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OFFSITE DOSE CALCULATION MANUAL

FOR

SOUTH CAROLINA ELECTRIC AND GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NON-CONTROLLED COPY

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1. Boegli, T.S., R.R. Bellamy, W.L. Britz, and R.L. Waterfield, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants" NUREG-0133 (October 1978).

2. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR 50, Appendix I", U.S. NRC Regulatory Guide 1.109 (March 1976).

3. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR 50, Appendix I", U.S. NRC Regulatory Guide 1.109, Rev. 1 (October 1977).

4. "Final Safety Analysis Report", South Carolina Electric and Gas Company, Virgil C. Summer Nuclear Station.

5. "Operating License Environmental Report", South Carolina Electric and Gas Company, Virgil C. Summer Nuclear Station.

6. Wahlig, B.G., "Estimation of the Radioactivity Release Rate/Equilibrium Concentration Relationship for the Parr Pumped Storage System", Applied Physical Technology, Inc., February 1981.

7. "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light - Water - Cooled Reactors", U.S. NRC Regulatory Guide 1.111 (March 1976).

8. "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light - Water - Cooled Reactors", U.S. NRC Regulatory Guide 1.111, Rev. 1 (July 1977).

9. Slade, D.H., (editor), "Meteorology and Atomic Energy"; U.S. Atomic Energy Commission, AEC TID-24190, 1968.

10. "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", U.S. NRC Regulatory Guide 1.21, Rev. 1 (June 1974).

11. "Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors", NUREG-0472, Revision 3 (January 1983).

12. "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment", USNRC Regulatory Guide 4.15, Revision 1 (February 1979).

13. "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake", NUREG-0172 (November 1977).

INTRODUCTION

The OFFSITE DOSE CALCULATION MANUAL (ODCM) is an implementing and supporting document of the RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICA-
CO2+ TIONS (RETS). In accordance with USNRC Generic Letter 89-01, entitled "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program", the procedural details for implementing the Radiological Limiting Conditions for Operation have been incorporated into the ODCM. The ODCM describes the methodology and parameters to be used in the calculation of offsite doses due to radioactive liquid and gaseous effluents and in the calculation of liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints. The ODCM contains a list and graphical description of the specific sample locations for the radiological environmental monitoring program. Configurations of the liquid and gaseous radwaste treatment systems are also included.

The ODCM will be maintained at the Station as the reference which details the Radiological Effluent Limiting Conditions for Operation of the V. C. Summer Nuclear Station. Additionally the ODCM will be maintained as the guide for accepted calculational methodologies. Changes in calculation methods or parameters will be incorporated into the ODCM in order to ensure that the ODCM represents the current methodology in all applicable areas. Computer software to perform described calculations will be maintained current with this ODCM.

RESPONSIBILITIES

The ODCM contains the radiological effluent limiting conditions for operation, their applicability, remedial actions, surveillance requirements, and their bases. Plant procedures implement responsibilities for compliance with the ODCM that include:

The Operations group is responsible for:

- Declaring radioactive liquid and gaseous effluent monitor channels operable or inoperable.
- Ensuring the minimum number of operable channels for radioactive liquid and gaseous effluent monitors.
- Notifying the responsible group to implement appropriate action if less than the minimum number of radioactive liquid and gaseous effluent monitor channels are operable.
- Initiating an Off Normal Occurence Report in accordance with SAP-132, when less than the minimum number of channels operable condition prevails for more than 30 days.
- Restoring to within limits, the concentration of liquid radioactive material exceeding ODCM limits released from the site.
- Ensuring radioactive liquid and gaseous effluent monitor setpoints are set as prescribed in the effluent release permit.
- Suspending release if radioactive liquid and gaseous effluent monitor setpoints are less conservative than ODCM requirements.
- Declaring liquid and gaseous radwaste treatment systems operable or inoperable.
- Ensuring operability of gaseous and liquid radwaste treatment systems and ventilation exhaust treatment system.
- Ensuring appropriate portions of the gaseous and liquid radwaste treatment systems are used to reduce the radioactive materials in liquid and gaseous waste prior to their discharge when the projected doses exceed limits specified by the ODCM.
- Initiating an Off Normal Occurrence Report in accordance with SAP-132, when liquid or gaseous radwaste system is inoperable for more than 31 days.
- Performing channel check and source check at the frequencies shown in Tables 1.1-2 and 1.2-2 for each radioactive liquid and gaseous effluent monitoring instrumentation channel.

Instrumentation and Controls group is responsible for:

- Performing channel calibration and analog channel operational test at the frequencies shown in Tables 1.1-2 and 1.2-2 for each radioactive liquid and gaseous effluent monitoring instrumentation channel.
- Informing the Operations group of surveillance test results.

The Health Physics group is responsible for:

- Establishing setpoints for radioactive liquid and gaseous effluent monitors, consistent with ODCM methodology, and providing setpoint information to Operations.
- Implementing remedial actions as requested by Operations. These actions include grab sampling and analysis and providing the results to Operations.
- Performing periodic radioactive effluent monitor checks to determine backgrounds, normal indications and verifying monitor correlation graphs, and providing this information as necessary to Operations.
- Implementing radioactive gaseous and liquid waste sampling and analysis program in accordance with ODCM Tables 1.1-4 and 1.2-3.
- Informing Operations when at least one Circulating Water Pump or the Circulating Water Jockey Pump is required to provide dilution to the discharge structure.
- Calculating cumulative dose contributions and performing dose projections from liquid and gaseous effluents in accordance with the ODCM and providing the information to Operations.
- Initiating an Off Normal Occurrence Report in accordance with SAP-132, when calculated dose from the discharge of radioactive materials in liquid or gaseous effluents are in excess of the limits specified by ODCM sections 1.1.3.1 or 1.2.3.1.
- Initiating an Off Normal Occurrence Report in accordance with SAP-132, when liquid or gaseous waste is discharged without treatment and is in excess of the limits specified by ODCM sections 1.1.4.1 or 1.2.3.1.
- Initiating an Off Normal Occurrence Report in accordance with SAP-132, when the dose or dose commitment to any member of the public due to releases of radioactivity and radiation is in excess of 25 mrem to the total body or any organ (except the thyroid, which shall be limited to less than or equal to 75 mrem) over 12 consecutive months.

- Implementing the Radiological Environmental Monitoring Program as specified in Section 1.4 of the ODCM.

- Initiating an Off Normal Occurrence Report in accordance with SAP-132, when the Radiological Environmental Monitoring Program limiting conditions for operation are exceeded.

- Preparation of the Semiannual Radioactive Effluent Release Report and the Annual Environmental Report.

1.2.2 Gaseous Effluents: Dose Rate

LIMITING CONDITION FOR OPERATION

1.2.2.1 The dose rate in unrestricted areas due to radioactive materials released in gaseous effluents from the site including effluents from oil incineration (see Technical Specification Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For all radioiodines and for all radioactive materials in particulate form and tritium with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABLE: At all Times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

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

1.2.2.3 The dose rate due to radioiodines, tritium and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of ODCM Section 3.2.2 by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 1.2-3.

TABLE 1.2-3

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type		Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A	Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	$1X10^{-4}$
B1	Reactor Building -36" Purge Line -6" Purge Line	P Each Purge[b,c]	P Each Purge[b]	Principal Gamma Emitters[g]	$1X10^{-4}$
				H-3	$1X10^{-6}$
B2	Reactor Building -6" Purge Line (if continuous)	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	$1X10^{-4}$
				H-3	$1X10^{-6}$
C	Main Plant Vent	M[b,e] Grab Sample	M[b]	Principal Gamma Emitters[g]	$1X10^{-4}$
				H-3	$1X10^{-6}$
D1.	Reactor Building Purge	Continuous Sampler[f]	W[d] Charcoal Sample	I-131 I-133	$1X10^{-12}$ $1X10^{-10}$
2.	Main Plant Vent	Continuous Sampler[f]	W[d] Particulate Sample	Principal Gamma Emitters[g] I-131, others	$1X10^{-11}$
		Continuous Sampler[f]	M Composite Particulate Sample	Gross Alpha	$1X10^{-11}$
		Continuous Sampler[f]	Q Composite Particulate Sample	Sr-89,Sr-90	$1X10^{-11}$
		Continuous Monitor	Noble Gas Monitor	Noble Gases Gross Beta	$2X10^{-6}$
E	Oil Incinerator	P Each Batch[h] Grab Sample	P Each Batch	Principal Gamma Emitters[g]	$5 X 10^{-7}$ [i]
				Noble Gases	1E-5 [i]
				I-131	1E-6 [i]
				H-3	3E-5 [i]
				Sr-89, Sr-90	3E-7 [i]
				Fe-55	1E-6 [i]

See Table 1.1-3 for explanation of frequency notation.

TABLE 1.2-3 (Continued)

TABLE NOTATION

a. See Table 1.1-4 notation (a) for definition of LLD.

b. Analyses shall be also be performed within 24 hours following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for a least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with ODCM Specifications 1.2.2.1, 1.2.3.1 and 1.2.4.1.

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

h. Prior to sampling for analysis, each batch of oil shall be isolated and representative samples obtained by methods described in ASTM D 4057-81, Volume 05.03, "Standard Practice for Manual Sampling of Petroleum and Petroleum Products".

i. This LLD refer to the liquid sample.

1.2.4 Gaseous Effluents: Dose - Radioiodines, Tritium, and Radioactive Materials in Particulate Form.

LIMITING CONDITION FOR OPERATION

1.2.4.1 The dose to an individual from radioiodines, tritium, and radioactive materials in particulate form, and radionuclides (other than noble gases) with half-lives greater than 8 days in gaseous effluents including effluents from oil incineration (see Technical Specification Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,
b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABLE: At all Times.

ACTION:

a. With the calculated dose from the release of tritium, radioiodines, and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents exceeding any of the above limits, in lieu of any other report required by ODCM Section 1.6, prepare and submit to the Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to releases and the proposed actions to be taken to assure that subsequent release will be in compliance with ODCM Specification 1.2.4.1.

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

1.2.4.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with ODCM Section 3.2.3 at least once per 31 days.

TABLE 1.4-2

Reporting Levels for Radioactivity Concentrations in Environmental Samples
Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases(pCi/m3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/Kg, wet)
H-3	20,000(a)	N.A.	N.A.	N.A.	N.A.
Mn-54	1,000	N.A.	30,000	N.A.	N.A.
Fe-59	400	N.A.	10,000	N.A.	N.A.
Co-58	1,000	N.A.	30,000	N.A.	N.A.
Co-60	300	N.A.	10,000	N.A.	N.A.
Zn-65	300	N.A.	10,000	N.A.	N.A.
Zr-95	400	N.A.	20,000	N.A.	N.A.
Nb-95	400	N.A.	20,000	N.A.	N.A.
I-131	2	0.9	N.A.	3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-140	200	N.A.	N.A.	300	N.A.
La-140	200	N.A.	N.A.	300	N.A.

(a) For drinking water samples. This is the 40 CFR Part 141 value.

1977, is titled "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I". Regulatory Guide 1.113, April 1977, is titled "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I".

B/1.1.4 Liquid Waste Treatment

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

B/1.2 GASEOUS EFFLUENTS

B/1.2.1 Radioactive Gaseous Effluent Monitoring Instrumentation

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

B/1.2.2 Dose Rate

This specification is provided to ensure that the dose at any time at the site boundary from gaseous effluents from all units as well as the oil incinerator on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentration of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20. 106 (b)). For individuals who may at times be within the site boundary,the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrem/year.

B/1.2.3 Dose - Noble Gases

This specification is provided to implement the requirements of Sections II.B, IIIA and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", section 5.3. NUREG-0133 implements Regulatory Guide 1.109, Revision 1, October 1977 and Regulatory Guide 1.111, Revision 1, July 1977. Regulatory Guide 1.109 is entitled "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, "Revision 1, October 1977 and Regulatory Guide 1.111 is entitled "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors, "Revision 1, July 1977. The ODCM equations provided for determining the air doses at the site boundary are based upon the historical average atmospheric conditions.

This specification applies to the release of gaseous effluents from all reactors at the site and from the incineration of oil.

B/1.2.4 Dose-Radioiodines, Tritium and Radioactive Materials in Particulate Form

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways in unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", section 5.3. NUREG-0133 implements Regulatory Guide 1.109, Revision 1, October 1977 and Regulatory Guide 1.111, Revision 1, July 1977. Regulatory Guide 1.109 is entitled "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, "Revision 1, October 1977 and Regulatory Guide 1.111 is entitled "Methods for Estimating Atmospheric Transport and Dispersion of of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, "Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines,

tritium, and radioactive materials in particulate form are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

This specification applies to the release of gaseous effluents from all reactors at the site and from the incineration of oil.

B/1.2.5 Gaseous Radwaste Treatment

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

B/1.3 RADIOACTIVE EFFLUENTS: TOTAL DOSE

The specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to a member of the public for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation which is part of the nuclear fuel cycle.

1.6.2 Semiannual Radioactive Effluent Release Report

1.6.2.1 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

1.6.2.2 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. The summary will also include quantities of radioactive gaseous effluent and solid waste (ash) released as a result of on-site oil incineration.

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station and oil incinerator during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figures 5.1-3 and 5.1-4 of the VCSNS Technical Specifications) during the year. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. Historical annual average meteorology or meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

In reality, all of these effluent pathways utilize the circulating water as dilution to the effluent stream, with the circulating water discharge canal being the point of release into an unrestricted area. Steam Generator Blowdown Effluent may be released to the Circulating Water either directly in the Condenser outflow (the normal flow path) or in the first hours following startup via the Industrial and Sanitary Waste System (ISWS) for chemical reasons. The Turbine Building Sump and Condensate Demineralizer Backwash Effluents enter Circulating Water via the sumps and ponds of the Industrial and Sanitary Waste System.

CO3↓ To ensure compliance with ODCM specification 1.1.2.1, normally no dilution is assumed for discharges to the Industrial and Sanitary Waste System. The assumption of no dilution limits discharges to $<$ 0.5 MPC and therefore ODCM specification 1.1.2.1 would not be compromised in the event circulating water dilution is lost. To add operational flexibility for abnormal conditions (radionuclide concentration in Turbine Building sump $>$ 0.5 MPC), discharges from the Turbine Building sump and concentrations in the ISWS may exceed the operational objective, 0.5 MPC, provided circulating water dilution is sufficient to ensure compliance with ODCM specification 1.1.2.1 and liquid effluents are being discharged in compliance with ODCM specification 1.1.4.1. Two separate setpoint calculations are given for Turbine Building sump discharges (RM-L8). Section 2.1.4.2.1 describes the setpoint calculation normally used, limiting discharges to 0.5 MPC. Section 2.1.4.2.2 provides an alternate setpoint methodology which may be used during abnormal conditions. RM-L8 set-points are considered in compliance with ODCM specification 1.1.1.1 provided the setpoints are
CO3↑ adequate to prevent releases in excess of ODCM specification 1.1.2.1.

Two mutually exclusive setpoint calculation processes are outlined below for steam generator blowdown. Section 2.1.4.1 is to be used whenever Steam Generator Blowdown is being released directly to the Circulating Water in the Condenser outflow, which is the normal mode. Section 2.1.4.2 is to be used whenever Steam Generator Blowdown is being released to the Industrial and Sanitary Waste System, or diverted to the Nuclear Blowdown Processing System, both of which are alternate modes.

Normally, water collected by the Nuclear Blowdown Processing System has very low specific activity. This water may be processed to the Turbine Building sump.

NOTE: When Circulating Water is unavailable for effluent dilution, releases containing activity above LLD (excluding tritium) should be discouraged via pathways

which lead to it. Steam Generator Blowdown should be diverted to the Nuclear Blowdown Processing System. Condensate Demineralizer Backwash may be diverted to the Turbine Building sump or not released. Turbine Building sump effluent should be processed through temporary demineralizers or diverted to the Excess Liquid Waste Processing System. (These steps are to keep the calculated dose to individuals as low as reasonably achievable.)

2.1.4.1 Steam Generator Blowdown Effluent Direct to Circulating Water (Normal Mode)

Equation (1) is again used to assure that effluents are in compliance with the aforementioned specification:

$$C \geq \frac{cf}{(F+f)}$$

The available dilution water flow (F_{dc}) is dependent upon the mode of operation of the Circulating Water System. Any change in this value will be accounted for in a recalculation of equation (1). The Steam Generator Blowdown flow rate (f_{ds}) and the Steam Generator Blowdown monitor setpoints (c_{S_a} and c_{S_b}) are set to meet the condition of equation (1).

RM-L3, the first monitor in the Steam Generator Blowdown discharge pathway, alarms and terminates release of the stream. The discharge is then automatically diverted to the Nuclear Blowdown Processing System. RM-L10, the last monitor in the Steam Generator Blowdown discharge pathway, alarms and terminates the release. Thus, RM-L10 is redundant to RM-L3 and the setpoint (c_{S_b}) will be determined in the same manner as RM-L3 (c_{S_a}).

The method by which the monitor setpoints are determined is as follows:

Where:

$\left[\sum_g C_g\right]_S$ = The isotopic concentration of the Steam Generator Blowdown effluent as obtained from the sum of the measured concentrations determined by the analysis required in ODCM Table 1.1-4, in uCi/ml.

CF_S = The Steam Generator Blowdown Effluent Concentration Factor from equation (29).

*See GENERAL NOTE under 2.1.

CO3→ 2.1.4.4 Turbine Building Sump (Abnormal Conditions)

Provided circulating water is available, 1 to 3 circulating water pumps, effluent exceeding 0.5 MPC may be released from the Turbine Building sump to the industrial and sanitary waste system, using the setpoint in this section, provided the following conditions are met:

1) Instantaneous release rate limits of ODCM Specification 1.1.2.1 are not exceeded in the circulating water discharge canal.

2) The average radionuclide concentration in the industrial and sanitary waste system (Pond 6B or 008) will not exceed 1.0 MPC when averaged over one year.

3) The limits of ODCM specification 1.1.4.1 will not be exceeded with actual liquid effluent releases over a 31 day period.

4) Average discharge flow does not exceed values used in setpoint determination.

In addition, the source of radioactivity should be identified and isolated. Radionuclide concentration in Turbine Building sump effluent should be restored to <0.5 MPC as soon as possible and normal setpoint reestablished. Radionuclide concentration in Pond 6B and 008 should be restored to < LLD (excluding tritium) using dilution as necessary (normal flow from the TBS would normally be adequate). Turbine Building sump samples should be obtained and analyzed every eight hours while the alternate setpoint is being used to ensure that the setpoint remains conservative with respect to the isotopic mixture and to ensure offsite doses are within ODCM limits.

Alterna... setpoint methodology for Turbine Building sump (RM-L8) is available to ensure operational flexibility in the event radioactivity is detected in the Turbine Building sump > 0.5 MPC and release would result in minimal offsite dose. The alternate setpoint methodology is not intended to be used continuously. To remove restrictions on operation of circulating water, pond concentrations should be restored to < LLD as soon as possible. The setpoint methodology follows:

2.1.4.4.1 <u>For RM-L8, Turbine Building Sump (alternate methodology)</u>

$$C_T \leq \frac{\sum_g C_g}{CF_T} X \frac{1}{F_k} \qquad (57)$$

where,

F_k = The near field dilution factor for C_i during release from Turbine Building sump.

= $\frac{\text{(average undiluted waste flow)}}{\text{(average flow from discharge structure)}}$

For purpose of implementing section 2.1.4.4 release condition 2, the following must be satisfied.

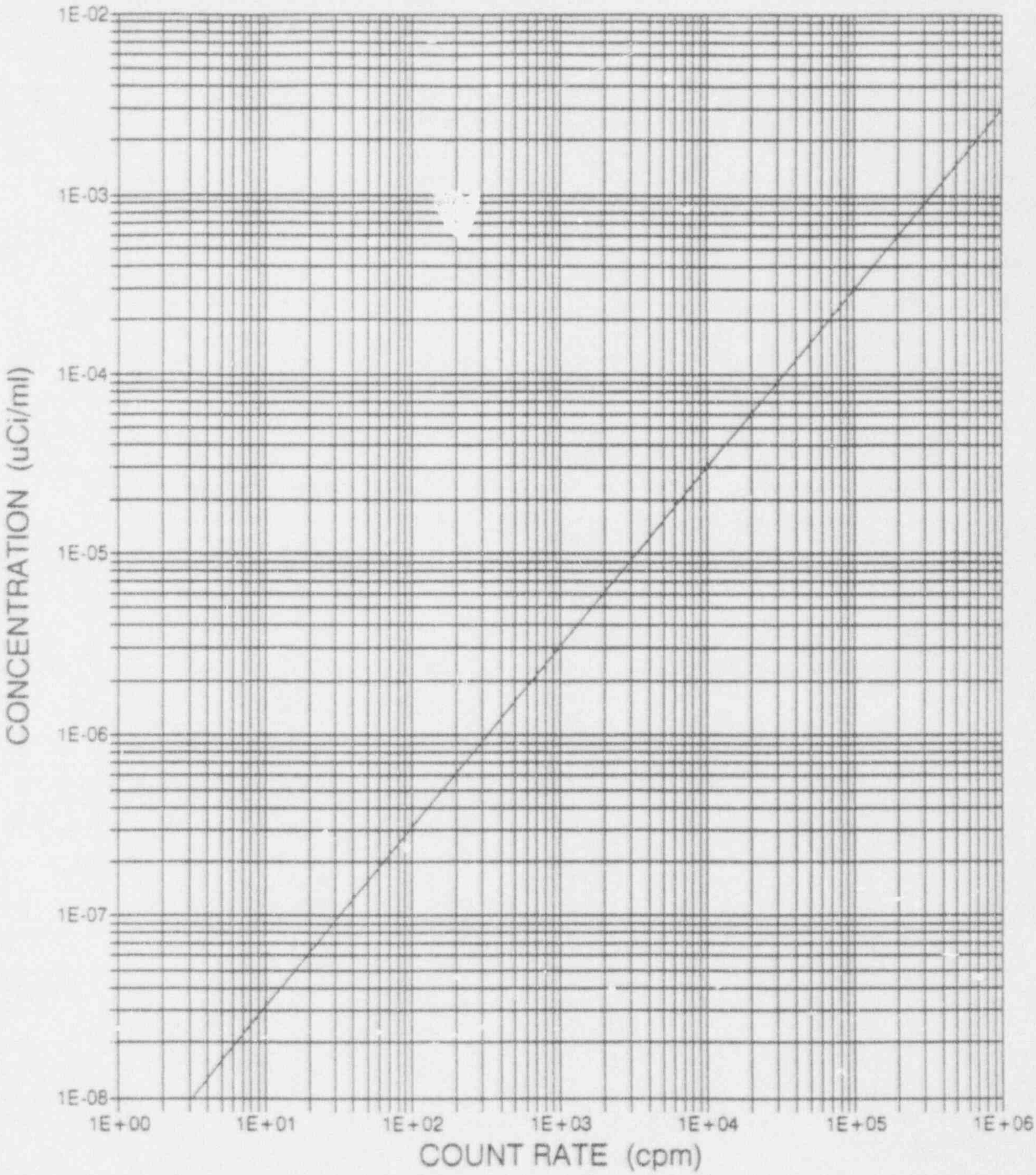
$$\frac{\sum_{j=1}^{n} \left\{ \left[\sum_{i=1}^{x} \left(C_i / MPC_i \right) \right]_{T_j} * V_j \right\}}{\sum_{j=1}^{n} V_j} < 1.0 \qquad (58)$$

where $[\Sigma(C_i / MPC_i)]_{T_j}$ = the sum of the ratios of the measured concentration of nuclide $_i$ to its limiting value MPC_i for the Turbine Building sump effluent for release permit $_j$, including proposed permit,

V_j = Release volume for Turbine Building sump release permit $_j$ (gal), and

$_j$ = index for batch release permits during the calendar year.

Figure 2.1-1
Example Liquid Effluent Monitor
Calibration Curve



2.2 Dose Calculation For Liquid Effluents

The method of this section is to be used in all cases for calculating doses to individuals from routine liquid effluents. Four notes at the end of the section confirm the values which certain parameters are to be assigned in some special cases.

2.2.1 Liquid Effluent Dose Calculation Parameters

Term		Definition	Section of Initial Use
$A_{i\tau}$	=	the site related ingestion dose commitment factor to the total body or any organ τ, for each identified principal gamma and beta emitter listed in Table 2.2-3 in mrem-ml per hr-µCi.	2.2.2
BF_i	=	Bioaccumulation Factor for nuclide i, in fish, pCi/Kg per pCi/l, from Table 2.2-1.	2.2.2
C_{ik}	=	the average concentration of radionuclide, i, in undiluted liquid effluent during time period Δt_k from any liquid released, in uCi/ml.	2.2.2
$DF_{i\tau}$	=	a dose conversion factor for nuclide, i, for adults in preselected organ, τ, in mrem/pCi found in Table 2.2-2.	2.2.2
D_τ	=	the cumulative dose commitment to the total body or any organ, τ, from the liquid effluents for the total time period, $\Sigma\Delta t_k$ in mrem (Ref. 1).	2.2.2
D_w	=	Dilution Factor from the near field area within one-quarter mile of the release points to the potable water intake for adult water consumption; for V. C. Summer, $D_w = 1$.	2.2.2
F_k	=	the near field average dilution factor for C_{ik} during any liquid effluent release.	2.2.2
K_o	=	1.14×10^5, units conversion factor = $(10^6$ pCi/uCi$)$ $(10^3$ ml/l$)$ $\div$ 8760 hr/yr	2.2.2

Liquid Effluent Dose Calculation Parameters (continued)

Term		Definition	Section of Initial Use
Δt_k	=	the length (in hours) of a time period over which concentrations and flow rates are averaged for dose calculations.	2.2.2
U_F	=	21 kg/yr, fish consumption (adult) (Reference 3).	2.2.2
U_w	=	730 kg/yr, water consumption (adult) (Reference 3).	2.2.2
Z	=	applicable near-field dilution factor when no additional dilution is to be considered; Z = 1.	2.2.2

2.2.2 Methodology

The dose contribution from all radionuclides identified in liquid effluents released to unrestricted areas is calculated using the following expression:

$$D_\tau = \sum_i \left[A_{i\tau} \sum_{k=1} \Delta t_k C_{ik} F_k \right] \tag{31}$$

$$A_{i\tau} = K_o \left((U_w/D_w) + U_F BF_i \right) DF_{i\tau} \tag{32}$$

$$F_k = \frac{\text{(average undiluted liquid waste flow)}}{\text{(average flow from the discharge structure) (Z)}} \tag{33}$$

NOTE 1: If radioactivity in the Monticello Reservoir (C_{ir}) becomes > the LLD specified in ODCM, Table 1.1-4, that concentration must be included in the Dose determination. For this part of the dose calculation, F_k = 1 and Δt_k = the entire time period for which the dose is being calculated.

NOTE 2: Prior to termination of Circulating Water Pumps, an assessment of the dose resulting from pond radioactivity concentrations and discharge flow rates from the Industrial And Sanitary Waste System (ISWS) will be performed as follows. Sampling of the liquid in the ISWS will be initiated.

and the measured concentrations of radionuclides will be used in the dose calculations with $F_k = 1$ and Δt_k = the entire time period for which the dose is being calculated.

NOTE 3: For releases through the ISWS pathway when circulating water is not available, dose projections for assessment of release acceptability should be based on the most representative samples obtained from in plant sumps. Normally sump samples are also used to assess actual release. However, due to the ultraconservative assumptions when circulating water is not available, i.e. dose calculations are based on radioactive material concentration in the discharge stream regardless of release volume, representative samples from the ISWS may be used to evaluate impact of releases.

NOTE 4: During periods when the Circulating Water Pumps are in operation, any releases to the ISWS are to be credited with dilution in Circulating Water for dose calculation purposes, even though such dilution is normally not claimed in the setpoint calculation. When taken in union with the note above, this procedure results in some overestimation of dose to the population because discharges made to the ISWS just before loss of Circulating Water will be counted twice in the dose calculation process.

NOTE 5: If radioactivity in the Service Water becomes > LLD as determined by the analysis required by ODCM, Table 1.1-4, that concentration must be included in the Dose determination. For this part of the dose calculation, $F_k = 1$ and Δt_k = the entire time since the last Service Water sample was taken.

TABLE 2.2-1
BIOACCUMULATION FACTORS*
(pCi/kg per pCi/liter)

ELEMENT	FRESHWATER FISH
H	9.0E-01
C	4.6E 03
F	1.0E 01
Na	1.0E 02
P	1.0E 05
Cr	2.0E 02
Mn	4.0E 02
Fe	1.0E 02
Co	5.0E 01
Ni	1.0E 02
Cu	5.0E 01
Zn	2.0E 03
Br	4.2E 02
Rb	2.0E 03
Sr	3.0E 01
Y	2.5E 01
Zr	3.3E 00
Nb	3.0E 04
Mo	1.0E 01
Tc	1.5E 01
Ru	1.0E 01
Rh	1.0E 01
Sb	1.0E 00
Te	4.0E 02
I	1.5E 01
Cs	2.0E 03
Ba	4.0E 00
La	2.5E 01
Ce	1.0E 00
Pr	2.5E 01
Nd	2.5E 01
W	1.2E 03
Np	1.0E 01

*Values in Table 2.2-1 are taken from Reference 3, Table A-1.

ADULT INGESTION DOSE FACTORS*
(mrem/pCi ingested)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	NO DATA	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07
C-14	2.84E-06	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07
†F-18	6.24E-07	NO DATA	6.92E-08	NO DATA	NO DATA	NO DATA	1.85E-08
NA-24	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.07E-06	1.70E-06
P-32	1.93E-04	1.20E-05	7.46E-06	NO DATA	NO DATA	NO DATA	2.17E-05
CR-51	NO DATA	NO DATA	2.66E-09	1.59E-09	5.86E-10	3.53E-09	6.69E-07
MN-54	NO DATA	4.57E-06	8.72E-07	NO DATA	1.36E-06	NO DATA	1.40E-05
MN-56	NO DATA	1.15E-07	2.04E-08	NO DATA	1.46E-07	NO DATA	3.67E-06
FE-55	2.75E-06	1.90E-06	4.43E-07	NO DATA	NO DATA	1.06E-06	1.09E-06
FE-59	4.34E-06	1.02E-05	3.91E-06	NO DATA	NO DATA	2.85E-06	3.40E-05
†CO-57	NO DATA	1.75E-07	2.91E-07	NO DATA	NO DATA	NO DATA	4.44E-06
CO-58	NO DATA	7.45E-07	1.67E-06	NO DATA	NO DATA	NO DATA	1.51E-05
CO-60	NO DATA	2.14E-06	4.72E-06	NO DATA	NO DATA	NO DATA	4.02E-05
NI-63	1.30E-04	9.01E-06	4.36E-06	NO DATA	NO DATA	NO DATA	1.88E-06
NI-65	5.28E-07	6.86E-08	3.13E-08	NO DATA	NO DATA	NO DATA	1.74E-06
CU-64	NO DATA	8.33E-08	3.91E-08	NO DATA	2.10E-07	NO DATA	7.10E-06
ZN-65	4.84E-06	1.54E-05	6.76E-06	NO DATA	1.03E-05	NO DATA	9.70E-06
ZN-69	1.03E-08	1.97E-08	1.37E-09	NO DATA	1.28E-08	NO DATA	2.96E-09
†Zn-69m‡	1.70E-07	4.08E-07	3.37E-08	NO DATA	2.47E-07	NO DATA	2.49E-05
†BR-82	NO DATA	NO DATA	2.26E-06	NO DATA	NO DATA	NO DATA	2.59E-06
BR-83‡	NO DATA	NO DATA	4.02E-08	NO DATA	NO DATA	NO DATA	5.79E-08
BR-84	NO DATA	NO DATA	5.21E-08	NO DATA	NO DATA	NO DATA	4.09E-13
BR-85	NO DATA	NO DATA	2.14E-09	NO DATA	NO DATA	NO DATA	LT E-24**
RB-86	NO DATA	2.11E-05	9.83E-06	NO DATA	NO DATA	NO DATA	4.16E-06
RB-88	NO DATA	6.05E-08	3.21E-08	NO DATA	NO DATA	NO DATA	8.36E-19
RB-89‡	NO DATA	4.01E-08	2.82E-08	NO DATA	NO DATA	NO DATA	2.33E-21
SR-89‡	3.08E-04	NO DATA	8.84E-06	NO DATA	NO DATA	NO DATA	4.94E-05
SR-90‡	7.58E-03	NO DATA	1.86E-03	NO DATA	NO DATA	NO DATA	2.19E-04
SR-91‡	5.67E-06	NO DATA	2.29E-07	NO DATA	NO DATA	NO DATA	2.70E-05
SR-92‡	2.15E-06	NO DATA	9.30E-08	NO DATA	NO DATA	NO DATA	4.26E-05
Y-90	9.62E-09	NO DATA	2.58E-10	NO DATA	NO DATA	NO DATA	1.02E-04
Y-91M‡	9.09E-11	NO DATA	3.52E-12	NO DATA	NO DATA	NO DATA	2.67E-10
Y-91	1.41E-07	NO DATA	3.77E-09	NO DATA	NO DATA	NO DATA	7.76E-05
Y-92	8.45E-10	NO DATA	2.47E-11	NO DATA	NO DATA	NO DATA	1.48E-05
Y-93	2.68E-09	NO DATA	7.40E-11	NO DATA	NO DATA	NO DATA	8.50E-05
ZR-95‡	3.04E-08	9.75E-09	6.60E-09	NO DATA	1.53E-08	NO DATA	3.09E-05
ZR-97‡	1.68E-09	3.39E-10	1.55E-10	NO DATA	5.12E-10	NO DATA	1.05E-04
NB-95	6.22E-09	3.46E-09	1.86E-09	NO DATA	3.42E-09	NO DATA	2.10E-05
†NB-97	5.22E-11	1.32E-11	4.82E-12	NO DATA	1.54E-11	NO DATA	4.87E-08
MO-99‡	NO DATA	4.31E-06	8.20E-07	NO DATA	9.76E-06	NO DATA	9.99E-06

‡Daughter contributions are included (see Reference 13).
†Values taken from Reference 13, Table 4.
*Values other than those footnoted in Table 2.2-2 are taken from Reference 3, Table E-11.
**Less than E-24.

TABLE 2.2-2 (continued)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
TC-99M	2.47E-10	6.98E-10	8.89E-09	NO DATA	1.06E-08	3.42E-10	4.13E-07
TC-101	2.54E-10	3.66E-10	3.59E-09	NO DATA	6.59E-09	1.87E-10	1.10E-21
RU-103‡	1.85E-07	NO DATA	7.97E-08	NO DATA	7.06E-07	NO DATA	2.16E-05
RU-105‡	1.54E-08	NO DATA	6.08E-07	NO DATA	1.99E-07	NO DATA	9.42E-06
RU-106‡	2.75E-06	NO DATA	3.48E-07	NO DATA	5.31E-06	NO DATA	1.78E-04
AG-110M‡	1.60E-07	1.48E-07	8.79E-08	NO DATA	2.91E-07	NO DATA	6.04E-05
†SB-124	2.80E-06	5.29E-08	1.11E-06	6.79E-09	NO DATA	2.18E-06	7.95E-05
†SB-125	1.79E-06	2.00E-08	4.26E-07	1.82E-09	NO DATA	1.38E-06	1.97E-05
†SB-126	1.15E-06	2.34E-08	4.15E-07	7.04E-09	NO DATA	7.05E-07	9.40E-05
†SB-127	2.58E-07	5.65E-09	9.90E-08	3.10E-09	NO DATA	1.53E-07	5.90E-05
TE-125M	2.68E-06	9.71E-07	3.59E-07	8.06E-07	1.09E-05	NO DATA	1.07E-05
TE-127M‡	6.77E-06	2.42E-06	8.25E-07	1.73E-06	2.75E-05	NO DATA	2.27E-05
TE-127	1.10E-07	3.95E-08	2.38E-08	8.15E-08	4.48E-07	NO DATA	8.68E-06
TE-129M‡	1.15E-05	4.29E-06	1.82E-06	3.95E-06	4.80E-05	NO DATA	5.79E-05
TE-129	3.14E-08	1.18E-08	7.65E-09	2.41E-08	1.32E-07	NO DATA	2.37E-08
TE-131M‡	1.73E-06	8.46E-07	7.05E-07	1.34E-06	8.57E-06	NO DATA	8.40E-05
TE-131‡	1.97E-08	8.23E-09	6.22E-09	1.62E-08	8.63E-08	NO DATA	2.79E-09
TE-132‡	2.52E-06	1.63E-06	1.53E-06	1.80E-06	1.57E-05	NO DATA	7.71E-05
I-130	7.56E-06	2.23E-06	8.80E-07	1.89E-04	3.48E-06	NO DATA	1.92E-06
I-131‡	4.16E-06	5.95E-06	3.41E-06	1.95E-03	1.02E-05	NO DATA	1.57E-06
I-132	2.03E-07	5.43E-07	1.90E-07	1.90E-05	8.65E-07	NO DATA	1.02E-07
I-133‡	1.42E-06	2.47E-06	7.53E-07	3.63E-04	4.31E-06	NO DATA	2.22E-06
I-134	1.06E-07	2.88E-07	1.03E-07	4.99E-06	4.58E-07	NO DATA	2.51E-10
I-135‡	4.43E-07	1.16E-06	4.28E-07	7.65E-05	1.86E-06	NO DATA	1.31E-06
CS-134	6.22E-05	1.48E-04	1.21E-04	NO DATA	4.79E-05	1.59E-05	2.59E-06
CS-136	6.51E-06	2.57E-05	1.85E-05	NO DATA	1.43E-05	1.96E-06	2.92E-06
CS-137‡	7.97E-05	1.09E-04	7.14E-05	NO DATA	3.70E-05	1.23E-05	2.11E-06
CS-138	5.52E-08	1.09E-07	5.40E-08	NO DATA	8.01E-08	7.91E-09	4.65E-13
BA-139	9.70E-08	6.91E-11	2.84E-09	NO DATA	6.46E-11	3.92E-11	1.72E-07
BA-140‡	2.03E-05	2.55E-08	1.33E-06	NO DATA	8.67E-09	1.46E-08	4.18E-05
BA-141‡	4.71E-08	3.56E-11	1.59E-09	NO DATA	3.31E-11	2.02E-11	2.22E-17
BA-142‡	2.13E-08	2.19E-11	1.34E-09	NO DATA	1.85E-11	1.24E-11	3.00E-26
LA-140	2.50E-09	1.26E-09	3.33E-10	NO DATA	NO DATA	NO DATA	9.25E-05
LA-142	1.28E-10	5.82E-11	1.45E-11	NO DATA	NO DATA	NO DATA	4.25E-07
CE-141	9.36E-09	6.33E-09	7.18E-10	NO DATA	2.94E-09	NO DATA	2.42E-05
CE-143‡	1.65E-09	1.22E-06	1.35E-10	NO DATA	5.37E-10	NO DATA	4.56E-05
CE-144‡	4.88E-07	2.04E-07	2.62E-08	NO DATA	1.21E-07	NO DATA	1.65E-04
PR-143	9.20E-09	3.69E-09	4.56E-10	NO DATA	2.13E-09	NO DATA	4.03E-05
PR-144	3.01E-11	1.25E-11	1.53E-12	NO DATA	7.05E-12	NO DATA	4.33E-18
ND-147‡	6.29E-09	7.27E-09	4.35E-10	NO DATA	4.25E-09	NO DATA	3.49E-05
W-187	1.03E-07	8.61E-08	3.01E-08	NO DATA	NO DATA	NO DATA	2.82E-05
NP-239	1.19E-09	1.17E-10	6.45E-11	NO DATA	3.65E-10	NO DATA	2.40E-05

TABLE 2.2-3
SITE RELATED INGESTION
DOSE COMMITMENT FACTOR, $A_{i\tau}$*
(mrem/hr per μCi/ml)
Page 1 of 2

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	NO DATA	8.96E+00	8.96E+00	8.96E+00	8.96E+00	8.96E+00	8.96E+00
C-14	3.15E+04	6.30E+03	6.30E+03	6.30E+03	6.30E+03	6.30E+03	6.30E+03
F-18	6.69E+01	NO DATA	7.42E+00	NO DATA	NO DATA	NO DATA	1.98E+00
NA-24	5.48E+02	5.48E+02	5.48E+02	5.48E+02	5.48E+02	5.48E+02	5.48E+02
P-32	4.62E+07	2.87E+06	1.79E+06	NO DATA	NO DATA	NO DATA	5.20E+06
CR-51	NO DATA	NO DATA	1.49E+00	8.94E-01	3.29E-01	1.98E+00	3.76E+02
MN-54	NO DATA	4.76E+03	9.08E+02	NO DATA	1.42E+03	NO DATA	1.46E+04
MN-56	NO DATA	1.20E+02	2.12E+01	NO DATA	1.52E+02	NO DATA	3.82E+03
FE-55	8.87E+02	6.13E+02	1.43E+02	NO DATA	NO DATA	3.42E+02	3.52E+02
FE-59	1.40E+03	3.29E+03	1.26E+03	NO DATA	NO DATA	9.19E+02	1.10E+04
CO-57	NO DATA	3.55E+01	5.91E+01	NO DATA	NO DATA	NO DATA	9.01E+02
CO-58	NO DATA	1.51E+02	3.39E+02	NO DATA	NO DATA	NO DATA	3.06E+03
CO-60	NO DATA	4.34E+02	9.58E+02	NO DATA	NO DATA	NO DATA	8.16E+03
NI-63	4.19E+04	2.91E+03	1.41E+03	NO DATA	NO DATA	NO DATA	6.07E+02
NI-65	1.70E+02	2.21E+01	1.01E+01	NO DATA	NO DATA	NO DATA	5.61E+02
CU-64	NO DATA	1.69E+01	7.93E+00	NO DATA	4.26E+01	NO DATA	1.44E+03
ZN-65	2.36E+04	7.50E+04	3.39E+04	NO DATA	5.02E+04	NO DATA	4.73E+04
ZN-69	5.02E+01	9.60E+01	6.67E+00	NO DATA	6.24E+01	NO DATA	1.44E+01
ZN-69m‡	8.28E+02	1.99E+03	1.82E+02	NO DATA	1.20E+03	NO DATA	1.21E+05
BR-82	NO DATA	NO DATA	2.46E+03	NO DATA	NO DATA	NO DATA	2.82E+03
BR-83‡	NO DATA	NO DATA	4.38E+01	NO DATA	NO DATA	NO DATA	6.30E+01
BR-84	NO DATA	NO DATA	5.67E+01	NO DATA	NO DATA	NO DATA	4.45E-04
BR-85	NO DATA	NO DATA	2.33E+00	NO DATA	NO DATA	NO DATA	1.09E-15
RB-86	NO DATA	1.03E+05	4.79E+04	NO DATA	NO DATA	NO DATA	2.03E+04
RB-88	NO DATA	2.95E+02	1.56E+02	NO DATA	NO DATA	NO DATA	4.07E-09
RB-89‡	NO DATA	1.95E+02	1.37E+02	NO DATA	NO DATA	NO DATA	1.13E-11
SR-89‡	4.78E+04	NO DATA	1.37E+03	NO DATA	NO DATA	NO DATA	7.66E+03
SR-90‡	1.18E+06	NO DATA	2.88E+05	NO DATA	NO DATA	NO DATA	3.48E+04
SR-91‡	8.79E+02	NO DATA	3.55E+01	NO DATA	NO DATA	NO DATA	4.19E+03
SR-92‡	3.33E+02	NO DATA	1.44E+01	NO DATA	NO DATA	NO DATA	6.60E+03
Y-90	1.38E+00	NO DATA	3.69E-02	NO DATA	NO DATA	NO DATA	1.46E+04
Y-91M‡	1.30E-02	NO DATA	5.04E-04	NO DATA	NO DATA	NO DATA	3.82E-02
Y-91	2.02E+01	NO DATA	5.39E-01	NO DATA	NO DATA	NO DATA	1.11E+04
Y-92	1.21E-01	NO DATA	3.53E-03	NO DATA	NO DATA	NO DATA	2.12E+03
Y-93	3.83E-01	NO DATA	1.06E-02	NO DATA	NO DATA	NO DATA	1.22E+04
ZR-95‡	2.77E+00	8.88E-01	6.01E-01	NO DATA	1.39E+00	NO DATA	2.82E+03
ZR-97‡	1.53E-01	3.09E-02	1.41E-02	NO DATA	4.67E-02	NO DATA	9.57E+03
NB-95	4.47E+02	2.49E+02	1.34E+02	NO DATA	2.46E+02	NO DATA	1.51E+06
NB-97	3.75E+00	9.49E-01	3.47E-01	NO DATA	1.11E+00	NO DATA	3.50E+03

‡Daughter contributions are included (see Reference 13).
*Calculated using equation (32) and Tables 2.2-1 and 2.2-2.

TABLE 2.2-3
SITE RELATED INGESTION
DOSE COMMITMENT FACTOR, $A_{i\tau}$*
(mrem/hr per μCi/ml)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
MO-99‡	NO DATA	4.62E+02	8.79E+01	NO DATA	1.05E+03	NO DATA	1.07E+03
TC-99M	2.94E-02	8.32E-02	1.06E+00	NO DATA	1.26E+00	4.07E-02	4.92E+01
TC-101	3.03E-02	4.36E-02	4.28E-01	NO DATA	7.85E-01	2.23E-02	1.31E-13
RU-103‡	1.98E+01	NO DATA	8.54E-01	NO DATA	7.57E+01	NO DATA	2.31E+03
RU-105‡	1.65E+00	NO DATA	6.52E-01	NO DATA	2.13E+01	NO DATA	1.01E+03
RU-106‡	2.95E+02	NO DATA	3.73E+01	NO DATA	5.69E+02	NO DATA	1.91E+04
AG-110M‡	1.42E+01	1.31E+01	7.80E+00	NO DATA	2.58E+01	NO DATA	5.36E+03
SB-124	2.40E+02	4.53E+00	9.50E+01	5.81E-01	NO DATA	1.87E+02	6.81E+03
SB-125‡	1.53E+02	1.71E+00	3.65E+01	1.56E-01	NO DATA	1.18E+02	1.69E+03
SB-126	9.85E+01	2.00E+00	3.55E+01	6.03E-01	NO DATA	6.04E+01	8.05E+03
SB-127	2.21E+01	4.84E-01	8.47E+00	2.65E-01	NO DATA	1.31E+01	5.05E+03
TE-125M	2.79E+03	1.01E+03	3.74E+02	8.39E+02	1.13E+04	NO DATA	1.11E+04
TE-127M‡	7.05E+03	2.52E+03	8.59E+02	1.80E+03	2.86E+04	NO DATA	2.36E+04
TE-127	1.14E+02	4.11E+01	2.48E+01	8.48E+01	4.66E+02	NO DATA	9.03E+03
TE-129M‡	1.20E+04	4.47E+03	1.89E+03	4.11E+03	5.00E+04	NO DATA	6.03E+04
TE-129	3.27E+01	1.23E+01	7.96E+00	2.51E+01	1.37E+02	NO DATA	2.47E+01
TE-131M‡	1.88E+03	8.81E+02	7.34E+02	1.39E+01	8.92E+03	NO DATA	8.74E+04
TE-131‡	2.05E+01	8.57E+00	6.47E+00	1.69E+01	8.98E+01	NO DATA	2.90E+00
TE-132‡	2.62E+03	1.70E+03	1.59E+03	1.87E+03	1.63E+04	NO DATA	8.02E+04
I-130	9.01E+01	2.66E+02	1.05E+02	2.25E+04	4.15E+02	NO DATA	2.29E+02
I-131‡	4.96E+02	7.09E+02	4.06E+02	2.32E+05	1.22E+03	NO DATA	1.87E+02
I-132	2.42E+01	6.47E+01	2.26E+01	2.26E+03	1.03E+02	NO DATA	1.22E+01
I-133‡	1.69E+02	2.94E+02	8.97E+01	4.32E+04	5.13E+02	NO DATA	2.64E+02
I-134	1.26E+01	3.43E+01	1.23E+01	5.94E+02	5.46E+01	NO DATA	2.99E-02
I-135‡	5.28E+01	1.38E+02	5.10E+01	9.11E+03	2.22E+02	NO DATA	1.56E+02
CS-134	3.03E+05	7.21E+05	5.89E+05	NO DATA	2.33E+05	7.75E+04	1.26E+04
CS-136	3.17E+04	1.25E+05	9.01E+04	NO DATA	6.97E+04	9.55E+03	1.42E+04
CS-137‡	3.88E+05	5.31E+05	3.48E+05	NO DATA	1.88E+05	5.99E+04	1.03E+04
CS-138	2.69E+02	5.31E+02	2.63E+02	NO DATA	3.90E+02	3.85E+01	2.27E-03
BA-139	9.00E+00	6.41E-03	2.64E-01	NO DATA	5.99E-03	3.64E-03	1.60E+01
BA-140‡	1.88E+03	2.37E+00	1.23E+02	NO DATA	8.05E-01	1.35E+00	3.88E+03
BA-141‡	4.27E+00	3.30E-03	1.48E-01	NO DATA	3.07E-03	1.87E-03	2.06E-09
BA-142‡	1.98E+00	2.03E-03	1.24E-01	NO DATA	1.72E-03	1.15E-03	2.78E-18
LA-140	3.58E-01	1.80E-01	4.76E-02	NO DATA	NO DATA	NO DATA	1.32E+04
LA-142	1.83E-02	8.33E-03	2.07E-03	NO DATA	NO DATA	NO DATA	6.08E+01
CE-141	8.01E-01	5.42E-01	6.15E-02	NO DATA	2.52E-01	NO DATA	2.07E+03
CE-143‡	1.41E-01	1.04E+02	1.16E-02	NO DATA	4.60E-02	NO DATA	3.90E+03
CE-144‡	4.18E+01	1.77E+01	2.24E+00	NO DATA	1.04E+01	NO DATA	1.41E+04
PR-143	1.32E+00	5.28E-01	6.52E-02	NO DATA	3.05E-01	NO DATA	5.77E+03
PR-144	4.31E-03	1.79E-03	2.19E-04	NO DATA	1.01E-03	NO DATA	6.19E-10
ND-147‡	9.00E-01	1.04E+00	6.22E-02	NO DATA	6.08E-01	NO DATA	4.99E+03
W-187	3.04E+02	2.55E+02	8.90E+01	NO DATA	NO DATA	NO DATA	8.34E+04
NP-239	1.28E-01	1.25E-02	6.91E-03	NO DATA	3.91E-02	NO DATA	2.57E+03

Figure 2.2
LIQUID RADWASTE TREATMENT SYSTEM

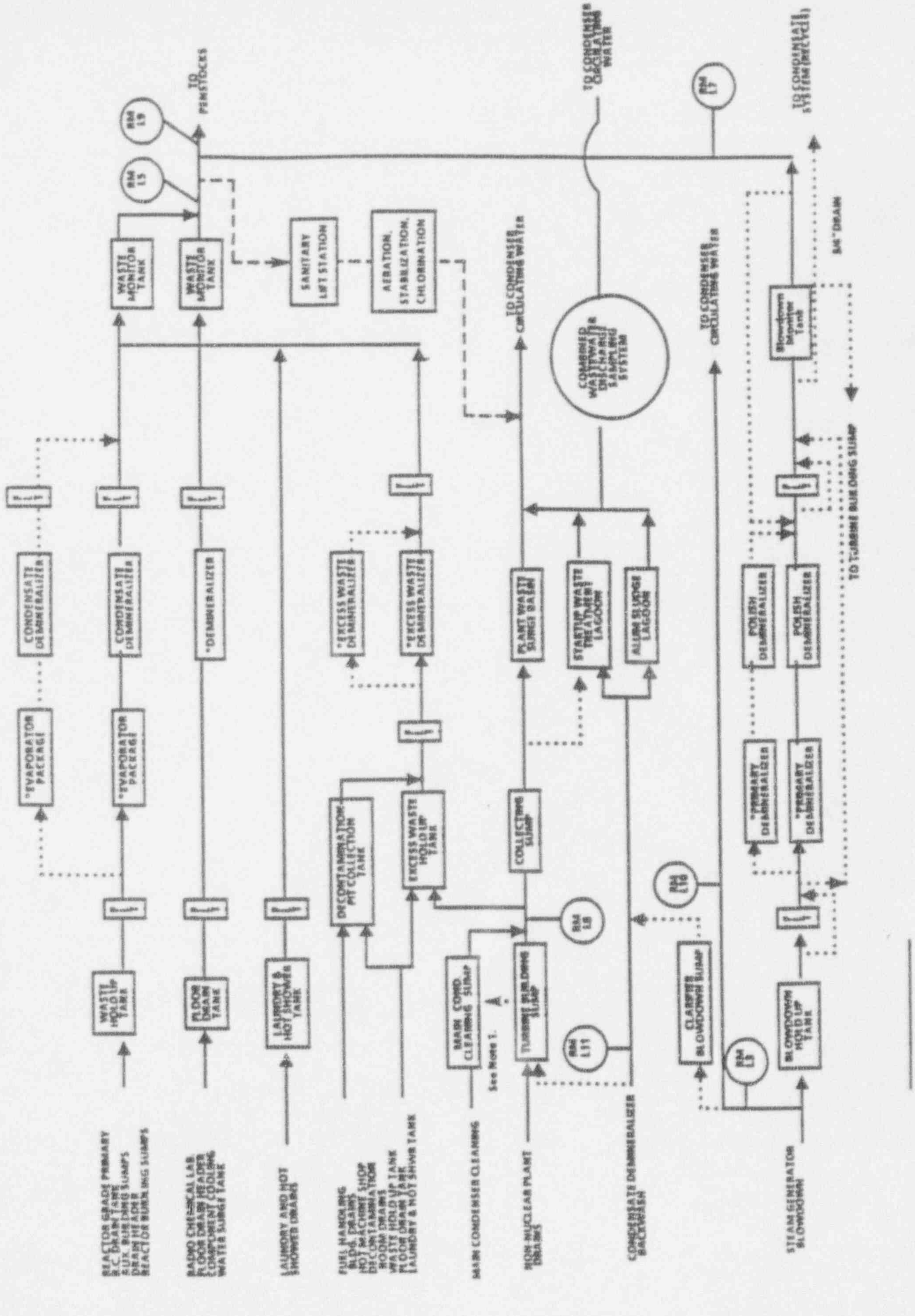


FIGURE 2.2-1

NOTES: 1. Turbine Building Sump contents may be processed to the main condenser cleaning sump through a portable demineralizer. This is an optional treatment pathway which provides processing flexibility in the event processing through excess liquid waste is not desirable. Since a temporary demineralizer is used for this optional treatment pathway, operability tests specified in ODCM specification 1.1.4.1 are not required. To ensure adequacy of the RM-L8 setpoint while using the alternate process pathway, samples must be obtained from the discharge side of the demineralizers or condenser cleaning sump and analyzed every eight hours.

Term	Definition	Section of Initial Use
S_d =	count rate of the waste gas decay system noble gas monitor at the alarm setpoint, in cpm.	(3.1.3)
S_v =	count rate of a station vent noble gas monitor at the alarm setpoint, in cpm.	(3.1.2)
S_{vc} =	count rate of the containment purge noble gas monitor at the alarm setpoint, in cpm.	(3.1.2)
S_{vp} =	count rate of the plant vent noble gas monitor at the alarm setpoint, in cpm.	(3.1.2)
X_{id} =	the concentration of noble gas radionuclide i in a waste gas decay tank, as corrected to the pressure of the discharge stream at the point of its flow measurement in uCi/cc.	(3.1.3)
X_{iv} =	the measured concentration of noble gas radionuclide i in the last grab sample analyzed for vent v in uCi/cc.	(3.1.2)
X_d' =	the total noble gas concentration in a waste gas decay tank, as corrected to the pressure of the discharge stream at the point of its flow measurement in uCi/cc.	(3.1.4)
X_v' =	a concentration of Xe-133 chosen to be in the operating range of the monitor on vent v in uCi/cc.	(3.1.4)
$\overline{X/Q}$ =	the highest annual average relative concentration in any sector, at the site boundary in sec/m^3.	(3.1.2)
1.1 =	mrem skin dose per mrad air dose	(3.1.2)
0.25 =	the safety factor applied to each of the two vent noble gas monitors (plant vent and containment purge) to assure that the sum of the releases has a combined safety factor of <u>0.5</u> which allows a 100 percent margin for cumulative uncertainties of measurements.	(3.1.2)

monthly. For the 6" and 36" containment purge lines, the sample is taken just prior to the release and also monthly, if the release is continuous.)

F_v = the flow rate in vent v, cc/sec. (1 cc/sec = 0.002119 cfm)

C_v = count rate, (cpm) of the monitor on station vent v corresponding to grab sample noble gas concentrations, X_{iv}, as determined from the monitor's calibration curve. i.e. product of the monitor response curve slope (^{cpm}/uCi/ml) and the sum of the noble gas concentrations in the grab sample (uCi/ml). (Initial calibration curves of the type shown in Figure 2.1-1 have been determined conservatively from families of response curves supplied by the monitor manufacturers. As releases occur, a historical correlation will be prepared and placed in service when sufficient data are accumulated.)

$\overline{X/Q}$ = the highest annual average relative concentration in any sector, at the site boundary (seven year average).

= 6.3E-6 sec/m^3 in the ENE sector.

K_i = total body dose factor due to gamma emissions from isotope i (mrem/yr per µCi/m^3) from Table 3.1-1.

L_i = skin dose factor due to beta emissions from isotope i (mrem/yr per µCi/m^3) from Table 3.1-1.

1.1 = mrem skin dose per mrad air dose.

M_i = air dose factor due to gamma emissions from isotope i (mrad/yr per µCi/m^3) from Table 3.1-1.

X_d' = the total concentration of noble gas radionuclides in the waste gas decay tank whose contents are to be discharged, as corrected to the pressure of the discharge stream at the point of the flow measurement.

c' = count rate in cpm of the waste gas decay system monitor corresponding to X_d' µCi/cc of Kr-85.

3.1.5 Oil Incineration

3.1.5.1 Releases from the oil incinerator will be limited such that

Eq. (60)

$$X/Q_{(oil)} \Sigma Pi\ Q_{(oil)} < 1500 \text{ mrem/yr.}$$

where:

$\overline{X/Q}_{(oil)}$ = highest annual average dispersion coefficient (sec/m^3) at the site boundary

= 3.3E-5 sec/m^3

P_i = dose parameter for radionuclide $_i$ for inhalation, from Table 3.2-1 (mrem / yr per uCi/m^3),

$\overline{\dot{Q}}_{(oil)}$ = $C_{i(oil)}$ X R

where:

$C_{i(oil)}$ = concentration of radionuclide $_i$ in oil (uCi/ml), and

R = burn rate (ml/s).

3.1.5.2 Incinerator operation will be administratively controlled such that the combination of gaseous releases from the station and oil incineration will be less than Specifications 1.2.2.1(b) and 1.2.5.1. If noble gases are detected in waste oil, an assessment of release acceptability should be performed using the general methodology described in sections 3.2.2.1 and 3.2.3.1.

3.1.6 Meteorological Release Criteria for Batch Releases

Planned gaseous batch releases (WGDT) and oil incineration will be performed during favorable meteorology. Limiting releases to favorable meteorology provides assurance that release conditions will be conservative with respect to annual average dispersion values ($\overline{X/Q}$, $\overline{X/Q}'$). Favorable meteorology is defined in Table 3.1-2.

D_β = air dose due to beta emissions from noble gas radionuclide i (mrad).

$$= 3.17 \times 10^{-8} \sum_i N_i \overline{X/Q}\, \tilde{Q}_i \quad (50)$$

where, N_i = air dose factor due to beta emission from noble gas radionuclide i (mrad/yr per uCi/m3) from Table 3.1-1.

3.2.3.2 For all gaseous effluents including oil incineration, dose to an individual from radioiodines and radioactive materials in particulate form and radionuclides (other than noble gases), with half-lives greater than eight (8) days (Calendar quarter: ≤7.5 mrem any organ, Calendar year: ≤15 mrem any organ) will be calculated for the purpose of implementation of section 1.2.4.1 as follows:

D_p = dose to an individual from radioiodines and radionuclides in particulate form, with half-lives greater than eight days (mrem)

$$= 3.17 \times 10^{-8} \sum_{ij} R_{ij}\, W_{ij}'\, \tilde{Q}_i' \quad (51)$$

where:

W_{ij}' = relative concentration or relative deposition for the maximum exposed individual, as appropriate for exposure pathway j and radionuclide i.

$\overline{X/Q}'$ for inhalation and all tritium pathways

= 3.5×10^{-6} sec/m^3

=

$\overline{D/Q}'$ for other pathways and non-tritium radionuclides

= 1.1×10^{-8} m^{-2}

(See the notes to Table 3.2-7 and 3.2-8 for the origin of these factors.)

NOTE: The controlling receptor in each sector was identified in the following way. Receptor locations and associated pathways were obtained from the August 1991 field survey. A child was assumed at each location, except that where a milk cow was listed, an infant was assumed. X/Q' and D/Q' for each candidate receptor was calculated using five year average meteorological data. XOQDOQ-82 software was used to analyze the meteo-rological data. Expected annual releases of each nuclide were taken from Table 5.2-2 of Reference 5. The specific dispersion values for each candidate are used with the methodology of ODCM section 3.2.3.2 to calculate a hypothetical dose. The controlling receptor for each sector was then chosen as the candidate receptor with the highest total annual dose of any candidate receptor in the given sector. All listed pathways are in addition to inhalation and ground plane exposure.

Table 3.2-8

ATMOSPHERIC DISPERSION PARAMETERS
FOR CONTROLLING RECEPTOR LOCATIONS*

SECTOR	X/Q'	D/Q'	DISTANCE (MILES/METERS)
N	2.3 E-7	6.3 E-10	3.8 / 6,100
NNE	2.9 E-7	8.5 E-10	3.3 / 5,300
NE	5.4 E-7	1.5 E-9	2.8 / 4,500
ENE	1.8 E-6	5.4 E-9	1.6 / 2,600
E	3.5 E-6	1.1 E-8	1.1 / 1,800
ESE	2.1 E-6	6.8 E-9	1.1 / 1,800
SE	6.5 E-7	2.4 E-9	1.5 / 2,400
SSE	1.2 E-7	5.3 E-10	2.7 / 4,300
S	7.6 E-8	3.5 E-10	3.9 / 6,300
SSW	1.2 E-7	7.0 E-10	3.4 / 5,500
SW	1.3 E-7	9.6 E-10	3.3 / 5,300
WSW	3.6 E-7	2.5 E-9	1.9 / 3,100
W	1.8 E-7	7.7 E-10	2.7 / 4,300
W	2.8 E-7	1.3 E-9	2.2 / 3,500
WNW	3.8 E-8	1.3 E-10	4.8 / 7,700
NW	9.8 E-8	2.8 E-10	4.1 / 6,600
NNW	3.3 E-7	9.0 E-10	3.0 / 4,800

* Annual average relative dispersion and deposition values for the receptor locations in Table 3.2-7. Values were calculated from 5 year averaged meteorological data using the XOQDOQ-82 software. Dispersion values were calculated assuming ground-level release, open terrain recirculation, dry depletion, and using decay with a half-life of 8.0 days. As a result of the analysis described in the note to Table 3.2-7, the location of the maximum exposed individual for the site is assumed to be the vegetable garden at 1.1 miles in the E sector. Therefore, the site X/Q' and D/Q' (Section 3.2.3.2 and following) are those from this table for that location.

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM
VIRGIL C. SUMMER NUCLEAR STATION
TABLE 4.0-1

Exposure Pathway and/or Sample	Criteria for Selection of Sample Number & Location	Sampling and Collection Frequency	Sample[1] Location	Locations Mi/Dir	Type & Frequency of Analysis
II. Radioiodine	A) 3 Indicator samples to be taken at two locations as given in I(A) above.	Continuous sampler operation with weekly canister collection.	2 5 10	1.2 SW 0.9 SE 2.5 NNE	Gamma Isotopic for I-131 weekly
	B) 1 Indicator sample to be taken at the location as given in I(B) above.	Continuous sampler operation with weekly canister collection.	6	1.0ESE	Gamma Isotopic for I-131 weekly
	C) 1 Indicator sample to be taken at the location as given in I(C) above.	Continuous sampler operation with weekly canister collection.	14	6.3 W	Gamma Isotopic for I-131 weekly
	D) 1 Control sample to be taken at a location similar in nature to I(D) above.	Continuous sampler operation with weekly canister collection.	17	24.7 SE	Gamma Isotopic for I-131 weekly
III. Direct	A) 13 Indicator stations to form an inner ring of stations in the 13 accessible sectors within 1 to 2 miles of the plant.	Monthly or quarterly exchange[5,7]; two or more dosimeters at each location.	1,2 3,4 5,6 7,8 9,10 29 30 47	1.2 S, 1.2 SW 1.2W,1.2 WNW 0.9 S E, 1.0 ESE 1.0 E, 1.5 ENE 2.2 NE, 2.5 NNE 0.9 WSW, 1.0 SSW 1.0 NW	Gamma dose monthly or quarterly.

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM
VIRGIL C. SUMMER NUCLEAR STATION
TABLE 4.0-1

Exposure Pathway and/or Sample	Criteria for Selection of Sample Number & Location	Sampling and Collection Frequency	Sample[1] Location	Locations Mi/Dir	Type & Frequency of Analysis
	E) 1 Control grass (forage) sample to be taken at the location of VII(B) above.	Monthly when available [5]	16	20.1 W	Gamma isotopic.
VIII. Food Products	A) 2 samples of broadleaf vegetation grown in the 2 nearest offsite locations of highest calculated annual average ground level D/Q if milk sampling is not performed within 3 km or if milk sampling is not performed at a location within 5-10 km where the doses are calculated to be greater than 1 mrem/yr.[10]	Monthly when available.[5]	6 7	1.0 ESE 1.0 E	Gamma Isotopic on edible portion.
	B) 1 Control sample for the same foods taken at a location at least 10 miles distance and not in the most prevalent wind direction if milk sampling is not performed within 3 km or if milk sampling is not performed at a location within 5-8 km where the doses are calculated to be greater than 1 mrem/yr.[10]	Monthly when available.[5]	18	16.5 S	Gamma Isotopic on edible portion.
IX. Fish	A) 1 Indicator sample to be taken at a location in the upper reservoir.	Semiannual[9] collection of the following specie types if available: bass; bream, crappie; catfish, carp; forage fish (shad).	23[3]	0.3-5	Gamma isotopic on edible portions semiannually.[9]