10^{-7}	1×10^{-6}	1×10^{-4}
		I	4×10^{-6}	3×10^{-6}	1×10^{-5}	1×10^{-4}
		S	8×10^{-7}	4×10^{-7}	3×10^{-6}	1×10^{-4}
Mercury (80)	Hg 197m	I	5×10^{-7}	3×10^{-7}	2×10^{-6}	1×10^{-4}
		S	7×10^{-7}	6×10^{-7}	3×10^{-6}	2×10^{-4}
		I	6×10^{-7}	5×10^{-7}	3×10^{-6}	2×10^{-4}
		S	1×10^{-6}	9×10^{-7}	4×10^{-6}	3×10^{-4}
		I	3×10^{-6}	1×10^{-6}	9×10^{-6}	5×10^{-4}
Mercury (80)	Hg 203	S	7×10^{-6}	5×10^{-6}	2×10^{-5}	2×10^{-5}
		I	1×10^{-3}	3×10^{-3}	4×10^{-3}	1×10^{-3}
		I	1×10^{-3}	3×10^{-3}	4×10^{-3}	1×10^{-3}

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

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1—Air ($\mu\text{Ci}/\text{m}^3$)	Col. 2— Water ($\mu\text{Ci}/\text{m}^3$)	Col. 1—Air ($\mu\text{Ci}/\text{m}^3$)	Col. 2— Water ($\mu\text{Ci}/\text{m}^3$)
Molybdenum (42)	Mo 99	S	7×10^{-7}	5×10^{-3}	3×10^{-6}	2×10^{-4}
		I	2×10^{-7}	1×10^{-3}	7×10^{-6}	4×10^{-3}
Neodymium (60)	Nd 144	S	8×10^{-11}	2×10^{-3}	3×10^{-12}	7×10^{-3}
		I	3×10^{-10}	2×10^{-3}	1×10^{-11}	8×10^{-3}
	Nd 147	S	4×10^{-7}	2×10^{-3}	1×10^{-6}	6×10^{-3}
		I	2×10^{-7}	2×10^{-3}	8×10^{-6}	6×10^{-3}
Neptunium (93)	Nd 149	S	2×10^{-6}	8×10^{-3}	6×10^{-6}	3×10^{-4}
		I	1×10^{-6}	8×10^{-3}	5×10^{-6}	3×10^{-4}
	Np 237	S	4×10^{-12}	9×10^{-4}	1×10^{-12}	3×10^{-6}
Neptunium (93)		I	1×10^{-10}	9×10^{-4}	4×10^{-12}	3×10^{-3}
	Np 239	S	8×10^{-7}	4×10^{-3}	3×10^{-6}	1×10^{-4}
		I	7×10^{-7}	4×10^{-3}	2×10^{-6}	1×10^{-4}
Nickel (28)	Ni 59	S	5×10^{-7}	6×10^{-3}	2×10^{-6}	2×10^{-4}
		I	8×10^{-7}	6×10^{-3}	3×10^{-6}	2×10^{-3}
	Ni 63	S	6×10^{-6}	8×10^{-4}	2×10^{-6}	3×10^{-3}
		I	3×10^{-7}	2×10^{-3}	1×10^{-6}	7×10^{-4}
Nickel (28)	Ni 65	S	9×10^{-7}	4×10^{-3}	3×10^{-6}	1×10^{-4}
		I	5×10^{-7}	3×10^{-3}	2×10^{-6}	1×10^{-4}
	Nb 93m	S	1×10^{-7}	1×10^{-3}	4×10^{-9}	4×10^{-4}
		I	2×10^{-7}	1×10^{-3}	5×10^{-9}	4×10^{-4}
Niobium (Columbium) (41)	Nb 95	S	5×10^{-7}	3×10^{-3}	2×10^{-6}	1×10^{-4}
		I	1×10^{-7}	3×10^{-3}	3×10^{-9}	1×10^{-4}
	Nb 97	S	6×10^{-6}	3×10^{-3}	2×10^{-7}	9×10^{-4}
		I	5×10^{-6}	3×10^{-3}	2×10^{-7}	9×10^{-4}
Osmium (76)	Os 185	S	5×10^{-7}	2×10^{-3}	2×10^{-6}	7×10^{-3}
		I	5×10^{-6}	2×10^{-3}	2×10^{-9}	7×10^{-3}
	Os 191m	S	2×10^{-5}	7×10^{-3}	6×10^{-7}	3×10^{-3}
		I	9×10^{-6}	7×10^{-3}	3×10^{-7}	2×10^{-3}
	Os 191	S	1×10^{-6}	5×10^{-3}	4×10^{-6}	2×10^{-4}
		I	4×10^{-7}	5×10^{-3}	1×10^{-6}	2×10^{-4}
Osmium (76)	Os 193	S	4×10^{-7}	2×10^{-3}	1×10^{-6}	6×10^{-3}
		I	3×10^{-7}	2×10^{-3}	9×10^{-9}	5×10^{-3}
	Pd 103	S	1×10^{-6}	1×10^{-3}	5×10^{-6}	3×10^{-4}
		I	7×10^{-7}	8×10^{-3}	3×10^{-6}	3×10^{-4}
Palladium (46)	Pd 109	S	6×10^{-7}	3×10^{-3}	2×10^{-6}	9×10^{-3}
		I	4×10^{-7}	2×10^{-3}	1×10^{-6}	7×10^{-3}
Phosphorus (15)	P 32	S	7×10^{-6}	5×10^{-4}	2×10^{-6}	2×10^{-3}
		I	8×10^{-6}	7×10^{-4}	3×10^{-6}	2×10^{-3}
Platinum (78)	Pt 191	S	8×10^{-7}	4×10^{-3}	3×10^{-6}	1×10^{-4}
		I	6×10^{-7}	3×10^{-3}	2×10^{-6}	1×10^{-4}
	Pt 193m	S	7×10^{-6}	3×10^{-3}	2×10^{-7}	1×10^{-3}
		I	5×10^{-6}	3×10^{-3}	2×10^{-7}	1×10^{-3}
	Pt 193	S	1×10^{-6}	3×10^{-3}	4×10^{-6}	9×10^{-4}
		I	3×10^{-7}	5×10^{-3}	1×10^{-6}	2×10^{-3}
	Pt 197m	S	6×10^{-6}	3×10^{-3}	2×10^{-7}	1×10^{-3}
	I	5×10^{-6}	3×10^{-3}	2×10^{-7}	9×10^{-4}	
Plutonium (94)	Pt 197	S	8×10^{-7}	4×10^{-3}	3×10^{-6}	1×10^{-4}
		I	6×10^{-7}	3×10^{-3}	2×10^{-6}	1×10^{-4}
	Pu 238	S	2×10^{-12}	1×10^{-4}	7×10^{-14}	5×10^{-6}
		I	3×10^{-11}	8×10^{-4}	1×10^{-12}	3×10^{-3}
	Pu 239	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	8×10^{-4}	1×10^{-12}	3×10^{-3}
	Pu 240	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	8×10^{-4}	1×10^{-12}	3×10^{-3}
	Pu 241	S	9×10^{-11}	7×10^{-3}	3×10^{-12}	2×10^{-4}
		I	4×10^{-6}	4×10^{-2}	1×10^{-9}	1×10^{-3}
	Pu 242	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	9×10^{-4}	1×10^{-12}	3×10^{-3}
	Pu 243	S	2×10^{-6}	1×10^{-3}	6×10^{-6}	3×10^{-4}
	I	2×10^{-6}	1×10^{-3}	8×10^{-6}	3×10^{-4}	
Pu 244	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	4×10^{-6}	
	I	3×10^{-11}	3×10^{-4}	1×10^{-12}	1×10^{-3}	
Polonium (84)	Po 210	S	5×10^{-10}	2×10^{-3}	2×10^{-11}	7×10^{-3}
		I	2×10^{-10}	8×10^{-4}	7×10^{-12}	3×10^{-3}

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

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2— Water ($\mu\text{Ci/ml}$)	Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2— Water ($\mu\text{Ci/ml}$)
Potassium (19)	K42	S	2×10^{-6}	9×10^{-3}	7×10^{-5}	3×10^{-4}
		I	1×10^{-3}	6×10^{-4}	4×10^{-9}	2×10^{-3}
Praseodymium (59)	Pr 142	S	2×10^{-3}	9×10^{-4}	7×10^{-9}	3×10^{-3}
		I	2×10^{-3}	9×10^{-4}	5×10^{-9}	3×10^{-3}
	Pr 143	S	3×10^{-3}	1×10^{-3}	1×10^{-9}	5×10^{-3}
		I	2×10^{-3}	1×10^{-3}	6×10^{-9}	5×10^{-3}
Promethium (61)	Pm 147	S	6×10^{-4}	6×10^{-3}	2×10^{-9}	2×10^{-4}
		I	1×10^{-3}	6×10^{-3}	3×10^{-9}	2×10^{-4}
	Pm 149	S	3×10^{-3}	1×10^{-3}	1×10^{-9}	4×10^{-3}
		I	2×10^{-3}	1×10^{-3}	8×10^{-9}	4×10^{-3}
Protoactinium (91)	Pa 230	S	2×10^{-9}	7×10^{-3}	6×10^{-11}	2×10^{-4}
		I	8×10^{-10}	7×10^{-3}	3×10^{-11}	2×10^{-4}
		S	1×10^{-12}	3×10^{-3}	4×10^{-14}	9×10^{-7}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	2×10^{-3}
	Pa 231	S	6×10^{-7}	4×10^{-3}	2×10^{-4}	1×10^{-4}
		I	2×10^{-3}	3×10^{-3}	6×10^{-9}	1×10^{-4}
		S	2×10^{-3}	4×10^{-3}	2×10^{-12}	3×10^{-3}
		I	2×10^{-3}	3×10^{-3}	2×10^{-9}	1×10^{-4}
Radium (88)	Ra 223	S	2×10^{-9}	2×10^{-3}	6×10^{-11}	7×10^{-3}
		I	2×10^{-10}	1×10^{-4}	8×10^{-12}	4×10^{-6}
		S	5×10^{-9}	7×10^{-3}	2×10^{-10}	2×10^{-6}
		I	7×10^{-10}	2×10^{-4}	2×10^{-11}	5×10^{-6}
	Ra 224	S	3×10^{-11}	4×10^{-3}	3×10^{-12}	3×10^{-6}
		I	5×10^{-11}	9×10^{-4}	2×10^{-12}	3×10^{-3}
		S	7×10^{-11}	8×10^{-3}	2×10^{-12}	3×10^{-3}
		I	4×10^{-11}	7×10^{-4}	1×10^{-12}	3×10^{-3}
Radon (86)	Rn 220	S	3×10^{-3}		1×10^{-6}	
	Rn 222	S	3×10^{-6}		3×10^{-9}	
Rhenium (75)	Re 183	S	3×10^{-6}	2×10^{-3}	9×10^{-9}	6×10^{-4}
		I	2×10^{-3}	8×10^{-3}	5×10^{-9}	3×10^{-4}
		S	6×10^{-3}	3×10^{-3}	2×10^{-9}	9×10^{-3}
		I	2×10^{-3}	1×10^{-3}	8×10^{-9}	5×10^{-3}
		S	9×10^{-6}	7×10^{-3}	3×10^{-7}	3×10^{-3}
		I	5×10^{-3}	4×10^{-3}	2×10^{-9}	2×10^{-3}
	Re 186	S	4×10^{-3}	2×10^{-3}	1×10^{-6}	6×10^{-3}
		I	2×10^{-3}	9×10^{-4}	6×10^{-9}	3×10^{-3}
		S	8×10^{-3}	4×10^{-3}	3×10^{-6}	1×10^{-3}
		I	6×10^{-3}	3×10^{-4}	2×10^{-6}	1×10^{-3}
Rhodium (45)	Rh 103m	S	8×10^{-3}	4×10^{-3}	3×10^{-6}	1×10^{-3}
		I	6×10^{-3}	3×10^{-4}	2×10^{-6}	1×10^{-3}
	Rh 105	S	8×10^{-3}	4×10^{-3}	3×10^{-6}	1×10^{-4}
		I	5×10^{-3}	3×10^{-3}	2×10^{-6}	1×10^{-4}
Rubidium (37)	Rb 86	S	3×10^{-3}	2×10^{-3}	1×10^{-9}	7×10^{-3}
		I	7×10^{-6}	7×10^{-4}	2×10^{-9}	2×10^{-3}
	Rb 87	S	5×10^{-3}	3×10^{-3}	2×10^{-6}	1×10^{-4}
		I	7×10^{-6}	5×10^{-3}	2×10^{-9}	2×10^{-4}
Ruthenium (44)	Ru 97	S	2×10^{-6}	1×10^{-3}	8×10^{-9}	4×10^{-4}
		I	2×10^{-6}	1×10^{-3}	6×10^{-9}	3×10^{-4}
		S	5×10^{-3}	2×10^{-3}	2×10^{-6}	8×10^{-3}
		I	8×10^{-4}	2×10^{-3}	3×10^{-9}	8×10^{-3}
		S	7×10^{-3}	3×10^{-3}	2×10^{-6}	1×10^{-4}
		I	5×10^{-3}	3×10^{-3}	2×10^{-6}	1×10^{-4}
	Ru 103	S	8×10^{-6}	4×10^{-4}	3×10^{-9}	1×10^{-3}
		I	6×10^{-9}	3×10^{-4}	2×10^{-10}	1×10^{-3}
		S	7×10^{-11}	2×10^{-3}	2×10^{-12}	6×10^{-3}
		I	3×10^{-10}	2×10^{-3}	9×10^{-12}	7×10^{-3}
Samarium (62)	Sm 147	S	6×10^{-8}	1×10^{-2}	2×10^{-9}	4×10^{-4}
		I	1×10^{-7}	1×10^{-2}	5×10^{-9}	4×10^{-4}
		S	5×10^{-7}	2×10^{-3}	2×10^{-6}	8×10^{-3}
Scandium (21)	Sc 46	S	4×10^{-3}	2×10^{-3}	1×10^{-6}	8×10^{-3}
		I	2×10^{-3}	1×10^{-3}	8×10^{-9}	4×10^{-3}
		S	2×10^{-6}	1×10^{-3}	8×10^{-10}	4×10^{-3}
		I	6×10^{-7}	3×10^{-3}	2×10^{-6}	9×10^{-3}
		S	5×10^{-7}	3×10^{-3}	2×10^{-6}	9×10^{-3}
		I	2×10^{-3}	8×10^{-4}	6×10^{-9}	3×10^{-3}
	Sc 48	S	1×10^{-3}	8×10^{-4}	5×10^{-9}	3×10^{-3}
		I	1×10^{-6}	9×10^{-3}	4×10^{-6}	3×10^{-4}
Selenium (34)	Se 75	S	1×10^{-3}	8×10^{-3}	4×10^{-9}	3×10^{-4}

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

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2—Water ($\mu\text{Ci/ml}$)	Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2—Water ($\mu\text{Ci/ml}$)
Silicon (14)	Si 31	S	6×10^{-6}	3×10^{-2}	2×10^{-3}	9×10^{-4}
		I	1×10^{-6}	6×10^{-3}	3×10^{-3}	2×10^{-3}
Silver (47)	Ag 105	S	6×10^{-7}	3×10^{-3}	2×10^{-6}	1×10^{-4}
		I	8×10^{-6}	3×10^{-3}	3×10^{-6}	1×10^{-4}
		S	2×10^{-7}	9×10^{-4}	7×10^{-7}	3×10^{-3}
		I	1×10^{-6}	9×10^{-4}	3×10^{-10}	3×10^{-3}
Sodium (11)	Na 22	S	3×10^{-7}	1×10^{-3}	1×10^{-6}	4×10^{-3}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	4×10^{-3}
		S	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-3}
		I	9×10^{-9}	9×10^{-4}	3×10^{-10}	3×10^{-3}
Strontium (38)	Sr 85m	S	1×10^{-6}	6×10^{-3}	4×10^{-9}	2×10^{-3}
		I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-3}
		S	4×10^{-5}	2×10^{-1}	1×10^{-6}	7×10^{-3}
		I	3×10^{-5}	2×10^{-1}	1×10^{-6}	7×10^{-3}
Sulfur (16)	S 35	S	2×10^{-7}	3×10^{-3}	8×10^{-9}	1×10^{-4}
		I	1×10^{-7}	5×10^{-3}	4×10^{-9}	2×10^{-4}
		S	3×10^{-6}	3×10^{-4}	3×10^{-10}	3×10^{-3}
		I	4×10^{-6}	8×10^{-4}	1×10^{-9}	3×10^{-3}
		S	1×10^{-9}	1×10^{-3}	3×10^{-11}	3×10^{-3}
		I	5×10^{-9}	1×10^{-3}	2×10^{-10}	4×10^{-3}
		S	4×10^{-7}	2×10^{-3}	2×10^{-6}	7×10^{-3}
		I	3×10^{-7}	1×10^{-3}	9×10^{-9}	5×10^{-3}
		S	4×10^{-7}	2×10^{-3}	2×10^{-6}	7×10^{-3}
		I	3×10^{-7}	2×10^{-3}	1×10^{-6}	6×10^{-3}
		S	3×10^{-7}	2×10^{-3}	9×10^{-9}	6×10^{-3}
		I	3×10^{-7}	8×10^{-3}	9×10^{-9}	3×10^{-4}
Tantalum (73)	Ta 182	S	4×10^{-6}	1×10^{-3}	1×10^{-9}	4×10^{-3}
		I	2×10^{-6}	1×10^{-3}	7×10^{-10}	4×10^{-3}
Technetium (43)	Tc 96m	S	8×10^{-5}	4×10^{-1}	3×10^{-6}	1×10^{-3}
		I	3×10^{-5}	3×10^{-1}	1×10^{-6}	1×10^{-3}
		S	6×10^{-7}	3×10^{-3}	2×10^{-9}	1×10^{-4}
		I	2×10^{-7}	1×10^{-3}	8×10^{-9}	5×10^{-3}
		S	2×10^{-6}	1×10^{-3}	8×10^{-4}	4×10^{-4}
		I	2×10^{-7}	5×10^{-3}	5×10^{-9}	2×10^{-4}
		S	1×10^{-5}	5×10^{-3}	4×10^{-7}	2×10^{-3}
		I	3×10^{-7}	2×10^{-3}	1×10^{-8}	8×10^{-4}
		S	4×10^{-5}	2×10^{-1}	1×10^{-6}	6×10^{-3}
		I	1×10^{-5}	8×10^{-2}	5×10^{-7}	3×10^{-3}
		S	2×10^{-6}	1×10^{-2}	7×10^{-6}	3×10^{-4}
		I	6×10^{-6}	5×10^{-3}	2×10^{-9}	2×10^{-4}
Tellurium (52)	Te 125m	S	4×10^{-7}	5×10^{-3}	1×10^{-9}	2×10^{-4}
		I	1×10^{-7}	3×10^{-3}	4×10^{-9}	1×10^{-4}
		S	1×10^{-7}	2×10^{-3}	5×10^{-9}	6×10^{-3}
		I	4×10^{-6}	2×10^{-3}	1×10^{-9}	5×10^{-3}
		S	2×10^{-6}	8×10^{-3}	6×10^{-6}	3×10^{-4}
		I	9×10^{-7}	5×10^{-3}	3×10^{-6}	2×10^{-4}
		S	8×10^{-6}	1×10^{-3}	3×10^{-9}	3×10^{-5}
		I	3×10^{-6}	6×10^{-4}	1×10^{-9}	2×10^{-5}
		S	5×10^{-6}	2×10^{-2}	2×10^{-7}	8×10^{-4}
		I	4×10^{-6}	2×10^{-2}	1×10^{-7}	8×10^{-4}
		S	4×10^{-7}	2×10^{-3}	1×10^{-9}	6×10^{-3}
		I	2×10^{-7}	1×10^{-3}	6×10^{-9}	4×10^{-3}
Terbium (65)	Tb 160	S	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
		I	1×10^{-7}	1×10^{-3}	3×10^{-9}	4×10^{-5}
Thallium (81)	Tl 200	S	3×10^{-4}	1×10^{-3}	1×10^{-9}	4×10^{-5}
		I	1×10^{-4}	7×10^{-3}	4×10^{-6}	2×10^{-4}
		S	2×10^{-4}	9×10^{-3}	7×10^{-6}	3×10^{-4}
		I	9×10^{-7}	5×10^{-3}	3×10^{-6}	2×10^{-4}
		S	8×10^{-7}	4×10^{-3}	3×10^{-6}	1×10^{-4}
		I	2×10^{-7}	2×10^{-3}	8×10^{-9}	7×10^{-5}
Thallium (81)	Tl 204	S	6×10^{-7}	3×10^{-3}	2×10^{-9}	1×10^{-4}
		I	3×10^{-6}	2×10^{-3}	9×10^{-10}	6×10^{-5}

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

[See footnotes at end of Appendix B]

Element (atomic number)	Isotope ¹		Table I		Table II	
			Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2— Water ($\mu\text{Ci/ml}$)	Col. 1—Air ($\mu\text{Ci/ml}$)	Col. 2— Water ($\mu\text{Ci/ml}$)
Thorium (90)	Th 227	S	3×10^{-10}	5×10^{-4}	1×10^{-11}	2×10^{-5}
		I	2×10^{-10}	5×10^{-4}	6×10^{-12}	2×10^{-5}
	Th 228	S	9×10^{-12}	2×10^{-4}	3×10^{-12}	7×10^{-6}
		I	6×10^{-12}	4×10^{-4}	2×10^{-12}	1×10^{-5}
	Th 230	S	2×10^{-12}	5×10^{-5}	8×10^{-14}	2×10^{-6}
		I	1×10^{-11}	9×10^{-4}	3×10^{-12}	3×10^{-5}
	Th 231	S	1×10^{-6}	7×10^{-3}	5×10^{-9}	2×10^{-4}
		I	1×10^{-6}	7×10^{-3}	4×10^{-9}	2×10^{-4}
	Th 232	S	3×10^{-11}	5×10^{-5}	1×10^{-12}	2×10^{-6}
		I	3×10^{-11}	1×10^{-3}	1×10^{-12}	4×10^{-5}
Th natural	S	6×10^{-11}	6×10^{-5}	2×10^{-12}	2×10^{-4}	
	I	6×10^{-11}	6×10^{-4}	2×10^{-12}	2×10^{-5}	
Th 234	S	6×10^{-4}	5×10^{-4}	2×10^{-9}	2×10^{-5}	
	I	3×10^{-6}	5×10^{-4}	1×10^{-9}	2×10^{-5}	
Thulium (69)	Tm 170	S	4×10^{-8}	1×10^{-3}	1×10^{-9}	5×10^{-5}
		I	3×10^{-8}	1×10^{-3}	1×10^{-9}	5×10^{-5}
Tm 171	S	1×10^{-7}	1×10^{-3}	4×10^{-9}	5×10^{-4}	
	I	2×10^{-7}	1×10^{-3}	8×10^{-9}	5×10^{-4}	
Tin (50)	Sn 113	S	4×10^{-3}	2×10^{-3}	1×10^{-6}	9×10^{-5}
		I	5×10^{-4}	2×10^{-3}	2×10^{-9}	8×10^{-5}
Sn 125	S	1×10^{-3}	5×10^{-4}	4×10^{-9}	2×10^{-5}	
	I	8×10^{-4}	5×10^{-4}	3×10^{-9}	2×10^{-5}	
Tungsten (Wolfram) (74)	W 181	S	2×10^{-6}	1×10^{-2}	8×10^{-6}	4×10^{-4}
		I	1×10^{-7}	1×10^{-2}	4×10^{-9}	3×10^{-4}
	W 185	S	8×10^{-7}	4×10^{-3}	3×10^{-9}	1×10^{-4}
		I	1×10^{-7}	3×10^{-3}	4×10^{-9}	1×10^{-4}
W 187	S	4×10^{-7}	2×10^{-3}	2×10^{-9}	7×10^{-5}	
	I	3×10^{-7}	2×10^{-3}	1×10^{-9}	6×10^{-5}	
Uranium (92)	U 230	S	3×10^{-10}	1×10^{-4}	1×10^{-11}	5×10^{-6}
		I	1×10^{-10}	1×10^{-4}	4×10^{-12}	5×10^{-6}
	U 232	S	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
		I	3×10^{-11}	8×10^{-4}	9×10^{-12}	3×10^{-5}
	U 233	S	5×10^{-10}	9×10^{-4}	2×10^{-11}	3×10^{-5}
		I	1×10^{-10}	9×10^{-4}	4×10^{-12}	3×10^{-5}
	U 234	S ⁴	6×10^{-10}	9×10^{-4}	2×10^{-11}	3×10^{-5}
		I	1×10^{-10}	9×10^{-4}	4×10^{-12}	3×10^{-5}
	U 235	S ⁴	5×10^{-10}	8×10^{-4}	2×10^{-11}	3×10^{-5}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	U 236	S	6×10^{-10}	1×10^{-3}	2×10^{-11}	3×10^{-5}
		I	1×10^{-10}	1×10^{-3}	4×10^{-12}	3×10^{-5}
	U 238	S ⁴	7×10^{-11}	1×10^{-3}	3×10^{-12}	4×10^{-5}
		I	1×10^{-10}	1×10^{-3}	5×10^{-12}	4×10^{-5}
	U 240	S	2×10^{-7}	1×10^{-3}	8×10^{-9}	3×10^{-5}
	I	2×10^{-7}	1×10^{-3}	6×10^{-9}	3×10^{-5}	
U-natural	S ⁴	1×10^{-10}	1×10^{-3}	5×10^{-12}	3×10^{-5}	
Vanadium (23)	V 48	S	2×10^{-7}	9×10^{-4}	6×10^{-9}	3×10^{-5}
		I	6×10^{-8}	8×10^{-4}	2×10^{-9}	3×10^{-5}
Xenon (54)	Xe 131m	Sub	2×10^{-5}		4×10^{-7}	
	Xe 133	Sub	1×10^{-5}		3×10^{-7}	
	Xe 133m	Sub	1×10^{-6}		3×10^{-7}	
	Xe 135	Sub	4×10^{-6}		1×10^{-7}	
Ytterbium (70)	Yb 175	S	7×10^{-7}	3×10^{-3}	2×10^{-9}	1×10^{-4}
		I	6×10^{-7}	3×10^{-3}	2×10^{-9}	1×10^{-4}
Yttrium (39)	Y 90	S	1×10^{-7}	6×10^{-4}	4×10^{-9}	2×10^{-5}
		I	1×10^{-7}	6×10^{-4}	3×10^{-9}	2×10^{-5}
	Y 91m	S	2×10^{-5}	1×10^{-1}	8×10^{-7}	3×10^{-3}
		I	2×10^{-5}	1×10^{-1}	6×10^{-7}	3×10^{-3}
	Y 91	S	4×10^{-6}	8×10^{-4}	1×10^{-9}	3×10^{-5}
		I	3×10^{-6}	8×10^{-4}	1×10^{-9}	3×10^{-5}
	Y 92	S	4×10^{-7}	2×10^{-3}	1×10^{-6}	6×10^{-5}
		I	3×10^{-7}	2×10^{-3}	1×10^{-6}	6×10^{-5}
	Y 93	S	2×10^{-7}	8×10^{-4}	6×10^{-9}	3×10^{-5}
	I	1×10^{-7}	8×10^{-4}	5×10^{-9}	3×10^{-5}	

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

[See footnotes at end of Appendix B]

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

¹Soluble (S); Insoluble (I).²"Sub" means that values given are for submersion in a semispherical infinite cloud of airborne material.

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

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

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

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

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

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

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

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

$$(C_A/\text{MPC}_A) + (C_B/\text{MPC}_B) + (C_C/\text{MPC}_C) \leq 1$$

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

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

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

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

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

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

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

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

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

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

$$(C_1/MPC_1 + C_2/MPC_2 + \dots) \leq 1/10$$