MAINE YANKEE

OFF-SITE DOSE CALCULATION MANUAL

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MAINE YANKEE ATOMIC POWER COMPANY
OFF-SITE DOSE CALCULATION MANUAL

ABSTRACT

The Maine Yankee Nuclear Power Station Off-Site Dose Calculation Manual (MY ODCM) contains the approved methods to estimate the doses and radionuclide concentrations occurring beyond the boundaries of the plant caused by normal plant operation. (The site boundary is shown in Appendix D, SITE BOUNDARY) With initial approval by the U.S. Nuclear Regulatory Commission and the MYNPS Plant Management and approval of subsequent revisions by the Plant Management (as per the Maine Yankee Quality Assurance Program (MYQAP), Appendix D and E, Change 27), this ODCM is suitable to show compliance where referred to by the MYQAP. Sufficient documentation of each method is provided to allow regeneration of the methods with few references to other material. Most of the methods are presented at two levels. The first, Method I, is a linear equation which provides an upper bound and the second, Method II, is an in-depth analysis which can provide more realistic estimates.

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1.0 INTRODUCTION

The purpose of this manual is to provide methods to ensure compliance with the dose requirements of Appendix I to 10 CFR Part 50 (Reference 1). Each method is based on a plant-specific application of the models presented in Regulatory Guide 1.109 (Reference 2).

Methods are included to calculate the doses to individuals from liquid releases from the plant. Under normal operations, experience has shown that the plant will be operated at a small fraction of the dose limits. For this reason, the dose evaluations are presented at different levels of sophistication. The first method being the most conservative, but simplest to use; the second method requiring a full analysis following the guidance presented in Regulatory Guide 1.109 (Reference 2).

The first method, Method I, is based on a critical organ, critical age group, and critical receptor location; as such, it provides a conservative estimate of the doses. If the dose limits are met by application of the first method, no further analysis will be required. If, however, it indicates that the dose limits may be approached or exceeded, a more realistic estimate may be obtained by application of the second method.

The second method, Method II, will calculate the dose to seven organs of four age groups for potentially critical individuals. It is based on measured releases for each nuclide and site-specific parameters. Method II is more accurate, but less conservative than Method I, and will be used to assess doses for the Estimated Dose Report.

Liquid effluent dose calculation methods are presented in Section 3 and are followed by the appropriate Method I dose equations. When necessary, Method II analyses may be performed by applying the site-specific parameters to the appropriate dose equations specified in Regulatory Guide 1.109 (Reference 2). The basis for each of the dose calculation methods is described in Appendix A.

2.0 RELEASE OF RADIOACTIVE EFFLUENTS

2.1 RELEASE OF LIQUID RADIOACTIVE EFFLUENTS

2.1.1 Applicability

The requirements in this section apply at all times to the release of all liquid waste discharged from the plant which may contain radioactive materials.

2.1.2 Objective

The objective is to establish conditions for the release of liquid waste containing radioactive materials and to assure that doses to the public resulting from all such releases are within the limits specified in 10 CFR Part 20, and also assure that the releases from the site of radioactive materials in liquid wastes (above background) are kept "as low as is reasonably achievable" in accordance with 10 CFR Part 50, Appendix I.

2.1.3 Liquid Effluents: Concentration

1. The concentration of radioactive material in liquid effluents released from the site to unrestricted areas shall be limited to not more than ten times the concentrations specified in 10 CFR, Part 20, Appendix B, Table 2, Column 2.

Remedial Action: With the concentration of radioactive material released from the site to unrestricted areas exceeding the above limits, without delay take action to restore the concentration to within the above limits.

Basis: These limitations apply to the concentration of radioactive materials released in the liquid waste effluents from the site to unrestricted areas at the point of discharge into the Back River. Concentration levels specified in 10 CFR 20, Appendix B, Table 2, Column 2 were established to control the dose to the public to within the limits specified in 10 CFR 20.1301 and 20.1302. Those values assure a continuous discharge at those concentrations (8760 hours per year). Pursuant to the requirements of 10 CFR 50.36a to maintain effluents as low as reasonably achievable (ALARA), Appendix I to 10 CFR 50 specifies dose values that are a small percentage of the dose limits in 10 CFR 20.1301. Consistent with Appendix I to 10 CFR 50, to allow operational flexibility, this specification in conjunction with the dose specification in Section 2.1.4.1 permits an instantaneous concentration release rate up to a factor of ten times greater than specified in 10 CFR 20, Appendix B, Table 2, Column 2 while continuing to limit the total annual discharge to a small fraction of the allowable annual dose as specified in Appendix I.

For determining compliance, it must be demonstrated that the concentrations of radionuclides in liquid effluent prior to discharge to the Back River (Reference Section 6.1) meet the limits specified. The release path for liquid effluent allows direct discharge into the Back River through a submerged offshore discharge line.

2.1.4 Liquid Effluents: Dose

1. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released from the site to unrestricted areas shall be limited:

   a. During any calendar quarter to less than or equal to 1.5 mrem to the total body, and to less than or equal to 5 mrem to any organ; and

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

Remedial Action: With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission a report within 30 days from the end of the quarter. The report shall identify the cause(s) for exceeding the limit(s) and define the corrective actions to be taken to reduce the releases and the corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

Remedial Action: With the calculated dose from the release of radioactive materials in liquid effluents exceeding twice the above limits, calculations should be made including direct radiation contributions from significant plant sources to determine whether the limits of 40 CFR 190 (Reference 4) have been exceeded.

If such is the case, prepare and submit a report to the Commission within 30 days. The report shall define the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits and include the schedule for achieving conformance with the limits.

If the release condition resulting in violation of 40 CFR Part 190, has not already been corrected, the report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190.

Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

Basis: These requirements are provided to implement the guidance of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The specification provides the required operating flexibility and, at the same time, assures that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable" as set forth in Section IV.A of Appendix I. In addition, since the facility is located on a saltwater estuary, the release of radioactive waste in liquids will not result in radionuclide concentrations in finished drinking water, which would be in excess of the requirements of 40 CFR Part 190.

The impact of discharging liquid effluent directly to the Back River has been modeled and assessed (Reference 11). The dilution/mixing model assumes that the discharge point is at least 20 feet off the low tide shoreline, has 20 feet of water over the end point at mean low tide, and has a maximum waste stream discharge flow rate of 300 gpm. These conditions assure that dose analyses of future releases maintain the conservatism originally included in prior ODCM dose models. It also maintains the original restriction on the size (approximately 13 acres) of the near-field mixing zone. The model does not allow the edge of the 10:1 dilution isopleth (which is equivalent to the original near field mixing credit) to reach the shoreline where members of the public could have access. The rationale behind this is to limit the shoreline exposure to gamma emitters that might be present in the liquid waste. If liquid effluent contains only low concentrations of tritium (beta emitter only), then the shoreline exposure limitation does not apply.

The dose calculations performed in accordance with the methods and parameters in this ODCM implement the guidance 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 remedial action requiring calculations when releases exceed two times the design objectives is included to assure that appropriate reports and requests for variance are made should effluents exceed the limits set forth in 40 CFR Part 190.

2.1.5 Liquid Radwaste Treatment

The Liquid Radwaste Treatment System shall be used in its designed modes of operation to reduce the radioactive materials in the liquid waste prior to its discharge when the estimated doses due to the liquid effluent from the site, when averaged with all other liquid releases over the last 31 days, would exceed 0.06 mrem to the total body, or 0.2 mrem to any organ.

Remedial Action: With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission a report with the next Annual Radioactive Effluent Release Report which includes the following information:

a. Explanation of why liquid waste was being discharged without treatment and in excess of the above limits, identification of any inoperable liquid waste equipment which prevented treatment prior to discharge, and the reason for the inoperability;

b. Actions taken to restore the inoperable equipment back to operable status; and

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

Basis: The requirement that the appropriate portions of the Liquid Radwaste System (as indicated in this ODCM) be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a and the design objective guidance given in Section II.D of Appendix I to 10 CFR Part 50.

The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

2.2 RELEASE OF GASEOUS RADIOACTIVE WASTE – Section Deleted

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2.2.2 Deleted

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2.2.6 Deleted

2.3 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

2.3.1 Applicability

The requirements in this section apply at all times to Radioactive Effluent Monitoring Systems which perform a surveillance, protective, or controlling function on the release of radioactive effluents from the plant.

2.3.2 Objective

The objective is to assure the operability of the Radioactive Effluent Monitoring Systems to perform their design functions.

2.3.3 Radioactive Liquid Effluent Instrumentation

The radioactive liquid effluent monitoring instrumentation channels shown in Table 2.1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Section 2.1.3.1 are not exceeded during periods of release of radioactive material through the pathway monitored. The liquid effluent monitoring system is not required for liquid discharges with pre-dilution concentrations that are less than those limited by 10CFR20 Table 2, Column 2.

The alarm/trip setpoints of these channels shall be determined in accordance with the methodology in this ODCM.

Remedial Action: With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits in Section 2.1.3.1 are met, without delay:

a. Take action to suspend the release of radioactive liquid effluents monitored by the affected channel, or

b. Declare the channel inoperable, or change the setpoint so it is acceptably conservative.

Remedial Action: With less than the minimum number of radioactive effluent monitoring instrumentation channels operable, take action shown in Table 2.1. Exert reasonable efforts to:

a. Return the instrument(s) to operable status within 30 days;and

b. If unsuccessful, explain in the next Annual Radioactive Effluent Release Report the reason for the delay in correcting the inoperability.

Basis: 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 are to [ensure that the alarm/trip will occur prior to exceeding 10 times the limits of 10 CFR [Part 20 Table 2, Column 2. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

[2.3.4 Deleted

2.3.5 Liquid Effluent Instrumentation Surveillance Requirements

Instrument Operation and Source Checks:

a. Daily*Check: Internal test signals used to check instrument operation. The Liquid Waste Effluent Monitor performs a self-diagnostic check.

b. Quarterly* Functional Test: Expose the detector with either an internal or an external radiation source.

c. 18-Month Calibration: Exposure to known radiation source.

*When required to be operable

TABLE 2.1

Radioactive Liquid Effluent Monitoring Instrumentation

	Instrument	Minimum Channels Operable	Remedial Action
	1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
	a. Liquid Radwaste Effluent Line	(1)	1
[	2. Flow Rate Measurement Devices [(a)]		
[	a. Liquid Radwaste Effluent Line [(a)]	(1)	2
[	b. Dilution Flow or Total Flow [(a)]	(1)	2

Table Notation

ACTION 1 With the number of channels operable less than required by the minimum channels operable requirement, effluent releases may continue provided that prior to initiating or continuing a release:

1. At least two independent samples are analyzed in accordance with Section 2.5, Table 2.6, and
2. At least two technically qualified members of the facility staff independently verify the release rate calculations, and
3. At least two technically qualified members of the facility staff independently verify the discharge valving.

Otherwise, suspend release of radioactive effluents via this pathway.

[ACTION 2 With the number of channels operable less than required by the minimum
[channels operable requirement, effluent releases may continue provided that
[flow is estimated by other means at least once per hour during the release. Other
[means may include but are not limited to insitu generated pump curves, or
[volume/time measurements.

[(a) Instrumentation required only if dilution flow is used to ensure discharge limit compliance

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TABLE 2.2

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2.4 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of the radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, <u>AND</u>

3) Participation in a Inter-laboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the FANP Quality Assurance Program for environmental monitoring.

2.4.1 <u>Applicability</u>

This section applies at all times to radiological environmental surveillance and land use census.

2.4.2 <u>Objective</u>

The objective of this section is to verify that plant operations have no significant radiological effect on the environment and that continued operation will not result in radiological effects detrimental to the environment. The program also shall verify that any measurable concentrations of radioactive materials related to plant operations are not significantly higher than expected based on effluent measurements and modeling of the environmental exposure pathways.

2.4.3 <u>Radiological Environmental Monitoring</u>

1. The Radiological Environmental Monitoring Program shall be conducted as specified in Table 2.3 with Lower Limits of Detection (LLDs) as specified in Table 2.4.

2. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 2.3, 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.

3. With the level of radioactivity in an environmental sampling medium at a location specified in Table 2.3 exceeding a reporting level of Table 2.5 when averaged over any calendar quarter, prepare and submit to the Commission with the next Annual Radioactive Effluent Release Report, following receipt of the laboratory analyses, a report which includes an evaluation of any release conditions, environmental factors, or other aspects which caused the limits of Table 2.5 to be exceeded. When more than one of the radionuclides in Table 2.5 are detected in the sampling medium, this report shall be submitted if:

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

   Exception: When radionuclides other than those in Table 2.5 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits in Sections 2.1.4 and 2.2.5. 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.

4. With milk samples no longer available from one or more of the sample locations required by Table 2.3, identify the new location(s) if available, for obtaining replacement samples and add to the Radiological Environmental Monitoring Program within 30 days. The specific location(s) from which samples were no longer available may then be deleted from the Monitoring Program. Identify the cause of the samples no longer being available and identify the new location(s) for obtaining available replacement samples in the next Annual Radiological Environmental Operating Report.

Basis: The radiological environmental monitoring required by this specification 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 individuals resulting from the station operation. This monitoring program 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 measurement and modeling of the environmental exposure pathways. Program changes may be initiated based on operational experience.

A two-zone sample collection network has been established for environmental surveillance. Samples are collected in Zone I at locations in the vicinity of the plant where concentrations of plant effluents may be detectable.

These samples are compared to samples which have been collected simultaneously at locations in Zone II where the concentration of plant effluents is expected to be negligible. The Zone II samples provide a running background which will make it possible to distinguish significant radioactivity introduced into the environment by the operation of the plant from that introduced by weapons testing or other sources.

[[Routine particulate sampling is performed in locations in Zone I when a building is no longer tied into existing ventilation systems, has large permanent openings to the environment, and is subject to active demolition activities. Demolition occurs after source term has been removed from these buildings and decontamination performed as necessary. No significant effluent release is expected during demolition activities. The number and location of sampling points is dependent on the activities performed. Air samplers are placed in or within close proximity of buildings undergoing demolition activities to provide reasonably assessment of the airborne activity that may be generated. Sampling during demolition activities is performed to validate models and assumptions used to bound demolition effluent releases.

The detection capabilities required by Table 2.4 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.

2.4.4 Land Use Census

1. An annual land use census within the distance of five miles shall be conducted to identify the location of the nearest milk animal, the nearest residence, and the nearest garden of 50 m^2.

   In lieu of a garden census, broad leaf vegetation of at least three different kinds may be sampled at or near the site boundary in two different sections.

2. With a land use census identifying a location(s) which yields a calculated dose commitment (via the same exposure pathway) at least twice than at a location from which samples are currently being obtained in accordance with Section 2.4.3.1, identify the new locations in the next Annual Radiological Environmental Operating Report.

   If permission from the owner to collect samples can be obtained and sufficient sample volume is available, then this new location shall be added to the Radiological Environmental Monitoring Program within 30 days. The sampling location having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted at this time.

3. The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

Basis: This specification is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of this census.

The addition of new sampling locations to Section 2.4.3.1 based on the land use census is limited to those locations which yield a dose commitment at least twice the calculated 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 a factor of 2 would not be expected to result in a significant increase in the ability to detect plant effluent-related nuclides. Changes in the location of monitoring locations are not to be done lightly since frequent changes disrupt time series and may make interpretation of data more difficult.

2.4.5 Interlaboratory Comparison Program

Analyses shall be performed on applicable radioactive environmental samples supplied as part of an interlaboratory comparison program which has been approved by NRC, if such a program exists.

If analyses are not performed as required above, a report shall be made in the next Annual Radiological Environmental Operating Report.

Basis: Participation in an NRC-approved interlaboratory comparison program (if one exists) provides quality assurance for the environmental laboratory, similar to programs in place for other environmental monitoring efforts, such as that for water quality.

TABLE 2.3

Radiological Environmental Surveillance Program(1)(2)(3)

Exposure Pathway and/or Sample	Number of Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis(4)
1. Airborne			
a. Demolition Particulate	Dependent†	Continuous operation+ of sampler with sample collection as required by dust loading but at least once weekly.	Particulate sampler. Analyze for gross beta radioactivity at least 24 hours following filer change. Perform gamma isotopic analysis on any filter indicating activity greater than 5 times the yearly mean of the control samples. Perform gamma isotopic analysis on a composite of the samples collected at least once per quarter.
2. Waterborne			
a. Surface (Estuary)	2	Weekly grab samples for a monthly composite sample*.	Gamma isotopic analysis of each monthly sample. Tritium analysis of composite sample at least once per quarter.

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* For the indicator station, grab samples shall be collected on the tide cycle when the direction of river flow is from the point of discharge toward the collection point.

† The number and location of sampling points is dependent on the activities performed. Air samplers are placed in or within close proximity of buildings undergoing demolition activities to provide reasonably assessment of the airborne activity that may be generated.

\+ Continuous operation during periods when a building is no longer tied into existing ventilation systems, has large permanent openings to the environment, and is subject to active demolition activities.

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TABLE 2.3 (Continued)

Radiological Environmental Surveillance Program[1][2][3]

Exposure Pathway and/or Sample	Number of Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis[4]
3. Ingestion			
a. Fish and Invertebrates	2	One sample in season, or semiannually if not seasonal, of each of at least two commercially or recreationally important species.	Gamma isotopic analysis on edible portions.

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(1) Specific sample locations for all media are specified in the Off-Site Dose Calculation Manual and reported in the Annual Radiological Environmental Operating Report.

(2) See Table 2.4 for maximum values for the lower limits of detection.

(3) Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, to seasonal unavailability or to malfunction of sampling equipment. If the latter occurs, every effort shall be made to complete corrective action prior to the end of the next sampling period.

(4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to effluents from the plant.

TABLE 2.4

Detection Capabilities for Environmental Sample Analysis(a)(b)(d)
Lower Limits of Detection

Analysis[e]	Water (pCi/1)	Airborne Particulate or Gas (pCi/m^3)	Fish and Invertegrates (pCi/kg/wet)
Gross Beta	4	0.01	
H-3	2000*		
Mn-54	15		130
[			
Co-58, Co-60	15		130
[			
Zr-Nb-95	15[c]		
[			
Cs-134	15	0.05	130
Cs-137	18	0.06	150
Ba-La-140	15[c,f]		

* If no drinking water pathway exists, a value of 3,000 pCi/l may be used.

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TABLE 2.4 (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% probability and that only a 5% probability exists of falsely concluding that a blank observation represents a "real" signal.

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

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

where:

LLD is the "a priori" lower limit of detection as defined above (as picocuries per unit mass or volume).

4.66 is a constant derived from the K_{alpha} and K_{beta} values for the 95% confidence level.

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

E is the counting efficiency (as counts per disintegration).

V is the sample size (in units of mass or volume).

2.22 is the number of disintegration per minute per picocuries.

Y is the fractional radiochemical yield (when applicable).

λ is the radioactive decay constant for the particular radionuclide.

Δt is the elapsed time between sample collection and analysis.

Typical values of E, V, Y, and Δt can be used in the calculation.

TABLE 2.4 (Continued)

Table Notation

[This equation results in an LLD in terms of picocuries. For the purposes of Section 2.5 (Tables 2.6 and 2.7), where the required LLD is set forth in microcuries, the terms 2.22 in the denominator should be replaced by 2.22E6, which is the number of disintegrations per minute per microcurie.

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).

The analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unavailable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

b. 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.

c. Parent only.

d. If the measured concentration minus the three standard deviation uncertainty is found to exceed the specified LLD, the sample does not have to be analyzed to meet the specified LLD.

e. This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the listed nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 5.9.1.5.

f. The Ba-140 LLD and concentration can be determined by the analysis of its short-lived daughter product, La-140, subsequent to an eight-day period following collection. The calculation shall be predicated on the normal ingrowth equations for a parent-daughter situation and the assumption that any unsupported La-140 in the sample would have decayed to an insignificant amount (at least 3.6% of its original value). The ingrowth equations will assume that the supported La-140 activity at the time of collection is zero.

TABLE 2.5

Reporting Levels for Radioactivity Concentrations
in Environmental Samples

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish and Invertebrates (pCi/kg/wet)
H-3	20,000[a]		
Mn-54	1,000		30,000
Co-58	1,000		30,000
Co-60	300		10,000
Zr-Nb-95[b]	400		
[Cs-134	30	10	1,000
[Cs-137	50	20	2,000
[Ba-La-140[b]	200		

[a] If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

[b] Parent only.

[

2.5 RADIOACTIVE EFFLUENT MONITORING

2.5.1 <u>Applicability</u>

This section applies to monitoring radioactive effluents, both liquid and gaseous.

2.5.2 <u>Objective</u>

The objective of this section is to specify the nature and frequency of radioactive effluent monitoring requirements.

2.5.3 <u>Liquid Effluents: Sampling and Analysis</u>

1. Liquid radioactive waste sampling and activity analysis shall be performed in accordance with Table 2.6.

2. The results of the radioactivity analysis shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Section 2.1.3.1.

3. Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in this ODCM at least once per 31 days.

2.5.4 <u>Liquid Effluents: Instrumentation</u>

Discharge of liquid radioactive effluents shall be continuously monitored with the alarm/trip setpoints of the monitor set in accordance with the methods outlined in the ODCM such that the requirements of Section 2.1.3 are met. The liquid effluent monitor is not required for liquid discharges that have pre-dilution concentrations that are less than those limited by 10CFR20 Table 2, Column 2.

2.5.5 Deleted

2.5.6 Deleted

2.5.7 Deleted

TABLE 2.6

Radioactive Liquid Waste Sampling and Analysis Program

Liquid Release Type	Minimum Sampling Frequency[h]	Analysis Frequency[h]	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a]
A. Batch Waste Release Tanks[d]	PR Each Batch	PR Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
	PR One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	PR Each Batch	M Composite[b]	H-3 Gross Alpha	1×10^{-5} 1×10^{-7}
	PR Each Batch	Q Composite[b]	Sr-89, Sr-90 Fe-55[g]	5×10^{-8} 1×10^{-6}

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TABLE 2.6 (Continued)

Table Notation

a. The Lower Limit of Detection (LLD) is defined in Table Notation a of Table 2.4 of Section 2.4.

b. 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.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected during release and composited in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

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

f. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Co-60, Cs-134, and Cs-137. This list does not mean that only these nuclides are to be considered. Other gamma peaks which are identifiable, together with the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level.

g. If, after a period of two years, the results indicate that Fe-55 is likely to contribute 1% or less of the total dose attributable to this pathway, the licensee may discontinue the analysis.

h. Frequency notations: PR = Prior to Release
D = Daily
W = Weekly
M = Monthly
Q = Quarterly

TABLE 2.7

Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Release Type	Minimum Sampling Frequency[d]	Analysis Frequency[d]	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a]
[A. Deleted				
B. Building Demolition	Notation [e]	Notation [e]	Notation [e]	Notation [e]

Table Notation

[a. Deleted

[b. Deleted

[c. Deleted

[d. Deleted

e. Prior to release of buildings for demolition after isolation from the Gaseous Radwaste Treatment System, contamination levels on all structural surfaces must meet the criteria specified in Reference 14, as follows:

Loose surface contamination
Average: Less than 5000 dpm/100 cm^2 β / γ

Maximum: Less than 20 dpm/100 cm^2 α

Fixed contamination
Average: Less than 500,000 dpm/100 cm^2 β / γ

[Average: Less than 100 dpm/100 cm^2 α

Loose surface contamination levels shall be counted on an instrument having a minimum detectable activity (MDA) of less than 100 dpm/100 cm^2 β and 8 dpm/100 cm^2 α. Alpha measurements are only required in plant areas of known or suspected alpha contamination or when the β / γ to α ratio is less than 5,000:1. Actual values of all results above MDA will be recorded. The average shall be established by taking the mean of all the samples in a given building, or any subdivision thereof, with the MDA value used for all samples that are less than MDA.

TABLE 2.8

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3.0 LIQUID EFFLUENT DOSE CALCULATIONS

3.1 LIQUID EFFLUENT DOSE TO AN INDIVIDUAL

Section 2.1.4.1 limits the dose or dose commitment to a member of the public from radioactive materials in liquid effluents released from the site to Back River:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body, and to less than or equal to 5 mrem to any organ; and

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

3.1.1.a Dose to the Total Body (Method I)

The total body dose, D_{tb}, in mrem for a liquid release is:

$$D_{tb} = 110 \sum_i Q_i \, DFL_{itb} \qquad (3\text{-}1)$$

where:

Q_i is the total activity released for radionuclide i, in Ci (for strontiums use the most recent measurement available).

DFL_{itb} is the site specific Total Body Dose Factor for radionuclide i, in mrem/Ci (see Table 3.1).

[110 is the dilution correction factor generated through effluent discharge modeling and dilution analysis (Reference 11). This factor provides conservative Method I dose estimates for liquid effluent discharge directly to the Back River.

3.1.1.b Dose to the Total Body (Method II)

Method II consists of the models, input data and assumptions (bioaccumulation factors, shore-width factor, dose conversion factors, and transport and buildup times) in Regulatory Guide 1.109, Rev. 1 (Reference 2), except where site-specific data or assumptions have been identified in the ODCM. The general equations (A-3 and A-7) taken from Regulatory Guide 1.109, and used in the derivation of the simplified Method I approach as described in the Bases Section A.1, are also applied to Method II assessments, except that doses calculated to the whole body from radioactive effluents are evaluated for each of the four age groups to determine the maximum whole body dose of an age-dependent individual via all existing exposure pathways. Table A-1 lists the usage factors for Method II calculations. During past periods when the Circulating/Service Water System provided dilution flow of effluent releases from the discharge diffuser to the Back River, the mixing
[ratio for the diffuser's nearfield mixing zone was set at 0.10. Under current
[decommissioning conditions that reflect the removal of the Circulating/Service
[Water Systems, the discharge forebay and the submerged diffuser discharge line,
[the mixing ratio may be reduced in Method II calculations when waste discharge
[flow rates are below 300 gpm. Reference 11, Table 1, demonstrates that a
[previously modeled mixing ratio of 0.024 remains conservative for the discharge
[conditions associated with a submerged, offshore, single point liquid discharge line
[situated at least 20 feet from the low tide shoreline with 20 feet or more of water
[over the end point at mean low tide. 06/03

3.1.2.a Dose to the Critical Organ (Method I)

The critical organ dose, D_{co}, in mrem for a liquid release is:

$$D_{co} = 110 \sum_i Q_i \, DFL_{ico} \tag{3-2}$$

where:

Q_i is the total activity released for radionuclide i, in Ci (for strontiums use the most recent measurement available).

DFL_{ico} is the site specific Critical Organ Dose Factor for radionuclide i, in mrem/Ci (see Table 3.1).

110 is as defined in Section 3.1.1.a.

3.1.2.b Dose to the Critical Organ (Method II)

Method II consists of the models, input data and assumptions (bioaccumulation factors, shore-width factor, dose conversion factors, and transport and buildup times) in Regulatory Guide 1.109, Revision 1 (Reference 2), except where site-specific data or assumptions have been identified in the ODCM. The general equations (A-3 and A-7) taken from Regulatory Guide 1.109, and used in the derivation of the simplified Method I approach as described in the Bases Section A.1, are also applied to Method II assessments, except that doses calculated to critical organs from radioactive effluents are evaluated for each of the four age groups to determine the maximum critical organ of an age-dependent individual via all existing exposure pathways. Table A-1 lists the usage factors for Method II calculations. During past periods when the Circulating/Service Water System provided dilution flow of effluent releases from the discharge diffuser to the Back River, the mixing ratio for the diffuser's nearfield mixing zone was set at 0.10. Under current decommissioning conditions that reflect the removal of the Circulating/Service Water Systems, the discharge forebay and the submerged diffuser discharge line, the mixing ratio may be reduced in Method II calculations when waste discharge flow rates are below 300 gpm. Reference 11, Table 1, demonstrates that a previously modeled mixing ratio of 0.024 remains conservative for the discharge conditions associated with a submerged, offshore, single point liquid discharge line situated at least 20 feet from the low tide shoreline with 20 feet or more of water over the end point at mean low tide.

TABLE 3.1

Maine Yankee Dose Factors for Liquid Releases

Nuclide	Total Body Dose Factor mrem/Ci DFL_{itb}	Critical Organ Dose Factor mrem/Ci DFL_{ico}
H-3	2.96E-07	2.96E-07
Mn-54	4.26E-03	2.55E-02
Fe-55	1.24E-02	7.53E-02
Co-60	4.79E-02	7.80E-02
Sr-90	3.16E-02	1.29E-01
Cs-134	2.79E-02	3.12E-02
Cs-137	2.92E-02	3.41E-02
Ag-110m	7.92E-03	6.26E-01
Sb-125	4.81E-03	6.81E-03
Other - β / γ	7.27E-02	4.02E+00

4.0 GASEOUS EFFLUENT DOSE CALCULATIONS

[4.1 GASEOUS EFFLUENT DOSE RATE – Section Deleted

4.1.1.a Deleted

4.1.1.b Deleted

4.1.2.a Deleted

4.1.2.b Deleted

[4.1.3.a Deleted

[4.1.3.b Deleted

[4.2 GASEOUS EFFLUENT DOSE FROM NOBLE GASES – Section Deleted

[4.2.1.a Deleted

[4.2.1.b Deleted

[4.2.2.a Deleted

[4.2.2.b Deleted

[4.3 GASEOUS EFFLUENT DOSE FROM TRITIUM AND RADIOACTIVE MATERIAL IN PARTICULATE FORM – Section Deleted

[4.3.1.a Deleted

[4.3.1.b Deleted

TABLE 4.1

[Deleted

TABLE 4.2

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TABLE 4.3

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TABLE 4.4

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5.0 ENVIRONMENTAL MONITORING

The Radiological Environmental Monitoring Stations are listed in Table 5.1. The locations of these stations with respect to the Maine Yankee facility are shown on the maps in Figures 5.1 through 5.4.

TABLE 5.1
Radiological Environmental Monitoring Stations[a]

Exposure Pathway and/or Sample	Sample Location and Designated Code[b]	Distance From the Plant (km)	Direction From the Plant
1. (DEMOLITION PARTICULATE)[c]	AP-3X	<.1	Various
2. DIRECT RADIATION (PLANT)	TL-36 Boothbay Harbor Fire Station	12.2	SSE
	TL-37 Bath Fire Station	10.7	WSW
	TL-38 Dresden Substation	20.1	N

Exposure Pathway and/or Sample	Sample Location and Designated Code[b]	Direction From the ISFSI
3. DIRECT RADIATION (ISFSI)	TL-I-01	N
	TL-I-02	NNE
	TL-I-03	NE
	TL-I-04	ENE
	TL-I-05	E
	TL-I-06	ESE
	TL-I-07	SE
	TL-I-08	SSE
	TL-I-09	S
	TL-I-10	SSW
	TL-I-11	SW
	TL-I-12	WSW
	TL-I-13	W
	TL-I-14	WNW
	TL-I-15	NW
	TL-I-16	NNW

TABLE 5.1 (Continued)

Radiological Environmental Monitoring Stations[a]

Exposure Pathway and/or Sample	Sample Location and Designated Code[b]	Distance From the Plant (km)	Direction From the Plant
4. WATERBORNE			
a. Surface (Estuary)	WE-14 Near Boat Dock	0.5	NE
	WE-20 Kennebec River	9.5	WSW
5. INGESTION			
a. Fish and Invertebrates[c]	FH/MU/CA/HA-11 Long Ledge Area	0.9	S
	FH/MU/CA/HA-24	11.1	S

Footnotes:

a Sample locations are shown on Figures 5.1 to 5.4.

b With the exception of DIRECT RADIATION locations, Station-1X's are indicator stations and Station-2X's are control stations.

c The station code letters will vary with the sample media collected. The sampling of all four media types is not required during each sampling period.

[d Deleted

e The number and location of Demolition sampling points is dependent on the activities performed. Air samplers are placed in or within close proximity of buildings undergoing demolition activities to provide reasonable assessment of the airborne activity that may be generated.

FIGURE 5.1

Environmental Radiological Sampling Locations
Within 1 Kilometer of Maine Yankee

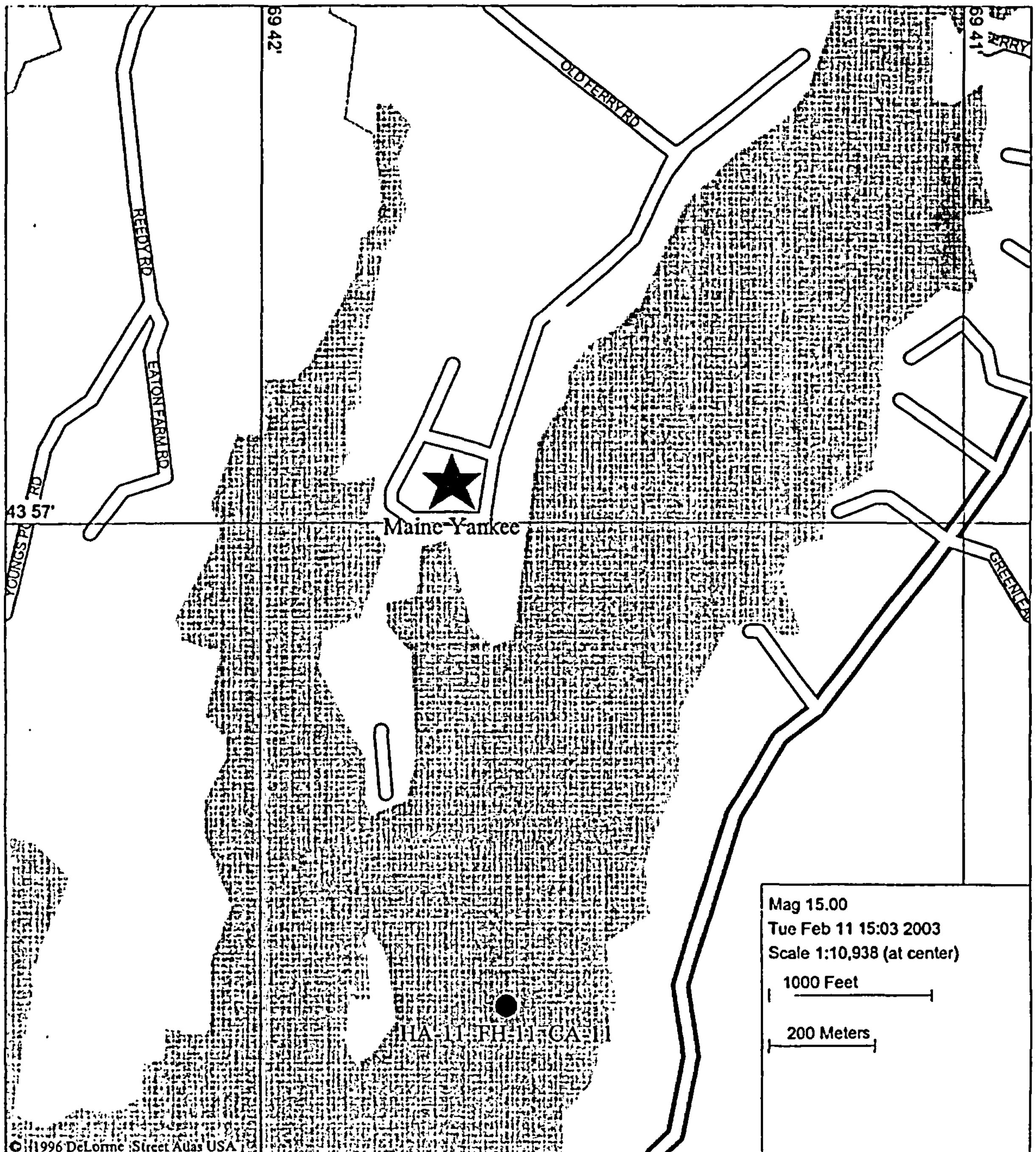

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FIGURE 5.2

Environmental Radiological Sampling Locations
Outside of 1 Kilometer of Maine Yankee

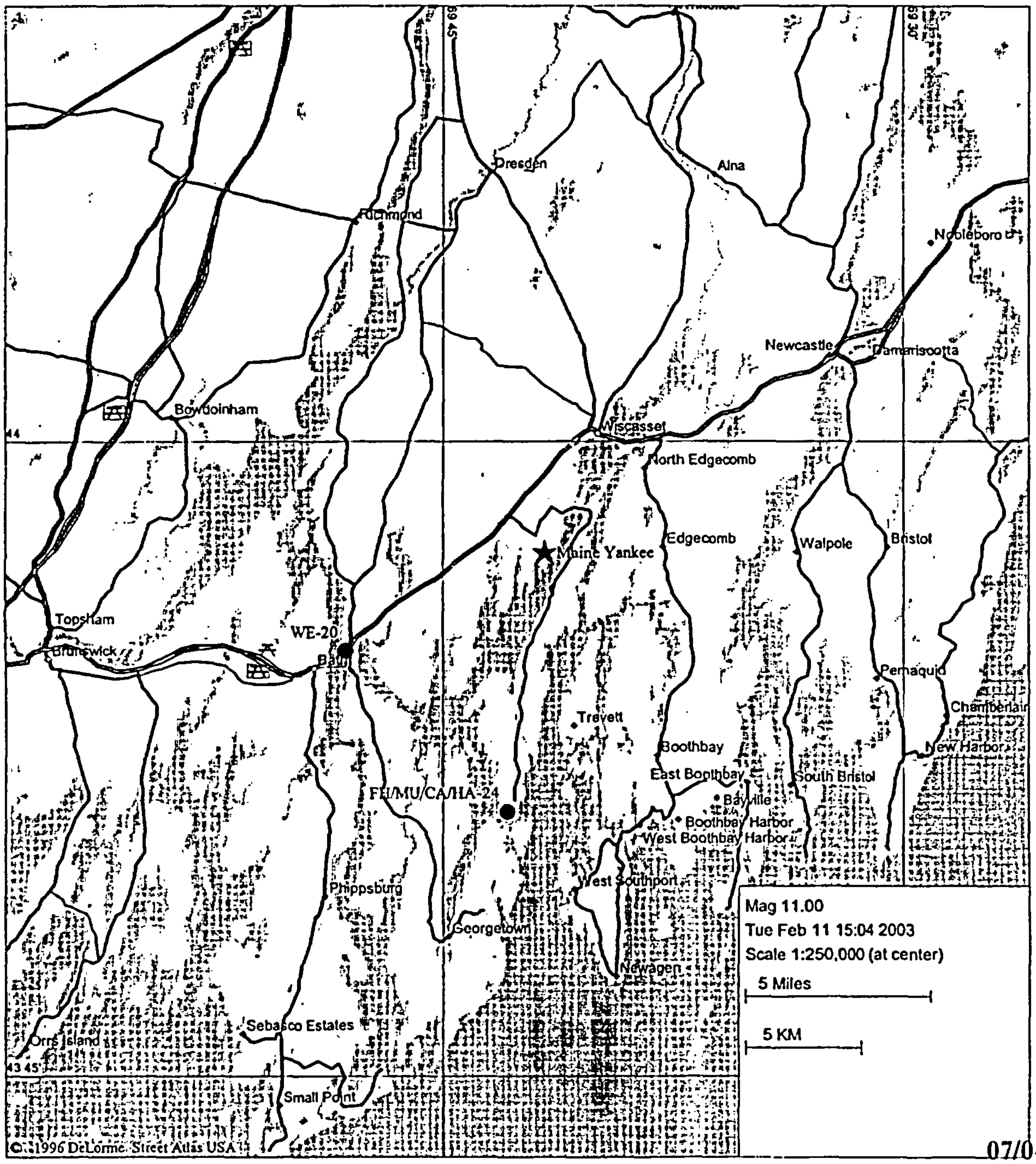

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FIGURE 5.3

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FIGURE 5.4

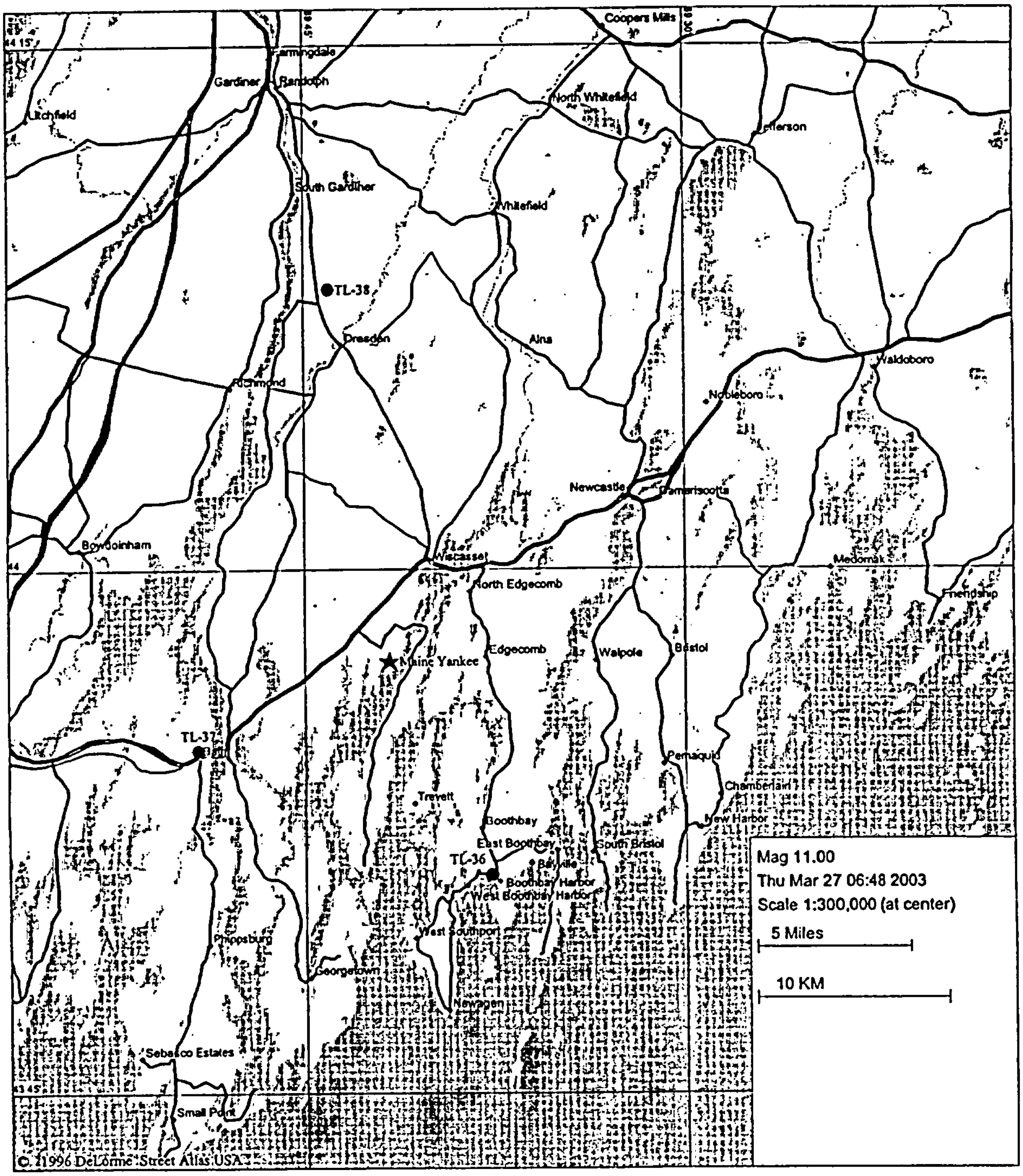

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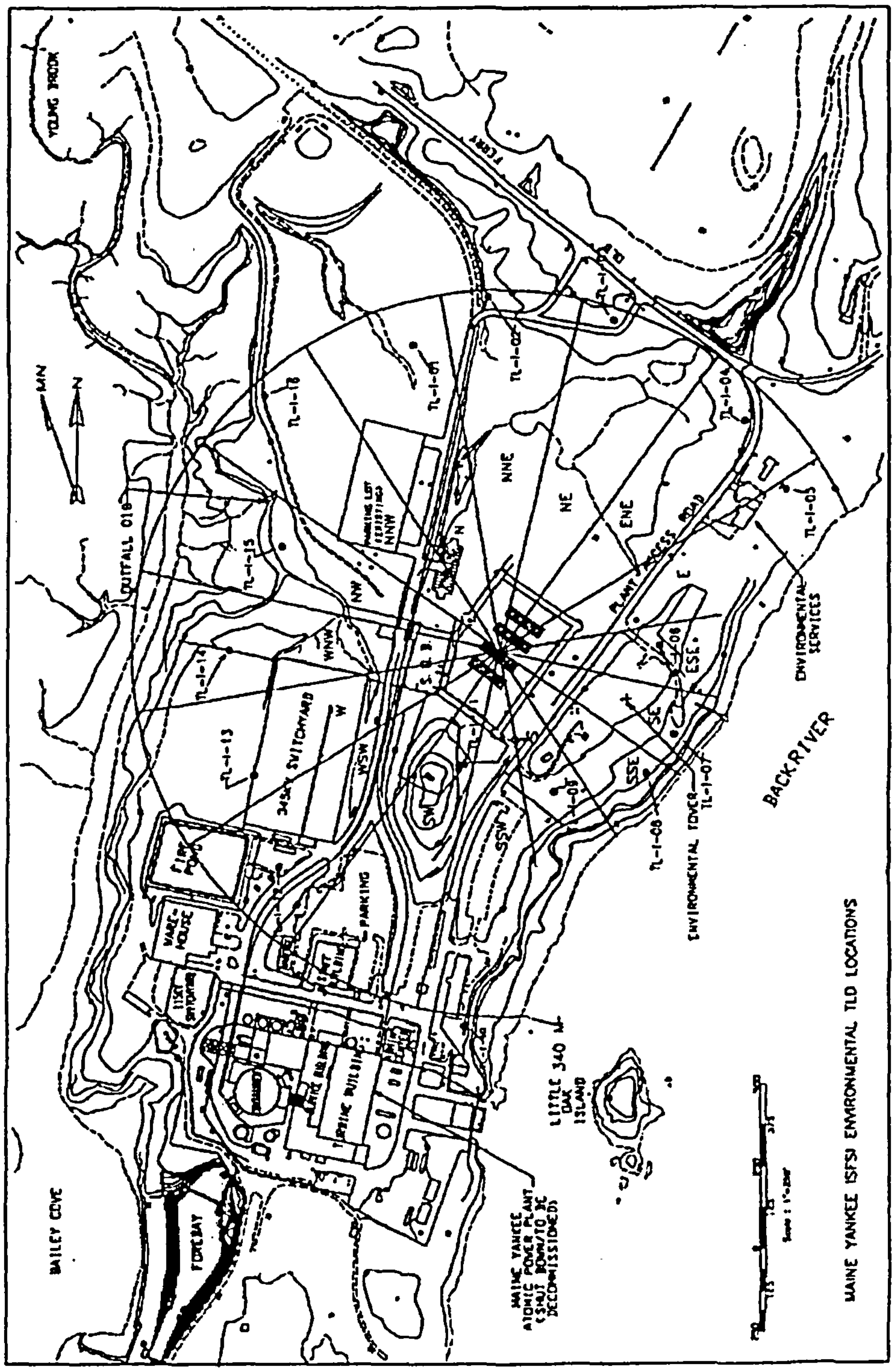
FIGURE 5.5

Direct Radiation Monitoring Locations
Within 340 Meters of the ISFSI



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6.0 MONITOR SETPOINTS

6.1 LIQUID EFFLUENT MONITOR SETPOINTS

This section describes the methodology to determine alarm/trip setpoints of liquid effluent monitors specified in Table 2.1, Radioactive Liquid Effluent Monitoring Instrumentation.

Consistent with Section 2.1.3.1, the total allowable concentration of radioactivity for all releases entering the Back River at any given time shall be limited to a total Effluent Concentration Limit Ratio, ECL Ratio, (R) equal to or less than ten when calculated as follows:

$$R = \sum R_i = \sum \frac{C_i}{ECL_i} \text{ shall be equal to or less than 10} \qquad (6.1)$$

Where:

R = Total ECL ratio (dimensionless)

R_i = ECL ratio (dimensionless) for each individual release "i"

C_i = concentration of each radionuclide (i), in μCi/ml, entering the Back River, and is equal to the undiluted concentration $(C_u)_i$ of radionuclide (i) times the flowrate through the monitored pathway (in gpm) (Q_i) divided by the total of the dilution flow (in gpm) (D_i) plus the release flowrate (Q_i). ($(C_u)_i$ includes non-gamma emitting isotopes such as Tritium)

$$= \frac{(C_u)_i * Q_i}{(D_i + Q_i)}$$

ECL_i = Effluent Concentration Limit (ECL) of radionuclide (i) in μCi/ml as specified in 10 CFR 20, Appendix B, Table 2, Column 2 (includes non-gamma emitters such as tritium).

6.1.1 Internal Setpoints

Internal monitor setpoints shall be established to monitor compliance with the release concentration limits specified in Section 2.1.3.1. Setpoints shall be calculated so as to alarm the monitor (and, if applicable, terminate the release) if the concentration in the discharge pathway may result in the concentration entering the Back River to exceed ten times the ECL (for the most limiting isotope)[1] using the relationship:

$$Setpoint_p = ECL_{gamma} * [(D + Q_p)/Q_p] * PF_p * RF * 10 \qquad (6.2)$$

Where:

$Setpoint_p$ = Monitor response (CPM) for the release pathway "p"

ECL_{gamma} = Effluent Concentration Limit (ECL) The most limiting[1] gamma emitting radionuclide (i) present in the release pathway (µCi/ml). The limiting radionuclide is the nuclide with the highest individual ECL ratio.

D = Minimum expected total Dilution Flow downstream from the monitor and prior to discharge into the Back River.

Q_p =Maximum expected release flowrate through the monitored release pathway, "p" (gpm).

PF_p = Pathway Factor (a value ≤ 1.0) applied to each monitor setpoint calculation. Application of the pathway factors shall be such that, allowing for instrument uncertainties, the total ECL ratio (R) resulting from releases via multiple pathways (R_i), should they exist, is maintained less than or equal to ten, such that:

$\sum PF_p$ shall be equal to or less than 1

RF = Radiation monitor response factor (sensitivity factor) (cpm/µCi/ml).

ECL = "$1/((1/ECL_i) + (C_j/(C_i*ECL_j)))$"

Where:

ECL = Surrogate effluent concentration used for RM-1664 internal setpoint calculation

ECL_i = The effluent concentration limit for limiting isotope[1] as specified in Table 2, Column 2 of 10CFR20

C_j = The pre-dilution release concentration of $isotope_j$

ECL_j = The effluent concentration limit for $isotope_j$ as specified in Table 2, Column 2 of 10CFR20

Example

ECL = 1/((1/3.0 E-6) + (3.41E-6 / (5.14E-5 * 1.0 E-6))
= 1/((3.33E+ 05 + (3.41E-6 / 5.14E-11))
= 1/(3.33E + 05 + 66342)

ECL = 2.50E-6µCi/mL

[1] The most limiting isotope (gamma emitting radionuclide (i)) may be selected based upon the isotope with the highest ECL ratio in the release pathway, as calculated in Section 6.1 (Eq 6.1). When selected in this manner, the ECL used for the internal setpoint shall be established as a surrogate ECL, which accounts for the contributions of other gamma emitting radionuclides in the release pathway.

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6.1.2 External Setpoints

Liquid Radwaste Monitors should also be equipped with an external alarm/trip setpoint. The intent of this setpoint is to provide assurance that the pre-release analysis is representative of the release being made through that monitor, and to alert the operator if a problem does exist. This setpoint shall be determined for each release as follows:

Calculate the expected radiation monitor response (ER), as follows:

$$ER = [\sum (C_u)_i - (C_u)_{non\text{-}gamma}] * RF$$

Where:

$ER =$ Expected radiation monitor response (CPM)

$\sum (C_u)_i =$ Sum of the undiluted activity concentration of each of the radionuclides (i) as determined by the pre-release analysis

$(C_u)_{non\text{-}gamma} =$ Undiluted activity concentration of Tritium and any other non-gamma emitters as determined by the pre-release analysis

$RF =$ Radiation monitor response factor (sensitivity factor) as determined by the most recent monitor calibration (CPM/µCi/ml)

Calculate the external setpoint as follows:

If $R \leq 5$, then:

$$Setpoint_{External} = 2 * ER + Background$$

- BUT -

If $R > 5$, then:

[$Setpoint_{External}$ shall not exceed $[(10/R) * ER] + Background$

The external setpoint shall not be set at a value greater than the internal setpoint.

If the $Setpoint_{External}$ calculates to a value less than 2000 cpm greater than background, the monitor setpoint may be set 2000 cpm above background, provided that setting is less than the internal setpoint.

In the event that the external setpoint alarms and/or trips a release, comply with ACTION 1 of the Table Notations for Table 2.1: Radioactive Liquid Effluent Monitoring Instrumentation.

- If the independent verification is in agreement with the initial analysis, the external setpoint may be established up to the value of the internal setpoint, and the release may proceed.

-but-

- If the independent verification is not in agreement with the initial analysis, the reason for the variation shall be determined, and appropriate corrective action shall be taken prior to recommencing the release. In this event, the external setpoint shall be recalculated and reestablished as described above before proceeding with the release.

Where appropriate, operator monitoring and response action may be an acceptable alternative to an external setpoint considering the following factors:

1. The close physical proximity of the radiation monitor indication from the operator control station (valve control switch).
2. The downstream location of the trip valve with respect to the rad monitor sensing location.
3. The methodology for calculating the expected monitor response is proceduralized.
4. During any release, an operator is stationed to observe the radiation monitor indication.
5. The criteria for taking operator action is proceduralized and available to the operator who is monitoring the release, and the action to be taken by the operator is simple and clearly prescribed in the procedure. (Reference 13)

[6.2 GASEOUS EFFLUENT MONITOR SETPOINTS – Section Deleted

[6.2.1 Deleted

[6.2.2 Deleted

[

FIGURE 6.1

Maine Yankee
Liquid Radwaste System

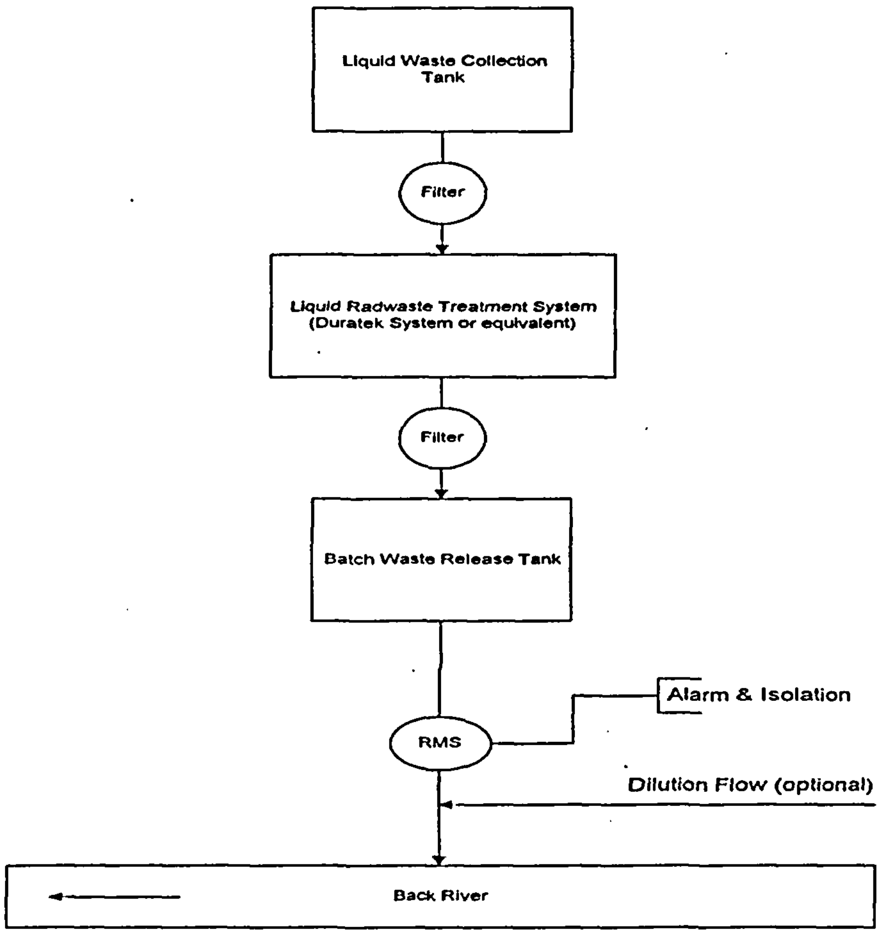

FIGURE 6.2

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[7.0 METEOROLOGY – Section Deleted

TABLE 7.1

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APPENDIX A

BASIS FOR THE DOSE CALCULATION METHODS

[

A.1 LIQUID EFFLUENT DOSES

Method I is used to demonstrate compliance with Section 2.1.4 which limits the dose commitment to a member of the public from radioactive materials in liquid effluents.

Liquid pathways contributing to individual doses at the Maine Yankee Nuclear Power Station are: ingestion of fish and shellfish, and direct exposure from shoreline deposits. The potable water pathway and the irrigated foods pathway are not considered since the receiving water is not suitable for either drinking or irrigation. Method I is derived from Equations A-3 and A-7 of Regulatory Guide 1.109 (Reference 2). Equation A-3 calculates radiation doses from aquatic foods. Equation A-7 from shoreline deposits.

The use of the methodology of Equations A-3 and A-7 for a 1 curie release of each radionuclide in liquid effluents yielded the dose impact to the critical organ. Table 3.1 lists the resulting site specific total body and critical organ dose conversion factors giving the number of millirem per curie released for each radionuclide. Since the dose factors of Table 3.1 represent a variety of critical organs, Method I conservatively calculates a critical organ dose consisting of the maximum critical organ for each radionuclide of any of the four age groups, and combines them into a composite individual independent of age.

APPENDIX A

Except for the site specific values noted below, the parameter values recommended in Regulatory Guide 1.109 (Reference 2) were used to derive the liquid dose factors for Method I. Table A-1 lists the usage factors for liquid pathways utilized in the dose analysis.

The original liquid effluents discharged from the plant when it was operating was via a submerged multi-port diffuser which extended approximately 1000 feet into the tidal estuary and had a design circulating water flow of 420,000 gpm (935 ft^3/sec). For the aquatic foods pathway, the dilution for the mixing effect of the diffuser based on that design flow was set at a minimum of 10 to 1 in the Method I dose factors (Reference 6). That dilution applied to the edge of the initial mixing zone where the effluent had undergone prompt dilution only. With the replacement of the Circulating/Service Water Systems, the discharge forebay and submerged diffuser pipe with a discharge configuration that includes a submerged (at least 20 feet below mean low tide), offshore (by 20 feet at low tide) hose with a maximum waste release discharge flow rate of 300 gpm, new discharge dilution modeling (Reference 11) has demonstrated that the existing Method I dose conversion factors (Table 3.1) and dose equations provide conservative results and can continue to be used. More detailed Method II dose calculations can also continue to conservatively use a previously estimated mixing ratio in the river of 0.024 since Reference 11 indicates that the expected mixing ratios for a range of pump discharge rates up to 300 gpm bound the original mixing credit within the near-field mixing zone, which is set at 13-acres based on the original plant operational discharge models. The continued use of those dilution factors is supported by a dilution analysis (Reference 11) leading to the conclusion that, when applied to liquid effluent discharged directly to the Back River, use of existing dilution factors provide the same or greater level of conservatism as in prior dose calculations. For shoreline deposits, the nearest point where tidal flats could be occupied on a recurring basis is in Bailey Cove which borders the site on the south and west. The estimated average dilution for Bailey Cove with respect to the discharge is conservatively 25 to 1 (Reference 6).

Shoreline activities in the vicinity of the site include a commercial worm digging industry along the tidal flats of Montsweag Bay. In the area of the plant (Bailey Cove), a commercial worm digger could occupy the mud flats for as long as 325 hours per year. This occupancy time is applied to both adults and teenagers in the dose calculations.

For Method I, the period of time for which sediment is exposed to the contaminated water is fifteen years. This time period represents the approximate mid-point of plant operating lifetime, and thus allows for the calculation of a plant lifetime average concentration of radioactivity in sediment. No credit is taken for the decay of activity in transit from the discharge point to the sediment in Bailey Cove.

APPENDIX A

TABLE A-1

Usage Factors for Various Liquid Pathways at Maine Yankee
(From Reference 1, Table E-5*, except as noted.
Zero where no pathway exists.)

AGE	VEG. (KG/YR)	VEG. (KG/YR)	LEAFY MILK (LITER/YR)	MEAT (KG/YR)	FISH (KG/YR)	INVERT. (KG/YR)	POTABLE WATER (LITER/YR)	SHORELINE (HR/YR)
[Adult	0.00	0.00	0.00	0.00	21.00	5.00	0.00	325.00**
Teen	0.00	0.00	0.00	0.00	16.00	3.80	0.00	67.00
Child	0.00	0.00	0.00	0.00	6.90	1.70	0.00	14.00
Infant	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00

*Regulatory Guide 1.109.

**Regional shoreline use associated with mudflats - Maine Yankee Atomic Power Station Environmental Report.

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APPENDIX A

A.2 TOTAL BODY DOSE RATE FROM NOBLE GASES – Section Deleted

A.3 SKIN DOSE RATE FROM NOBLE GASES – Section Deleted

[A.4 CRITICAL ORGAN DOSE RATE FROM PARTICULATES – Section Deleted

A.5 GAMMA AIR DOSE – Section Deleted

A.6 BETA AIR DOSE – Section Deleted

[A.7 DOSE FROM PARTICULATES – Section Deleted

APPENDIX A

A.8 DIRECT DOSE CALCULATION

[With the calculated dose from the release of radioactive materials in liquid effluents exceeding twice the limits in Section 2.1.4 or two times the gaseous limits in Section 2.2.5, calculations should be made including direct radiation contributions from significant plant sources to determine whether the limits of 40 CFR 190 have been exceeded.

The dose to the opposite shoreline of Bailey Cove from fixed direct radiation sources located in or next to the primary structures in the original plant protected area can be calculated from:

$$D_{dir} = 0.087 * E_{tld} * OT$$

Where:

D_{dir} = Estimate of direct dose from fixed facility sources in the Protected Area during the period for which area TLD measurements for E_{tld} are included (mrem).

0.087 = Proportionality factor to change on-site TLD field measurements in mR to mrem along the opposite shoreline of the Bailey Cove (maximum off-site location).

E_{tld} = The net average exposure rate of on-site TLD locations AM-31, AM-39, AM-71, and AM-78 (in mR). In calculating the net average exposure rate, the background exposure rate can be taken as the average of the TLD locations that make up the outer ring of TLDs in the Radiological Environmental Monitoring Program (REMP) as reported in the Maine Yankee annual REMP report.

OT = Assumed occupancy time along the opposite (western) shoreline of Bailey Cove as the nearest off-site boundary with highest expected impact potential (hours/year).

Basis:

An extensive network of on-site TLDs provides a near field direct measurement of radiation from adjacent fixed sources in the plant's original Protected Area. Historical TLD measurements were used to normalize a source–distance computer model using the MCNP4C computer code to predict exposures at a distance typically beyond the ability of direct measurement. The closest off-site land boundary of predicted maximum potential exposures is approximately 1000 feet along the WNW/NW opposite shoreline of Bailey Cove. The predicted relationship between on-site TLD measurements and the far shoreline are used to estimate the direct dose from primary plant structures and materials located in the original Protected Area. Federal regulations (40 CFR 190) state that a dose limit of 25 mrem/year to the total body or any organ from all uranium fuel cycle sources (including direct radiation) applies to real individuals in the areas impacted at or beyond the site boundary. This allows for consideration of occupancy time that members of the public may be subject to exposures. A time of 8760 hours/year (hypothetical full time occupancy) would provide an upper estimate of potential impact. Table A-1 provides time for worm diggers while working on the mudflats during low tide. Other time estimates based on projections of actual land use may be used.

Milk and meat animals are assumed to be on pasture 50 percent of the time, consuming 100 percent of their feed from pasture during that period. This assumption is conservative since most dairy operations use supplemental feeding of animals when on pasture or actually restrict animals to full time silage feeding throughout the year.

APPENDIX A

TABLE A-2

Deleted

APPENDIX A

TABLE A-3

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APPENDIX B

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07/04

APPENDIX C

ROUTINE REPORTS

1. ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

[The Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period, and an assessment of the environmental impact of plant operation, if any. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50. The reports shall also include the results of the land use censuses required by Section 2.4.4 of the ODCM.

The Annual Radiological Environmental Operating Reports shall include summarized and tabulated results of radiological environmental samples taken during the report period pursuant to the tables and figures in the ODCM. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program including a map of all sampling locations keyed to a table giving distances and directions from the reactor; and a discussion of all analyses in which the LLD required by Table 2.4 of the ODCM was not achievable.

2. ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

[[[The Annual Radioactive Effluent Release Report covering the activities of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10CFR50.36a.

The report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the unit summarized on a quarterly basis. The report shall also include a summary of the solid waste released from the unit summarized on a semiannual basis. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped off-site during the report period:

a. Container volume.

b. Total curie quantity (specify whether determined by measurement or estimate).

c. Principal radionuclides (specify whether determined by measurement or estimate).

d. Source waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms).

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity).

f. Solidification agent or absorbent (e.g., cement, asphalt, "Dow").

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

The Radioactive Effluent Release Reports shall include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Section 2.4.4 of the ODCM.

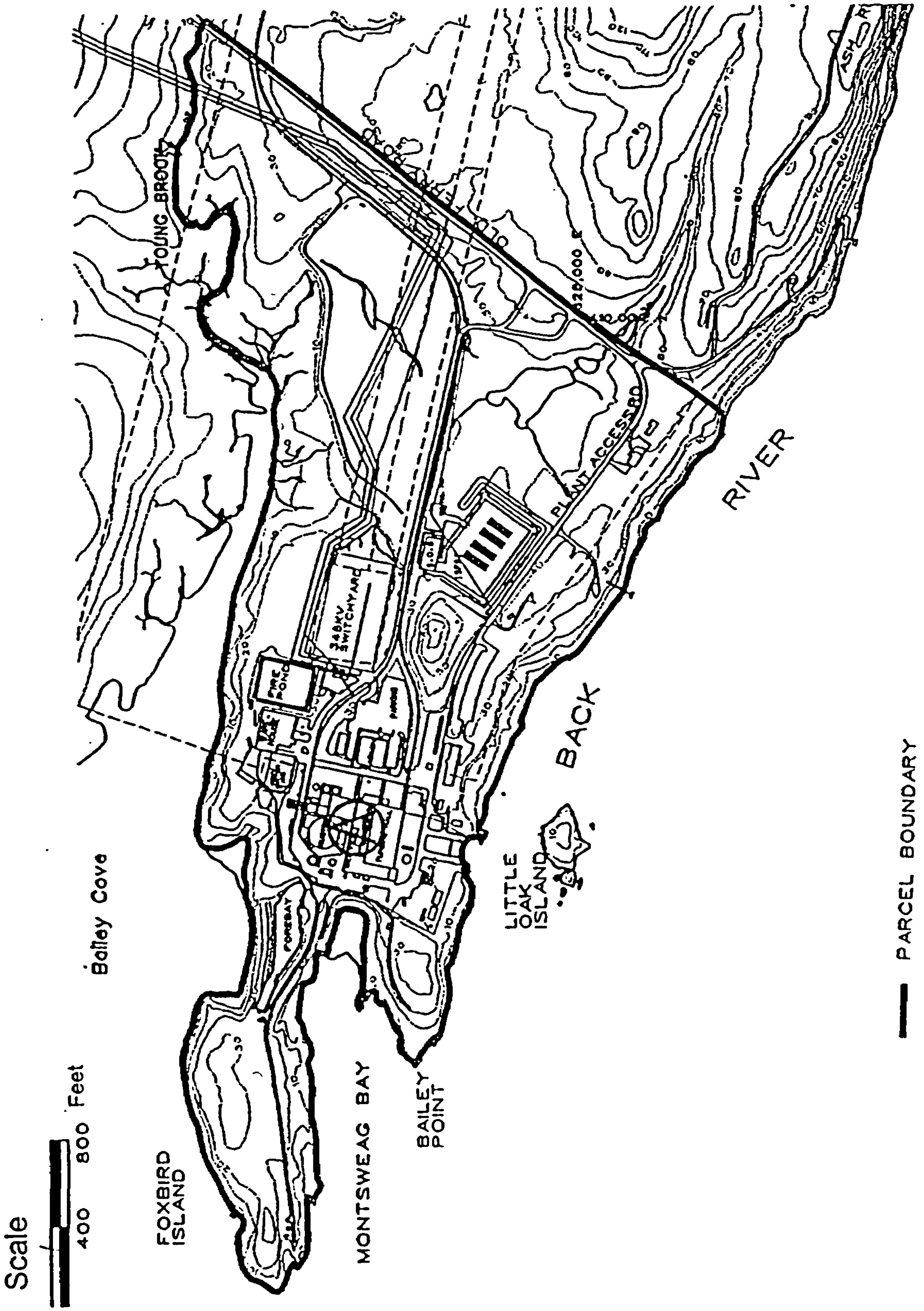
The Radioactive Effluent Release Report shall include changes to the ODCM in the form of a complete, legible copy of the entire ODCM in accordance with the Quality Assurance Program, Appendix D.

3. <u>ESTIMATED DOSE REPORT</u>

A report of the estimated maximum potential dose to the members of the public from radioactive effluent releases for the previous calendar year shall be submitted within 120 days after January 1 of each year. The assessment of the radiation doses shall be performed in accordance with the Off-Site Dose Calculation Manual (ODCM). Site historical meteorological data used in calculating the annual public doses shall be included with the report.

APPENDIX D

SITE BOUNDARY



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