

**OFFSITE DOSE CALCULATION MANUAL
FOR
{CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 3}**

TABLE OF CONTENTS

	Page
TABLE OF CONTENTS	i
LIST OF TABLES	iv
LIST OF FIGURES	v
1.0 INTRODUCTION	1-1
1.1 Revisions to ODCM	1-1
2.0 DEFINITIONS	2-1
3.0 CONTROL APPLICABILITY	3-1
3.1 Instrumentation	3.1.1-1
3.1.1 Radioactive Liquid Effluent Monitoring Instrumentation	3.1.1-1
3.1.2 Radioactive Gaseous Effluent Monitoring Instrumentation	3.1.2-1
3.2 Radioactive Effluents	3.2.1-1-1
3.2.1 Liquid Effluents	3.2.1.1-1
3.2.2 Gaseous Effluents	3.2.2.1-1
3.2.3 Total Dose	3.2.3-1
3.3 Radiological Environmental Monitoring	3.2.3-1
3.3.1 Radiological Environmental Monitoring Program	3.3.1-1
3.3.2 Land Use Census	3.3.2-1
3.3.3 Interlaboratory Comparison Program	3.3.3-1
3.4 Radiological Effluent Controls Bases	3.4-1
4.0 REPORTING REQUIREMENTS	4.1-1
4.1 Annual Radiological Environmental Operating Report	4.1-1
4.2 Radioactive Effluent Release Report	4.2-1
4.3 Major Changes to Radioactive Liquid, Gaseous, and Solid Waste Treatment Systems	4.3-1
4.4 Ground Water Monitoring Reports	4.4-1
5.0 METHOD TO CALCULATE OFFSITE LIQUID CONCENTRATIONS	5.1-1
5.1 Method to Determine F_1^{ENG}	5.1-1
5.2 Method to Determine Radionuclide Concentration for Each Liquid Effluent Pathway	5.2-1
5.2.1 Liquid Waste Monitoring Tanks	5.2-1

TABLE OF CONTENTS (continued)

	Page
5.2.2 Turbine Building Drains	5.2-1
6.0 OFFSITE DOSE CALCULATION METHODS	6.1-1
6.1 Introductory Concepts.....	6.1-1
6.2 Method to Calculate the Total Body Dose from Liquid Releases	6.2-1
6.2.1 Method I	6.2-2
6.2.2 Method II	6.2-2
6.3 Method to Calculate Maximum Organ Dose from Liquid Releases.....	6.3-1
6.3.1 Method I	6.3-1
6.3.2 Method II	6.3-2
6.4 Method to Calculate the Total Body Dose Rate from Noble Gases	6.4-1
6.4.1 Method I	6.4-1
6.4.2 Method II	6.4-2
6.5 Method to Calculate the Skin Dose Rate from Noble Gases	6.5-1
6.5.1 Method I	6.5-1
6.5.2 Method II	6.5-2
6.6 Method to Calculate the Critical Organ Dose Rate from Iodines, Tritium, and Particulates with Half-Lives Greater than 8 Days.....	6.6-1
6.6.1 Method I	6.6-1
6.6.2 Method II	6.6-1
6.7 Method to Calculate Gamma Air Dose from Noble Gases.....	6.7-1
6.7.1 Method I	6.7-1
6.7.2 Method II	6.7-2
6.8 Method to Calculate Beta Air Dose from Noble Gases	6.8-1
6.8.1 Method I	6.8-1
6.8.2 Method II	6.8-2
6.9 Method to Calculate Critical Organ Dose from Iodines, Tritium, and Particulates	6.9-1
6.9.1 Method I	6.9-1
6.9.2 Method II	6.9-2
6.10 Receptor Points and Annual Average Atmospheric Dispersion Factors for Important Exposure Pathways.....	6.10-1
6.10.1 Receptor Locations	6.10-1
6.10.2 {Calvert Cliffs Nuclear Power Plant Unit 3 Atmospheric Dispersion Model.....	6.10-1
6.11 Method to Calculate Direct Dose from Plant Operations.....	6.11-1
6.12 Cumulative Dose	6.12-1
7.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM.....	7.1-1
7.1 {REMP Modifications CCNPP UNIT 3	7.1-1

TABLE OF CONTENTS (continued)

	Page
7.2 Ground Water Protection Program	7.1-2
8.0 SETPOINT DETERMINATIONS	8.1-1
8.1 Liquid Effluent Instrumentation Setpoints	8.1-1
8.1.1 Introduction.....	8.1-1
8.1.2 Liquid Radwaste Effluent Line Monitor and Setpoints	8.1-1
8.1.3 Turbine Building Effluent Line Monitor and Setpoints.....	8.1-4
8.1.4 Component Cooling System Water Monitor and Setpoints.....	8.1-4
8.2 Gaseous Effluent Instrumentation Setpoints	8.2-1
8.2.1 Introduction.....	8.2-1
8.2.2 Main Vent Stack Effluent Line Monitor and Setpoints.....	8.2-1
8.2.3 Delay Bed Monitor and Setpoint	8.2-8
8.2.4 Reactor Building Ventilation Monitor and Setpoint (KLA/KLB).....	8.2-8
8.2.5 Turbine Building Condenser Monitor and Setpoint	8.2-8
9.0 BASES FOR DOSE CALCULATION METHODOLOGIES	9.1-1
9.1 Liquid Release Dose Calculations.....	9.1-1
9.1.1 Dose to the Total Body.....	9.1-2
9.1.2 Dose to the Critical Organ.....	9.1-4
9.2 Gaseous Release Dose Calculations	9.2-1
9.2.1 Total Body Dose Rate from Noble Gases.....	9.2-1
9.2.2 Skin Dose Rate from Noble Gases	9.2-2
9.2.3 Critical Organ Dose Rate from Iodines, Tritium and Particulates with Half-Lives Greater than Eight Days.....	9.2-4
9.2.4 Gamma Dose to Air from Noble Gases.....	9.2-5
9.2.5 Beta Dose to Air from Noble Gases	9.2-7
9.2.6 Dose to Critical Organ from Iodines, Tritium and Particulates with Half-Lives Greater than Eight Days.....	9.2-8
10.0 REFERENCES	10-1

LIST OF TABLES

		Page
Table 2.0-1	MODES	2-6
Table 3.1.1-1	Radioactive Liquid Effluent Monitoring Instrumentation	3.1.1-3
Table 3.1.2-1	Radioactive Gaseous Effluent Monitoring Instrumentation	3.1.2-6
Table 3.2.1.1-1	Radioactive Liquid Waste Sampling and Analysis Program	3.2.1.1-3
Table 3.2.2.1-1	Radioactive Gaseous Waste Sampling and Analysis Program	3.2.2.1-3
Table 3.3.1-1	Radioactive Environmental Monitoring Program.....	3.3.1-6
Table 3.3.1-2	Reporting Levels for Radioactivity Concentrations in Environmental Samples(a).....	3.3.1-11
Table 3.3.1-3	Detection Capabilities for Environmental Sample Analysis(a) Lower Limit of Detection (LLD) (b)(c)	3.3.1-12
Table 6-1	{Dose Factors Specific for CCNPP Unit 3 Liquid Releases}	6.2-3
Table 6-2	{CCNPP Unit 3 Dose Factors for Noble Gas Releases}	6.4-3
Table 6-3	{Dose and Dose Rate Factors Specific for CCNPP Unit 3 for Iodine, Tritium or Particulate Releases}	6.6-3
Table 6-4	{Calvert Cliffs Nuclear Power Plant Unit 3 Site Boundary Distances}	6.10-3
Table 6-5	{Calvert Cliffs Nuclear Power Plant Unit 3 Long Term Average Dispersion Factors - Primary Vent Stack}.....	6.10-4
Table 7.1-1	{Existing CCNPP Units 1 and 2 Radiological Environmental Monitoring Program Locations}.....	7.1-4
Table 7.1-2	{Operational CCNPP Unit 3 Radiological Environmental Monitoring Program Locations}.....	7.1-5

LIST OF FIGURES

	Page
Figure 7.1-1	{CCNPP Unit 3 Sampling Locations Inside the Protected Area Boundary} 7.1-7
Figure 7.1-2	{CCNPP Sampling Locations 0-2 Miles (0-3.2 km)}..... 7.1-8
Figure 7.1-3	{CCNPP Sampling Locations 0-10 Miles (0-16 km)}..... 7.1-9
Figure 8.1-1	{Liquid Radwaste Effluent Flow Paths} 8.1-6
Figure 8.2-1	Gaseous Radwaste Effluent Flow Paths..... 8.2-10

1.0 INTRODUCTION

The Offsite Dose Calculation Manual (ODCM) describes the methodology and parameters used in the calculation of offsite doses from radioactive liquid and gaseous effluents, in the calculation of liquid and gaseous effluent monitoring instrumentation alarm/ trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program (REMP). The ODCM also includes the Radiological Effluent Controls Program required by Technical Specifications 5.5.3 at {Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 3} for the purpose of ensuring that the amount of radioactive materials released to the environment are as low as reasonable achievable (ALARA). The ODCM also describes the environmental monitoring activities. Requirements are also established for the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required the {CCNPP Unit 3} Technical Specifications. The methodology and parameters described in the ODCM utilize guidance provided in NUREG-0133 (Reference [1]), Regulatory Guide 1.109 (Reference [2]), and Generic Letter 89-01 (Reference [3]). This ODCM has been prepared following the guidance provided in the draft template NEI 07-09 (Reference [4]). In addition, the Radiological Effluent Controls have been formatted and revised to be consistent with the {CCNPP Unit 3} Technical Specifications and plant-specific design.

The Radiological Effluent Control requirements are provided in ODCM Sections 2.0 through 4.0. The provisions of {CCNPP Unit 3} Technical Specifications Section 1.2, "Logical Connectors," Section 1.3, "Completion Times," and Section 1.4, "Frequency," are applicable to the Radiological Effluent Controls Requirements.

1.1 REVISIONS TO ODCM

All changes to the ODCM will be documented and records of reviews performed will be retained. This documentation will contain sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent dose or setpoint calculations. All changes to the ODCM must be approved by the Plant Manager prior to implementation. All approved changes shall be submitted to the NRC for their information in the Radioactive Effluent Release Report for the period in which the change(s) was made effective in the form of a complete, legible copy of the entire ODCM. Each change will be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and will indicate the date (i.e., month and year) the change was implemented. The plant's Records Management/Document Control Manager shall maintain the current version of the ODCM and issue under controlled distribution all approved changes to it.

2.0 DEFINITIONS

-----NOTE-----

The defined terms in this section appear in capitalized type and shall be applicable throughout the Radiological Effluent Controls.

<u>Term</u>	<u>Definition</u>
ACTUATING DEVICE OPERATIONAL TEST (ADOT)	An ADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the division required for trip actuating device OPERABILITY. The ADOT may be performed by means of any series of sequential, overlapping, or total division steps.
CALIBRATION	A CALIBRATION shall be the adjustment, as necessary, of the sensor output such that it responds within the necessary range and accuracy to known values of the parameter that the sensor monitors. The CALIBRATION shall encompass all devices in the division required for sensor OPERABILITY. The CALIBRATION may be performed by means of any series of sequential, overlapping, or total steps.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status to other indications or status derived from independent instrumentation channels measuring the same parameter.
COMPENSATORY MEASURES	COMPENSATORY MEASURES shall be part of a Radiological Effluent Control (i.e., Control) that prescribes Required Compensatory Measures to be taken under designated Conditions within specified Completion Times.
COMPOSITE SAMPLE	A COMPOSITE SAMPLE is a combination of individual samples obtained at intervals that are very short (e.g., hourly) in relation to the compositing time interval (e.g., monthly) to assure obtaining a representative sample. The sample volume should be proportionate to the volume, either liquid or gas, flowing through the system where practical.

2.0 DEFINITIONS

DOSE EQUIVALENT I-131 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

GAMMA ISOTOPIC ANALYSIS A GAMMA ISOTOPIC ANALYSIS is an analytical method of measurement used for identification and quantification of gamma-emitting radionuclides.

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

LIQUID WASTE PROCESSING SYSTEM A LIQUID WASTE PROCESSING SYSTEM is the system installed and designed to reduce radioactive liquid effluents.

LOWER LIMIT OF DETECTION (LLD) The LLD is defined, for purposes of these Radiological Effluent Controls, as 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 with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system which may include radiochemical separation:

$$LLD = \frac{4.66 S_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta \tau)}$$

where:

LLD is the "a priori" lower limit of detection as defined above as microcuries per unit mass or volume,

4.66 is a constant derived from K_{α} and K_{β} 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

2.0 DEFINITIONS

counts per minute,

E is the counting efficiency as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield when applicable,

λ is the radioactive decay constant for the particular radionuclides and

$\Delta\tau$ for samples is the lapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and $\Delta\tau$ should be used in the calculation.

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.

MEMBER(S) OF THE PUBLIC

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

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 2.0-1 with fuel in the reactor vessel.

OFFSITE DOSE
CALCULATION MANUAL
(ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and fixed setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain the radioactive

2.0 DEFINITIONS

	effluent controls, radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Environmental Operating and Radioactive Effluent Release Reports.
OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PROCESS CONTROL PROGRAM (PCP)	The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid waste will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State and Local regulations governing the disposal of the radioactive waste.
PURGE - PURGING	PURGE - PURGING shall be the controlled process of discharging air or gas from a containment to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the containment.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 4590 MWt.
SENSOR OPERATIONAL TEST (SOT)	A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the input circuit required for OPERABILITY. The SOT shall include the verification of the accuracy and time constants of the analog input modules. The SOT may be performed by means of any series of sequential, overlapping, or total steps.
SITE BOUNDARY	The SITE BOUNDARY shall be that line beyond which the land is neither owned, leased, nor otherwise controlled by the licensee.
SOURCE CHECK	A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

2.0 DEFINITIONS

UNRESTRICTED AREA	An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY for which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the site boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.
VENTILATION EXHAUST TREATMENT SYSTEM	A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.
VENTING	VENTING shall be the controlled process of discharging air or gas from a containment to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

Table 2.0-1 (Page 1 of 1)

MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER ^(a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown ^(b)	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown ^(b)	< 0.99	NA	≤ 200
6	Refueling ^(c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

3.0 CONTROL APPLICABILITY

CONTROL 3.0.1 Controls shall be met during the MODES or other specified conditions in the Applicability, except as provided in CONTROL 3.0.2.

CONTROL 3.0.2 Upon discovery of a failure to meet a Control, the Required Compensatory Measures of the associated Conditions shall be met, except as provided in CONTROL 3.0.4.

If the Control is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Compensatory Measure(s) is not required unless otherwise stated.

CONTROL 3.0.3 When a Control is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

- a. When the associated COMPENSATORY MEASURES to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; or
- b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Control are stated in the individual Controls; or
- c. When an allowance is stated in the individual value, parameter, or other Control.

This Control shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with COMPENSATORY MEASURES.

CONTROL 3.0.4 Equipment removed from service or declared inoperable to comply with COMPENSATORY MEASURES may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to CONTROL 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual Controls, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Control. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the Control except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this requirement are stated in the individual Controls.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the Control not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Control must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Control must immediately be declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of a Control shall only be made when the Control's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When a Control is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with CONTROL 3.0.3.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with COMPENSATORY MEASURES.

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
C. As required by Compensatory Measure A.1 and referenced in Table 3.1.1-1.	C.1 Analyze a grab sample for radioactivity (beta or gamma) of the associated pathway. The LLD shall be as specified in Table 3.2.1-1.	Once per 12 hours
	<u>AND</u> C.2 Restore the channel to OPERABLE status.	30 days
D. As required by Compensatory Measure A.1 and referenced in Table 3.1.1-1.	D.1 -----NOTE----- Pump performance curves generated in place may be used to estimate flow rate. ----- Estimate the flow rate through the associated pathway.	At the beginning of the release and once per 4 hours during releases through the associated line
	<u>AND</u> D.2 Restore the channel to OPERABLE status.	30 days
E. Required Compensatory Measure B.1, B.2, C.1, or D.1 and associated Completion Time not met.	E.1 Suspend releases through the associated pathway.	Immediately
F. Required Compensatory Measure B.3, C.2, or D.2 and associated Completion Time not met.	F.1 Prepare and submit, in the Radioactive Effluent Release Report, the reason the channel was not restored to operable status within 30 days.	Upon submittal of current calendar year Radioactive Effluent Release Report

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.1.1.2	Perform SOURCE CHECK.	Prior to each radioactive release
SR 3.1.1.3	Perform SOURCE CHECK.	31 days
SR 3.1.1.4	Perform SENSOR OPERATIONAL TEST.	92 days
SR 3.1.1.5	<p style="text-align: center;">-----NOTE-----</p> <p>The ACTUATING DEVICE OPERATIONAL TEST shall demonstrate, for Function 1, that automatic isolation of the associated pathway and Control Room alarm annunciation occurs and shall demonstrate, for Function 3, that Control Room alarm annunciation occurs if any of the following condition(s) exists:</p> <ul style="list-style-type: none"> • Instrument indicates measured levels above the Alarm/Trip setpoint. • Circuit failure. • Instrument indicates a downscale failure. • Instrument controls not set in operate mode. <p style="text-align: center;">-----</p> <p>Perform ACTUATING DEVICE OPERATIONAL TEST.</p>	92 days
SR 3.1.1.6	<p style="text-align: center;">-----NOTE-----</p> <p>For Functions 1, 2, and 3, the initial CALIBRATION shall be performed using one or more of the reference standards traceable to NIST or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system within its intended range of energy and measurement range. For subsequent CALIBRATION, sources that have been related to the initial CALIBRATION can be used.</p> <p style="text-align: center;">-----</p> <p>Perform CALIBRATION.</p>	18 months

Table 3.1.1-1

Radioactive Liquid Effluent Monitoring Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM COMPENSATORY MEASURE A.1	SURVEILLANCE REQUIREMENTS	ALARM/ TRIP SETPOINT
1. Liquid Radwaste Effluent Line Gross Radioactivity Monitor	(a)	1	B	SR 3.1.1.1 SR 3.1.1.2 SR 3.1.1.4 SR 3.1.1.5 SR 3.1.1.6	(b)
2. Turbine Building Drains Effluent Line Gross Radioactivity Monitor	(c)	1	C	SR 3.1.1.1 SR 3.1.1.3 SR 3.1.1.4 SR 3.1.1.5 SR 3.1.1.6	(b)
3. Component Cooling Water System Gross Radioactivity Monitor	(c)	1 per train	C	SR 3.1.1.1 SR 3.1.1.3 SR 3.1.1.4 SR 3.1.1.5 SR 3.1.1.6	(b)
4. Liquid Radwaste Effluent Line Flow Rate Monitor	(a)	1	D	SR 3.1.1.1 SR 3.1.1.4 SR 3.1.1.6	(b)
5. Turbine Building Drains Effluent Line Flow Rate Monitor	(a)	1	D	SR 3.1.1.1 SR 3.1.1.4 SR 3.1.1.6	(b)

- (a) When radioactive effluents are being discharged through this pathway.
- (b) The alarm/trip setpoint shall be set to ensure the limits of CONTROL 3.2.1.1 are not exceeded. The alarm/trip setpoint shall be determined and adjusted in accordance with the methodology and parameters described in the ODCM Section 8.1.
- (c) When there is flow in the system identified by this function

3.1 INSTRUMENTATION

3.1.2 Radioactive Gaseous Effluent Monitoring Instrumentation

CONTROL 3.1.2.1 The radioactive gaseous effluent monitoring instrumentation channels in Table 3.1.2-1 shall be OPERABLE.

APPLICABILITY: In accordance with Table 3.1.2-1.

COMPENSATORY MEASURES

NOTE

Separate condition entry is allowed for each channel.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Enter the condition referenced in Table for the channel.	Immediately
B. As required by Compensatory Measure A.1 and referenced in Table 3.1.2-1.	B.1 Verify the Condition C Compensatory Measures are being met.	Immediately
	<u>OR</u> B.2.1 Verify equivalent radiation monitor is OPERABLE	Immediately
	<u>AND</u> B.2.2 Restore the channel to OPERABLE status.	30 days

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
C. As required by Compensatory Measure B.1.	C.1 Take a grab sample at the associated sample location.	12 hours <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> C.2 Analyze the grab sample for gross activity. The LLD shall be as specified in Table 3.2.2.1-1.	24 hours after completion of Required Compensatory Measure C.1
	<u>AND</u> C.3 Restore the channel to OPERABLE status.	30 days
D. As required by Compensatory Measure A.1 and referenced in Table 3.1.2-1.	D.1 Estimate the flow rate through the associated pathway.	4 hours <u>AND</u> Once per 4 hours thereafter
	<u>AND</u> D.2 Restore the channel to OPERABLE status.	30 days
E. As required by Compensatory Measure A.1 and referenced in Table 3.1.2-1.	E.1 Establish auxiliary sampling equipment to continuously collect samples from the associated effluent release pathway as required in Table 3.2.2-1.	4 hours <u>AND</u> Continuously thereafter
	<u>AND</u> E.2 Restore the channel to OPERABLE status.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.1 -----NOTE----- For Functions 1.d(1) and 1.d(2), CHANNEL CHECK shall consist of verifying indication of flow. ----- Perform CHANNEL CHECK.	24 hours
SR 3.1.2.2 Perform SOURCE CHECK.	24 hours
SR 3.1.2.3 Perform CHANNEL CHECK.	7 days
SR 3.1.2.4 Perform SOURCE CHECK.	31 days
SR 3.1.2.5 Perform SENSOR OPERATIONAL TEST.	92 days
SR 3.1.2.6 -----NOTE----- The ACTUATING DEVICE OPERATIONAL TEST shall demonstrate, for Function 3, that automatic isolation of the associated pathway and Control Room alarm annunciation occurs and shall demonstrate for Functions 1.a, 2.a, and 4.a, that Control Room alarm annunciation occurs if any of the following condition(s) exists: <ul style="list-style-type: none"> • Instrument indicates measured levels above the Alarm/Trip setpoint. • Circuit failure. • Instrument indicates a downscale failure. • Instrument controls not set in operate mode. ----- Perform ACTUATING DEVICE OPERATIONAL TEST.	92 days

SUREVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.2.7</p> <p>-----NOTE-----</p> <p>For Functions 1.a, 2.a, 3, and 4.a, the initial CALIBRATION shall be performed using one or more of the reference standards traceable to NIST or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system within its intended range of energy and measurement range. For subsequent CALIBRATION, sources that have been related to the initial calibration can be used.</p> <p>-----</p> <p>Perform CALIBRATION.</p>	<p>18 months</p>

Table 3.1.2-1 (Page 1 of 2)

Radioactive Gaseous Effluent Monitoring Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM COMPENSATORY MEASURE A.1	SURVEILLANCE REQUIREMENTS	ALARM/ TRIP SETPOINT
1. Main Vent Stack					
a. Noble Gas Activity Monitor	(a)	1	B	SR 3.1.2.1 SR 3.1.2.4 SR 3.1.2.5 SR 3.1.2.6 SR 3.1.2.7	(b)
b. Particulate Sampler	(a)	1	E	SR 3.1.2.3	N/A
c. Iodine Sampler	(a)	1	E	SR 3.1.2.3	N/A
d. Flow Rate Measurement Device					
(1) Main Vent Stack	(a)	1	D	SR 3.1.2.1 SR 3.1.2.5 SR 3.1.2.7	(b)
(2) Sample Line Flow	(a)	1	D	SR 3.1.2.1 SR 3.1.2.5 SR 3.1.2.7	(b)
2. Gas Treatment System (Delay Beds)					
a. Noble Gas Activity Monitor	(c)	1	G	SR 3.1.2.1 SR 3.1.2.2 SR 3.1.2.5 SR 3.1.2.6 SR 3.1.2.7	(b)
b. Flow Rate Measurement Device	(c)	1	D	SR 3.1.2.1 SR 3.1.2.5 SR 3.1.2.7	(b)
3. Reactor Building Ventilation System Gamma Activity Monitor					
	(d)	1	F	SR 3.1.2.1 SR 3.1.2.2 SR 3.1.2.5 SR 3.1.2.6 SR 3.1.2.7	(b)

- (a) At all times.
- (b) The alarm/trip setpoint shall be set to ensure the limits of CONTROL 3.2.2.1 are not exceeded. The alarm/trip setpoint shall be determined and adjusted in accordance with the methodology and parameters described in the ODCM Section 8.2.
- (c) Whenever the flowpath is unisolated.
- (d) During Reactor Building Ventilation system operation.

Table 3.1.2-1 (Page 2 of 2)

Radioactive Gaseous Effluent Monitoring Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM COMPENSATORY MEASURE A.1	SURVEILLANCE REQUIREMENTS	ALARM/ TRIP SETPOINT
4. Turbine Building Condenser Air Evacuation					
a. Noble Gas Activity Monitor	(c)	1	B	SR 3.1.2.1 SR 3.1.2.4 SR 3.1.2.5 SR 3.1.2.6 SR 3.1.2.7	(b)
b. Flow Rate Measurement Device	(c)	1	D	SR 3.1.2.1 SR 3.1.2.5 SR 3.1.2.7	(b)

- (b) The alarm/trip setpoint shall be set to ensure the limits of CONTROL 3.2.2.1 are not exceeded. The alarm/trip setpoint shall be determined and adjusted in accordance with the methodology and parameters described in the ODCM Section 8.2.
- (c) Whenever the flowpath is unisolated.

3.2 RADIOACTIVE EFFLUENTS

3.2.1 Liquid Effluents

3.2.1.1 Liquid Concentration

CONTROL 3.2.1.1 The concentration of radioactive material released in liquid effluents at the point of discharge {from the multi-port diffuser} shall be limited to ten times the concentration specified in Table 2, Column 2 of Appendix B to 10 CFR Part 20 for radionuclides other than dissolved or entrained noble gases.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Concentration of radioactive material released in liquid effluent at the point of discharge {from the multi-port diffuser} not within limits.	A.1 Initiate action to restore concentration to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1.1	The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 3.2.1.1-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of CONTROL 3.2.1.1.	In accordance with Table 3.2.1.1-1
SR 3.2.1.1.2	Post-release analysis of samples composited from batch releases shall be performed in accordance with Table 3.2.1.1-1. The results of the post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of CONTROL 3.2.1.1.	In accordance with Table 3.2.1.1-1

Table 3.2.1.1-1 (Page 1 of 3)

Radioactive Liquid Waste Sampling and Analysis Program

LIQUID RELEASE TYPE	SAMPLE FREQUENCY	MINIMUM SAMPLE ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD)	
1. Liquid Radwaste Monitoring Tanks	Prior to release of each batch ^(a)	Prior to release of each batch	Principal Gamma Emitters ^(b)	$5 \times 10^{-7} \mu\text{Ci/ml}$	
			I-131	$1 \times 10^{-6} \mu\text{Ci/ml}$	
	Prior to release of each batch ^(a)	31 days	Dissolved and entrained gases (gamma emitters)		$1 \times 10^{-5} \mu\text{Ci/ml}$
				<u>AND</u>	
	One batch ^(a) every 31 days				
	Prior to release of each batch ^(a)	31 days	COMPOSITE SAMPLE	H-3	$1 \times 10^{-5} \mu\text{Ci/ml}$
Prior to release of each batch ^(a)	31 days	COMPOSITE SAMPLE	Gross Alpha	$1 \times 10^{-7} \mu\text{Ci/ml}$	
2. Turbine Building Drains Effluent Line	7 days Grab Sample	7 days COMPOSITE SAMPLE	Principal Gamma Emitters ^(b)	$5 \times 10^{-7} \mu\text{Ci/ml}$	
			I-131	$1 \times 10^{-6} \mu\text{Ci/ml}$	
	31 days Grab Sample	31 days	Dissolved and entrained gases (gamma emitters)		$1 \times 10^{-5} \mu\text{Ci/ml}$
	7 days Grab Sample	31 days COMPOSITE SAMPLE	H-3		$1 \times 10^{-5} \mu\text{Ci/ml}$
				Gross Alpha	$1 \times 10^{-7} \mu\text{Ci/ml}$
	7 days Grab Sample	92 days COMPOSITE SAMPLE	Sr-89, Sr-90		$5 \times 10^{-8} \mu\text{Ci/ml}$
				Fe-55	$1 \times 10^{-6} \mu\text{Ci/ml}$

- (a) Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling.
- (b) The principal gamma emitters for which the LLD control applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6} . This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report.

Table 3.2.1.1-1 (page 2 of 3)

Radioactive Liquid Waste Sampling and Analysis Program

LIQUID RELEASE TYPE	SAMPLE FREQUENCY	MINIMUM SAMPLE ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD)
3. Component Cooling Water System	7 days Grab Sample	7 days	Principal Gamma Emitters ^(b)	$5 \times 10^{-7} \mu\text{Ci/ml}$
			I-131	$1 \times 10^{-6} \mu\text{Ci/ml}$
	7 days Grab Sample	31 days	Dissolved and entrained gases (gamma emitters)	$1 \times 10^{-5} \mu\text{Ci/ml}$
			H-3	$1 \times 10^{-5} \mu\text{Ci/ml}$
			Gross Alpha	$1 \times 10^{-7} \mu\text{Ci/ml}$
7 days Grab Sample	92 days	Sr-89, Sr-90	$5 \times 10^{-8} \mu\text{Ci/ml}$	
		Fe-55	$1 \times 10^{-6} \mu\text{Ci/ml}$	
4. Essential Service Water System ^(f)	7 days Grab Sample ^{(c)(d)(e)}	7 days ^{(d)(e)}	Principal Gamma Emitters ^(b)	$5 \times 10^{-7} \mu\text{Ci/ml}$
			I-131	$1 \times 10^{-6} \mu\text{Ci/ml}$
	7 days Grab Sample ^(c)	31 days	Dissolved and entrained gases (gamma emitters)	$1 \times 10^{-5} \mu\text{Ci/ml}$
			H-3	$1 \times 10^{-5} \mu\text{Ci/ml}$
	7 days Grab Sample ^(c)	31 days	Gross Alpha	$1 \times 10^{-7} \mu\text{Ci/ml}$
			Sr-89, Sr-90	$5 \times 10^{-8} \mu\text{Ci/ml}$
7 days Grab Sample ^(c)	92 days	Fe-55	$1 \times 10^{-6} \mu\text{Ci/ml}$	

(b) The principal gamma emitters for which the LLD control applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6} . This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report.

(c) A grab sample can be considered as a combination of aliquots taken from each Essential Service Water System train during the same collection cycle or as individual samples taken from each train in service.

(d) Whenever Component Cooling Water System activity exceeds $1 \times 10^{-3} \mu\text{Ci/cc}$, the Essential Service Water System shall be sampled and analyzed once per 24 hours for principal gamma emitters.

(e) With a confirmed leak between the Component Cooling Water System and the Essential Service Water System and Component Cooling Water System activity in excess of $1 \times 10^{-4} \mu\text{Ci/cc}$, the Essential Service Water System shall be sampled and analyzed once per 12 hours for principal gamma emitters.

(f) The setpoint on the Component Cooling Water System head tank liquid rate-of-change alarm shall be set to ensure that its sensitivity to detect a leak between the Component Cooling Water System and the Essential Service Water System is equal to or greater than that of an Essential Service Water System radiation monitor, located in the unit's Essential Service Water System discharge, with an LLD of $1 \times 10^{-8} \mu\text{Ci/cc}$. If this sensitivity cannot be achieved, the Essential Service Water System will be sampled and analyzed once per 12 hours for principal gamma emitters.

Table 3.2.1.1-1 (page 3 of 3)

Radioactive Liquid Waste Sampling and Analysis Program

LIQUID RELEASE TYPE	SAMPLE FREQUENCY	MINIMUM SAMPLE ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD)
5. Subsurface Dewatering	31 days Grab Sample	31 days	Principal Gamma Emitters ^(b)	$5 \times 10^{-7} \mu\text{Ci/ml}$
			H-3	$2 \times 10^{-6} \mu\text{Ci/ml}$
	31 days Grab Sample	31 days	Gross Alpha	$1 \times 10^{-7} \mu\text{Ci/ml}$

(b) The principal gamma emitters for which the LLD control applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6} . This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report.

3.2 RADIOACTIVE EFFLUENTS

3.2.1 Liquid Effluents

3.2.1.2 Liquid Dose

CONTROL 3.2.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from {CCNPP Unit 3} to UNRESTRICTED AREAS shall be limited to:

- a. ≤ 1.5 mrem to the whole body and ≤ 5 mrem to any organ during any calendar quarter; and
- b. ≤ 3 mrem to the whole body and ≤ 10 mrem to any organ during any calendar year.

APPLICABILITY: When radioactive liquid effluents are released.

COMPENSATORY MEASURES

-----NOTE-----
Enter applicable Conditions and Required Compensatory Measures of CONTROL 3.2.3, "Total Dose," when the calculated dose from liquid effluents exceeds twice the limits of CONTROL 3.2.1.2.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Calculated dose from the release of radioactive materials in liquid effluents to UNRESTRICTED AREAS not within limits.	A.1 Submit a Special Report to the NRC that identifies causes for exceeding limits, corrective actions taken to reduce releases, corrective actions to assure that subsequent releases will be in compliance with the required limits.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2.1 Verify the cumulative dose contributions from radioactive liquid effluents for the current calendar quarter and current calendar year are within limits in accordance with the methodology and parameters in the ODCM.	31 days

3.2 RADIOACTIVE EFFLUENTS

3.2.1 Liquid Effluents

3.2.1.3 Liquid Waste Processing System

CONTROL 3.2.1.3 The LIQUID WASTE PROCESSING SYSTEM shall be OPERABLE. Appropriate portions of the system shall be used to reduce the radioactive materials in liquid waste prior to discharge when the projected doses due to the liquid effluent, from {CCNPP Unit 3} to UNRESTRICTED AREAS, would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
<p>A. Liquid waste being discharged without treatment when CONTROL 3.2.1.3 limits are exceeded.</p>	<p>A.1 Submit a Special Report to the NRC that includes explanation of why the LIQUID WASTE PROCESSING SYSTEM was not OPERABLE and why liquid radwaste was being discharged without treatment. The Special Report shall identify any required inoperable equipment and the reasons for the inoperability, the corrective actions taken to restore the required inoperable equipment to OPERABLE status, and the corrective actions to prevent recurrence.</p>	<p>30 days</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.1.3.1</p> <p>-----NOTE-----</p> <p>Not required to be performed when the LIQUID WASTE PROCESSING SYSTEM is being fully utilized to reduce radioactive material in liquid waste.</p> <p>-----</p> <p>Dose due to liquid releases to UNRESTRICTED AREAS shall be projected in accordance with the methodology and parameters in the ODCM.</p>	<p>31 days</p>
<p>SR 3.2.1.3.2</p> <p>The installed LIQUID WASTE PROCESSING SYSTEM shall be demonstrated OPERABLE by meeting CONTROLS 3.2.1.1 and 3.2.1.2.</p>	<p>In accordance with CONTROLS 3.2.1.1 and 3.2.1.2.</p>

3.2 RADIOACTIVE EFFLUENTS

3.2.2 Gaseous Effluents

3.2.2.1 Dose Rate

CONTROL 3.2.2.1 The dose rate at and beyond the SITE BOUNDARY due to radioactive materials released in gaseous effluents from the site shall be limited to the following:

- a. For nobles gases, < 500 mrem per year to the whole body and < 3000 mrem per year to the skin; and
- b. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days, < 1500 mrem per year to any organ.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Dose rate from the release of radioactive materials in gaseous effluents at and beyond the SITE BOUNDARY not within limits.	A.1 Initiate action to restore dose rate to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1.1	Verify the dose rate due to noble gases in gaseous effluents is within limits in accordance with methodology and parameters in the ODCM.	In accordance with Table 3.2.2.1-1
SR 3.2.2.1.2	Verify dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents is within limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses.	In accordance with Table 3.2.2.1-1

Table 3.2.2.1-1 (Page 1 of 1)

Radioactive Gaseous Waste Sampling and Analysis Program

GASEOUS RELEASE TYPE	SAMPLE FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD)	
1. Main Vent Stack	31 days ^{(a)(b)} Grab Sample	31 days	Principal Gamma Emitters ^(c)	1 X 10 ⁻⁴ μCi/ml	
			H-3	1 X 10 ⁻⁶ μCi/ml	
	Continuous ^(d)	7 days ^(e) Charcoal Sample	I-131	1 X 10 ⁻¹² μCi/ml	
			Principal Gamma Emitters ^(c)	1 X 10 ⁻¹¹ μCi/ml	
	Continuous ^(d)	31 days COMPOSITE SAMPLE Particulate Sample	Gross Alpha	1 X 10 ⁻¹¹ μCi/ml	
			Sr-89, Sr-90	1 X 10 ⁻¹¹ μCi/ml	
	Continuous ^(d)	92 days COMPOSITE SAMPLE Particulate Sample	Noble Gases Gross Beta or Gamma	1 X 10 ⁻⁶ μCi/ml	
			Monitor	1 X 10 ⁻⁴ μCi/ml	
	2. Condenser Air Removal/ Turbine Gland Seal Exhaust	31 days ^(f) Grab Sample	31 days ^(f) Noble Gases	Principal Gamma Emitters ^(c)	1 X 10 ⁻⁴ μCi/ml
				H-3	1 X 10 ⁻⁶ μCi/ml
3. Containment PURGE	Prior to each Containment PURGE ^(a) Grab Sample	Prior to each Containment PURGE	Principal Gamma Emitters ^(c)	1 X 10 ⁻⁴ μCi/ml	
			H-3	1 X 10 ⁻⁶ μCi/ml	

- (a) Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 % of RATED THERMAL POWER within a one hour period unless: 1) analysis shows that the DOSE EQUIVALENT I-131 concentrations in the primary coolant has not increased more than a factor of 3; 2) the noble gas activity monitor for the plant vent has not increased by more than a factor of 3. For containment PURGE, requirements apply only when PURGING.
- (b) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.
- (c) The principal gamma emitters for which the LLD control applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report. Isotopes which are not detected should be reported as "not detected." Values determined to be below detectable levels are not used in dose calculations.
- (d) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with CONTROLS 3.2.2.1, 3.2.2.2, and 3.2.2.3.
- (e) 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 at least 7 days following each shutdown, startup, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a one hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if 1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.
- (f) Sample shall be taken prior to startup of the condenser air removal system when there have been indications of a primary to secondary leak.

3.2 RADIOACTIVE EFFLUENTS

3.2.2 Gaseous Effluents

3.2.2.2 Dose - Noble Gases

CONTROL 3.2.2.2 The air dose at and beyond the SITE BOUNDARY from noble gases in gaseous effluents from {CCNPP Unit 3} shall be limited to the following:

- a. < 5 mrad gamma radiation and < 10 mrad beta radiation during any calendar quarter; and
- b. < 10 mrad gamma radiation and < 20 mrad beta radiation during any calendar year.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

-----NOTE-----

Enter applicable Conditions and Required Compensatory Measures of CONTROL 3.2.3, Total Dose, when gaseous effluent (noble gas) dose exceeds twice the limits of CONTROL 3.2.2.2.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Calculated air dose from radioactive noble gases in gaseous effluents in the UNRESTRICTED AREA not within limits.	A.1 Submit a Special Report to the NRC that identifies causes for exceeding the limits, corrective actions taken to reduce releases, and corrective actions to assure that subsequent releases are within limits.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.2.1 Verify the cumulative dose contributions from noble gases in gaseous effluents for the current calendar quarter and current calendar year are within limits in accordance with the methodology and parameters in the ODCM.	31 days

3.2 RADIOACTIVE EFFLUENTS

3.2.2 Gaseous Effluents

3.2.2.3 Dose - I-131, I-133, Tritium, and Radionuclides in Particulate Form

CONTROL 3.2.2.3 The dose to any organ of a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from {CCNPP Unit 3} to the UNRESTRICTED AREA shall be limited to the following:

- a. ≤ 7.5 mrems during any calendar quarter; and
- b. ≤ 15 mrems during any calendar year.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

-----NOTE-----

Enter applicable Conditions and Required Compensatory Measures of CONTROL 3.2.3, Total Dose, when gaseous effluent dose from I-131, I-133, Tritium, and Radionuclides in Particulate Form exceeds twice the limits of CONTROL 3.2.2.3.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives > 8 days, in gaseous effluents in the UNRESTRICTED AREA not within limits.	A.1 Submit a Special Report to the NRC that identifies causes for exceeding the limits, corrective actions taken to reduce releases, and corrective actions to assure that subsequent releases are within limits.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.3.1 Verify the cumulative dose contributions from iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives > 8 days, in gaseous effluents for the current calendar quarter and current calendar year are within limits in accordance with the methodology and parameters in the ODCM.	31 days

3.2 RADIOACTIVE EFFLUENTS

3.2.2 Gaseous Effluents

3.2.2.4 Gaseous Waste Processing System

CONTROL 3.2.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and GASEOUS WASTE PROCESSING SYSTEM shall be OPERABLE. Appropriate portions of these systems shall be used to reduce the radioactive materials in gaseous waste prior to release when the projected doses due to the gaseous effluent, from {CCNPP Unit 3} at or beyond the SITE BOUNDARY, would exceed:

- a. 0.2 mrad to air from gamma radiation, or
- b. 0.4 mrad to air from beta radiation, or
- c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Gaseous waste being released without treatment when CONTROL 3.2.2.4 limits are exceeded.	A.1 Submit a Special Report to the NRC that includes explanation of why the GASEOUS WASTE PROCESSING SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM were not OPERABLE and why gaseous radwaste was being released without treatment. The Special Report shall identify any required inoperable equipment and the reasons for the inoperability, the corrective actions taken to restore the required inoperable equipment to OPERABLE status, and the corrective actions to prevent recurrence.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.2.4.1 -----NOTE----- Not required to be performed when the GASEOUS WASTE PROCESSING SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM are being fully utilized to reduce radioactive material in gaseous waste. ----- Dose due to gaseous releases from {CCNPP Unit 3} to areas at or beyond the SITE BOUNDARY shall be projected in accordance with the methodology and parameters in the ODCM.</p>	<p>31 days</p>
<p>SR 3.2.2.4.2 The installed GASEOUS WASTE PROCESSING SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by meeting CONTROLS 3.2.2.1, 3.2.2.2, and 3.2.2.3.</p>	<p>In accordance with CONTROLS 3.2.2.1, 3.2.2.2, and 3.2.2.3</p>

3.2 RADIOACTIVE EFFLUENTS

3.2.3 Total Dose

- CONTROL 3.2.3 The annual (calendar year) dose or dose commitment to a MEMBER OF THE PUBLIC due to releases of radioactivity and radiation from uranium fuel cycle sources shall be limited to:
- a. ≤ 25 mrem to the whole body or any organ except the thyroid; and
 - b. ≤ 75 mrem to the thyroid.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
<p>A. Calculated dose to any MEMBER OF THE PUBLIC in the UNRESTRICTED AREA from {CCNPP Units 1, 2 or 3} exceeds twice the limits of CONTROLS 3.2.1.2 or 3.2.2.2 or 3.2.2.3.</p>	<p>A.1 -----NOTE----- Calculations shall include direct radiation contributions from {all CCNPP units, outside radwaste storage areas, and the Independent Spent Fuel Storage Installation (ISFSI)}.</p> <p>----- Calculate the dose or dose commitment to a MEMBER OF THE PUBLIC from all uranium fuel cycle sources to determine if specified limits of CONTROL 3.2.3 were exceeded.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with SRs 3.2.1.2.1, 3.2.2.2.1, and 3.2.2.3.1 and in accordance with the methodology and parameters in the ODCM.	31 days

3.3 RADIOLOGICAL ENVIRONMENTAL MONITORING

3.3.1 Radiological Environmental Monitoring Program

CONTROL 3.3.1 The Radiological Environmental Monitoring Program shall be as follows:

- a. Radiological Environmental Monitoring samples shall be collected at locations and analyzed as specified in Table 3.3.1-1;
- b. The level of radioactivity as the result of plant effluents for a single radionuclide in each environmental sampling medium at a required location shall be less than the limits specified in Table 3.3.1-1, when averaged over the calendar quarter;
- c. The total level of radioactivity in multiple detected radionuclides as the result of plant effluents in each environmental sampling medium at a required location shall be less than the limit specified in Table 3.3.1-2, when averaged over the calendar quarter; and
- d. The potential annual dose to the MEMBER OF THE PUBLIC from all radionuclides other than those in Table 3.3.1-2 in each environmental sampling medium at a required location shall be less than the calendar year limits of CONTROLS 3.2.1.2, 3.2.2.2, and 3.2.2.3.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. One or more samples not collected or analyzed as specified in Table 3.3.1-1.	A.1 Prepare and submit, in the Annual Radiological Environmental Operating Report, a description for not conducting the Radiological Environmental Monitoring sampling and analysis requirements as required and the corrective actions to prevent recurrence.	Upon submittal of current Annual Radiological Environmental Operating Report

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
<p>C. -----NOTE----- Separate Condition entry is allowed for each sample location. -----</p> <p>One or more sample locations with the level of radioactivity for one radionuclide as the result of plant effluents in an environmental sampling medium not within the limits of Table 3.2.1-2 when averaged over the calendar quarter.</p> <p><u>OR</u></p> <p>One or more sample locations with the total level of radioactivity for more than one radionuclide as a result of plant effluents in an environmental sampling medium not within the limits of Table 3.3.1-2 when averaged over the calendar quarter.</p>	<p>C.1 -----NOTE----- Only required if the radionuclides are the result of plant effluents. -----</p> <p>Submit a Special Report to the NRC and state/local officials which includes the cause(s) for exceeding the limit(s) and the corrective actions to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year reporting limits of CONTROLS 3.2.1.2, 3.2.2.2, and 3.2.2.3.</p>	<p>30 days from receipt of laboratory analyses</p>

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
<p>D. One or more sample locations with the potential annual dose to the MEMBER OF THE PUBLIC from all radionuclides not within limits.</p>	<p>D.1 -----NOTE----- Only required if the radionuclides are the result of plant effluents. ----- Submit a Special Report to the NRC which includes the cause(s) for exceeding the limit(s) and the corrective actions to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of CONTROLS 3.2.1.2, 3.2.2.2, and 3.2.2.3.</p> <p><u>AND</u></p> <p>D.2 -----NOTE----- Only required if the radionuclides are not the result of plant effluents. ----- Describe the condition in the Annual Radiological Environmental Operating Report.</p>	<p>30 days</p> <p>Upon submittal of the current calendar year Annual Radiological Environmental Operating Report</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Verify radiological environmental monitoring samples collected at the locations and analyzed as specified in Table 3.3.1-1 are within limits. Detection capabilities for the analysis are specified in Table.	In accordance with Table 3.3.1-1

Table 3.3.1-1 (Page 1 of 5)

Radioactive Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. DIRECT RADIATION(b)	<p>23 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows:</p> <p>An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY.</p> <p>An outer ring of stations, one in each of the meteorological sectors in the 4 to 5 mi (6 to 8 km) range from the site.</p> <p>The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and one area to serve as control stations.</p>	Quarterly	Gamma dose Quarterly

- (a) Specific parameters of distance and direction sector relative to the reactor are provided for each sample location in Table 7.1-1 and Table 7.1-2. Refer to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment.
- (b) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered to be equivalent to two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. (Due to geographical limitations, 9 sectors are monitored around the CCNPP site. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.)

Table 3.3.1-1 (Page 2 of 5)

Radioactive Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. AIRBORNE Radioiodine and Particulates	<p>Samples from 5 locations^(c):</p> <p>3 samples from close to the 3 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground-level D/Q.</p> <p>1 sample from the vicinity of a community having the highest calculated annual average ground-level D/Q.</p> <p>1 sample from a control location, as for example 9 to 19 mi (15 to 30 km) distant and in a non-prevalent wind direction.</p>	<p>Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.</p>	<p>Radioiodine Canister: I-131 analysis weekly.</p> <p>Particulate Sampler: Gross beta radioactivity analysis following filter change^(d)</p> <p>GAMMA ISOTOPIC ANALYSIS of composite (by location) quarterly.</p>

- (a) Specific parameters of distance and direction sector relative to the reactor are provided for each sample location in Table 7.1-1 and Table 7.1-2. Refer to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment.
- (c) Optional air sampling locations as based not only on D/Q, but on factors such as population in the area, year-round access to the site, and availability of power.
- (d) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thorium daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, GAMMA ISOTOPIC ANALYSIS shall be performed on the individual samples.

Table 3.3.1-1 (Page 3 of 5)

Radioactive Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS ^(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. WATERBORNE			
a. Surface	1 sample at intake area 1 sample at discharge area	COMPOSITE SAMPLE over one month period. ^(e)	GAMMA ISOTOPIC ANALYSIS monthly Composite for tritium analysis quarterly.
b. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value.	Semi-annually.	GAMMA ISOTOPIC ANALYSIS semi-annually Tritium analysis semi-annually.

(a) Specific parameters of distance and direction sector relative to the reactor are provided for each sample location in Table 7.1-1 and Table 7.1-2. Refer to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment.

(e) A COMPOSITE SAMPLE is one in which the quantity (aliquot) of liquid is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program, COMPOSITE SAMPLE aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure a representative sample is obtained.

Table 3.3.1-1 (Page 4 of 5)

Radioactive Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS ^(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. INGESTION			
a. Milk - if available ^(f)	<p>Samples from milking animals in three locations within 3 mi (5 km) distance having the highest dose potential. If there are none, then one sample from milking animals in each of three areas between 3 to 8 mi (5 to 8 km) distant where doses are calculated to be greater than 1 mrem per year.^(g)</p> <p>One sample from milking animals at a control location, 9 to 19 mi (15 - 30 km) distant and in a non-prevalent wind direction.</p>	Semi-monthly when animals are on pasture, monthly at other times.	GAMMA ISOTOPIC ANALYSIS semi-monthly and I-131 analysis semi-monthly when animals are in pasture; monthly at other times.
b. Fish and Invertebrates	<p>3 samples of commercially and/or recreationally important species (2 fish species and 1 invertebrate species) in vicinity of plant discharge area.</p> <p>3 samples of same species in areas not influenced by plant discharge.</p>	Sample in season, or semi-annually if they are not in season.	GAMMA ISOTOPIC ANALYSIS on edible portions.

(a) Specific parameters of distance and direction sector relative to the reactor are provided for each sample location in Table 7.1-1 and Table 7.1-2. Refer to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment.

(f) Broad leaf vegetation sampling is performed in lieu of milk sampling if the required minimum number of milk locations is not available in the site area. Milk samples need to be collected and analyzed if the milk is commercially available in quantities greater than 130 liters (34.3.gal) per year.

(g) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

Table 3.3.1-1 (Page 5 of 5)

Radioactive Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS ^(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. INGESTION (continued)			
c. Food Products	Samples of 3 different kinds of broad leaf vegetation grown near the SITE BOUNDARY at two different locations of highest predicted annual average ground-level D/Q.(f)(h)(i)	Monthly during growing season.	GAMMA ISOTOPIC ANALYSIS and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 9 to 19 mi (15-30 km) distant in the least prevalent wind direction if milk sampling is not performed.	Monthly during growing season.	GAMMA ISOTOPIC ANALYSIS and I-131 analysis.

- (a) Specific parameters of distance and direction sector relative to the reactor are provided for each sample location in Table 7.1-1 and Table 7.1-2. Refer to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment.
- (f) Broad leaf vegetation sampling is performed in lieu of milk sampling if the required minimum number of milk locations is not available in the site area. Milk samples need be collected and analyzed if the milk is commercially available in quantities greater than 130 liters (34.3.gal) per year.
- (h) If broad leaf vegetation is unavailable, other vegetation will be sampled. Attention shall be paid to including samples of tuberous and root food products.
- (i) Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census.}

Table 3.3.1-2 (Page 1 of 1)

Reporting Levels for Radioactivity
Concentrations in Environmental Samples^(a)

ANALYSIS	WATER (pCi/L)	AIRBORNE PARTICULATE OR GASES (pCi/m ³)	FISH (pCi/kg, wet)	MILK (pCi/L)	FOOD PRODUCTS (pCi/kg, wet)
H-3 ^(b)	2 x 10 ⁴				
Mn-54	1 x 10 ³		3 x 10 ⁴		
Fe-59	4 x 10 ²		1 x 10 ⁴		
Co-58	1 x 10 ³		3 x 10 ⁴		
Co-60	3 x 10 ²		1 x 10 ⁴		
Zn-65	3 x 10 ²		2 x 10 ⁴		
Zr-Nb-95	4 x 10 ²				
I-131	2	0.9		3	1 x 10 ²
Cs-134	30	10	1 x 10 ³	60	1 x 10 ³
Cs-137	50	20	2 x 10 ³	70	2 x 10 ³
Ba-La-140	2 x 10 ²			3 x 10 ²	

(a) The limits are for samples that have only one radionuclide detected. When a sample contains more than one radionuclide, the total level of radioactivity limit is

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \dots \leq 1.0$$

(b) For drinking water samples. The value given is the 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/L may be used.

Table 3.3.1-3 (Page 1 of 1)

Detection Capabilities for Environmental Sample Analysis^(a)
Lower Limit of Detection (LLD)^{(b)(c)}

ANALYSIS	WATER (pCi/L)	AIRBORNE PARTICULATE OR GASES (pCi/m ³)	FISH (pCi/kg, wet)	MILK (pCi/L)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross beta	4	1 x 10 ⁻²				
H-3	2000 ^(d)					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1 ^(e)	7 x 10 ⁻²		1	60	
Cs-134	15	5 x 10 ⁻²	130	15	60	150
Cs-137	18	6 x 10 ⁻²	150	18	80	180
Ba-La-140	15			15		

- (a) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report.
- (b) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13, Revision 1, July 1977.
- (c) The LLD is defined in the ODCM definitions section. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.
- (d) If no drinking water pathway exists, a value of 3,000 pCi/L may be used.
- (e) LLD for drinking water samples. If no drinking water pathway exists, the LLD for GAMMA ISOTOPIC ANALYSIS may be used.

3.3. RADIOLOGICAL ENVIRONMENTAL MONITORING

3.3.2 Land Use Census

CONTROL 3.3.2 A Land Use Census shall be conducted and:

- a. Shall identify the location of the nearest milk animal, nearest permanent residence, and nearest garden of greater than 500 ft² (50 m²) producing broad leaf vegetation in each of the 16 meteorological sectors within a distance of 5 miles;
- b. The calculated dose and dose commitment at each newly identified census location shall be less than the most recent values calculated by SR 3.2.2.3.1 at the previous census location.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
<p>A. Land Use Census not conducted.</p> <p><u>OR</u></p> <p>All required locations not identified.</p>	<p>A.1 Prepare and submit the reasons for not conducting the Land Use Census and the corrective actions to prevent recurrence.</p>	<p>Upon submittal of current calendar year Annual Radiological Environmental Operating Report</p>
<p>B. One or more newly identified census locations with the calculated dose or dose commitment greater than the values calculated by SR 3.2.2.3.1 at the current census locations.</p>	<p>B.1 Identify new location(s) in the Radioactive Effluent Release Report.</p>	<p>Upon submittal of the current calendar year Radioactive Effluent Release Report</p>

3.3. RADIOLOGICAL ENVIRONMENTAL MONITORING

3.3.3 Interlaboratory Comparison Program

CONTROL 3.3.3 Analyses shall be performed on all radioactive materials supplied as part of an Interlaboratory Comparison Program approved by the NRC.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Requirements of CONTROL 3.3.3 not met.	A.1 Prepare and submit, in the Annual Radiological Environmental Operating Report, corrective actions to prevent recurrence.	Upon submittal of current calendar year Annual Radiological Environmental Operating Report

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.3.1 Perform analysis on all radioactive material supplied as part of the Interlaboratory Comparison Program as described in the OCDM and submit the results in the next annual Radiological Environmental Operating Report.	12 months

3.4 RADIOLOGICAL EFFLUENT CONTROLS BASES

BASES

B 3.1.1 Radioactive Liquid Effluent Monitoring Instrumentation

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 shall be calculated and adjusted in accordance with the methodology and parameters 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 3.1.2 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 and adjusted in accordance with the methodology and parameters 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. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of CONTROL 3.2.2.2 shall be such that concentrations as low as 1×10^{-6} $\mu\text{Ci/cc}$ are measurable.

B 3.2.1.1 Liquid Concentration

This CONTROL is provided to ensure that the concentration of radioactive materials released in liquid effluents at the point of discharge {from the multiport diffuser} will be less than ten times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2, for radionuclides other than dissolved or entrained noble gases. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of Appendix I, 10 CFR 20.1301 and 20.1302 to the population. Those values assume a continuous discharge at those concentrations (8760 hours per year). Pursuant to the requirements of 10 CFR 50.36a to maintain effluent concentrations as low as reasonably achievable (ALARA), Appendix I to 10 CFR 50 specifies that dose values be a small percentage of the dose limits of 10 CFR 20.1301. Consistent with Appendix I to 10 CFR Part 50, to allow operational flexibility, this CONTROL in conjunction with the dose control in CONTROL 3.2.1.2 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.

BASES (continued)

B 3.2.1.2 Liquid Dose

This CONTROL is provided to implement the requirements of Sections II.A., III.A, and IV.A of Appendix I to 10 CFR Part 50. The CONTROL implements the guides set forth in Section II.A of Appendix I. The associated condition and required compensatory measure 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 liquid effluents to UNRESTRICTED AREAS will be kept ALARA. The dose calculation methodology and parameters in the ODCM 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 a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109 (Reference [2]), and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

B 3.2.1.3 Liquid Waste Processing System

The OPERABILITY of the LIQUID WASTE PROCESSING 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 ALARA. This CONTROL implements the requirements of 10 CFR 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 WASTE PROCESSING SYSTEM were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix A to 10 CFR Part 50 for liquid effluents.

B 3.2.2.1 Dose Rate (Gaseous Effluents)

This CONTROL is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table 2 of 10 CFR Part 20. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/yr to the whole body and to less than or equal to 3000 mrem/yr 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/yr.

BASES (continued)

B 3.2.2.2 Dose – Noble Gases

This CONTROL is provided to implement the requirements of Sections II.B.1, III.A, and IV.A of Appendix I to 10 CFR Part 50. The CONTROL implements the guides set forth in Section II.B.1 of Appendix I. The associated condition and required compensatory measure provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I at the SITE BOUNDARY that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept ALARA. 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 a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses from the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109 (Reference [2]), and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977.

B 3.2.2.3 Dose – I-131, I-133, Tritium, and Radionuclides in Particulate Form

This CONTROL is provided to implement the requirements of Section II.C, III.A, and IV.A of Appendix I to 10 CFR Part 50. The CONTROLS are the guides set forth in Section II.C of Appendix I. The associated condition and required compensatory measure 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 releases of radioactive materials in gaseous effluents at the SITE BOUNDARY will be kept ALARA. The ODCM calculation 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 a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109 (Reference [2]) and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The simplified Method I equations provide for determining the actual doses based upon the historical annual average atmospheric conditions and are dependent upon the existing or potential radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of the Method I calculations were:

1. individual inhalation of airborne radionuclides,
2. deposition of radionuclides onto green leafy vegetation with subsequent consumption by man,
3. deposition of radionuclides onto grassy areas where milk animals and meat-producing animals graze followed by human consumption of that milk and meat, and
4. deposition of radionuclides on the ground followed by the subsequent human exposure.

BASES (continued)

B 3.2.2.4 Gaseous Waste Processing Systems

The OPERABILITY of the GASEOUS WASTE PROCESSING 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 ALARA. This CONTROL implements the requirements of 10 CFR 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 to 10 CFR Part 50, for gaseous effluents.

B 3.2.3 Total Dose

This CONTROL is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR 20 by 46FR18525. The CONTROL requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrem to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units (including outside storage tanks, etc.) are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within 40 CFR Part 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 cycles is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site that are 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 Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.2203(a)(4), is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in CONTROLS 3.2.1.1 and 3.2.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

BASES (continued)

B 3.3.1 Radiological Environmental Monitoring Program

The Radiological Environmental Monitoring Program (REMP) required by this CONTROL provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50, and thereby supplements the effluent control program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979 (Reference [5]). {Locations for CCNPP Unit 3 consider the existing site area REMP conducted for CCNPP Units 1 and 2 and utilizes common sampling locations and media when the selection criteria of Table 3.3.1-1 also satisfy the CCNPP Unit 3 location.}

B 3.3.2 Land Use Census

This CONTROL is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the REMP given in the ODCM are made if required by the results of this census. Information from methods such as the door-to-door survey, from aerial survey, or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 ft² provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored, since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 (Reference [2]) for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad-leaf vegetation (i.e., similar to lettuce and cabbage), and (2) there was a vegetation yield of 2 kg/m².

B 3.3.3 Interlaboratory Comparison Program

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the Quality Assurance Program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

4.0 REPORTING REQUIREMENTS

4.1 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

Routine Annual Radiological Environmental Operating Reports covering the operation of the station during the previous calendar year shall be submitted prior to May 15 of each year pursuant to Technical Specification 5.6.1.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program (REMP) for the report period, including a comparison with preoperational studies, with operational controls, as appropriate, and with previous environmental operating reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by CONTROL 3.3.2. The material provided shall be consistent with the objectives outlined in the ODCM and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3 and IV.C.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in Section 7.0, as well as summarized and tabulated results of these analyses and measurements in the format similar to the table in the Radiological Assessment Branch Technical Position (Reference [5]). The reports shall include all on-site ground water sample results and a description of any significant on-site leaks/spills into the ground water. In the event that some individual 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; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required by CONTROL 3.3.3; reason for not conducting the Radiological Environmental Monitoring Program as required by CONTROL 3.3.1, and discussion of all deviations from the sampling schedule; discussion of environmental sample measurements that exceed the reporting levels; and discussion of all analyses in which the LLD required was not achievable.

* One map shall cover locations near the SITE BOUNDARY; the more distant locations shall be covered by one or more additional maps

4.0 REPORTING REQUIREMENTS

4.2 RADIOACTIVE EFFLUENT RELEASE REPORT

A routine Radioactive Effluent Release Report covering the operation of the station during the previous calendar year of operation shall be submitted by May 1 of each year, pursuant to Technical Specification 5.6.2.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the station 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 similar to that of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity) and solidification agent or absorbent (e.g., cement), if any.

The Radioactive Effluent Release Report 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 on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. 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 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, if any, during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time, and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the ODCM.

The Radioactive Effluent Release Report shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109 (Reference [2]).

4.2 Radioactive Effluent Release Report (continued)

The Radioactive Effluent Release Report shall include a report of solid waste transported from the site during the year as specified in Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

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

The Radioactive Effluent Release Report shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and the ODCM, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Section 4.3. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to CONTROL 3.3.2.

The Radioactive Effluent Release Report shall also include an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in CONTROL 3.1.1.1 or 3.1.2.1, respectively.

4.0 REPORTING REQUIREMENTS

4.3 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

- a. Shall be reported to the NRC in the Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Independent Review Committee (IRC).

The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made without prior NRC approval;
 2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
 3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;
 4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;
 5. An evaluation of the change, which shows the expected maximum exposures to a member of the public at the site boundary and to the general population that differ from those previously estimated in the license application and amendments thereto;
 6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;
 7. An estimate of the exposure to plant operating personnel as a result of the change; and
 8. Documentation of the fact that the change was reviewed and found acceptable by the IRC.
- b. Shall become effective upon approval and acceptance by the Plant Manager.

* Licensee may choose to submit the information called for in this section as part of the annual FSAR update.

4.0 REPORTING REQUIREMENTS

4.4 GROUND WATER MONITORING REPORTS

In accordance with NEI 07-07, (Reference [6]) a written 30-day report will be submitted to the NRC and designated State/Local officials documenting any on-site water sample result for on-site ground water that is or may be used as a source of drinking water that exceeds any of the criteria in the existing REMP. The Special Report should include:

- A statement that the report is being submitted in support of the Ground Water Protection Initiative (GPI),
 - A list of the contaminant(s) and the verified concentration(s),
 - Description of the action(s) taken,
 - An estimate of the potential or bounding annual dose to a member of the public, and
 - Corrective action(s), if necessary, that will be taken to reduce the projected annual dose to a member of the public to less than the limits in 10 CFR 40 Appendix I.
-

5.0 METHOD TO CALCULATE OFFSITE LIQUID CONCENTRATIONS

Section 5.0 contains the basis for plant procedures used to demonstrate compliance with CONTROL 3.2.1.1, which limits the total fraction of Effluent Concentration Limit (ECL) in liquid pathways, excluding noble gases, denoted here as F_1^{ENG} at the point of discharge at any time. F_1^{ENG} is limited to less than or equal to ten, i.e.,

$$F_1^{ENG} \leq 10$$

Evaluation of F_1^{ENG} is required concurrent with the sampling and analysis program in Table 3.2.1.1-1.

5.1 METHOD TO DETERMINE F_1^{ENG}

Determine the total fraction of combined effluent concentrations at the point of discharge in liquid pathways (excluding noble gases), denoted F_1^{ENG} , as follows:

Equation 5-1

$$F_1^{ENG} = \sum_i \frac{C_{pi}}{ECL_i} \leq 10$$

where :

- F_1^{ENG} = Total sum of the fractions of each radionuclide concentration in liquid effluents (excluding noble gases) at the point of discharge from the {multiport diffuser}, divided by each radionuclides ECL value,
- C_{pi} = Concentration at point of discharge {from the multiport diffuser} of radionuclide "i", except for dissolved and entrained noble gases, from any tank or other significant source, p, from which a discharge may be made in $\mu\text{Ci/ml}$. This concentration is determined by dividing the product of the measured radionuclide concentration in liquid waste monitoring tanks, PCCW, Turbine Building Drains Effluent Line or other effluent streams times their discharge flow rate by the total available dilution water flow rate at the time of release ($\mu\text{Ci/ml}$),
- ECL_i = Effluent concentration limit (ECL) for radionuclide "i" (except for dissolved and entrained noble gases) in $\mu\text{Ci/ml}$ as specified in 10 CFR 20, Appendix B, Table 2,

5.2 METHOD TO DETERMINE RADIONUCLIDE CONCENTRATION FOR EACH LIQUID EFFLUENT PATHWAY

5.2.1 Liquid Waste Monitoring Tanks

C_{pi} is determined for each radionuclide detected from the activity in a representative grab sample of any of the liquid waste monitoring tanks and the predicted flow at the point of discharge.

The batch releases are normally made from two 18,500 gallon capacity monitoring tanks. These tanks normally hold liquid waste which may have been processed through the liquid waste processing equipment. The waste monitoring tanks can also contain other waste such as liquid taken directly from nuclear island vent and drain systems when that liquid does not require processing, or flushing water from the Steam Generator Blowdown System when that system must discharge liquid off site.

Prior to discharge, each waste monitoring tank is analyzed for principal gamma emitters in accordance with the liquid sample and analysis program outlined in Table 3.2.1.1-1. In addition, radiation sensors continually measure and record the total actual activity and the activity release rate during each release of processed liquid waste effluents to the environment. Each radiation sensor can generate control signals that stop the discharge pump and isolate the release path if the sensor detects activity in excess of the anticipated level or release rate.

5.2.2 Turbine Building Drains

The Turbine Building collects leakage from the turbine building floor drains and discharges the liquid unprocessed to the {retention basin}. Sampling of this potential source is normally done once per week for determining the radioactivity released to the environment (See Table 3.2.1.1-1)

6.0 OFFSITE DOSE CALCULATION METHODS

Section 6.0 provides the basis for plant procedures required to meet the 10 CFR Part 50, Appendix I ALARA dose objectives and the 40 CFR Part 190 total dose limits to MEMBERS OF THE PUBLIC in UNRESTRICTED AREAS, as stated in the Radiological Effluent Controls Program (RECP) (implementing the requirements of Technical Specification 5.5.3). A simple, conservative method (called Method I) is presented for each of the requirements in the RECP in Sections 0 through 6.9. In addition, reference is provided to more sophisticated but still conservative methods (called Method II) for use when more accurate results are needed. This section provides the methods, data, and reference material with which the operator can calculate the needed doses and dose rates. Setpoint methods for effluent monitor alarms are described in Section 8.0 .

6.1 INTRODUCTORY CONCEPTS

The Radiological Effluent Controls Program (Technical Specifications 5.5.3) either limits dose or dose rate. The term “dose” for ingested or inhaled radioactivity means the dose commitment, measured in mrem, which results from the exposure to radioactive materials that, because of uptake in the body, will continue to expose the body to radiation for some period of time after the source of radioactivity is stopped. The time frame over which the dose commitment is evaluated is 50 years. The phrases “annual dose” or “dose in one year” then refers to the fifty-year dose commitment from one year’s worth of releases. The term “dose,” with respect to external exposures, such as to noble gas clouds, refers only to the doses received during the actual time period of exposure to the radioactivity released from the plant. Once the source of the radioactivity is removed, there is no longer any additional accumulation to the dose commitment.

“Dose rate” is the total dose or dose commitment divided by the exposure period. For example, an individual who is exposed via the ingestion of milk for one year to radioactivity from plant gaseous effluents and receives a 50-year dose commitment of 10 mrem is said to have been exposed to a dose rate of 10 mrem/year, even though the actual dose received in the year of exposure may be less than 10 mrem.

In addition to limits on dose commitment, gaseous effluents from the plant are also controlled such that the maximum “dose rates” at the SITE BOUNDARY at any time are limited to 500 mrem/yr. This instantaneous dose rate limit allows for operational flexibility when off-normal occurrences may temporarily increase gaseous effluent release rates from the plant, while still providing controls to ensure that licensees meet the dose objectives of Appendix I to 10 CFR 50.

It should be noted that a dose rate due to noble gases that exceeds for a short time period (less than one hour in duration) the equivalent 500 mrem/year dose rate limit stated in CONTROL 3.2.2.1a does not necessarily, by itself, constitute a Licensee Event Report (LER) under 10 CFR Part 50.73, unless it is determined that the air concentration of radioactive effluents in UNRESTRICTED AREAS has also exceeded 20 times applicable concentration limits specified in Appendix B to 20.1001-20.2402, Table 2, Column 1 (four-hour notification per 10 CFR 50.72, and 30 day LER per 10 CFR 50.73).

The Method I equations provide an upper bound dose for a given release because of the conservative margin built into the dose factors and the selection and definition of critical receptors. The radionuclide specific dose factors in each Method I dose equation

represent the greatest dose to any organ of any age group. (Organ dose is a function of age because organ mass and intake are functions of age.) The critical receptor assumed by "Method I" equations is then generally a hypothetical individual whose behavior - in terms of location and intake - results in a dose which is higher than any real individual is likely to receive. Method II also allows for a more exact dose calculation for each individual if necessary.

The dose rate (\dot{D}) is calculated using the station's effluent monitoring system reading and an annual or long-term average atmospheric dispersion factor. \dot{D} predicts the maximum offsite annual dose if the peak observed radioactivity release rate from the plant stack continued for one entire year.

Each of the methods to calculate dose or dose rate is presented in the following subsections. Each dose type has two levels. Method I is the simplest and contains many conservative factors. Method II is a more realistic analysis (i.e., use of actual location of individuals and receptor pathways as identified in the most recent Land Use Census, concurrent meteorology with period of release and summation of dose by specific organs for all nuclides identified in a release) which also makes use of the models in Regulatory Guide 1.109 (Reference [2]), as noted in each subsection of Section 6.0 for the various exposure types. A detailed description of the methodology, assumptions, and input parameters to the dose models that are applied in each Method II calculation, if not already explicitly described in the ODCM, shall be documented and provided when this option is used for NRC reporting.

6.2 METHOD TO CALCULATE THE TOTAL BODY DOSE FROM LIQUID RELEASES

CONTROL 3.2.1.2 limits the total body dose commitment to a MEMBER OF THE PUBLIC from radioactive material in liquid effluents to 1.5 mrem per quarter and 3 mrem per year for {CCNPP Unit 3}. CONTROL 3.2.1.3 requires liquid radwaste treatment when the total body dose estimate exceeds 0.06 mrem in any 31-day period. CONTROL 3.2.3 limits the total body dose commitment to any real MEMBER OF THE PUBLIC from all station sources (including liquids) to 25 mrem in a year.

Use Method I first to calculate the maximum total body dose from a liquid release from {CCNPP Unit 3} as it is simpler to execute and more conservative than Method II.

Use Method II if a more refined calculation of total body dose is needed, i.e., Method I indicates the dose might be greater than Control limits, or if Method I cannot be applied.

To evaluate the total body dose, use Equation 6-1 to estimate the dose from the planned release and add this to the total body dose accumulated from prior releases during the month.

6.2.1 Method I

The total body dose (D_{tb}) from a liquid release is:

Equation 6-1

$$D_{tb} = k \sum_i Q_i \text{DFL}_{itb}$$

where:

- DFL_{itb} = Site-specific total body dose factor (mrem/Ci) for a liquid release. It is the highest of the four age groups. See Table 6-1.
- Q_i = Total activity (Ci) released for radionuclide "i". (For strontiums and Fe-55, use the most recent measurement available.)
- k = {39.3/F_d; where F_d is the average (typically monthly average) dilution flow at the point of discharge from the multipoint diffuser (in ft³/sec). For normal operations with a discharge flow of 39.3 ft³/sec, k is equal to 1.}

Equation 6-1 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Liquid releases via the {multipoint diffuser} to UNRESTRICTED AREAS (at the edge of the initial mixing or prompt dilution zone that corresponds to at least a factor of {13} dilution), and
2. Any continuous or batch release over any time period up to 1 year. For annual dose estimates, the annual average discharge flow should be used as the dilution flow estimate.

6.2.2 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 (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 (Reference [2]), and used in the derivation of the simplified Method I approach as described in the Section 9.1, 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 9-1 lists the usage factors for various liquid effluent exposure pathways. As noted in Section 9.1, {the mixing ratio associated with fish and invertebrate ingestion and boating pathways is 0.075 and the mixing ratio associated with swimming and shoreline exposure pathways is 0.014.}

Table 6-1 (Page 1 of 1)

{Dose Factors Specific for CCNPP Unit 3 Liquid Releases}

Nuclide	Maximum Organ Dose Factor (DFL_{imo}) (mrem/Ci)	Total Body Dose Factor (DFL_{itb}) (mrem/Ci)
H-3	6.76E-06	6.76E-06
NA-24	2.13E-03	2.13E-03
CR-51	2.57E-02	1.76E-04
MN-54	4.19E-01	4.21E-02
FE-55	1.34E+00	2.21E-01
CO-57	6.93E-02	7.12E-03
CO-58	2.31E-01	3.24E-02
FE-59	1.17E+01	1.53E+00
CO-60	8.35E-01	3.05E-01
ZN-65	9.59E+00	4.79E+00
SR-89	1.39E-01	3.96E-03
SR-90	2.73E+00	5.49E-02
Y-93	2.02E-01	5.03E-05
MO-99	4.76E-03	6.45E-04
TC-99M	9.41E-05	6.35E-05
AG-110M	1.11E+01	5.48E-02
SB-124	1.53E-01	9.52E-03
SB-125	6.11E-02	2.55E-02
I-131	1.77E+00	3.48E-03
TE-129M	6.06E+01	2.69E+00
TE-131M	5.16E+01	5.54E-01
TE-132	6.66E+01	1.59E+00
I-133	2.07E-01	6.91E-04
CS-134	3.82E-01	3.22E-01
CS-136	5.31E-02	3.89E-02
CS-137	3.42E-01	2.55E-01
BA-140	6.07E-02	2.70E-03
LA-140	7.25E-01	1.24E-03
CE-141	1.63E-01	1.82E-04
PR-143	4.53E-01	7.83E-06
CE-144	1.13E+00	1.01E-03
PR-144	2.81E-05	1.42E-05
W-187	2.39E-02	2.67E-04
OTHERS	2.77E+01	5.46E-01

6.3 METHOD TO CALCULATE MAXIMUM ORGAN DOSE FROM LIQUID RELEASES

CONTROL 3.2.1.2 limits the maximum organ dose commitment to a MEMBER OF THE PUBLIC from {CCNPP Unit 3} radioactive material in liquid effluents to 5 mrem per quarter and 10 mrem per year. CONTROL 3.2.1.3 requires liquid radwaste treatment when the maximum organ dose estimate exceeds 0.2 mrem in any month. CONTROL 3.2.3 limits the maximum organ dose commitment to any real MEMBER OF THE PUBLIC from all station sources (including liquids) to 25 mrem in a year except for the thyroid, which is limited to 75 mrem in a year. Dose evaluation is required at least once per month if releases have occurred. If the LIQUID WASTE PROCESSING SYSTEM is not being used, dose evaluation is required before each release.

Use Method I first to calculate the maximum organ dose from a liquid release from {CCNPP Unit 3} as it is simpler to execute and more conservative than Method II.

Use Method II if a more refined calculation of organ dose is needed (i.e., Method I indicates the dose might be greater than Control limits), or if Method I cannot be applied.

If the LIQUID WASTE PROCESSING SYSTEM is not operating, the maximum organ dose must be estimated prior to a release (CONTROL 3.2.1.3). To evaluate the maximum organ dose, use Equation 6-2 to estimate the dose from the planned release and add this to the maximum organ dose accumulated from prior releases during the month.

6.3.1 Method I

For any liquid release, during any period, the dose from radionuclide "i" to the maximum organ (D_{mo}) is:

Equation 6-2

$$D_{mo} = k \sum Q_i DFL_{imo}$$

where:

DFL_{imo} = Site-specific maximum organ dose factor (mrem/Ci) for a liquid release. See Table 6-1.

Q_i = Total activity (Ci) released for radionuclide "i".

K = $\{39.3\}/F_d$ (dimensionless); where F_d is the average dilution flow at the point of discharge {from the multiport diffuser} (in ft^3/sec).

Equation 6-2 can be applied under the following conditions (otherwise, justify Method I or consider Method II):

1. Liquid releases {via the multiport diffuser} to UNRESTRICTED AREAS (at the edge of the initial mixing or prompt dilution zone that corresponds to at least a factor of {13} dilution), and
2. Any continuous or batch release over any time period up to 1 year. For annual dose estimates, the annual average discharge flow should be used as the dilution flow estimate.

6.3.2 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 (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 (Reference [2]), and used in the derivation of the simplified Method I approach as described in the Section 9.1, 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 dose of an age-dependent individual via all existing exposure pathways. Table 9-1 lists the usage factors associated with the various liquid exposure pathways. As noted in Section 9.1, {the mixing ratio associated with fish and invertebrate ingestion and boating pathways is 0.075 and the mixing ratio associated with swimming and shoreline exposure pathways is 0.014.}

6.4 METHOD TO CALCULATE THE TOTAL BODY DOSE RATE FROM NOBLE GASES

CONTROL 3.2.2.1.a limits the instantaneous dose rate at any time to the total body from all release sources of noble gases at any location at or beyond the SITE BOUNDARY to equal to or less than 500 mrem/year.

Use Method I first to calculate the total body dose rate from the peak release rate.

Use Method II if Method I predicts a dose rate greater than the Control limit (i.e., use of actual meteorology over the period of interest) to determine if, in fact, CONTROL 3.2.2.1.a had actually been exceeded during a short time interval.

Compliance with the dose rate limits for noble gases are continuously demonstrated when effluent release rates are below the plant stack noble gas activity monitor alarm setpoint by virtue of the fact that the alarm setpoint is based on a value which corresponds to the offsite dose rate limit of CONTROL 3.2.2.1.a.

Determinations of dose rate for compliance with CONTROL 3.2.2.1.a are performed when the effluent monitor alarm setpoint is exceeded and the corrective action required by CONTROL 3.2.2.1.a is unsuccessful.

6.4.1 Method I

Method I was derived from the general Equation B-8 in Regulatory Guide 1.109 (Reference [2]) as follows:

Equation 6-3

$$\dot{D}_{tb} = 1E+6 [\chi/Q] \sum_i \dot{Q}_i DFB_i$$

where:

\dot{D}_{tb} = the total body dose rate from noble gases in mrem/year.

1E+06 = the number of pCi per μ Ci.

$[\chi/Q]$ = Maximum offsite receptor location long-term average undepleted atmospheric dispersion factor (sec/m^3).

\dot{Q}_i = Release rate to the environment of noble gas "i" ($\mu\text{Ci}/\text{sec}$).

DFBi = Gamma total body dose factor, $\left(\frac{\text{mrem}\cdot\text{m}^3}{\text{pCi}\cdot\text{yr}} \right)$. See Table 6-2.

Incorporating the {CCNPP Unit 3} limiting site boundary long-term average undepleted atmospheric dispersion factor from Section 6.10, Equation 6-3 takes the form:

Equation 6-4

$$\dot{D}_{tb} = (1E+6) * \{(1.05E-06)\} * \sum_i (\dot{Q}_i * DFB_i)$$

which reduces to:

Equation 6-5¹

$$\dot{D}_{tb} = \{1.05\} * \sum_i (\dot{Q}_i * DFB_i)$$

The selection of critical receptor, outlined in Section 6.10 is inherent in the derived Method I since the maximum expected offsite long-term average atmospheric dispersion factor is used. All noble gases in Table 6-2 should be considered.

6.4.2 Method II

Method II consists of the model and input data (whole body dose factors) in Regulatory Guide 1.109 (Reference [2]), except where site-specific data or assumptions have been identified in the ODCM. The general equation (B-8) taken from Regulatory Guide 1.109 (Reference [2]), and used in the derivation of the simplified Method I approach as described in the Section 9.2.1, is also applied to a Method II assessment. No credit for a shielding factor (S_F) associated with residential structures is assumed. Concurrent meteorology with the release period may be utilized for the undepleted atmospheric dispersion factor at the limiting site boundary location not bounded by water identified in Equation 9-3 (Section 9.2.1), and determined as indicated in Section 6.10 for the release point from which recorded effluents have been discharged.

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

Table 6-2 (Page 1 of 1)

{CCNPP Unit 3 Dose Factors for Noble Gas Releases}

Nuclide	Gamma Total Body Dose Factor $DF_{Bi}^{\gamma} \left(\frac{mrem - m^3}{pCi - yr} \right)$	Beta Skin Dose Factor $DF_{Si}^{\beta} \left(\frac{mrem - m^3}{pCi - yr} \right)$	Combined Skin Dose Rate Factor $DF_i^{\prime} \left(\frac{mrem}{Ci} \right)$	Beta Air Dose Factor $DF_i^{\beta} \left(\frac{mrad - m^3}{pCi - yr} \right)$	Gamma Air Dose Factor $DF_i^{\gamma} \left(\frac{mrad - m^3}{pCi - yr} \right)$
Ar-41	8.84E-03	2.69E-03	4.33E-04	3.28E-03	9.30E-03
Kr-83m	7.56E-08	-----	7.14E-07	2.88E-04	1.93E-05
Kr-85m	1.17E-03	1.46E-03	9.37E-05	1.97E-03	1.23E-03
Kr-85	1.61E-05	1.34E-03	4.49E-05	1.95E-03	1.72E-05
Kr-87	5.92E-03	9.73E-03	5.49E-04	1.03E-02	6.17E-03
Kr-88	1.47E-02	2.37E-03	6.41E-04	2.93E-03	1.52E-02
Kr-89	1.66E-02	1.01E-02	9.73E-04	1.06E-02	1.73E-02
Kr-90	1.56E-02	7.29E-03	8.44E-04	7.83E-03	1.63E-02
Xe-131m	9.15E-05	4.76E-04	2.15E-05	1.11E-03	1.56E-04
Xe-133m	2.51E-04	9.94E-04	4.49E-05	1.48E-03	3.27E-04
Xe-133	2.94E-04	3.06E-04	2.32E-05	1.05E-03	3.53E-04
Xe-135m	3.12E-03	7.11E-04	1.48E-04	7.39E-04	3.36E-03
Xe-135	1.81E-03	1.86E-03	1.32E-04	2.46E-03	1.92E-03
Xe-137	1.42E-03	1.22E-02	4.58E-04	1.27E-02	1.51E-03
Xe-138	8.83E-03	4.13E-03	4.77E-04	4.75E-03	9.21E-03

6.5 METHOD TO CALCULATE THE SKIN DOSE RATE FROM NOBLE GASES

CONTROL 3.2.2.1.a limits the instantaneous dose rate at any time to the skin from all release sources of noble gases at any location at or beyond the SITE BOUNDARY to 3000 mrem/year.

Use Method I first to calculate the skin dose rate from the peak release rate.

Use Method II if Method I predicts a dose rate greater than the Control limits (i.e., use of actual meteorology over the period of interest) to determine if, in fact, CONTROL 3.2.2.1.a had actually been exceeded during a short time interval.

Compliance with the dose rate limits for noble gases are continuously demonstrated when effluent release rates are below the plant stack noble gas activity monitor alarm setpoint by virtue of the fact that the alarm setpoint is based on a value which corresponds to the offsite dose rate limit of CONTROL 3.2.2.1.a.

Determinations of dose rate for compliance with CONTROL 3.2.2.1.a are performed when the effluent monitor alarm setpoint is exceeded and the corrective action required by CONTROL 3.2.2.1.a is unsuccessful.

6.5.1 Method I

The skin dose rate due to noble gases from {CCNPP Unit 3} is determined as follows:

Equation 6-6¹

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

where:

\dot{D}_{skin} = the offsite skin dose rate (mrem/year) due to noble gases in an effluent discharge,

\dot{Q}_i = the noble gas release rate for nuclide "i" in Ci/yr,

DF_i' = the combined skin dose rate factor in mrem/Ci from Table 6-2

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

- 1) Normal operations (non-emergency event), and
- 2) Noble gas release via the plant stack.

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

6.5.2 Method II

Method II consists of the model and input data (skin dose factors) in Regulatory Guide 1.109 (Reference [2]), except where site-specific data or assumptions have been identified in the ODCM. The general equation (B-9) taken from Regulatory Guide 1.109 (Reference [2]), and used in the derivation of the simplified Method I approach is also applied to a Method II assessment. No credit for a shielding factor (S_F) associated with residential structures is assumed. Concurrent meteorology with the release period may be utilized for the atmospheric dispersion factor at the limiting site boundary not bounded by water identified in Equation 9-6, and determined as indicated in Section 6.10 for the release point from which recorded effluents have been discharged.

6.6 METHOD TO CALCULATE THE CRITICAL ORGAN DOSE RATE FROM IODINES, TRITIUM, AND PARTICULATES WITH HALF-LIVES GREATER THAN 8 DAYS

CONTROL 3.2.2.1.b limits the dose rate to any organ, denoted \dot{D}_{co} , from all release sources of I-131, I-133, H-3 and radionuclides present in particulate form with half-lives greater than 8 days to 1500 mrem/year to any organ.

Use Method I first to calculate the critical organ dose rate from the peak release rate.

Use Method II if Method I predicts a dose rate greater than the Control limits (i.e., use of concurrent meteorology over the period of interest) to determine if, in fact, CONTROL 3.2.2.1.b had actually been exceeded during the sampling period.

6.6.1 Method I

The critical organ dose rate to an offsite receptor from {CCNPP Unit 3} can be determined as follows:

Equation 6-7¹

$$\dot{D}_{co} = \sum_i (\dot{Q}_i * DFG'_{ico})$$

where

\dot{D}_{co} = The offsite critical organ dose rate (mrem/yr) due to iodine (I-131, I-133), tritium, and particulates ($T_{1/2} > 8$ days) from the plant stack,

\dot{Q}_i = the activity release rate of radionuclide "i" in $\mu\text{Ci}/\text{sec}$ (i.e., total activity measured of radionuclide "i" averaged over the time period for which the filter/charcoal sample collector was in the effluent stream. For $i = \text{Sr-89}$ or Sr-90 , use the best estimates, such as most recent measurements), and

DFG'_{ico} = the site-specific critical organ dose rate factor for plant stack gaseous release $\left(\frac{\text{mrem} \cdot \text{sec}}{\mu\text{Ci} \cdot \text{yr}} \right)$ (See Table 6-3).

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

6.6.2 Method II

Method II consists of the model and input data in Regulatory Guide 1.109 (Reference [2]), except where site-specific data or assumptions have been identified in the ODCM (see Table 9-2 and Table 9-3). The critical organ dose rate will be determined based on the location (site boundary, nearest resident, or farm) of receptor pathways as identified in the most recent annual Land Use Census, or by conservatively assuming the existence of all pathways (ground plane, inhalation, ingestion of stored and leafy vegetables, milk, and meat) at an offsite location of maximum potential dose. Concurrent meteorology with the release period may be utilized for determination of atmospheric dispersion factors in accordance with Section 6.10 for the release point from which recorded effluents have been discharged. The maximum critical organ dose rates will consider the four age groups independently, and take no credit for a shielding factor (S_F) associated with residential structures.

Table 6-3 (Page 1 of 1)

{Dose and Dose Rate Factors Specific for CCNPP Unit 3 for Iodine, Tritium or Particulate
 Releases}

Nuclide	Critical Organ Dose Factor (DFG_{ico}) (mrem/Ci)	Critical Organ Dose Rate Factor (DFG'_{ico}) (mrem-sec/μCi-yr)
H-3	1.56E-04	7.60E-04
CR-51	5.24E-03	1.99E-02
MN-54	7.58E-01	1.88E+00
FE-55	3.33E-01	1.18E-01
FE-59	7.42E-01	1.45E+00
CO-57	1.82E-01	5.55E-01
CO-58	3.78E-01	1.27E+00
CO-60	8.42E+00	8.26E+00
ZN-65	1.64E+00	1.18E+00
SR-89	1.35E+01	2.29E+00
SR-90	7.45E+02	3.66E+01
ZR-95	7.70E-01	2.55E+00
NB-95	1.56E+00	7.13E-01
MO-99	1.32E-02	2.55E-01
RU-103	2.36E+00	7.44E-01
AG-110M	3.50E+00	6.40E+00
SB-124	1.23E+00	3.66E+00
SB-125	1.33E+00	2.60E+00
I-131	1.20E+02	1.54E+01
I-133	1.22E+00	3.66E+00
CS-134	4.19E+01	1.07E+00
CS-137	4.04E+01	8.61E-01
BA-140	8.72E-02	1.93E+00
LA-140	3.43E-02	4.64E-01
CE-141	1.71E-01	5.83E-01
CE-144	4.12E+00	1.27E+01
OTHER	3.24E+00	3.75E+00

6.7 METHOD TO CALCULATE GAMMA AIR DOSE FROM NOBLE GASES

CONTROL 3.2.2.2 limits the gamma dose to air from noble gases at any location at or beyond the SITE BOUNDARY to 5 mrad in any quarter and 10 mrad in any year for {CCNPP Unit 3}. Dose evaluation is required at least once per 31 days.

Use Method I first to calculate the gamma air dose from the station gaseous effluent releases during the period.

Use Method II if a more refined calculation is needed (i.e., use of actual release point parameter with annual or actual meteorology to obtain release-specific χ/Q_s), or if Method I predicts a dose greater than the Control limit to determine if it had actually been exceeded. See Section 9.2.4 for basis.

6.7.1 Method I

The general form of the gamma air dose equation is:

Equation 6-8

$$D_{air}^{\gamma} = 3.17E+04 * \left[\chi/Q \right] * \sum_i (Q_i * DF_i^{\gamma})$$

where

- D_{air}^{γ} = the gamma air dose in mrad,
- 3.17E+04 = the number of pCi per Ci divided by the number of second per year,
- $[\chi/Q]$ = the maximum offsite receptor location long-term average undepleted atmospheric dispersion factor in sec/m^3 ,
- Q_i = the total activity in Ci of each noble gas "i" released to the atmosphere from the station gaseous effluent release point during the period of interest, and
- DF_i^{γ} = the gamma dose factor to air for noble gas "i" in $(\text{mrad} \cdot \text{m}^3)/(\text{pCi} \cdot \text{yr})$ (see Table 6-2).

Incorporating the limiting site boundary long-term average undepleted atmospheric dispersion factor from Table 6-5 ($\{1.05E-06 \text{ sec}/\text{m}^3\}$) into Equation 6-8 yields the following:

Equation 6-9¹

$$D_{air}^{\gamma} = \{3.33E-02\} \sum_i Q_i * DF_i^{\gamma}$$

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

Equation 6-9 can be applied under the following conditions (otherwise justify Method I or consider Method II):

1. Normal operations (non-emergency event), and
2. Noble gas releases via the plant stack to the atmosphere.

6.7.2 Method II

Method II consists of the models, input data (dose factors) and assumptions in Regulatory Guide 1.109 (Reference [2]), except where site-specific data or assumptions have been identified in the ODCM. The general equations (B-4 and B-5) taken from Regulatory Guide 1.109 (Reference [2]), and used in the derivation of the simplified Method I approach as described in Section 9.2.4 are also applied to Method II assessments. Concurrent meteorology with the release period may be utilized for the atmospheric dispersion factor identified in Equation 6-8, and determined as indicated in Section 6.10 for the release point from which recorded effluents have been discharged.

6.8 METHOD TO CALCULATE BETA AIR DOSE FROM NOBLE GASES

CONTROL 3.2.2.2 limits the beta dose to air from noble gases at any location at or beyond the SITE BOUNDARY to 10 mrad in any quarter and 20 mrad in any year for {CCNPP Unit 3}. Dose evaluation is required at least once per 31 days.

Use Method I first to calculate the beta air dose from gaseous effluent releases during the period. Method I applies at all dose levels.

Use Method II if a more refined calculation is needed (i.e., use of actual release point parameters with annual or actual meteorology to obtain release-specific χ/Q_s) or if Method I predicts a dose greater than the Control limit to determine if it had actually been exceeded. See Section 9.2.5 for basis.

6.8.1 Method I

The general form of the beta air dose equation is:

Equation 6-10

$$D_{air}^{\beta} = 3.17E + 04 * [\chi / Q] \sum_i Q_i * DF_i^{\beta}$$

where:

- D_{air}^{β} = the beta air dose in mrad,
- 3.17E+04 = the number of pCi per Ci divided by the number of seconds per year,
- $[\chi / Q]$ = the maximum offsite receptor location long-term average undepleted atmospheric dispersion factor in sec/m^3 ,
- Q_i = the number of curies of noble gas "i" released and
- DF_i^{β} = The beta air dose factor for a uniform semi-infinite cloud of radionuclide "i" in $(\text{mrad}\cdot\text{m}^3)/(\text{pCi}\cdot\text{yr})$ (see Table 6-2).

Incorporating the long-term average undepleted atmospheric dispersion factor from Table 6-5 ($\{1.05E-06 \text{ sec/m}^3\}$) into Equation 6-10 yields the following:

Equation 6-11¹

$$D_{air}^{\beta} = \{3.33E - 02\} \sum_i Q_i * DF_i^{\beta}$$

Equation 6-11 can be applied under the following conditions (otherwise justify Method I or consider Method II):

1. Normal operations (non-emergency event), and
2. Noble gas releases via the plant stack to the atmosphere.

6.8.2 Method II

Method II consists of the models, input data (dose factors) and assumptions in Regulatory Guide 1.109 (Reference [2]), except where site-specific data or assumptions have been identified in the ODCM. The general equations (B-4 and B-5) taken from Regulatory Guide 1.109 (Reference [2]), and used in the derivation of the simplified Method I approach as described in the Section 9.2.5 are also applied to Method II assessments. Concurrent meteorology with the release period may be utilized for the atmospheric dispersion factor identified in Equation 6-10, and determined as indicated in Section 6.10 for the release point from which recorded effluents have been discharged.

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

6.9 METHOD TO CALCULATE CRITICAL ORGAN DOSE FROM IODINES, TRITIUM, AND PARTICULATES

CONTROL 3.2.2.3 limits the critical organ dose to a MEMBER OF THE PUBLIC from radioactive iodines, tritium, and particulates with half-lives greater than 8 days in gaseous effluents to 7.5 mrem per quarter and 15 mrem per year per unit. CONTROL 3.2.3 limits the total body and organ dose to any real MEMBER OF THE PUBLIC from all {CCNPP Unit 3} sources (including gaseous effluents) to 25 mrem in a year except for the thyroid, which is limited to 75 mrem in a year.

Use Method I first to calculate the critical organ dose from gaseous effluent releases as it is simpler to execute and more conservative than Method II.

Use Method II if a more refined calculation of critical organ dose is needed (i.e., Method I indicates the dose is greater than the limit). See Section 9.2.6 for basis.

6.9.1 Method I

The general form of the equation for calculating doses associated with release of iodines, tritium and particulates is:

Equation 6-12¹

$$D_{co} = \sum_i (Q_i * DFG_{ico})$$

where:

- D_{co} = the critical organ dose in mrem associated with the release of iodines (I-131, I-133), particulates ($T_{1/2} > 8$ days) and tritium,
- Q_i = the total activity (Ci) released from the stack of radionuclide "i" and
- DFG_{ico} = the site-specific critical organ dose factor (mrem/Ci) for the maximum SITE BOUNDARY for nuclide "i" released from the plant vent stack. For each radionuclide, this dose factor represents the age group and organ with the largest dose. (from Table 6-3).

Equation 6-12 can be applied under the following conditions (otherwise justify Method I or consider Method II):

1. Normal operations (non-emergency event),
2. Iodine, tritium, and particulate releases via the plant stack to the atmosphere and
3. Any continuous or batch release over any time period.

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

6.9.2 Method II

Method II consists of the models, input data and assumptions in Appendix C of Regulatory Guide 1.109 (Reference [2]), except where site-specific data or assumptions have been identified in the ODCM (see Table 9-2 and Table 9-3). The critical organ dose will be determined based on the location (SITE BOUNDARY, nearest resident, or farm) of receptor pathways, as identified in the most recent annual Land Use Census, or by conservatively assuming the existence of all pathways (ground plane, inhalation, ingestion of stored and leafy vegetables, milk and meat) at an offsite location of maximum potential dose. Concurrent meteorology with the release period may be utilized for determination of atmospheric dispersion factors in accordance with Section 6.10 for the release point from which recorded effluents have been discharged. The maximum critical organ dose will consider the four age groups independently, and use a shielding factor (S_F) of 0.7 associated with residential structures.

6.10 RECEPTOR POINTS AND ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS FOR IMPORTANT EXPOSURE PATHWAYS

The gaseous effluent dose equations for Method I have been simplified by assuming an individual whose location and living habits lead to a higher dose than anyone else. The following exposure pathways to gaseous effluents listed in Regulatory Guide 1.109 (Reference [2]) have been considered for radiodines, tritium and particulates with half-lives greater than 8 days:

1. Direct exposure to contaminated ground;
2. Inhalation of air;
3. Ingestion of vegetables;
4. Ingestion of goat's milk; and
5. Ingestion of meat.

Section 6.10.1 details the selection of important offsite locations and receptors. Section 6.10.2 describes the atmospheric model used to convert meteorological data into atmospheric dispersion factors and presents the maximum atmospheric dispersion factors calculated at each of the offsite receptor locations.

6.10.1 Receptor Locations

The most limiting SITE BOUNDARY location in which individuals are, or are likely to be located as a place of residence was assumed to be the receptor for all the gaseous pathways considered (SITE BOUNDARIES bordered by waters of the {Chesapeake Bay} were not considered due to controlled occupancy). This provides a conservative estimate of the dose to an individual from existing and potential gaseous pathways for the Method I analysis. Distances from the {CCNPP Unit 3} stack to the SITE BOUNDARY are provided in Table 6-4. {The SITE BOUNDARY locations in the NNW through ESE sectors are located over water, and consequently are not used as locations for determining maximum offsite receptors.}

6.10.2 {Calvert Cliffs Nuclear Power Plant Unit 3 Atmospheric Dispersion Model

The time average atmospheric dispersion factors for use in both Method I and Method II are computed for routine releases using a methodology consistent with US NRC Regulatory Guide 1.111 (Revision 1) criteria and the methodology for calculating routine release diffusion factors as represented by the XOQDOQ computer code (NUREG/CR-2919). The primary vent stack is treated as a "mixed-mode" release, as defined in Regulatory Guide 1.111. Effluents are considered to be part-time ground level/part-time elevated releases depending on the ratio of the primary vent stack effluent exit velocity relative to the speed of the prevailing wind.

In addition, Regulatory Guide 1.111 (Revision 0) discusses that constant mean wind direction models do not describe spatial and temporal variations in airflow such as the recirculation of airflow which can occur during prolonged periods of

atmospheric stagnation. For sites near large bodies of water like Calvert Cliffs, the onset and decay of sea breezes can also result in airflow reversals and curved trajectories. Consequently, Regulatory Guide 1.111(Revision 0) states that adjustments to constant mean wind direction model outputs may be necessary to account for such spatial and temporal variations in air flow trajectories. Recirculation correction factors have been applied to the diffusion factors. The recirculation correction factors used are compatible to the "default open terrain" recirculation correction factors used by the XOQDOQ computer code.

The relative deposition rates, D/Q values, were derived using the relative deposition rate curves presented in Regulatory Guide 1.111 (Revision 1). These curves provide estimates of deposition rates as a function of plume height, stability class, and plume travel distance.

Based on the above, the following average atmospheric dispersion factors for each location were generated:

1. Undepleted χ/Q dispersion factors for evaluating ground level concentrations of noble gases;
2. Depleted χ/Q dispersion factors for evaluating ground level concentrations of iodines and particulates; and
3. D/Q deposition factors for evaluating dry deposition of elemental radioiodines and other particulates.

Actual measured meteorological data for the six-year period, 2000 through 2005, were analyzed to determine all the values and locations of the maximum offsite long-term average atmospheric dispersion factors. Each dose and dose rate calculation incorporates the maximum applicable offsite long-term average atmospheric dispersion factor. The values for the dispersion factors used in the development of the ODCM Method I dose models are summarized in Table 6-5. The most limiting SITE BOUNDARY location is in the southeast sector, 1413 meters (0.88 miles) from the center of the CCNPP Unit 3 stack for undepleted χ/Q , depleted χ/Q and D/Q calculations.}

Table 6-4 (Page 1 of 1)

{Calvert Cliffs Nuclear Power Plant Unit 3 Site Boundary Distances}

Downwind Sector	Distance to Site Boundary from Plant Stack
N*	623.4 m
NNE*	429.4 m
NE*	443.3 m
ENE*	471.0 m
E*	554.1 m
ESE*	692.7 m
SE	1413 m
SSE	1607 m
S	1385 m
SSW	1371 m
SW	1759 m
WSW	1662 m
W	1732 m
WNW	2313 m
NW	1662 m
NNW*	761.9 m

* Site boundary for this sector borders the water of the Chesapeake Bay.

Table 6-5

{Calvert Cliffs Nuclear Power Plant Unit 3 Long-Term Average Dispersion Factors* - Primary Vent Stack}

	Dose Rate to Individual			Dose to Air		Dose to Critical Organ
	Total Body	Skin	Critical Organ	Gamma	Beta	Thyroid
χ/Q depleted $\left(\frac{\text{sec}}{\text{m}^3}\right)$	-	-	9.49E-07	-	-	9.49E-07
χ/Q undepleted $\left(\frac{\text{sec}}{\text{m}^3}\right)$	1.05E-06	1.05E-06	-	1.05E-06	1.05E-06	-
D/Q $\left(\frac{1}{\text{m}^2}\right)$	-	-	1.05E-08	-	-	1.05E-08

* {SE site boundary, 1413 meters (0.88 miles) from the CCNPP Unit 3 stack.}

6.11 METHOD TO CALCULATE DIRECT DOSE FROM PLANT OPERATIONS

CONTROL 3.2.3 restricts the dose to the whole body or any organ to any MEMBER OF THE PUBLIC from all uranium fuel cycle sources to 25 mrem in a calendar year (except the thyroid, which is limited to 75 mrem). Direct radiation from contained sources is required to be included in the assessment of compliance with this standard.

The direct dose from the station will be determined by obtaining the dose from Thermoluminescent Dosimeter (TLD) locations situated on-site near potential sources of direct radiation, as well as those TLDs near the SITE BOUNDARY which are part of the environmental monitoring program, and subtracting out the dose contribution from background. Additional methods to calculate the direct dose may also be used to supplement the TLD information, such as high pressure ion chamber measurements, or analytical design calculations of direct dose from identified sources (such as solid waste storage facilities).

The dose determined from direct measurements or calculations will be related to the nearest real person offsite to assess the contribution of direct radiation to the total dose limits of CONTROL 3.2.3 in conjunction with liquid and gaseous effluents.

6.12 CUMULATIVE DOSE

CONTROL 3.2.3 restricts the annual dose to the whole body or any organ of a MEMBER OF THE PUBLIC from all uranium fuel cycle sources (including direct radiation) to 25 mrem (except the thyroid, which is limited to 75 mrem). These cumulative dose contribution limits from liquids and gaseous effluents, and direct radiation, implement the Environmental Protection Agency (EPA) 40CFR190, "Environmental Standards for the Uranium Fuel Cycle."

Compliance with the Controls dose objectives for the maximum individual, as calculated by the methods described in Sections 6.2, 1.1, 6.7, 6.8 and 6.9 also demonstrates compliance with the EPA limits to any MEMBER OF THE PUBLIC. This indirect determination of compliance is based on the fact that the liquid and gaseous dose objectives are taken from 10 CFR 50, Appendix I, and represent lower values than the 40CFR190 dose limits. Direct radiation dose from contained sources is not expected to be a significant contributor to the total dose to areas beyond the SITE BOUNDARY. If the operational dose objectives in CONTROLS 3.2.1.2.a, 3.2.1.2.b, 3.2.2.2.a, 3.2.2.2.b, 3.2.2.3.a or 3.2.2.3.b are determined to be exceeded, a Special Report must be prepared. The purpose of this Special Report is to determine by direct assessment if the cumulative dose (calendar year) to any MEMBER OF THE PUBLIC (real individual) from all sources is within the limits of the Total Dose CONTROL 3.2.3.

In addition, Section 4.2, "Radioactive Effluent Release Report," requires that an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from all effluent and direct radiation sources be included for the previous calendar year to show compliance with 40CFR190.

When required, the total dose to a MEMBER OF THE PUBLIC will be calculated for all significant effluent release points for all real pathways, including direct radiation. This will include contribution from any other uranium fuel cycle facilities within five miles. The calculations will be based on the liquid and gaseous Methods II dose models as described in Section 6.0, including usage factors and other documented site-specific parameters reflecting realistic assumptions, where appropriate. The liquid and gaseous effluent Method II models are derived from the methods given in Regulatory Guide 1.109 (Reference [2]).

The direct radiation component from the facility can be determined using environmental TLD results projected to areas where MEMBERS OF THE PUBLIC could be located as noted in Section 6.11 (or alternately, high pressure ion chamber measurements or analytical design calculations for estimating the direct radiation dose from identified contained radioactive sources within the facility).

7.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The existing {(pre-Unit 3 operations)} radiological environmental monitoring stations are listed in Table 7.1-1. The locations of the stations with respect to the {Calvert Cliffs Nuclear Power Plant Unit 3} are shown on the maps in Figure 7.1-2 to Figure 7.1-3.

All radiological analyses for environmental samples are performed at a contractor laboratory. The contractor laboratory participates in an Interlaboratory Comparison Program for all relevant species in an aqueous (water) matrix. An independent vendor who participates in measurement assurance activities traceable to NIST supplies the remaining cross check samples. These samples are presented on an air filter and in milk and water matrices. This independent vendor will participate in measurement assurance activities with NIST.

Pursuant to Surveillance 3.3.2.1, the Land Use Census will be conducted "during the growing season" at least once per 12 months. The growing season is defined, for the purposes of the Land Use Census, as the period from {June 1 to October 1}. The method to be used for conducting the census will consist of one or more of the following, as appropriate: door-to-door survey, visual inspection from roadside, aerial survey, or consulting with local agricultural authorities.

Section 4.1 requires that the results of the Radiological Environmental Monitoring Program be summarized in the Annual Radiological Environmental Operating Report "in a similar format of the table in the Radiological Assessment Branch Technical Position, Revision 1, 1979."

7.1 {REMP MODIFICATIONS CCNPP UNIT 3

Table 7.1-2 lists the location of the operational CCNPP Unit 3 radiological environmental sampling locations. The operational program shares many of the same sampling locations with those used for CCNPP Units 1 and 2, along with several additional locations specific to CCNPP Unit 3 for various media.

Changes to the existing CCNPP Unit 1 and 2 REMP results from the location on the site property of CCNPP Unit 3. CCNPP Unit 3 is centered approximately 0.5 miles (0.8 km) south-southeast (SSE) of the center line between CCNPP Units 1 and 2. This places the CCNPP Unit 3 construction footprint in the site area where an existing Units 1 and 2 REMP air particulate and radioiodine sampler (Station A1) and TLD (DR7) are situated. This requires the relocation of the monitoring equipment to an area outside of that portion of the site that is involved with CCNPP Unit 3 construction. Prior to initiation of construction activities for CCNPP Unit 3, replacement sampling equipment will be located in the southern sector from CCNPP Units 1 and 2 near the SITE BOUNDARY (as power availability and road access permit). In addition, three vegetation species sample locations (Ib4, Ib5 and Ib6) also are impacted by the CCNPP Unit 3 construction footprint and will be relocated to be near the new site of A1 air particulate and radioiodine collection equipment, as shown on Figure 7.1-2.

In addition to Unit 1 and 2 sample locations directly impacted by the placement of Unit 3, one additional air particulate and iodine sampler (including TLD) location, as shown on Figure 7.1-2 is included in the operational program for CCNPP Unit 3 REMP. This air sampler will begin operations at least two years prior to CCNPP Unit 3 startup to cover the south-southwest (SSW) SITE BOUNDARY area as viewed from the CCNPP Unit 3

location. This sampler addition will provide coverage to satisfy REMP siting criteria which stipulates that there are at least three samplers close to CCNPP SITE BOUNDARY locations of highest calculated annual average ground-level deposition rates (D/Q's). Sample collections from this airborne monitoring location will include the same sample collection frequency, type of analysis and detection limits as applied to all other airborne samples as detailed in Section 3.3.

An additional surface water sampling site near the CCNPP Unit 3 discharge location in the Chesapeake Bay as shown on Figure 7.1-2 will be added to the CCNPP Unit 3 REMP since the CCNPP Unit 3 discharge point is several thousand feet south of the existing sampling location for the discharge from CCNPP Units 1 and 2. Sample collections from this surface station will be initiated at least two years prior to CCNPP Unit 3 startup, and will include the sample collection frequency, type of analysis and detection limits as applied to all other water samples as detailed in Section 3.3.}

7.2 GROUND WATER PROTECTION PROGRAM

Prior to fuel load, {CCNPP Unit 3} shall develop a written Ground Water Protection Initiative (GPI) program describing the approach to assure timely detection and effective response to situations involving inadvertent radiological releases to ground water following the guidance provided in NEI 07-07, "Industry Ground Water Protection Initiative – Final Guidance Document." This program shall include the following:

- Analysis of site hydrology and geology to determine predominant ground water flow characteristics and gradients and potential pathways for ground water migration from on-site locations to offsite locations including periodic reviews to identify possible changes in site hydrology.
- Performance of a site risk assessment that evaluates all systems, structures, or components (SSCs) that contain or could contain licensed material and for which there is credible mechanism for the licensed material to reach ground water along with work practices involving licensed material for which there is credible mechanism for the licensed material to reach ground water.
- Establishment of on-site ground water monitoring to ensure timely detection of inadvertent radiological releases to ground water. Sampling and analysis protocols shall be established, including analytical sensitivity requirements, for ground water and soil.
- Establishment of a remediation protocol to prevent migration of licensed material offsite and to minimize decommissioning impacts.
- Establishment of a record keeping program to record leaks, spills and remediation efforts to meet the requirements of 10 CFR 50.75(g).
- Communication with state/local officials, with follow-up notification to the NRC, regarding significant on-site leaks/spills into ground water and on-site or offsite water sample results exceeding the criteria given in Table 3.3.1-2.

The current predominant ground water flow characteristics and gradients are described in {CCNPP Unit 3} FSAR Sections 2.4.12 and 2.4.13. Preliminary ground water monitoring locations have been developed based on current site hydrological characteristics along with review of buildings containing considerable volume of radioactive liquid. These preliminary ground water sampling locations are given in Table

7.1-2 and Figures 7.1-1 and 7.1-2. The placement of these ground water monitoring locations may change based on future (post-construction) site hydro-geologic studies. The sampling and analysis protocol associated with these ground water monitoring locations shall be established after on-site construction is complete.

Table 7.1-1 (Page 1 of 1)

{Existing CCNPP Units 1 and 2 Radiological Environmental Monitoring Program Locations}

Sample Site/Type	Sector	Distance ^a		Description
		km	mi	
DR1	NW	0.6	0.4	Onsite, Along Cliffs
DR2	WNW	2.7	1.7	Rt. 765, Auto Dump
DR3	W	2.3	1.4	Rt. 765, Giovanni's Tavern (Knotty Pine)
DR4	WSW	2.0	1.2	Rt. 765, Across from White Sand Drive
DR5	SW	2.4	1.5	Rt. 765 at Johns Creek
DR6, A4	SSW	2.9	1.8	Rt. 765 at Lusby, Frank's Garage
DR7, A 1, lb4, lb5, lb6	S	0.7	0.5	Onsite, before entrance to Camp Conoy
DR8, A2	SSE	2.5	1.5	Camp Conoy Road at Emergency Siren
DR9, A3	SE	2.6	1.6	Bay Breeze Road
DR10	NW	6.4	4.0	Calvert Beach Rd & Decatur St.
DR11	WNW	6.6	4.1	Dirt Road off Mackall Rd & Parran Rd
DR12	W	6.7	4.2	Bowen Rd & Mackall Rd
DR13	WSW	6.1	3.8	Mackall Rd near Wallville
DR14	SW	6.4	4.0	Rodney Point
DR15	SSW	6.2	3.9	Mill Bridge Rd & Turner Rd
DR16	S	6.5	4.1	Across from Appeal School
DR17	SSE	5.9	3.7	Cove Point Rd & Little Cove Point Rd
DR18	SE	7.1	4.5	Cove Point
DR19	NW	4.4	2.8	Long Beach
DR20	NNW	0.4	0.3	Onsite, near shore
DR21, A5, lb7, lb8, lb9	WNW	19.3	12.1	Emergency Operations Facility
DR22	S	12.5	7.8	Solomons Island
DR23	ENE	12.6	7.9	Taylor's Island, Carpenter's Property
Wa1	NNE	0.2	0.1	Intake Area
Wa2, la1, la2	N	0.3	0.2	Discharge Area
Wb1	ESE	0.6	0.4	Shoreline at Barge Road
lb 1, lb2, lb3,	SSE	2.6	1.6	Garden Plot off Bay Breeze Rd
la4, la5	(Area not influenced by the plant discharge)			Patuxent River
la3	E	0.9	0.6	Camp Conoy
la6	NNW	10.7	6.7	Kenwood Beach
la10	SSE	15.3	9.5	Hog Island

^a Distance and direction are from the central point between the CCNPP Unit 1 and 2 containment buildings.

Key:

- # The sequential number of the sampling station)
- DR# Direct Radiation, TLD Station
- A# Airborne Sampling Station
- Wa# Waterborne Sampling Station at Intake (Wa1) and Discharge (Wa2)
- Wb1 Waterborne Sediment Sampling Station
- la# Fish and Invertebrates Sampling Station
- lb# Broad Leaf Sampling Station

Table 7.1-2 (Page 1 of 2)

{Operational CCNPP Unit 3 Radiological Environmental Monitoring Program Locations}

Sample Site/Type	Sector	Distance ^a		Description
		km	mi	
DR1	NW	0.6	0.4	Onsite, Along Cliffs
DR2	WNW	2.7	1.7	Rt. 765, Auto Dump
DR3	W	2.3	1.4	Rt. 765, Giovanni's Tavern (Knotty Pine)
DR4	WSW	2.0	1.2	Rt. 765, Across from White Sand Drive
DR5	SW	2.4	1.5	Rt. 765 at Johns Creek
DR6, A4	SSW	2.9	1.8	Rt. 765 at Lusby, Frank's Garage
DR7*, A1*, lb4*, lb5*, lb6*	S	2.9	1.8	Relocated from footprint of Unit 3 to near site boundary
DR8, A2	SSE	2.5	1.5	Camp Conoy Road at Emergency Siren
DR9, A3	SE	2.6	1.6	Bay Breeze Road
DR10	NW	6.4	4.0	Calvert Beach Rd & Decatur St.
DR11	WNW	6.6	4.1	Dirt Road off Mackall Rd & Parran Rd
DR12	W	6.7	4.2	Bowen Rd & Mackall Rd
DR13	WSW	6.1	3.8	Mackall Rd near Wallville
DR14	SW	6.4	4.0	Rodney Point
DR15	SSW	6.2	3.9	Mill Bridge Rd & Turner Rd
DR16	S	6.5	4.1	Across from Appeal School
DR17	SSE	5.9	3.7	Cove Point Rd & Little Cove Point Rd
DR18	SE	7.1	4.5	Cove Point
DR19	NW	4.4	2.8	Long Beach
DR20	NNW	0.4	0.3	Onsite, near shore
DR21, A5, lb7, lb8, lb9	WNW	19.3	12.1	Emergency Operations Facility
DR22	S	12.5	7.8	Solomons Island
DR23	ENE	12.6	7.9	Taylor's Island, Carpenter's Property
DR24*, A6*	SSW	2.9	1.8	New Air sampler (TLD) specific Unit 3
Wa1	NNE	0.2	0.1	Intake Area
Wa2, la1, la2	N	0.3	0.2	Discharge Area (Unit 1 and 2)
Wa3*	E	1.0	0.6	Near Discharge area of Unit 3

^a Distance and direction are from the central point between the CCNPP Unit 1 and 2 containment buildings.

Key: # The sequential number of the sampling station. An asterisk (*) following a station number indicates location changes due to the Unit 3 operational REMP.

- DR# Direct Radiation, TLD Station
- A# Airborne Sampling Station
- Wa# Waterborne Sampling Station at Intake and Discharges
- Wb# Waterborne Sediment Sampling Station
- la# Fish and Invertebrates Sampling Station)
- lb# Broad Leaf Sampling Station)
- Wg# Ground water Sampling Station

Table 7.1-2 (Page 2 of 2)

{Operational CCNPP Unit 3 Radiological Environmental Monitoring Program Locations}

Sample Site/Type	Sector	Distance ^a		Description
		km	mi	
Wb1	ESE	0.6	0.4	Shoreline at Barge Road
Wg1*	Protected Area ^b			Near Nuclear Island (see Figure 7.1-1)
Wg2*	Protected Area ^b			Near Nuclear Island (see Figure 7.1-1)
Wg3*	Protected Area ^b			Near Nuclear Island (see Figure 7.1-1)
Wg4*	Protected Area ^b			Near Nuclear Island (see Figure 7.1-1)
Wg5*	Protected Area ^b			Near Nuclear Island (see Figure 7.1-1)
Wg6*	Protected Area ^b			Near Nuclear Island (see Figure 7.1-1)
Wg7*	Protected Area ^b			Near Nuclear Island (see Figure 7.1-2)
Wg8*	Protected Area ^b			Near Retention Basin (see Figure 7.1-2)
Ib1, Ib2, Ib3,	SSE	2.6	1.6	Garden Plot off Bay Breeze Rd
Ia4, Ia5	(Area not influenced by Plant Discharge)			Patuxent River
Ia3	E	0.9	0.6	Camp Conoy.
Ia6	NNW	10.7	6.7	Kenwood Beach
Ia10	SSE	15.3	9.5	Hog Island

^a Distance and direction are from the central point between the CCNPP Unit 1 and 2 containment buildings.

^b Ground water sampling locations shall be located down gradient (ground water flow) of facilities at a depth sufficient to monitor the aquifer.

Key:

- # The sequential number of the sampling station. An asterisk (*) following a station number indicates location changes due to the Unit 3 operational REMP.
- DR# Direct Radiation, TLD Station
- A# Airborne Sampling Station
- Wa# Waterborne Sampling Station at Intake and Discharges
- Wb# Waterborne Sediment Sampling Station
- Ia# Fish and Invertebrates Sampling Station)
- Ib# Broad Leaf Sampling Station)
- Wg# Ground Water Sampling Station

Figure 7.1-2
{CCNPP Sampling Locations 0-2 Miles (0-3.2 km)}

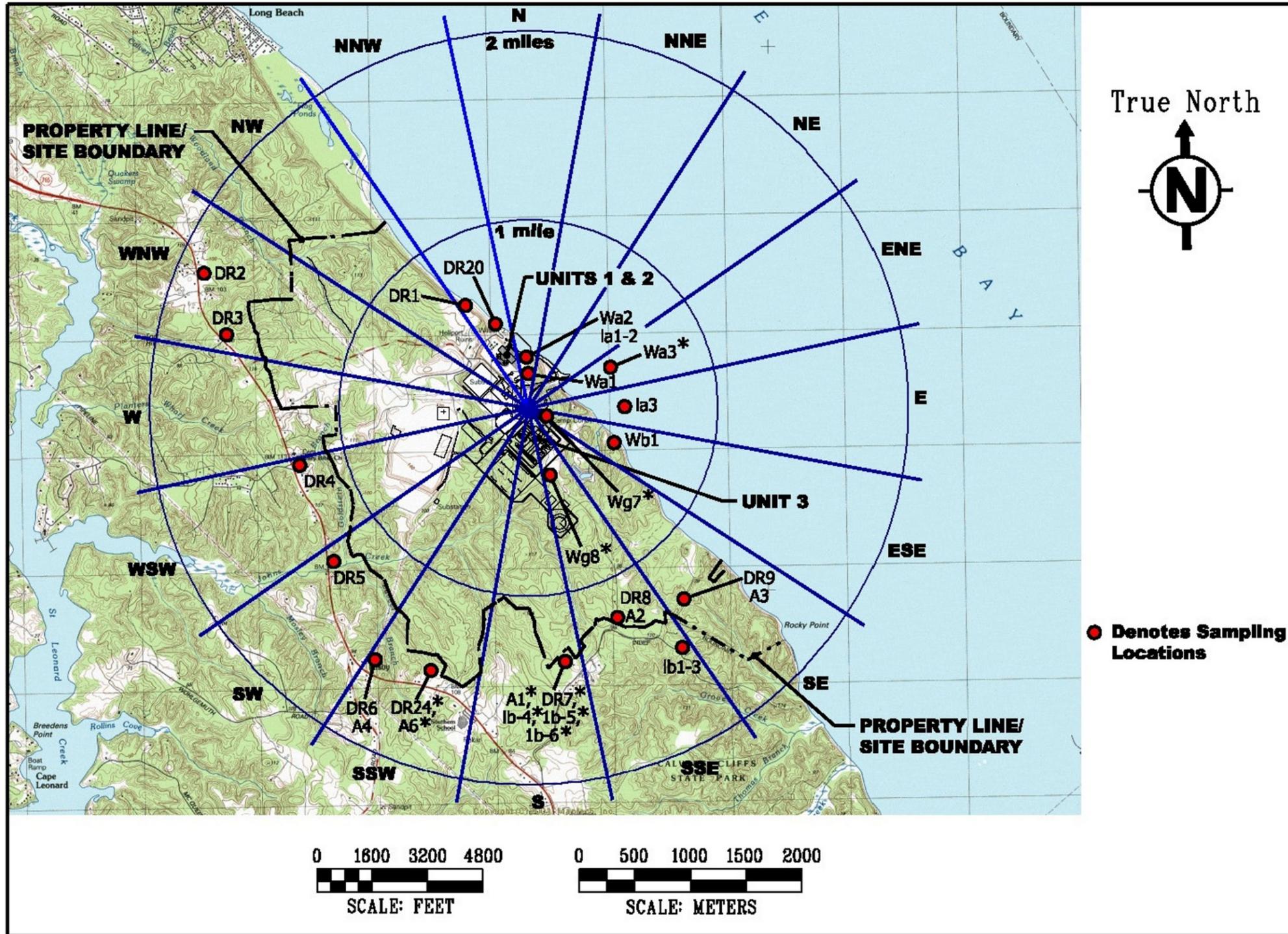
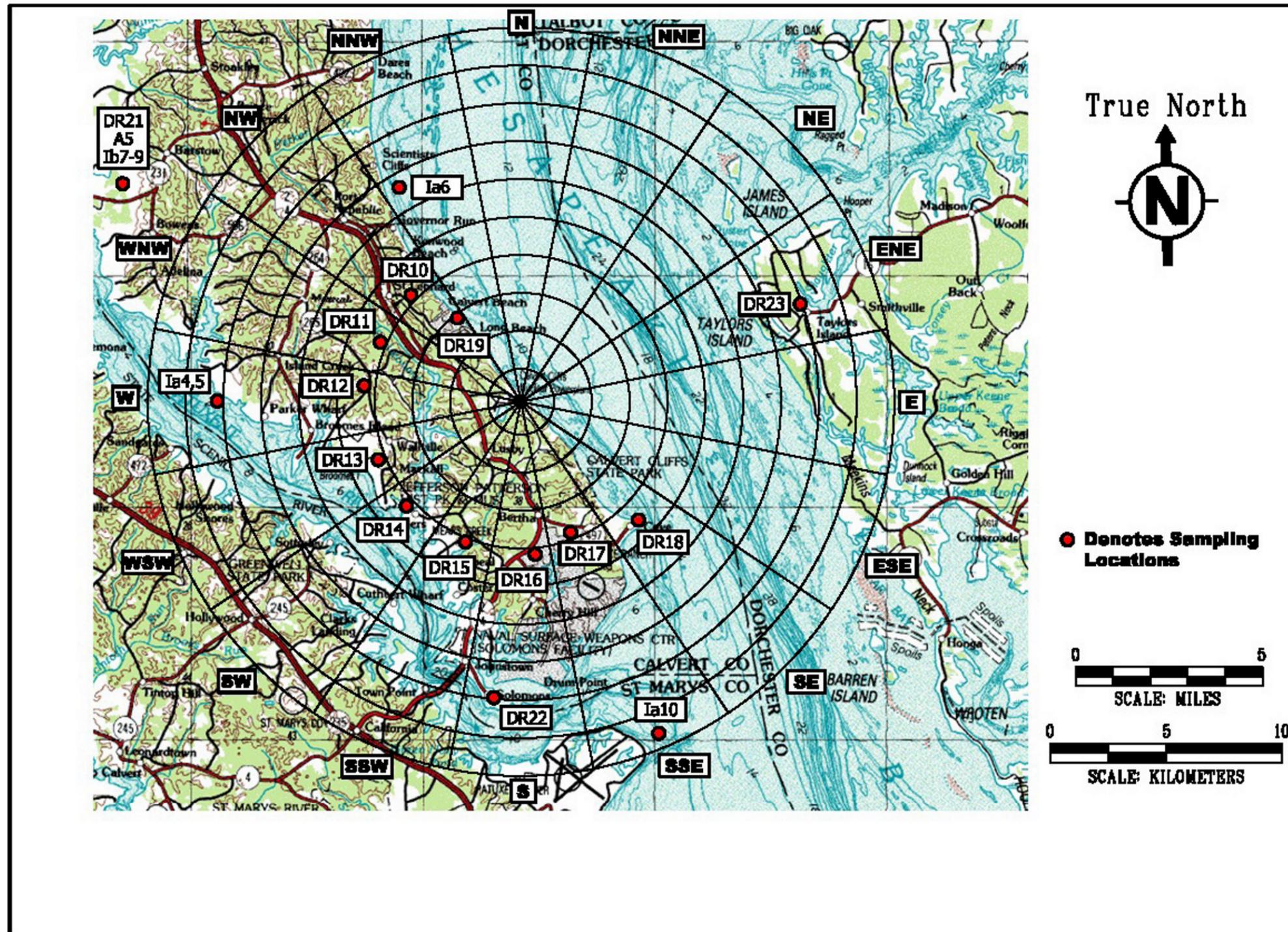


Figure 7.1-3

{CCNPP Sampling Locations 0-10 Miles (0-16 km)}



8.0 SETPOINT DETERMINATIONS

8.1 LIQUID EFFLUENT INSTRUMENTATION SETPOINTS

8.1.1 Introduction

Radiation monitors are used to:

Measure the radioactivity level in liquid effluent streams directly released to UNRESTRICTED AREAS (liquid effluent line and turbine building effluent line) and provide alarm/isolation function on high radioactivity, or

Provide indication of abnormally high levels of radioactivity within specific plant systems (SG Blowdown Line and Component Cooling Water System). These systems do not have a direct flow path to UNRESTRICTED AREAS, but may divert their flow to the LIQUID WASTE PROCESSING SYSTEM.

A schematic diagram of the liquid radwaste effluent flow paths is provided in Figure 8.1-1.

8.1.2 Liquid Radwaste Effluent Line Monitor and Setpoints

8.1.2.1 Flow Rate

The liquid effluent flow to the environment is made in a batch mode from the LIQUID WASTE PROCESSING SYSTEM and is controlled so that the instantaneous concentration of radionuclides in the liquid effluent released to UNRESTRICTED AREAS is less than 10 times the effluent concentration limit of 10 CFR 20 Appendix B, Table 2, Column 2. All monitoring tanks are mixed, with a representative sample taken prior to release to determine radionuclide content. The required dilution factor is then determined by comparing the known radionuclide concentration to the ECL:

Equation 8-1

$$DF = \sum_i \frac{C_i}{(10 * ECL_i)}$$

where:

DF = required dilution factor to be applied to the undiluted liquid effluent flow (unitless)

C_i = the known concentration of the radionuclide, i , in the monitoring tanks ($\mu\text{Ci/ml}$)

ECL_i = effluent concentration limit for the radionuclide, i , from 10 CFR 20 Appendix B, Table 2, Column 2 ($\mu\text{Ci/ml}$).

If $DF \leq 1.0$, then no dilution is required and the release rate is unrestricted.

If $DF > 1.0$, then dilution flow is required and is calculated by:

Equation 8-2

$$f \leq \left(\frac{F}{(DF-1)} \right) * (SF)$$

where:

f = maximum release rate of undiluted liquid effluent (gpm)

F = the dilution flow available (gpm)

SF = Safety Factor, 0.8

8.1.2.2

Liquid Radwaste Effluent Line Setpoints

The liquid effluent instrumentation alarm setpoint is determined so that the instantaneous concentration of radioactive material released in liquid effluents at the boundary of the UNRESTRICTED AREA does not exceed 10 times the effluent concentration limits in 10 CFR 20 Appendix B, Table 2, Column 2 (10 CFR 20.1302(b)(2)(i)).

The contents of the monitoring tanks are discharged to the environment in a batch mode only after a sample has been taken, analyzed in the laboratory, and met the release criteria. The discharge valves can only be opened by a key on the control panel, and the maximum flow rate from the tank is determined per Equation 8-2.

The liquid effluent line radiation monitors continuously measure the activity contained in the liquid waste discharge line and automatically suspend discharges from the liquid waste monitoring system by closing the shut-off valves (30KPD29AA001/002) whenever the setpoint is exceeded.

The alarm setpoint is calculated by:

Equation 8-3

$$SP \leq \sum_i \left(\frac{(F + f)(10 * ECL_i)}{f} \right) + bkg$$

where:

SP = Setpoint ($\mu\text{Ci/ml}$)

f = calculated in Equation 8-2 (gpm)

bkg = background reading for the monitor ($\mu\text{Ci/ml}$)

Additional lower setpoints may be used in order to provide additional information to the plant operators of abnormal conditions warranting investigation prior to exceeding any release limits.

Instrument Number/Location: 30KPK29CR001/CR002 in 30UKS01 077.

Range: 5E-06 to 1E-03 µCi/ml

8.1.2.3

Example Setpoint Calculation

The radioactivity concentration in the LIQUID WASTE PROCESSING SYSTEM monitoring tank is analyzed prior to release. The following is an example setpoint calculation based on the following data:

Nuclide (<i>i</i>)	C_i (µCi/ml)	ECL_i (µCi/ml)	$10 * ECL_i$ (µCi/ml)
Cs-134	1.2E-05	9E-07	9E-06
Cs-137	3.5E-05	1E-06	1E-05
Co-60	1.1E-05	3E-06	3E-05
H-3	9.0E-02	1E-03	1E-02

For this example, the dilution flow available (F) is set at 9,000 gpm, and gamma background is 1E-05 µCi/ml.

The required dilution factor for all nuclides (gamma + beta emitters) is:

$$DF = \sum_i \frac{C_i}{(10 * ECL_i)} = \left[\left(\frac{1.2E - 05}{9E - 06} \right) + \left(\frac{3.5E - 05}{1E - 05} \right) + \left(\frac{1.1E - 05}{3E - 05} \right) + \left(\frac{9.0E - 02}{1E - 02} \right) \right] = 14.2$$

As the DF is > 1.0, the maximum release rate is calculated by:

$$f \leq \left(\frac{F}{(DF - 1)} \right) * (SF) = \frac{9,000\text{gpm}}{(14.2 - 1)} * 0.8$$

$$f \leq 545\text{gpm}$$

The nominal flow rate from the monitoring tanks is approximately 175 gpm based on pump parameters. Since the nominal flow rate is less than the maximum release rate, the nominal flow rate is used to determine the setpoint.

The setpoint is calculated using only the gamma-emitters, as the in-line monitor only detects gammas:

$$\begin{aligned}
 SP &\leq \sum_i \left(\frac{(F + f)(10 * ECL_i)}{f} \right) + bkg \\
 &= \frac{(9,000 \text{ gpm} + 175 \text{ gpm})[(9E - 06) + (1E - 05) + (3E - 05)]}{(175 \text{ gpm})} + 1E - 05 \mu\text{Ci/ml} \\
 SP &\leq 3E - 03 \mu\text{Ci/ml}
 \end{aligned}$$

8.1.3 Turbine Building Effluent Line Monitor and Setpoints

The sources of water to the turbine building drains are from the secondary steam system and the condensate polishing demineralizer. Leakage collected in the floor drain sump is directly discharged to the environment without treatment. Neither of these systems normally have detectable levels of radioactivity, and higher levels of radioactivity would indicate the presence of a significant primary-to-secondary leak.

The setpoint is set at 2-3 times background levels. Additional lower setpoints may be used in order to provide additional information to the plant operators of abnormal conditions warranting investigation.

When the setpoint is exceeded, the monitor isolates the sump for further evaluation.

Instrument Number/Location: 30UMA01CR001 in Turbine Island

Range: 3E-06 to 1E-02 $\mu\text{Ci/ml}$

8.1.4 Component Cooling System Water Monitor and Setpoints

NOTE: This is not a direct effluent path to the environment.

Gamma detectors monitor the concentration of radioactive material in the component cooling water system. The component cooling water system does not normally contain radioactive material, and higher levels of radioactivity provides indication of leakage in any of the heat exchangers within the safeguards systems Safety Injection System/Residual Heat Removal (SIS/RHR), Extra Borating System (EBS), Emergency Feed Water System (EFWS), Containment Heat Removal System (CHRS), or the high pressure coolers for the volume control system.

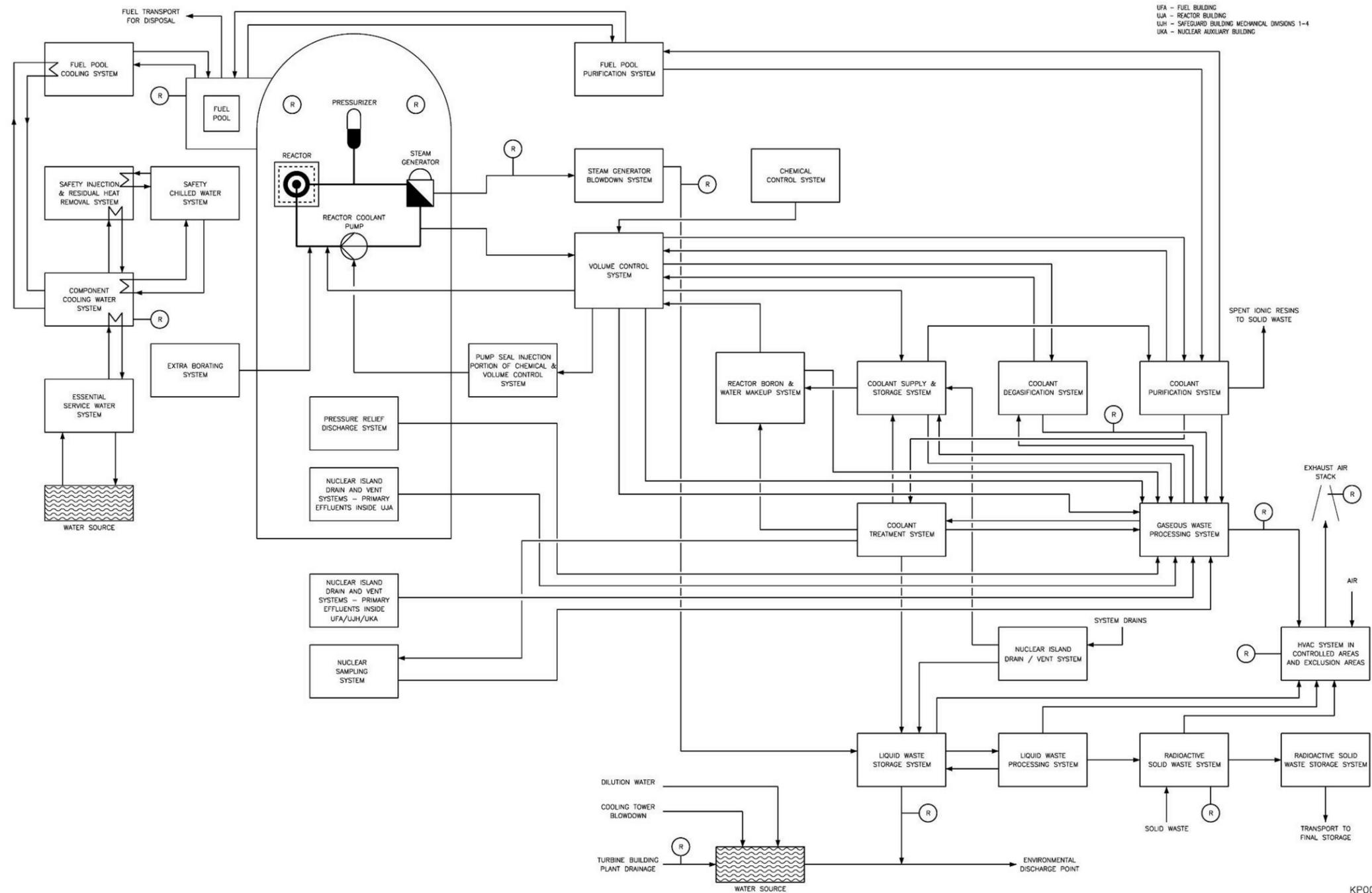
The setpoint is set at 2-3 times background levels. Additional lower setpoints may be used in order to provide additional information to the plant operators of abnormal conditions warranting investigation.

When the setpoint is exceeded, the radiation monitor initiates an alarm in the main Control Room.

Instrument Number/Location: KAA10CR001 in 31UJH01026, KAA40CR00 in 34UJH01026, KAA20CR001 in 32UJH01038, and KAA30CR001 in 33UJH01038.

Range: 1E-06 to 1E-03 $\mu\text{Ci/ml}$

Figure 8.1-1
{Liquid Radwaste Effluent Flow Paths}



KP0001T2

8.2 GASEOUS EFFLUENT INSTRUMENTATION SETPOINTS

8.2.1 Introduction

Radiation monitors are used to:

Measure the radioactivity level in gaseous effluent streams directly released to UNRESTRICTED AREAS, either directly through the main vent stack, or major processes that feed into the main vent stack (delay beds, air evacuation, and turbine condenser), or

Provide indication of abnormally high levels of radioactivity within specific plant systems (containment ventilation system).

A schematic diagram of the gaseous radwaste effluent flow paths is provided in Figure 8.2-1.

8.2.2 Main Vent Stack Effluent Line Monitor and Setpoints

The gaseous effluent instrumentation alarm setpoint is determined so that the instantaneous concentration of radioactive material released in gaseous effluents at the boundary of the UNRESTRICTED AREA from all release points does not exceed any of the following:

The CONTROL Levels specified in 3.2.2.1.a of:

For noble gases: ≤ 500 mrem/yr to the whole body and
 ≤ 3000 mrem/yr to the skin.

The setpoint used for the vent stack monitor is based on the most restrictive setpoint calculated below in 8.2.2.1 or 8.2.2.2. {The allowable effluent concentration is reduced by the presence of additional release points on site, i.e., units 1 and 2 with two separate vent stacks. The gaseous effluent line radiation monitors continuously measure the activity contained in the CCNPP Unit 3 main vent stack.}

8.2.2.1 Noble Gas Whole Body Dose rate Control Level Setpoint

For the noble gas whole body dose rate control levels, Equation 6-5 is modified to find an overall setpoint, representing the total noble gas release rate corresponding to 500 mrem/yr. A composite gamma total body dose factor is defined as:

Equation 8-4

$$DFB_c * Q_c = \sum Q_i * DFB_i$$

where:

DFB_c = Composite gamma total body dose factor in (mrem-m³)/(pCi-yr)

Q_c = total noble gas release rate = setpoint SP = $\sum \dot{Q}_i$ (μCi/sec)

\dot{Q}_i = Release rate of noble gas "i" in the mixture (μCi/sec)

DFB_i = Gamma total body dose factor for nuclide "i" from Table 6-2 (mrem-m³/pCi-yr)

Substituting this into Equation 6-5, including a factor to address multiple release points, and solving for the setpoint (Q_c) yields:

Equation 8-5

$$SP_{tb} \leq \frac{500 * \sum \dot{Q}_i}{\{1.05\} * f * \sum (\dot{Q}_i * DFB_i)}$$

where:

SP_{tb} = Setpoint (μCi/sec)

500 = Offsite limiting total body dose rate (mrem/yr)

{1.05} = The maximum offsite long-term average atmospheric dispersion factor and units conversion factor = {1.05E-06 sec/m³} x (1E+06 pCi/μCi)

f = # release points/stacks {= 3 for CCNPP}

8.2.2.2

Noble Gas Skin dose rate Control Level Setpoint

Similarly, for the noble gas skin dose rate control levels, Equation 6-6 is modified to find an overall setpoint, representing the total noble gas release rate corresponding to 3000 mrem/yr. A composite skin dose factor is defined as:

Equation 8-6

$$DF'_c * Q_c = \sum \dot{Q}_i * DF'_i$$

where:

DF'_c = Composite skin dose factor in (mrem-m³)/(pCi-yr)

Q_c = Total noble gas release rate = setpoint SP = $\sum \dot{Q}_i$ (μCi/sec)

\dot{Q}_i = Release rate of noble gas "i" in the mixture ($\mu\text{Ci}/\text{sec}$)

DF'_i = Combined skin dose rate factor for nuclide "i" in mrem/Ci from Table 6-2

Substituting this into Equation 6-6, including a factor to address multiple release points, and solving for the setpoint (Q_c) yields:

Equation 8-7

$$\text{SP}_{\text{skin}} \leq \frac{3000 * \sum \dot{Q}_i}{f * \sum (\dot{Q}_i * \text{DF}'_i)} * 3.16\text{E} - 02$$

where:

SP_{skin} = Setpoint ($\mu\text{Ci}/\text{sec}$)

3000 = Offsite limiting total skin dose rate (mrem/yr)

3.16E-02 = Conversion factor (Ci/yr to $\mu\text{Ci}/\text{sec}$)

8.2.2.3

Example Setpoint Calculation

The most restrictive setpoint for the nuclides being monitored is used for each individual instrument. The following are example setpoint calculations based on the following data:

Nuclide	\dot{Q}_i ($\mu\text{Ci/sec}$)	Gamma Total Body Dose Factor From Table 6-2 $DFB_i \left(\frac{\text{mrem} - m^3}{\text{pCi} - \text{yr}} \right)$	Combined Skin Dose Rate Factor From Table 6-2 $DF'_i \left(\frac{\text{mrem}}{\text{Ci}} \right)$
Kr-83m	4.2E+00	7.56E-08	{7.14E-07}
Kr-85m	4.5E+01	1.17E-03	{9.37E-05}
Kr-85	4.0E+00	1.61E-05	{4.49E-05}
Kr-87	7.8E+01	5.92E-03	{5.49E-04}
Kr-88	1.2E+02	1.47E-02	{6.41E-04}
Kr-89	1.4E+02	1.66E-02	{9.73E-04}
Kr-90	1.1E+02	1.56E-02	{8.44E-04}
Xe-131m	1.1E+00	9.15E-05	{2.15E-05}
Xe-133m	5.0E+01	2.51E-04	{4.49E-05}
Xe-133	3.2E+02	2.94E-04	{2.32E-05}
Xe-135m	6.6E+01	3.12E-03	{1.48E-04}
Xe-135	6.9E+01	1.81E-03	{1.32E-04}
Xe-137	9.1E+01	1.42E-03	{4.58E-04}
Xe-138	2.5E+02	8.83E-03	{4.77E-04}
Total	1.35E+03	---	---

From this table $\Sigma \dot{Q}_i = 1.35\text{E}+03 \mu\text{Ci/sec}$

$$\begin{aligned} \Sigma \dot{Q}_i DFB_i &= (4.2\text{E}+00)(7.56\text{E}-08) + (4.5\text{E}+01)(1.17\text{E}-03) \\ &\quad + (4.0\text{E}+00)(1.61\text{E}-05) + (7.8\text{E}+01)(5.92\text{E}-03) + \\ &\quad + (1.2\text{E}+02)(1.47\text{E}-02) + (1.4\text{E}+02)(1.66\text{E}-02) + \\ &\quad + (1.1\text{E}+02)(1.56\text{E}-02) + (1.1\text{E}+00)(9.15\text{E}-05) + \\ &\quad + (5.0\text{E}+01)(2.51\text{E}-04) + (3.2\text{E}+02)(2.94\text{E}-04) + \\ &\quad + (6.6\text{E}+01)(3.12\text{E}-03) + (6.9\text{E}+01)(1.81\text{E}-03) + \\ &\quad + (9.1\text{E}+01)(1.42\text{E}-03) + (2.5\text{E}+02)(8.83\text{E}-03) \\ &= 9.09\text{E}+00 \end{aligned}$$

$$\begin{aligned} \Sigma \dot{Q}_i DF'_i &= (4.2\text{E}+00)\{7.14\text{E}-07\} + (4.5\text{E}+01)\{9.37\text{E}-05\} \\ &\quad + (4.0\text{E}+00)\{4.49\text{E}-05\} + (7.8\text{E}+01)\{5.49\text{E}-04\} + \end{aligned}$$

$$\begin{aligned}
 & (1.2\text{E}+02)(\{6.41\text{E}-04\}) + (1.4\text{E}+02)(\{9.73\text{E}-04\}) + \\
 & (1.1\text{E}+02)(\{8.44\text{E}-04\}) + (1.1\text{E}+00)(\{2.15\text{E}-05\}) + \\
 & (5.0\text{E}+01)(\{4.49\text{E}-05\}) + (3.2\text{E}+02)(\{2.32\text{E}-05\}) + \\
 & (6.6\text{E}+01)(\{1.48\text{E}-04\}) + (6.9\text{E}+01)(\{1.32\text{E}-04\}) + \\
 & (9.1\text{E}+01)(\{4.58\text{E}-04\}) + (2.5\text{E}+02)(\{4.77\text{E}-04\}) \\
 & = \{5.43\text{E}-01\}
 \end{aligned}$$

For the noble gas whole body:

Equation 8-8

$$\text{SP}_{\text{tb}} \leq \frac{500 * 1.35\text{E} + 03}{\{1.05\} * \{3\} * 9.09\text{E} + 00} \text{ or}$$

$$\text{SP}_{\text{tb}} \leq \{2.36\text{E} + 04\} \mu\text{Ci}/\text{sec}$$

For the noble skin dose:

Equation 8-9

$$\text{SP}_{\text{skin}} \leq \frac{3000 * 1.35\text{E} + 03}{\{3\} * (\{5.43\text{E} - 01\})} (3.16\text{E} - 02) \text{ or}$$

$$\text{SP}_{\text{skin}} \leq \{7.86\text{E} + 04\} \mu\text{Ci}/\text{sec}$$

The most restrictive of the above is used to determine the monitor setpoint ($\{2.4\text{E}4 \mu\text{Ci}/\text{sec}\}$). Alternatively, using the most limiting expected nuclide in the mix (i.e., Kr-89 in above example) would result in the following setpoint:

Equation 8-10

$$\text{SP}_{\text{tb}} \leq \frac{500 * 1.4\text{E} + 02}{\{1.05\} * \{3\} * (1.4\text{E} + 02)(1.66\text{E} - 02)} \text{ or}$$

$$\text{SP}_{\text{tb}} \leq \{9.56\text{E} + 03\} \mu\text{Ci}/\text{sec}$$

This is bounding compared to the 10 CFR 20 Appendix B, Table 2, Column 1 limits, as shown by:

Equation 8-11

$$SP \leq \frac{\left(\frac{2.119 \text{ E} + 03 \text{ sec} - \text{ft}^3/\text{min} - \text{m}^3}{(\chi/Q)(FR)} \right) \sum_i ECL_i}{\# \text{ Release vents}} * SF + \text{bkg}$$

where:

- SP = 10 CFR 20 Setpoint ($\mu\text{Ci/ml}$)
- 2.119E+03 = Conversion factor ($\text{sec-ft}^3/(\text{min-m}^3)$)
- χ/Q = The highest calculated annual average dispersion parameter at the SITE BOUNDARY {not bordered by water} (sec/m^3)
= {1.05E-6 sec/m^3 for CCNPP Unit 3} from Table 6-5
- FR = Vent Stack Flow Rate (ft^3/min), typically 157,000 cfm
- ECL_i = Effluent concentration limit for the radionuclide, "i", from 10 CFR 20 Appendix B, Table 2, Column 1 ($\mu\text{Ci/ml}$).
- # Release vents = {3 for CCNPP}
- SF = Safety Factor, 0.8
- bkg = Background reading for the monitor ($\mu\text{Ci/ml}$)

Using the following data:

Nuclide	\dot{Q}_i ($\mu\text{Ci}/\text{sec}$)	Critical Organ Dose Rate Factor From Table 6-3 (DFG'_{ico}) ($\text{mrem}\text{-sec}/\mu\text{Ci}\text{-yr}$)	ECL_i from 10CFR 20 Appendix B, Table 2, Column 1 ($\mu\text{Ci}/\text{ml}$)
H-3	4.2E-03	{7.60E-04}	1E-07
CO-58	6.2E-04	{1.27E+00}	2E-09
CO-60	6.3E-04	{8.26E+00}	2E-10
I-131	2.3E-09	{1.54E+01}	2E-10
I-133	1.5E-08	{3.66E+00}	1E-09
Total	5.5E-03	---	1E-07

$$\begin{aligned} \sum \dot{Q}_i \text{DFG}'_{\text{ico}} &= (4.2\text{E-}03)(\{7.60\text{E-}04\}) + (6.2\text{E-}04)(\{1.27\text{E+}00\}) + (6.3\text{E-} \\ &04)(\{8.26\text{E+}00\}) + (2.3\text{E-}09)(\{1.54\text{E+}01\}) + (1.5\text{E-} \\ &08)(\{3.66\text{E+}00\}) \\ &= \{5.99\text{E-}03\} \end{aligned}$$

For the 10 CFR 20 Appendix B, Table 2, Column 1 limits:

$$\sum \text{ECL}_i = 1\text{E-}07 \mu\text{Ci}/\text{ml}$$

And assuming a background of 0.001 $\mu\text{Ci}/\text{ml}$

Equation 8-12

$$\text{SP} \leq \frac{\left(\frac{2.119 \text{E} + 03 \text{ sec} \cdot \text{ft}^3/\text{min} \cdot \text{m}^3}{(\{1.05\text{E} - 06\} \text{ sec}/\text{m}^3)(157,000 \text{ ft}^3/\text{min})} \right) (1\text{E} - 07)}{\{3\}} * 0.8 + 0.001$$

$$\text{SP} \leq \{1.3\text{E} - 03\} \mu\text{Ci}/\text{ml}$$

$$\text{SP} < \{9.6\text{E} + 04\} \mu\text{Ci}/\text{sec} \text{ using the flow rate of } 157,000 \text{ cfm } (7.41\text{E} + 07 \text{ ml}/\text{sec})$$

8.2.2.4

Instruments

Noble Gas (Kr-85, Xe-133)

Instrument Number/Location: KLK70CR001/CLK90CR001 (beta) in 30UKA34 072/092.

KLK90CR002 (gamma) in 30UKA34 092.

Range: 3E-06 to 1E+02 $\mu\text{Ci}/\text{cc}$ (3E+04 to 1E+09 $\mu\text{Ci}/\text{hr}$) (beta)

1-50,000 counts/sec (gamma)

Particulate Activity Monitor

Instrument Number/Location: KLK70CR031 in 30UKA34 072.

Range: 1E-10 to 1E-06 $\mu\text{Ci/cc}$

Iodine Sampler (I-131)

Instrument Number/Location: KLK70CR071 in 30UKA34 027.

Range: 5E-11 to 3E-07 $\mu\text{Ci/cc}$ (1E+01 to 1E+04 $\mu\text{Ci/hr}$)

8.2.3 Delay Bed Monitor and Setpoint

The KPL process monitor measures the downstream noble gas activity (beta - Kr-85) from the gas delay beds leading to the vent stack.

The setpoint is determined as in Equation 8-11, with the flow rate (FR) set to 0.0826 ft^3/min for normal operation and 7.105 ft^3/min for surge operations. Additional lower setpoints may be used in order to provide additional information to the plant operators of abnormal conditions warranting investigation.

Instrument Number/Location: KPL83CR001 in 30UKA29 001.

Range: 1E-06 to 1E+02 $\mu\text{Ci/ml}$

8.2.4 Reactor Building Ventilation Monitor and Setpoint (KLA/KLB)

The reactor building ventilation process monitors measure the exhaust air of the containment ventilation KLA2 after being filtered and prior to entering the main vent stack (for discharge to UNRESTRICTED AREAS), and suspends discharges from the containment whenever the setpoint is exceeded.

The setpoint is set at 2-3 times background levels. Additional lower setpoints may be used in order to provide additional information to the plant operators of abnormal conditions warranting investigation.

Instrument Number/Location: KLK13CR001/002 in UFA24 095.

Range: 1E-05 to 1E+00 rad/hr

8.2.5 Turbine Building Condenser Monitor and Setpoint

The condenser air removal system exhaust gas monitor measures the noble gas effluent concentration at the condenser exhaust during and following an accident in order to detect a breach of the fuel cladding, primary coolant boundary and containment. The condenser air removal exhaust combines with turbine gland seal exhaust in a common header before being discharged through the Nuclear Auxiliary Building Ventilation System to the environment. This monitor is placed downstream of where the two exhausts combine. Neither of these systems normally have detectable levels of radioactivity.

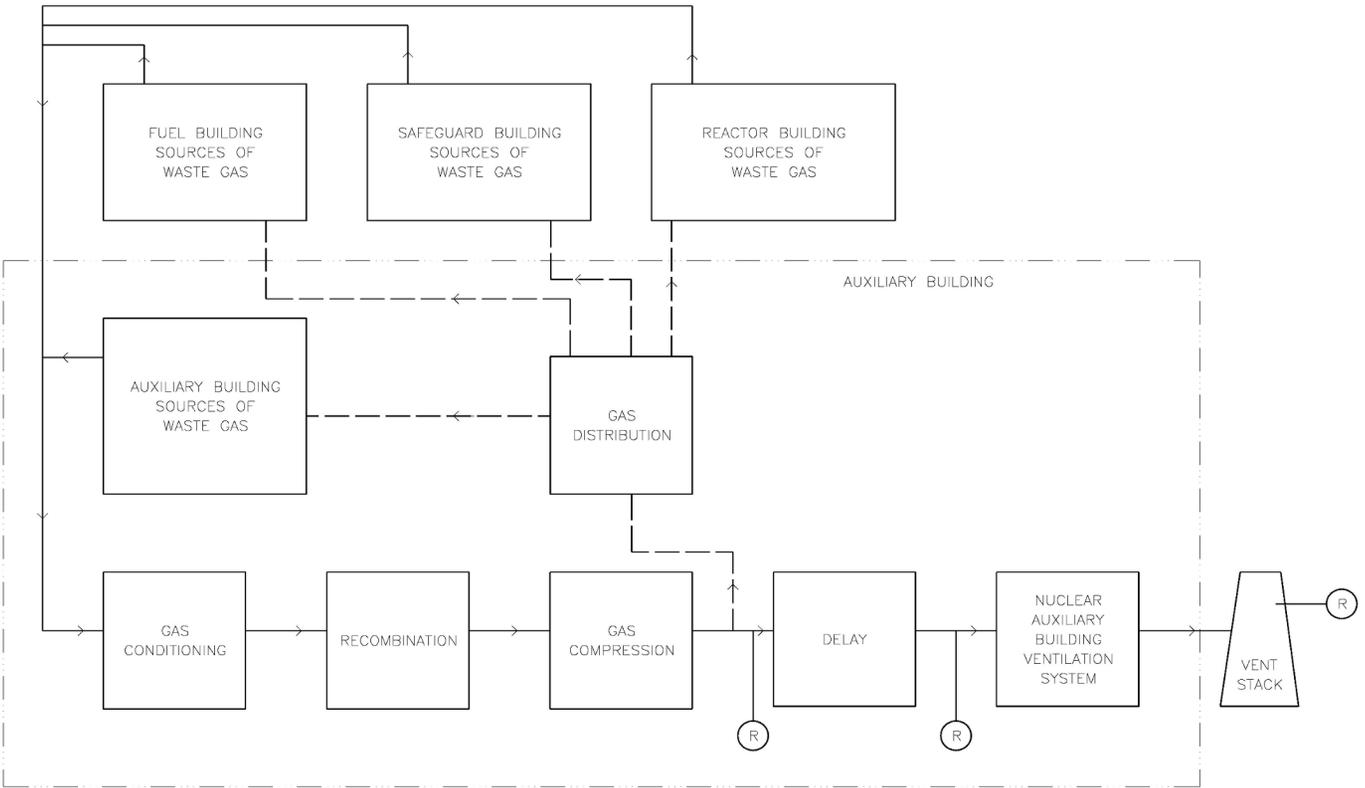
The setpoint is set at 2-3 times background levels. Additional lower setpoints may be used in order to provide additional information to the plant operators of abnormal conditions warranting investigation.

Instrument Number/Location: MAQ90CR001 in UMA01 011.

Range: 3E-06 to 1E-02 $\mu\text{Ci/cc}$

Figure 8.2-1

Gaseous Radwaste Effluent Flow Paths



KPL0712

9.0 BASES FOR DOSE CALCULATION METHODOLOGIES

This section serves: (1) to document the development and conservative nature of Method I equations to provide background information to Method I users, and (2) to identify the general equations, parameters and approaches to Method II-type dose assessments.

9.1 LIQUID RELEASE DOSE CALCULATIONS

Method I may be used to show that the Controls which limit offsite total body dose from liquids (CONTROLS 3.2.1.2 and 3.2.1.3) have been met for releases over the appropriate periods. The quarterly and annual dose limits in CONTROL 3.2.1.2 are based on the ALARA design objectives in 10 CFR 50, Appendix I Subsection II A. The minimum dose values noted in CONTROL 3.2.1.3 are "appropriate fractions," as determined by the NRC, of the design objective to ensure that radwaste equipment is used as required to keep offsite doses ALARA.

Method I was developed such that "the actual exposure of an individual ... is unlikely to be substantially underestimated" (10 CFR 50, Appendix I). The definition, below, of a single "critical receptor" (a hypothetical or real individual whose behavior results in a maximum potential dose) provides part of the conservative margin to the calculation of total body dose in Method I. Method II allows that actual individuals, associated with identifiable exposure pathways, be taken into account for any given release. In fact, Method I was based on a Method II analysis for a critical receptor assuming all principal pathways present instead of any real individual. That analysis was called the "base case"; it was then reduced to form Method I. The general equations used in the base case analysis are also used as the starting point in Method II evaluations. The base case, the method of reduction, and the assumptions and data used are presented below.

The steps performed in the Method I derivation follow. First, the dose impact to the critical receptor [in the form of dose factors DFL_{itb} (mrem/Ci)] for a unit activity release of each radioisotope in liquid effluents was derived. The base case analysis uses the general equations, methods, data and assumptions in Regulatory Guide 1.109 (Equations A-3 and A-7, Reference [2]). The liquid pathways contributing to an individual's dose are consumption of fish and invertebrates, shoreline activities, and swimming and boating near the discharge point. A nominal operating plant discharge flow rate of {39.3 ft³/sec} was used with a mixing ratio of {0.075 for exposures associated with fish and invertebrate ingestion and boating pathways and a mixing ratio of 0.014 for swimming and shoreline exposure pathways}. {These mixing ratios are based on a submerged, multi-port diffuser (with three nozzles), a discharge line situated approximately 550 feet off the near shoreline with the nozzles directed out into the Chesapeake Bay and into the overhead water column.}

The requirements for the determination of radiological impacts resulting from releases in liquid effluents is derived from 10 CFR 50, Appendix I. Section III.A.2 of Appendix I indicates that in making the assessment of doses to hypothetical receptors, "The Applicant may take account of any real phenomenon or factors actually affecting the estimate of radiation exposure, including the characteristics of the plant, modes of discharge of radioactive materials, physical processes tending to attenuate the quantity of radioactive material to which an individual would be exposed, and the effects of averaging exposures over time during which determining factors may fluctuate." In accessing the liquid exposure pathways that characterize {CCNPP Unit 3}, the design

and physical location of the {Cooling Tower Blowdown Water System} needs to be considered within the scope of Appendix I.

{CCNPP Unit 3 utilizes an offshore submerged multiport diffuser discharger for rapid dissipation and mixing of thermal effluents in the Chesapeake Bay. The multi-port diffuser (with three nozzles) is located approximately 1200 feet south of the CCNPP Unit 3 intake structure, extending about 550 feet into the Chesapeake Bay a depth of -10 ft msl. The diffuser will consist of three nozzles located approximately 3 feet off the bottom.

The multiport diffuser is designed to achieve a 13.3 to 1 dilution in the near field jet plume, and a 69 to 1 dilution for the nearest shore with the minimum tidal average mixing. }

The dose assessment models utilized in the ODCM are taken from NRC Regulatory Guide 1.109 (Reference [2]). The liquid pathway equations include a parameter (M_p) to account for the mixing ratio (reciprocal of the dilution factor) of effluents in the environment at the point of exposure. Table 1, in Regulatory Guide 1.109 (Reference [2]), defines the point of exposure to be the location that is anticipated to be occupied during plant lifetime, or have potential land and water usage and food pathways as could actually exist during the term of plant operation. {Although the liquid effluents are discharged directly to the brackish waters of the Chesapeake Bay and, therefore, drinking water is not expected to be a significant contributor to dose, the potential for desalination of Chesapeake Bay water for potable water use onsite and by ships using the bay have been included in the dose assessment.}

{The transit time used for the aquatic food pathway was 24 hours. For ingestion of potable water and shoreline, swimming and boating activity, the transit time used was 0.0 hours.} Table 9-1 outlines the human consumption and use factors used in the analysis. The resulting, site-specific, total body dose factors appear in Table 6-1.

9.1.1 Dose to the Total Body

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

Equation 9-1

$$D_{tb} = k \sum Q_i DFL_{itb}$$

where:

D_{tb} = The total body dose (mrem).

DFL_{itb} = Site-specific total body dose factor (mrem/Ci) for a liquid release. It is the highest of the four age groups. See Table 6-1.

Q_i = Total activity (Ci) released for radionuclide "i".

k = $\{39.3\}/F_d$ (dimensionless); where F_d is the average dilution flow at the point of discharge from the {multiport diffuser} (in ft^3/sec).

Method I is more conservative than Method II in the region of the dose limits because the dose factors DFL_{itb} used in Method I were chosen for the base case to be the highest of the four age groups (adult, teen, child and infant) for that radionuclide. In effect, each radionuclide is conservatively represented by its own critical age group.

9.1.2 Dose to the Critical Organ

The methods to calculate maximum organ dose parallel the total body dose methods (see Section 9.1.1).

For each radionuclide, a dose factor (mrem/Ci) was determined for each of seven organs and four age groups. The largest of these was chosen to be the maximum organ dose factor (DFL_{imo}) for that radionuclide. DFL_{imo} also includes the external dose contribution to the critical organ.

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

Equation 9-2

$$D_{mo} = k \sum Q_i DFL_{imo}$$

where:

- D_{mo} = the maximum organ dose from liquid release (mrem),
- DFL_{imo} = Site-specific maximum organ dose factor (mrem/μCi) for a liquid release. See Table 6-1,
- Q_i = Total activity (μCi) released for radionuclide "i", and
- k = {39.3}/F_d (dimensionless); where F_d is the average dilution flow at the point of discharge from the {multiport diffuser} (in ft³/sec).

Table 9-1:

{Usage Factors for Various Liquid Pathways at CCNPP Unit 3}

AGE	FISH	INVERT.	POTABLE WATER	SHORELINE	SWIMMING	BOATING
	(KG/YR)	(KG/YR)	(LITER/YR)	(HR/YR)	(HR/YR)	(HR/YR)
Adult	21	5.0	730	200	100	200
Teen	16	3.8	510	200	100	200
Child	6.9	1.7	510	200	100	200
Infant	0.0	0.0	330	200	100	200

9.2 GASEOUS RELEASE DOSE CALCULATIONS

9.2.1 Total Body Dose Rate from Noble Gases

Method I may be used to show that the Controls which limit total body dose rate from noble gases released to the atmosphere (CONTROL 3.2.2.1) has been met for the peak noble gas release rate.

Method I was derived from general Equation B-8 in Regulatory Guide 1.109 (Reference [2]) as follows:

Equation 9-3

$$\dot{D}_{tb} = 1E+06 [\chi/Q] \sum_i \dot{Q}_i * DFB_i$$

where:

\dot{D}_{tb} = the total body dose rate from noble gases (mrem/yr)

1E+06 = the number of pCi per μ Ci.

$[\chi/Q]$ = the maximum offsite receptor location long-term average undepleted atmospheric dispersion factor for a semi-infinite cloud.

\dot{Q}_i = Release rate to the environment of noble gas "i" (μ Ci/sec).

DFB_i = Total body dose factor, $\left(\frac{\text{mrem-m}^3}{\text{pCi-yr}} \right)$. See Table 6-2. (Regulatory Guide 1.109 (Reference [2]), Table B-1).

Substituting the long-term average atmospheric dispersion factor for a stack gaseous effluent release point and off site receptor from Section 6.10, Equation 9-3 takes the form:

Equation 9-4

$$\dot{D}_{tb} = (1E+06) * \{1.05 E-06\} * \sum_i (\dot{Q}_i * DFB_i)$$

which reduces to:

Equation 9-5

$$\dot{D}_{tb}^1 = \{1.05\} * \sum_i (\dot{Q}_i * DFB_i)$$

The selection of critical receptor, outlined in Section 6.10 is inherent in the derived Method I, since the maximum expected off site long-term average

atmospheric dispersion factor is used. All noble gases in Table 6-2 should be considered.

A Method II analysis could include the use of actual concurrent meteorology to assess the dose rates as the result of a specific release.

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

9.2.2 Skin Dose Rate from Noble Gases

Method I may be used to show that the Controls which limit skin dose rate from noble gases released to the atmosphere (CONTROL 3.2.2.1) have been met for the peak noble gas release rate.

The annual skin dose limit is 3,000 mrem (from NBS Handbook 69 (Reference [7], pages 5 and 6, is 30 rem/10). The factor of 10 reduction is to account for nonoccupational dose limits.

Using Method I, the calculated skin dose is the skin dose commitment to the critical, or most limiting, offsite receptor assuming long-term site average meteorology and a constant release rate reading over the entire year.

Method I was derived from the general Equation B-9 in Regulatory Guide 1.109 (Reference [2]) as follows:

Equation 9-6

$$\dot{D}^S = 1.11 * S_F * D_{air}^\gamma + 3.17E + 04 \sum_i \dot{Q}_i * [\chi / Q] * DFS_i$$

where:

\dot{D}^S = The skin dose rate from noble gases (mrem/yr),

1.11 = The average ratio of tissue to air absorption coefficient to convert mrad in air to mrem in tissue,

S_F = The shielding factor = 1.0 for dose rate determination,

DFS_i = The beta skin dose factor for a semi-infinite cloud of radionuclide "i" which includes the attenuation by the outer "dead" layer of the skin [(mrem-m³)/(pCi-yr)],

D_{air}^γ = $3.17E + 04 * [\chi / Q] \sum_i Q_i * DF_i^\gamma$,

DF_i^γ = The gamma air dose factor for a uniform semi-infinite cloud of radionuclide "i" in (mrad-m³)/(pCi-yr) from Table B-1 of Regulatory Guide 1.109 (Reference [2]) (see Table 6-2),

3.17E+04 = Number of pCi per Ci divided by the number of seconds per year.

\dot{Q}_i = The radionuclide release rate in Ci/sec, and

$[\chi/Q]$ = the maximum offsite receptor location long-term average undepleted atmospheric dispersion factor = {1.05E-06 sec/m³} from Table 6-5.

Substituting the long-term average undepleted atmospheric dispersion factor into Equation 9-6 yields the following:

Equation 9-7

$$\begin{aligned} \dot{D}^S &= \{0.037\} \sum_i \dot{Q}_i DF_i^\gamma + \{0.033\} \sum_i \dot{Q}_i * DFS_i \\ &= \sum_i \dot{Q}_i [\{0.037\} DF_i^\gamma + \{0.033\} DFS_i] \end{aligned}$$

Defining the combined skin dose rate factor, DF_i' , as follows:

Equation 9-8

$$DF_i' = \{0.037\} DF_i^\gamma + \{0.033\} DFS_i$$

simplifies Equation 9-7 into the following:

Equation 9-9¹

$$\dot{D}^S = \sum_i \dot{Q}_i * DF_i'$$

The calculated combined skin dose rate factor, DF_i' , is shown for various nuclides in Table 6-2.

The selection of critical receptor, outlined in Section 6.10, is inherent in the derived Method I, as it is based on the determined maximum expected offsite atmospheric dispersion factors. All noble gases in Table 6-2 must be considered.

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode

release maximum offsite receptor location long-term average atmospheric dispersion factors.}

9.2.3 Critical Organ Dose Rate from Iodines, Tritium and Particulates with Half-Lives Greater than Eight Days

Method I may be used to show that the Controls which limit organ dose rate from iodines, tritium and radionuclides in particulate form with half lives greater than 8 days released to the atmosphere (CONTROL 3.2.2.1.b) have been met for the peak release rates. The annual organ dose limit is 1500 mrem/yr (from NBS Handbook 69, (Reference [7], pages 5 and 6). It is evaluated by looking at the critical organ dose commitment to the most limiting offsite receptor assuming long-term site average meteorology.

The equation for the critical organ dose rate from iodines, tritium and particulates with half-lives greater than 8 days (\dot{D}_{co}) is derived from a form of Equation 6-12 in Section 6.9.1 by applying the conversion factor, 3.154E+07 (sec/yr) and converting the total activity released, Q, to a release rate, \dot{Q} , in $\mu\text{Ci}/\text{sec}$:

Equation 9-10

$$\dot{D}_{co} = 3.15 \text{ E}+ 07 * \sum_i (\dot{Q}_i * DFG_{ico})$$

Equation 9-10 is rewritten in the form:

Equation 9-11¹

$$\dot{D}_{co} = \sum_i (\dot{Q}_i * DFG'_{ico})$$

where:

Equation 9-12

$$DFG'_{ico} = DFG_{ico} * 3.154 \text{ E}+ 07 / 1\text{E} + 06$$

DFG'_{ico} incorporates the conversion constant of 31.54 and has assumed that the shielding factor (S_F) applied to the direct exposure pathway from radionuclides deposited on the ground plane is equal to 1.0 in place of the shielding factor value of 0.7 assumed in the determination of DFG_{ico} for the integrated dose over time.

The selection of critical receptor (based on the inhalation pathway) as outlined in Section 6.10 is inherent in Method I, as are the expected atmospheric dispersion factors which are based on long-term site-specific meteorology.

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

In accordance with the Bases Statement 3/4.11.2.1 in NUREG-1301 (Reference [8]), and the base's section for the organ dose rate limit given for CONTROL 3.2.2.1, a Method II dose rate calculation, for compliance purposes, can be based on restricting the inhalation pathway to a child's thyroid to less than or equal to 1500 mrem/year. Concurrent meteorology with time of release may also be used to assess compliance for a Method II calculation.

9.2.4 Gamma Dose to Air from Noble Gases

Method I may be used to show that the CONTROL 3.2.2.2 which limits offsite gamma air dose from gaseous effluents has been met for releases over appropriate periods. This Control is based on the objective in 10 CFR 50, Appendix I, Subsection II.B.1, which limits the estimated gamma air dose in offsite UNRESTRICTED AREAS.

For any noble gas release, in any period, the dose is calculated using the following equation which is derived using Equations B-4 and B-5 of Regulatory Guide 1.109 (Reference [2]):

Equation 9-13

$$D_{air}^{\gamma} = 3.17E+04 [\chi/Q] \sum_i Q_i * DF_i^{\gamma}$$

where:

- D_{air}^{γ} = the gamma dose to air from noble gases (mrad),
- 3.17E+04 = the number of pCi per Ci divided by the number of seconds per year,
- $[\chi/Q]$ = the maximum offsite receptor location long-term average undepleted atmospheric dispersion factor = {1.05E-06 } sec/m³ from Table 6-5,
- Q_i = Number of curies of noble gas "i" released and
- DF_i^{γ} = Gamma air dose factor for a uniform semi-infinite cloud of radionuclide "i" in (mrad-m³)/(pCi-yr) from Table 6-2.

Incorporating the long-term average undepleted atmospheric dispersion factor into Equation 9-13 yields the following:

Equation 9-14¹

$$D_{air}^{\gamma} = \{3.33E-02\} \sum_i Q_i * DF_i^{\gamma}$$

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

The major difference between Method I and Method II is that Method II would use actual or concurrent meteorology with a specific noble gas release spectrum to determine $[\chi/Q]$ rather than use the site's long-term average meteorological dispersion values.

9.2.5 Beta Dose to Air from Noble Gases

Method I may be used to show that CONTROL 3.2.2.2, which limits offsite beta air dose from gaseous effluents, has been met for releases over appropriate periods. This CONTROL is based on the objective in 10 CFR 50, Appendix I, Subsection II.B.1, which limits the estimated beta air dose in offsite UNRESTRICTED AREA locations.

For any noble gas release, in any period, the beta air dose is calculated using Equations B-4 and B-5 of Regulatory Guide 1.109 (Reference [2]) as follows:

Equation 9-15

$$D_{air}^{\beta} = 3.17E+04 * \chi/Q \sum_i (Q_i * DF_i^{\beta})$$

where:

D_{air}^{β} = the beta dose to air from noble gases (mrad),

3.17E+04 = the number of pCi per Ci divided by the number of seconds per year,

$[\chi/Q]$ = the maximum offsite receptor location long-term average undepleted atmospheric dispersion factor = {1.05E-06 sec/m³} from Table 6-5,

Q_i = Number of curies of noble gas "i" released,

DF_i^{β} = Beta air dose factor for a uniform semi-infinite cloud of radionuclide "i" in (mrad-m³)/(pCi-yr) from Table 6-2.

Substituting the long-term average undepleted atmospheric dispersion factor into Equation 9-15 yields the following:

Equation 9-16¹

$$D_{air}^{\beta} = \{3.33E - 02\} \sum_i Q_i * DF_i^{\beta}$$

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

9.2.6 Dose to Critical Organ from Iodines, Tritium and Particulates with Half-Lives Greater than Eight Days

Method I may be used to show that the Controls which limit offsite organ dose from gases (CONTROL 3.2.2.3 and 3.2.3) have been met for releases over the appropriate periods. CONTROL 3.2.2.3 is based on the ALARA objectives in 10 CFR 50, Appendix I, Subsection II C. CONTROL 3.2.3 is based on Environmental Standards for Uranium Fuel Cycle in 40CFR190, which applies to direct radiation as well as liquid and gaseous effluents. These methods apply only to iodine, tritium, and particulates in gaseous effluent contribution.

Method I was developed such that "the actual exposure of an individual ... is unlikely to be substantially underestimated" (10 CFR 50, Appendix I). The use below of a single "critical receptor" provides part of the conservative margin to the calculation of critical organ dose in Method I. Method II allows that actual individuals, associated with identifiable exposure pathways, be taken into account for any given release. In fact, Method I was based on a Method II analysis of a critical receptor assuming all pathways present. That analysis was called the "base case"; it was then reduced to form Method I. The base case, the method of reduction, and the assumptions and data used are presented below.

The steps performed in the Method I derivation follow. First, the dose impact to the critical receptor [in the form of dose factors DFG_{ico} (mrem/ μ Ci)] for a unit activity release of each iodine, tritium, and particulate radionuclide with half lives greater than eight days to gaseous effluents was derived. Six exposure pathways (ground plane, inhalation, stored vegetables, leafy vegetables, milk, and meat ingestion) were assumed to exist at the SITE BOUNDARY (not over water or marsh areas) which exhibited the highest long-term χ/Q . Doses were then calculated to six organs (bone, liver, kidney, lung, GI-LLI, and thyroid), as well as for the whole body and skin for four age groups (adult, teenager, child, and infant) due to the combined exposure pathways. For each radionuclide, the highest dose per unit activity release for any organ (or whole body) and age group was then selected to become the Method I site-specific dose factors. The base case, or Method I analysis, uses the general equations methods, data, and assumptions in Regulatory Guide 1.109 (Equation C-2 for doses resulting from direct exposure to contaminated ground plane; Equation C-4 for doses associated with inhalation of all radionuclides to different organs of individuals of different age groups; and Equation C-13 for doses to organs of individuals in different age groups resulting from ingestion of radionuclides in produce, milk, meat, and leafy vegetables in Reference [2]). Table 9-2 and Table 9-3 outline human consumption and environmental parameters used in the analysis. It is conservatively assumed that the critical receptor lives at the "maximum offsite atmospheric dispersion factor location" as defined in Section 6.10.

The resulting site-specific dose factors are for the maximum organ which combine the limiting age group with the highest dose factor for any organ with each nuclide. These critical organ, critical age dose factors are given in Table 6-3.

For iodine (I-131, I-133), tritium, and particulate ($T_{1/2} > 8$ days) in gas release, during any period, the dose from radionuclide "i" is:

Equation 9-17¹

$$D_{ico} = Q_i * DFG_{ico}$$

where:

DFG_{ico} = the critical dose factor for radionuclide "i" from Table 6-3, and

Q_i = the activity of radionuclide "i" released in microcuries.

¹ {If release is ground level (i.e., from startup condenser evacuation), multiply by an additional factor of 16, which is a bounding ratio of the ground level to mixed mode release maximum offsite receptor location long-term average atmospheric dispersion factors.}

Table 9-2

{Environmental Parameters for Gaseous Effluents at Calvert Cliffs Nuclear Power Plant Unit 3}

Variable	Vegetables		Cow Milk		Goat Milk		Meat	
	Stored	Leafy	Pasture	Stored	Pasture	Stored	Pasture	Stored
Agricultural productivity (kg/m ²)	2.0	2.0	0.7	2.0	0.7	2.0	0.7	2.0
Soil surface density (kg/m ²)	240	240	240	240	240	240	240	240
Transport time to user (hrs)			48	48	48	48	480	480
Soil exposure time (hrs)	131400	131400	131400	131400	131400	131400	131400	131400
Crop exposure time to plume (hrs)	1440	1440	720	1440	720	1440	720	1440
Holdup after harvest (hrs)	1440	24	0	2160	0	2160	0	2160
Animals daily feed (kg/day)			50	50	6	6	50	50
Fraction of year on pasture			0.58		0.58		0.58	
Fraction of pasture when on pasture			1.00		1.00		1.00	
Fraction of stored vegetables grown in garden	0.76							
Fraction of leafy vegetables grown in garden		0.58						
Fraction of elemental iodine = 0.50								
Absolute humidity = 8.4 g/m ³								
Fractional equilibrium ratio for C-14 = 1.0								
Shield Factor = 0.7								
Occupancy Correction Factor = 1.0								

Table 9-3

{Usage Factors for Gaseous Effluent Pathway
at Calvert Cliffs Nuclear Power Plant Unit 3}
[values from Table E-5 of Regulatory Guide 1.109 (Reference [2])]

	Vegetables (kg/yr)	Leafy Vegetables (kg/yr)	Milk (l/yr)	Meat (kg/yr)	Inhalation (m³/yr)
Adult	520	64	310	110	8000
Teen	630	42	400	65	8000
Child	520	26	330	41	3700
Infant	0	0	330	0	1400

10.0 REFERENCES

1. US NRC NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978.
2. US NRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," U.S. Nuclear Regulatory Commission, Revision 1, October 1977.
3. Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications (RETS) 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."
4. NEI 07-09, "Draft Generic FSAR Template Guidance for OffSite Dose Calculation Manual (ODCM) Program Description," August, 2007.
5. "Radiological Assessment Branch Technical Position," Revision 1, U.S. NRC, November 1979.
6. NEI 07-07, "Industry Ground Water Protection Initiative – Final Guidance Document," August, 2007.
7. National Bureau of Standards, "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure," Handbook 69, June 5, 1959.
8. US NRC NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors," April 1991.