VERMONT YANKEE NUCLEAR POWER STATION

OFF-SITE DOSE CALCULATION MANUAL

REVISION 20

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TABLE 1.1-1
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Summary of Radiological Effluent Technical Specifications and Implementing Equations

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

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

(2) Technical Specification 3.8.M.2 requires this evaluation only if twice the limit of Equations 3-1, 3-3, 3-21, 3-23, or 3-25 is reached. If this occurs a Method II calculation shall be made considering available information for pathways of exposure to real individuals from liquid, gaseous, and direct radiation sources.

TABLE 1.1-12

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

	Stack Release		Ground Level Release*	
Radio-nuclide	Critical Organ Dose Factor DFG_{sico} $\left(\frac{mrem}{Ci}\right)$	Critical Organ Dose Rate Factor DFG'_{sico} $\left(\frac{mrem\text{-}sec}{yr\text{-}\mu Ci}\right)$	Critical Organ Dose Factor DFG_{gico} $\left(\frac{mrem}{Ci}\right)$	Critical Organ Dose Rate Factor DFG'_{gico} $\left(\frac{mrem\text{-}sec}{yr\text{-}\mu Ci}\right)$
H-3	3.13E-04	9.87E-03	1.06E-02	3.34E-01
C-14	1.90E-01	5.99E+00	6.43E+00	2.03E+02
Cr-51	6.11E-03	2.11E-01	4.16E-02	1.43E+00
Mn-54	7.01E-01	2.77E+01	4.71E+00	1.84E+02
Fe-55	3.17E-01	1.00E+01	2.05E+00	6.47E+01
Fe-59	6.99E-01	2.32E+01	4.60E+00	1.52E+02
Co-57	2.18E-01	8.23E+00	1.41E+00	5.33E-01
Co-58	3.62E-01	1.30E+01	2.39E+00	8.52E+01
Co-60	7.63E+00	3.41E+02	4.99E+01	2.16E+03
Zn-65	3.71E+00	1.20E+02	2.36E+01	7.63E+02
Se-75	2.41E+00	7.76E+01	1.53E+01	4.92E+02
Sn-113	1.03E+00	3.25E+01	6.58E+00	2.08E+02
Sr-89	1.14E+01	3.60E+02	7.27E+01	2.29E+03
Sr-90	4.31E+02	1.36E+04	2.82E+03	8.89E+04
Zr-95	6.91E-01	2.28E+01	4.51E+00	1.49E+02
Sb-124	1.26E+00	4.23E+01	8.35E+00	2.79E+02
Sb-125	1.25E+00	4.89E+01	8.01E+00	3.13E+02
I-131	7.71E+01	2.43E+03	5.02E+02	1.58E+04
I-133	8.22E-01	2.59E+01	8.30E+00	2.62E+02
Cs-134	1.58E+01	5.27E+02	1.02E+02	3.37E+03
Cs-137	1.63E+01	5.55E+02	1.04E+02	3.53E+03
Ba-140	1.13E-01	3.66E+00	2.18E+00	6.94E+01
Ce-141	1.70E-01	5.42E+00	1.19E+00	3.78E+01
Ce-144	3.85E+00	1.22E+02	2.52E+01	7.98E+02

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

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

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

where:

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

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

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

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

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

F = Stack flow rate (cc/sec).

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

DFB_i = Total body gamma dose factor (see Table 1.1-10).

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

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

where:

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

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

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

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

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

F = Stack flow rate (cc/sec).

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

DF'_{is} = combined skin dose factor (see Table 1.i-10) for stack release.

3.11 Method to Calculate Direct Dose From Plant Operation

Technical Specification 3.8.M.1 restricts the dose to the whole body or any organ to any member of the public from all station sources (including direct radiation from fixed sources on-site) to 25 mrem in a calendar year (except the thyroid, which is limited to 75 mrem).

3.11.1 Turbine Building

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

$$D_d = K_{N16}(L) * E \qquad (3\text{-}27)$$

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

where:

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

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

$K_{N16}(L)$ = The N-16 dose conversion factor for (L) equal to either:
(1) 3.17E-06 for the maximum west site boundary; or
(2) 1.26E-06 for the closest residence ($mrem/MW_e h$).

3.11.2 North Warehouse

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

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

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

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

3.11.4 Total Direct Dose Summary

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

3.11.5 Other Fixed Sources

In addition to the fixed sources noted above (Turbine Building, North Warehouse, and LLW Storage Pad), other identified temporary or fixed sources that are created due to plant operations will be included in the total direct summary of 3.11.4 if the projected annual dose contribution would add any notable addition to the reported total (i.e., $\geq$ 0.1 mrem/yr).

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

4.0 ENVIRONMENTAL MONITORING PROGRAM

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

4.1 Intercomparison Program

All routine radiological analyses for environmental samples are performed at the Yankee Atomic Environmental Laboratory (YAEL). The YAEL participates in several government and commercial intercomparison quality assurance programs (QAPs) that are traceable to the National Institute of Standards and Technology (NIST). These include: Nuclear Energy Institute (NEI)/NIST Measurement Assurance Program for the Nuclear Power Industry; U.S. Environmental Protection Agency's Intercomparison Program for Drinking Water; and Analytics, Inc. (commercial). YAEL also participates in the QAP for environmental media conducted by the Environmental Measurements Laboratory of the U.S. Department of Energy.

4.2 Airborne Pathway Monitoring

The environmental sampling program is designed to achieve several major objectives, including sampling air in predominant up-valley and down-valley wind directions, and sampling air in nearby communities and at a proper control location, while maintaining continuity with two years of preoperational data and 18 years of operational data (as of 1990). The chosen air sampling locations are discussed below.

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

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

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

The down-valley direction is monitored by two stations - at River Station Number 3.3 (AP/CF-11, SSE, 1.9 km) and at Northfield, Massachusetts (AP/CF-14, SSE, 11.3 km). They both reside in the sector with the maximum wind frequency and they bound the down-valley point of calculated maximum ground level D/Q (the second highest overall ground level D/Q for any location in any sector). Station AP/CF-11 is approximately one mile from this point, between it and the plant. Station AP/CF-14 also serves as a community monitor for Northfield, Massachusetts. Both stations have been in operation since the preoperational period.

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

The control air sampler was located at Spofford Lake (AP/CF-21, NNE, 16.1 km) due to its distance from the plant and the low frequency for wind blowing in that direction based on the long-term (five-year) meteorological history. Sectors in the general west to southwest direction, which would otherwise have been preferable due to lower wind frequencies, were not chosen since they approached the region surrounding the Yankee Atomic plant in Rowe, Massachusetts.

An additional air sampler is maintained at the Tyler Hill site (AP/CF-15, WNW, 3.4 km), which is along the western side of the valley in general proximity of historical dairy operations. (The sixth location is not a specific Technical Specification requirement.)

Table 4.1

Radiological Environmental Monitoring Stations[1]

Exposure Pathway and/or Sample		Sample Location and Designated Code[2]		Distance (km)[5]	Direction[5]
1. AIRBORNE (Radioiodine and Particulate)					
		AP/CF-11	River Station No. 3.3	1.9	SSE
		AP/CF-12	N. Hinsdale, NH	3.6	NNW
		AP/CF-13	Hinsdale Substation	3.1	E
		AP/CF-14	Northfield, MA	11.3	SSE
		AP/CF-15	Tyler Hill Road[4]	3.2	WNW
		AP/CF-21	Spofford Lake	16.1	NNE
2. WATERBORNE					
	a. Surface	WR-11	River Station No. 3.3	1.9	Downriver
		WR-21	Rt. 9 Bridge	12.8	Upriver
	b. Ground	WG-11	Plant Well	--	On-Site
		WG-12	Vernon Nursing Well	2.0	SSE
		WG-22	Skibniowsky Well	14.3	N
	c. Sediment From Shoreline	SE-11	Shoreline Downriver	0.8	SSE
		SE-12	North Storm Drain Outfall[3]	0.15	E
3. INGESTION					
	a. Milk[8]	TM-11	Miller Farm	0.8	WNW
		TM-14	Brown Farm	2.1	S
		TM-16	Meadow Crest Farm	4.4	WNW/NW
		TM-18	Blodgett Farm[4]	3.4	SE
		TM-24	County Farm	22.5	N
	b. Mixed Grasses	TG-11	River Station No. 3.3	1.9	SSE
		TG-12	N. Hinsdale, NH	3.6	NNW
		TG-13	Hinsdale Substation	3.1	E
		TG-14	Northfield, MA	11.3	SSE
		TG-15	Tyler Hill Rd.[4]	3.2	WNW
		TG-21	Spofford Lake	16.1	NNE

Table 4.1
(Continued)

Radiological Environmental Monitoring Stations[1]

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

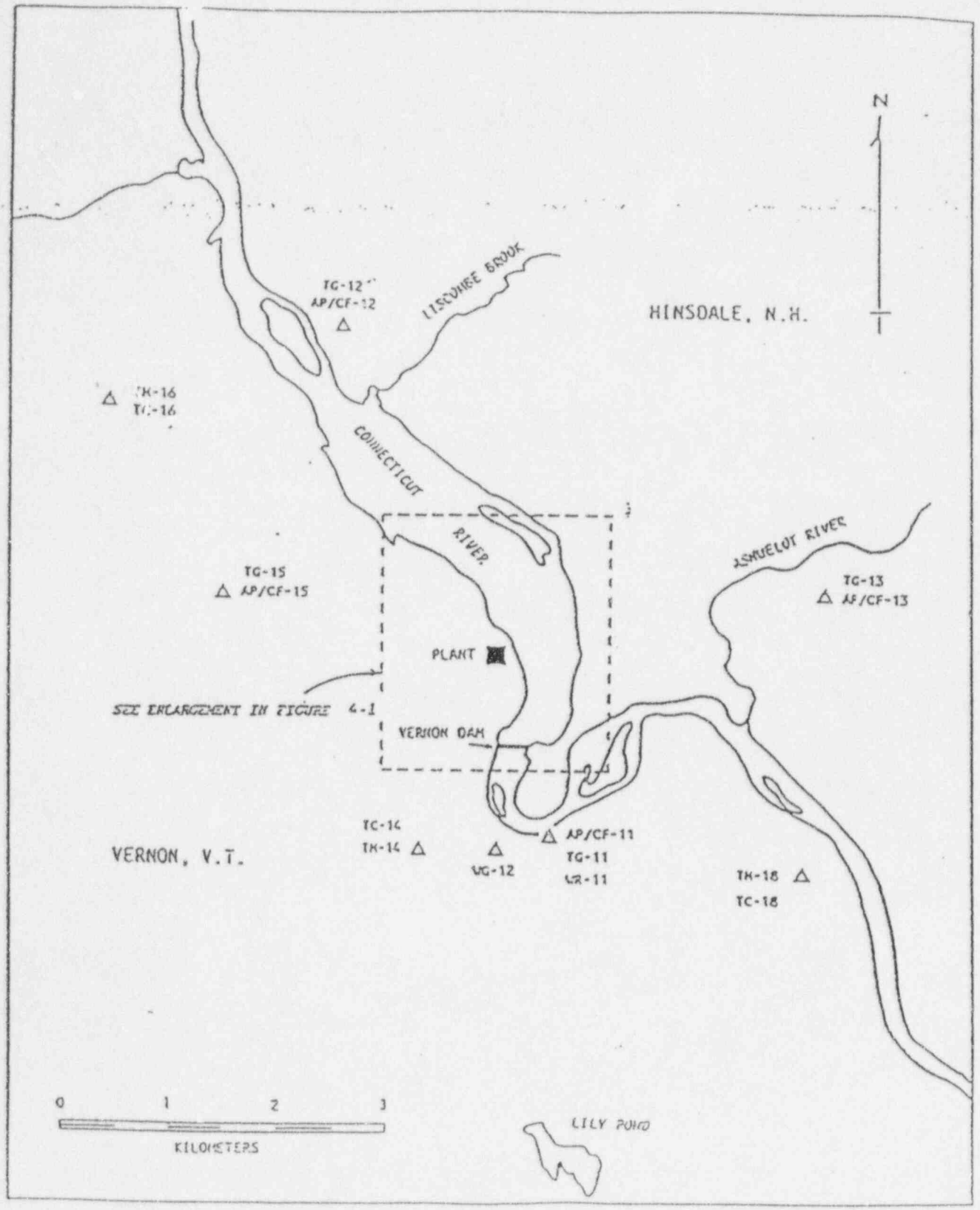

Figure 4-2 Environmental Sampling Locations Within 5 km of Plant

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

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

5.2.1.2 Plant Stack Noble Gas Activity Monitor Setpoint Example

The following setpoint example for the plant stack noble gas activity monitors demonstrates the use of Equations 5-9 and 5-10 for determining setpoints.

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

When monitor responses indicate that activity levels are below the LLDs at the stack (or AOG) monitors, the relative contribution of each noble gas radionuclide can conservatively be approximated by analysis of a sample of off-gas obtained during plant operations at the steam jet air ejector (SJAE). This setpoint example is based on the following data (see Table 1.1-10 for DFB_i and DF'_i):

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

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

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

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

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

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

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

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

$$= 39.348 \text{ cpm}$$

Next:

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

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

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

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

$$R_{skin}^{spt} = 3.000 \ S_g \ \frac{1}{F} \ \frac{1}{DF'_c}$$

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

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

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

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

The setpoints of the plant stack and AOG system noble gas activity monitors must ensure that Technical Specification 3.8.E.1.a is not exceeded. Sections 3.4 and 3.5 show that Equations 3-5 and 3-7 are acceptable methods for determining compliance with that Technical Specification. Which equation (i.e., dose to total body or skin) is more limiting depends on the noble gas mixture. Therefore, each equation must be considered separately. The

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

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

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

where:

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

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

Substituting the right half of Equation 5-14 into Equation 5-13 for C_{mi} yields:

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

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

The detector calibration procedure establishes a counting efficiency for a reference radionuclide, Xe-133 (half life 5.24 days). For routine conditions where offgas is processed through the AOG, all short lived gases are decayed away before discharge leaving only long lived radionuclides as the significant contributors to the monitor response. In this case, Xe-133 as the reference radionuclide for the detector counting efficiency is representative of the expected release conditions. For off normal conditions that might lead to inclusion of short lived radioactivity in the gas stream being released, Xe-133 as the reference radionuclide is expected to lead to a conservative response factor for the detectors since the short lived noble gases tend to have higher energies that can cause them to over respond. Therefore, in Equation 5-15, one may substitute S_g for S_{gi}, where S_g represents the detector counting efficiency determined from the Xe-133 calibration. If necessary, the actual concentration and discharge rate of individual gases being released

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

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

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

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

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

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

Where:

$\dot{R}_{tbs}$	=	total body dose rate (mrem/yr) due to noble gases from stack release
0.61	=	(1.0E+06) x (6.11E-07) (pCi-sec/μCi-m^3)
1E + 06	=	number of pCi per μCi (pCi/μCi)
6.11E - 07	=	$[X/Q]^{\gamma}$, maximum long term average gamma atmospheric dispersion factor (sec/m^3)
$\dot{Q}_i$	=	the release rate of noble gas "i" in the mixture for each noble gas identified (μCi/sec) (Equivalent to $\dot{Q}_i^{ST}$ for noble gases released at the plant stack.)
DFB_i	=	total body dose factor (see Table 1.1-10) (mrem-m^3/pCi-yr)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ONSITE DISPOSAL OF CONTAMINATED SOIL

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

INTRODUCTION

By letters dated November 18, 199', and July 10, 1992, the Vermont Yankee Nuclear Power Corporation (the licensee) requested approval pursuant to Section 20.2002 of Title 10 of the Code of Federal Regulations (CFR) for the disposal of licensed material not previously considered in the Vermont Yankee Final Environmental Statement (FES), dated 1972.

The Vermont Yankee request contains: (a) a detailed description of the licensed material to be disposed of, including the physical and chemical properties important to risk evaluation, and the proposed manner and conditions of waste disposal; (b) an analysis and evaluation of pertinent information on the nature of the environment; (c) the nature and location of other potentially affected licensed and unlicensed facilities; and (d) analyses and procedures to ensure that doses are maintained as-low-as is reasonably achievable and within the dose limits in 10 CFR Part 20.

DESCRIPTION OF WASTE

In 1991, a leak was discovered in a chemistry laboratory drain inside the radiation control area (RCA) that allowed discharge from the chemistry laboratory sink to seep directly into the structural fill soil beneath the floor slab. The fill soil is a 15 foot layer of fine-grained sand with some silt and minor gravel. The area is confined on three sides by existing foundations and on the bottom by bedrock. All of the soil volume under the 150-foot length of buried pipe is contaminated; the total volume is about 58,500 cubic feet. The end of the pipe has been capped and the area of excavation has been backfilled with concrete to the original floor line so that the line is inaccessible.

New piping for the sink has been run above the floor to the collection tank. This new piping is accessible over its full length for periodic inspection to preclude a repeat of this event. The licensee has no way of determining how long the drain has been leaking; in order to bound the potential impacts associated with the leakage, the licensee assumed that the drain line had been leaking for 10 years. Samples of soil from grade to bedrock were obtained

from a split-spoon boring through the floor of the chemistry laboratory. Samples were analyzed for chemical and radionuclide distribution and concentration. Estimated amount of the principal radionuclides bound in the contaminated soil are listed in Table 1. The activity remaining after a 20 year decay period are also presented in the table.

Table 1 Radionuclide Activity and Concentration

Nuclide (half-life) (in years)	* Activity μCi	** Activity μCi
H-3 (12.2)	8.0E+04	2.6E+04
Mn-54 (0.85)	5.4E+01	4.9E-06
Fe-55 (2.7)	4.4E+02	2.6E+00
Co-60 (5.27)	4.1E+02	3.0E+01
Cs-134 (2.06)	3.9E+01	4.8E-02
Cs-127 (30.17)	1.4E+02	8.7E+01
Sr-90 (28.6)	3.2E-01	2.0E-01

* Activity after 10 years of weekly "batch" releases
** That activity after a 20 year decay period.

The chemistry laboratory is located in the lower level of the office building at the north end of the turbine building complex. During plant construction, this area was excavated to bedrock, 15 feet below the chemistry laboratory (El. 233 feet). The area under the laboratory was then filled to its current grade and the concrete laboratory floor was poured. It is impractical to remove this contaminated material because it is located underneath building structures.

PROPOSED DISPOSAL METHOD

The licensee proposed to leave the contaminated soil in place. By terminating the release of liquids into the failed drain line, there is no significant driving force to cause any further movement of the activity now in the soil below the chemistry laboratory floor any deeper toward the groundwater level. The natural groundwater surface appears to be below the bedrock surface beneath the chemistry laboratory. The total quantity now present is sufficiently small that it does not present a direct radiation exposure hazard in the chemistry laboratory. To remove the material would, however, require major excavation under the laboratory floor in proximity to the reactor building foundation and other critical structures, and would directly expose workers performing the excavation to the hazard. The direct exposure, as well

as potential airborne exposures to workers performing remediation, outweigh the risk of leaving the contamination in place, and exceed by far the potential risk to a future population from leaving the contaminated soil where it is. There is no practical way for this material to be removed from the plant at this time.

RADIOLOGICAL IMPACTS

The licensee evaluated the following potential exposure pathways to members of the general public from the radionuclides in the contaminated soil: (1) external exposure caused by farming on the contaminated grounds, (2) internal exposure caused by inhaling of resuspended radionuclides, and (3) internal exposure from ingesting groundwater, and water from onsite potable wells.

Table 2 presents the doses calculated by the licensee for the maximum exposed member of the public from the contaminated soil under the floor of the chemistry laboratory. These doses are based on the radionuclide activities in Table 1. The doses were calculated for an inadvertent intruder for the following pathways: ingestion of food from crops raised on contaminated land, ingestion of milk from cows grazing on the contaminated land, and inhalation of suspended material. It is also assumed that the family and animals raised on the land also drink water from the contaminated land and breathe only air affected by the contaminated area.

Table 2 Intruder Exposures

Pathway	Whole Body (mrem)	Organ (mrem)
Drinking water ingestion	2.5E-05	6.3E-05
Irrigation exposure pathway	1.2E-04	4.0E-04
Well water ingestion	3.8E-01	1.9E-01
Direct ground plane	2.7E-01	0.0E-00
Inhalation (resuspension)	1.1E-01	6.5E-01
Leafy vegetable	2.5E-02	2.4E-01
Cow milk	1.6E-01	1.5E-01

The licensee conservatively calculated these values with the assumption that the total exhumation of the 58,500 cubic feet of radioactive material and spreading in a layer equivalent to the plow depth, results in a continuous annual exposure of less than 1 mrem. This is a small fraction of the 300 mrem received annually by members of the general public in the United States and Canada from sources of natural background radiation.

The guidelines used by the NRC staff for onsite disposal of licensed material and the staff evaluation of how each guideline has been satisfied are given in Table 3.

The staff has reviewed the licensee's calculational methods and assumptions and find that they are consistent with NUREG-1101, "Onsite Disposal of Radioactive Waste," Volumes 1 and 2, November 1986 and February 1987, and Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluent for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I," Revision 1 (October 1977). The staff finds the assessment methodology acceptable.

On this basis, the staff finds the licensee's procedures and amendments acceptable as documented in this safety evaluation. This safety evaluation will be added to the licensee's Offsite Dose Calculation Manual (ODCM). No future modifications are necessary prior to decommissioning of the plant.

The licensee's proposal to dispose of the contaminated soil under the chemistry laboratory (onsite) in a manner described in the Vermont Yankee submittal dated July 10, 1992, is acceptable.

Table 3 Guidelines for Onsite Disposal of Licensed Material

20.2002 GUIDELINE FOR ONSITE DISPOSAL	STAFF'S EVALUATION
The radioactive material should be disposed of in such a manner that it is unlikely that the material would be recycled.	The nature of the disposed material makes it unlikely that it would be recycled to the general public.
Doses to the total body and any body organ of a minimally exposed individual (a member of the general public or a non-occupationally exposed worker) from the probable pathways of exposure to the disposed material should be less than 1 mrem/year.	This guideline is addressed in Table 2.
Doses to the total body and any body organ of an inadvertent intruder from the probable pathways of exposure should be less than 5 mrem/year.	Because the material is insitu, the staff considers the maximally exposed individual scenario to also address the intruder scenario.
Doses to the total body and any body organ of an individual from assumed recycling of the disposed material at the time the disposal site is released from regulatory control from all likely pathways of exposure should be less than 1 mrem.	Even if recycling were to occur after release from regulatory control, the dose to a maximally exposed member of the public is not expected to exceed 1 mrem/year, based on exposure scenarios considered in this analysis.

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Date:

APPENDIX I

RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS

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

Response: There were no licensee-initiated major changes to the radioactive waste systems during this reporting period.