ENCLOSURE (2)

OFFSITE DOSE CALCULATION MANUAL FOR

CALVERT CLIFFS NUCLEAR POWER PLANT,

Revision 00802

Calvert Cliffs Nuclear Power Plant, LLC
July 13, 2012

OFFSITE DOSE CALCULATION MANUAL

Revision 00802

Calvert Cliffs Nuclear Power Plant

PREPARED BY:	Catherine L. Clarke Technical Writer	DATE:	12-20-11
REVIEWED BY:	John D'Souza RETS Program Manager	DATE:	12/20/11
REVIEWED BY:	Jeffrey A. York General Supervisor – Chemistry	DATE:	12/20/11
REVIEWED BY:	**N/A Editorial Correction** POSRC Meeting Number	DATE:	
APPROVED BY:	Christopher Costanzo Plant General Manager	DATE:	12/21/11

EXECUTIVE SUMMARY OF CHANGES

REV	CHG	PAGE	DESCRIPTION
8	0	117	The TE-001 test is now STP-M-462-2, the footnote was changed to reflect this change.
		121	Simplified equation 29G by rewriting it into an equivalent form.
		122	Deleted parameters no longer in equation 29G and changed the units on background to microcuries per second.
		124	Simplified equation 29G by rewriting it into an equivalent form.
		125	Deleted parameters no longer in equation 29G and changed the units on background to microcuries per second.
		134	The TE-001 test is now STP-M-462-2, the footnote was changed to reflect this change.
		136	Parentheses were changed in Equation 27G.
		139	Parentheses were changed in Equation 27G.
		232	Changed the flowrate for Unit 2 Main Plant Vent Stack to accurately reflect current plant conditions as measured by STP-M-462-2.
		314	Changed the description of several sample stations to accurately describe proper sector designations. These changes are made based on satellite imagery.
		316	Added 2 new TLD Sample locations to map. The new sample locations ensure that each meteorological sector has a TLD.
008	01		
		7 59	**PCR-09-06778** – Step E, added "including the Independent Spent Fuel Storage Installation (ISFSI) to end of sentence. 3.b) – removed FSAR and added UFSAR. Reason: Editorial correction proposed by NPT.
		9	**PCR-10-07370** – 2nd paragraph, Dose Equivalent Iodine-131, Deleted sentence, "The thyroid dose conversion factors used for…… Replaced sentence with: The TEDE (Total Effective Dose Equivalent) inhalation dose conversion factors used for this calculation shall be those listed in the Table 2.1 in the column headed "effective" of Federal Guidance Report 11, ORNL, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." Reason: Tech Spec Amendment 281/258.
		35	**PCR-09-06032** - Table 4.11-2, Replaced the first character of the units associated with the last column (Lower Limit of Detection (LLD) on Table 4.11-2 from j to m and the period (.) to a slash (/). Reason: Editorial change to correct typo associated with units.

EXECUTIVE SUMMARY OF CHANGES (CONTINUED)

REV	CHG	PAGE	DESCRIPTION
008	02	12	**PCR-11-04924** – Change ODCM Rated Thermal Power definition to reflect Appendix K Uprate from 2700 to 2737 mWt. REASON: Editorial change. Comply with Tech Spec Amendments 291/267.
		308	**PCR-11-05428** – Attachment 13, DR23, changed Distance km from 12.6 to 12.4, changed Distance mi from 7.9 to 7.7, changed Description of property from Carpenter's to Anderson's where TLD was located. REASON: Editorial change. Align ODCM to reflect location of TLD.
		220	**PCR-11- 06095** - 4.c. – reworded last sentence to clarify expectation for how changes to the ODCM should be handled and submitted to the NRC. REASON: Editorial change. To clarify ODCM change expectations prior to submitting next Radioactive Effluent Release Report to the NRC.

LIST OF EFFECTIVE PAGES

PAGES	REVISION/CHANGE
1-318	08

PROCEDURAL ALTERATIONS

PAGES	REVISION/CHANGE
12, 220, 308	008/02

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

ATTACHMENTS:

PART 1.0: PURPOSE AND APPLICABILITY/SCOPE

PURPOSE

A. The ODCM lists the radiological effluent controls established by Technical Specifications 5.5.1 and 5.5.4 at Calvert Cliffs Nuclear Power Plant (CCNPP) for the purpose of ensuring the amount of radioactive materials released to the environment are as low as reasonably achievable.

B. The ODCM defines parameters and methodologies for calculating projected doses and dose rates resulting from the release of radioactive materials in liquid and gaseous effluents.

C. The ODCM defines parameters and methodologies for calculating alarm and fixed setpoints for Technical Specification related effluent radiation monitoring systems.

D. The ODCM defines and establishes controls for the Radiological Environmental Monitoring Program.

APPLICABILITY/SCOPE

A. The information presented in this manual is applicable to any division, department, section, or unit within the Calvert Cliffs Nuclear Power Plant (CCNPPI) which is either wholly or partly responsible for performing any of the activities listed in the PURPOSES section of this procedure. Responsible organizations include, but are not limited to, the following:

1. CCNPPI, CHEMISTRY SECTION
2. Constellation Generation Group, Technical Services, Analytical Chemistry Unit
3. CCNPPI, Electrical and Controls (E&C) Section
4. CCNPPI, Operations Section
5. CCNPPI, Radiation Safety Section

B. This manual is applicable to the determination of alarm and fixed setpoints for the following radioactive gaseous effluent monitoring instrumentation:

1. 0-RE-2191
2. 1/2-RE-5415
3. 1/2-RE-5416

C. This manual is applicable to the determination of alarm and fixed setpoints for the following radioactive liquid effluent monitoring instrumentation:

1. 0-RE-2201
2. 1/2-RE-4014
3. 1/2-RE-4095

D. This manual is applicable to the determination of the offsite doses and/or offsite dose rates due to the following:

1. Radioactive material in gaseous waste discharged from CCNPP
2. Radioactive material in liquid waste discharged from CCNPP
3. Radioactive material contained in outside storage tanks at CCNPP

E. This manual is applicable to the determination of the radiological effects on the environment due to the presence of Calvert Cliffs Nuclear Power Plant (including the Independent Spent Fuel Storage Installation (ISFSI).

Jan 2011

F. The ODCM is based on Technical Specifications and CCNPPIs interpretation of industry standards and practices.

PART 2.0 : DEFINITIONS AND REFERENCES

DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Controls.

ABNORMAL AND/OR UNANTICIPATED RADIOACTIVE RELEASE:

Any unplanned or uncontrolled release of radioactive material from the **SITE BOUNDARY**.

ACTION:

That part of a control which prescribes remedial measures required under designated conditions.

BATCH RELEASE:

A **BATCH RELEASE** is the discharge of liquid (or gaseous) waste of a discrete volume. (NUREG-0133, page 14)

CHANNEL CALIBRATION:

A **CHANNEL CALIBRATION** shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The **CHANNEL CALIBRATION** shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the **CHANNEL FUNCTIONAL TEST**. The **CHANNEL CALIBRATION** may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK:

A **CHANNEL CHECK** shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST:

a. An analog **CHANNEL FUNCTIONAL TEST** shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify **OPERABILITY** including alarm and/or trip functions.
b. A bistable **CHANNEL FUNCTIONAL TEST** shall be injection of a simulated signal into the channel sensor to verify **OPERABILITY** including alarm and/or trip functions.

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 of fluid, either liquid or gas, flowing through the system.

CONTINUOUS RELEASE

A **CONTINUOUS RELEASE** is the discharge of liquid (or gaseous) wastes of a nondiscreet volume; e.g., from a volume or system that has an input flow during the **CONTINUOUS RELEASE**. (NUREG-0133, P. 14.)

CONTINUOUS SAMPLING FREQUENCY

A **CONTINUOUS SAMPLING FREQUENCY** is a sampling arrangement that provides for uninterrupted sampling flow under normal operating conditions. During periods of sampling interruption, the requirement for continuous sampling is considered to be met, provided the interruption is less than one hour AND plant conditions are such that no adverse trend in effluents would be present during the period of interruption. Interruptions in continuous sampling during changing plant conditions OR interruptions in continuous sampling for greater than one hour must be evaluated to determine if an ODCM violation has occurred.

DOSE EQUIVALENT IODINE-131:

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The TEDE (Total Effective Dose Equivalent) inhalation dose conversion factors used for this calculation shall be those listed in the Table 2.1 in the column headed "effective" of Federal Guidance Report 11, ORNL, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

Jan 2011

FREQUENCY NOTATION:

The frequency notation specified for the performance of Surveillance Requirements shall correspond to the intervals defined in the following Table.

Frequency Notation Table

NOTATION	FREQUENCY
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days
SA	At least once per 6 months
R	At least once per 18 months
S/U	Prior to each reactor startup
P	Completed prior to each release
N.A.	Not applicable
Refueling Interval	At least once per 24 months

GAMMA ISOTOPIC ANALYSIS:

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

GASEOUS RADWASTE PROCESSING SYSTEM:

A **GASEOUS RADWASTE 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.

GSC:

GSC stands for General Supervisor Chemistry.

LIQUID RADWASTE PROCESSING SYSTEM:

A **LIQUID RADWASTE PROCESSING SYSTEM** is the system installed and designed to reduce radioactive liquid effluents. The minimum components necessary for reducing liquid radioactive effluents is either 13 or 14 Reactor Coolant Waste Ion Exchange; however, other equipment in the system may be used in the maintenance of ALARA for liquid radioactive effluents.

LOWER LIMIT OF DETECTION:

The LLD is the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

MEMBERS 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.

OFFSITE DOSE CALCULATION MANUAL:

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 Environmental Radiological Monitoring Program. The ODCM shall also contain the radioactive effluent controls, radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Technical Specification 5.6.2 and 5.6.3.

OPERABILITY:

A system, subsystem, train, component or device shall be operable or have **OPERABILITY** when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other required auxiliary equipment that are required for the system, subcomponent or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE:

An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified below.

Operational Modes

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
POWER OPERATION	≥0.99	>5%	≥300 °F
STARTUP	≥0.99	≤5%	≥300 °F
HOT STANDBY	<0.99	0	≥300 °F
HOT SHUTDOWN	<0.99	0	300 °F > T_{avg} > 200 °F
COLD SHUTDOWN	<0.99	0	≤200 °F
REFUELING**	≤0.95	0	≤140 °F

* Excluding decay heat.
** Reactor vessel head unbolted or removed and fuel in the vessel.

PROCESS CONTROL PROGRAM:

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 wastes 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.

PROJECTED DOSE:

A reasonable estimate of dose expected as a result of future radioactive releases.

PURGE OR PURGING:

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER:

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2737 MWt.

Dec2011

SIMULTANEOUS RELEASES:

Simultaneous liquid releases are discharges of liquid radwaste which occur at the same time. Simultaneous gaseous releases are discharges of gaseous radwaste which occur at the same time excluding main vent discharges.

SITE BOUNDARY:

The **SITE BOUNDARY** shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.
(The **SITE BOUNDARY** is depicted on Attachment 18, "Environmental Monitoring Sites")

SOURCE CHECK:

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

THERMAL POWER:

The THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TYPICAL RADWASTE RELEASES (OR TYPICAL RADWASTE EFFLUENTS):

TYPICAL RADWASTE RELEASES are defined as (1) all releases conducted during any calendar quarter while either Unit 1 or Unit 2 is in mode 1, and also includes (2) all releases conducted during any calendar quarter following mode 1 operation of either Unit 1 or Unit 2.

UNRESTRICTED AREA:

An UNRESTRICTED AREA shall be any area at or beyond the **SITE BOUNDARY** access to which 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 of/for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST PROCESSING SYSTEM:

A **VENTILATION EXHAUST PROCESSING SYSTEM** is 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 High Efficiency Particulate Air (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 Feature (ESF) atmospheric cleanup systems are not considered to be **VENTILATION EXHAUST TREATMENT SYSTEM** components.

VENTING:

VENTING is the controlled process of discharging air or gas from a confinement 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.

WASTE GAS HOLDUP SYSTEM:

See **GASEOUS RADWASTE PROCESSING SYSTEM**.

REFERENCES

DEVELOPMENTAL REFERENCES

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

2. Regulatory Guide 1.109, "Calculation Of Annual Doses To Man From Routine Release Of Reactor Effluents For The Purpose Of Compliance With 10 CFR Part 50, Appendix I," Revision 1, (October 1977).

3. 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).

4. CCNPP System Description Number 14B, "Reactor Coolant Waste Processing System Description."

5. Updated Final Safety Analysis Report, Chapter 11, "Waste Processing And Radiation Protection."

6. CCNPP System Description Number 14D, "Miscellaneous Liquid Waste Processing System Description."

7. OI-17D, "Miscellaneous Waste Processing System"

8. OI-17C, "Reactor Coolant Waste Processing System"

9. Title 10 of the Code of Federal Regulations, Part 20, Jan 1990 and May 1991

10. Title 10 of the Code of Federal Regulations, Part 50

11. Calvert Cliffs Nuclear Power Plant Semi-Annual Radiological Effluent Release Report (1986, 1987, 1988).

12. Radioactive Decay Data Tables, David C. Kocher, 1981.

13. Radiological Health Handbook, Bureau of Radiological Health, Jan. 1970.

14. TE-001, "Main Vent Stack Flow Measurement"

15. ETP-87-16, "Main Vent Stack Flow Measurement"

16. Verification And Validation Of The Gaseous Release Permit Portion Of The CCNPP EMS Computer Code, J. S. Bland And Associates, July 10, 1990.

17. Land Use Survey In The Vicinity Of The Calvert Cliffs Nuclear Power Plant September 1991, (for the year 1990), Baltimore Gas And Electric Company, Environmental Programs Section, C. Key, B. E. Helbing

18. CP-607, Revision 1, "Offsite Dose Calculation Manual"

19. NO-1-201, Calvert Cliffs Operating Manual

20. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites"

21. "Plant Data Book", BGE CCNPP Units 1 and 2, Bechtel Power Corporation, Volume 1, Job 6750.

22. 50.59 Log No. 82-B-999-028-R00, Safety Analysis No. 2, FCR 82-1053, Supplement 1 (Component Cooling System) **[B527]**

23. 50.59 Log No. 90-0-074-011-R2, Activity MASE 90-7 (Plant Nitrogen System) **[B527]**

24. 50.59 Log No. 90-0-029-045-R1, Activity MASE 90-6 (Plant Heating System) **[B527]**

25. 50.59 Log No. 90-0-037-044-R1, Activity MASE 90-5 (Demineralized Water System) **[B527]**

26. 50.59 Log No. 90-B-012-043-R1, Activity MASE 90-4 (Service Water System) **[B527]**

27. 50.59 Log No. 90-0-027-037-R2, Activity MASE 90-8 (Auxiliary Boiler System) **[B527]**

28. Bechtel Power Corporation, Calculation Number M-90-20, "Allowable Radioactive Contamination Levels in the Plant Heating System" **[B527]**

29. Bechtel Power Corporation, Calculation Number M-90-18, "Allowable Radioactive Contamination Levels in the Nitrogen System Header" **[B527]**

30. Bechtel Power Corporation, Calculation Number M-90-04, "Allowable Radioactive Contamination Levels in Auxiliary Boiler Water" **[B527]**

31. Bechtel Power Corporation, Calculation Number M-90-21, "Allowable Radioactive Contamination Levels in the Demineralized Water System" **[B527]**

32. Bechtel Power Corporation, Calculation Number M-90-19, "Allowable Radioactive Contamination Levels in the Service Water System" **[B527]**

33. Bechtel Power Corporation, Job Number 11865, Calculation Number 7.4.3-15, "Allowable Radioactive Contamination Levels in the Condensate Storage Tank" **[B527]**

34. CNG-CM-1.01-1003, Engineering Services Process Overview

35. OI-8A, Blowdown System

36. NUREG-1301, "**OFFSITE DOSE CALCULATION MANUAL**: Standard Radiological Effluent Controls for Pressurized Water Reactors", W. W. Meinke, and T. H. Essig, (Published April 1991).

37. Radiological Environmental Monitoring Program Annual Report for the Calvert Cliffs Nuclear Power Plant Units 1 and 2 January 1 - December 31, 1991", Baltimore Gas And Electric Company, March 1992

38. 50.59 Log No. 90-B-037-120-R2, Activity MASE 90-11 (Condensate Storage Tank) **[B527]**

39. Regulatory Guide 4.13, Performance Testing and Procedural Specifications for Thermoluminescence Dosimetry; Environmental Applications

PERFORMANCE REFERENCES

1. STP-M-462-1 and STP-M-462-2, Main Vent Stack Flow Measurement

2. Technical Specifications

3. Technical Requirements Manual

4. Regulatory Guide 4.13, Performance Testing and Procedural Specifications for Thermoluminescence Dosimetry; Environmental Applications

5. TE-006, Containment Purge Exhaust System HEPA Filter Test

PART 3.0 / 4.0 : CONTROLS AND SURVEILLANCE REQUIREMENTS

CONTROLS

3.0.1 Compliance with the Controls contained in the succeeding controls is required during the **OPERATIONAL MODES** or other conditions specified therein; except that upon failure to meet the Control, the associated **ACTION** requirements shall be met.

3.0.2 Noncompliance with a Control shall exist when the requirements of the Control and associated **ACTION** requirements are not met within the specified time intervals. If the Control is restored prior to expiration of the specified time intervals, completion of the **ACTION** requirements is not required.

3.0.3 When a Control is not met, except as provided in the associated **ACTION** requirements, within one hour **ACTION** shall be initiated to place the unit in a MODE in which the control does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the **ACTION** requirements, the **ACTION** may be taken in accordance with the specified time limits as measured from the time of failure to meet the Control. Exceptions to these requirements are stated in the individual controls.

3.0.4 Entry into an **OPERATIONAL MODE** or other specified condition shall not be made unless the conditions of the Control are met without reliance on provisions contained in the **ACTION** requirements. This provision shall not prevent passage through **OPERATIONAL MODES** as required to comply with **ACTION** requirements. Exceptions to these requirements are stated in the individual controls.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Control, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are **OPERABLE**, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within 2 hours, action shall be initiated to place the unit in a **MODE** in which the applicable Control does not apply by placing it, as applicable in:

1. At least HOT STANDBY within the next 6 hours,

2 At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 24 hours.

This specification is not applicable in **MODES** 5 or 6.

SURVEILLANCE

4.0.1 Surveillance Requirements shall be applicable during the **OPERATIONAL MODES** or other conditions specified for individual Control unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Control 4.0.2, shall constitute noncompliance with the **OPERABILITY** requirements for a Control. The time limits of the **ACTION** requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. However, this time of applicability may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the **ACTION** requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an **OPERATIONAL MODE** or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Control have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to **OPERATIONAL MODES** as required to comply with **ACTION** requirements.

MONITORING INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT

CONTROLS

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-12 shall be **OPERABLE** with their alarm or alarm-with-automatic-termination setpoints set to ensure that the limits of Control 3.11.2.1 are not exceeded. The setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: **As shown in Table 3.3-12.**

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm setpoint less conservative than required by the above Control, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels **OPERABLE**, take the **ACTION** shown in Table 3.3-12. Exert best efforts to return the instruments to **OPERABLE** status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated **OPERABLE** by performance of the **CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION** and **CHANNEL FUNCTIONAL TEST** operations at the frequencies shown in Table 4.3-11.

TABLE 3.3-12
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	35
b. Effluent System Flow Rate Measuring Device	1	*	36
2. MAIN VENT SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	38
c. Particulate Sampler	1	*	38
d. Tritium Sampler	1	*	38

* At all times

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 35 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, the contents of the tank(s) may be released to the environment:

a. Using the main vent monitor as a backup and recording RMS readings every 15 minutes during the release, or

b. Provided that prior to initiating the release, at least two independent samples of the tank's contents are analyzed, and at least two technically qualified members of the Facility Staff independently verify the release rate calculations and two qualified operators verify the discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, effluent releases via this pathway may continue provided either (1) grab samples are taken and analyzed for gross activity at least once per 24 hours, or (2) an equivalent monitor is provided.

ACTION 38 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, effluent releases via the affected pathway may continue provided samples are continuously collected as required in Table 4.11-2 with auxiliary sampling equipment.

TABLE 4.3-11
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	SA(1)	*
b. Effluent System Flow Rate Measuring Device	D(4)	NA	R	NA	*
2. MAIN VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	SA(2)	*
b. Iodine Sampler	W	NA	NA	NA	*
c. Particulate Sampler	W	NA	NA	NA	*
d. Tritium Sampler	W	NA	NA	NA	*

* At all times other than when the line is valued out and locked

TABLE 4.3-11 (Continued)

TABLE NOTATION

(1) The **CHANNEL FUNCTIONAL TEST** shall also demonstrate the automatic isolation of this pathway and/or Control Room alarm annunciation occurs if the appropriate following condition(s) exists:

1. Instrument indicates measured levels above the alarm-with-automatic-termination setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The **CHANNEL FUNCTIONAL TEST** shall also demonstrate that Control Room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(3) The initial **CHANNEL 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 **CHANNEL CALIBRATION**, sources that have been related to the initial calibration can be used.

(4) The **CHANNEL CHECK** shall consist of verifying indication of flow during periods of release and shall be made at least once per 24 hours during periods in which effluent releases are made.

RADIOACTIVE LIQUID EFFLUENT

CONTROLS

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-13 shall be **OPERABLE** with their alarm or alarm-with-automatic-termination setpoints set to ensure that the limits of Control 3.11.1.1 are not exceeded. The setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the **OFFSITE DOSE CALCULATION MANUAL (ODCM)**.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Control, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels **OPERABLE**, take the **ACTION** shown in Table 3.3-13. Exert best efforts to return the instruments to **OPERABLE** status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated **OPERABLE** by performance of the **CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION** and **CHANNEL FUNCTIONAL TEST** operations at the frequencies shown in Table 4.3-12.

TABLE 3.3-13

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE			
a. Liquid Radwaste Effluent Line	1	(1)	28
b. Steam Generator Blowdown Effluent Line	1	(2)	29
2. FLOW RATE MEASUREMENT DEVICES			
a. Liquid Radwaste Effluent Line	1	(1)	30
b. Steam Generator Blowdown Effluent Line	1	(2)	30

Note: (1) At all times.
(2) At all times while process flow is present.

TABLE 3.3-13 (Continued)

ACTION STATEMENTS

ACTION 28 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Control 4.11.1.1.1, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and two qualified operators verify the discharge valve line up.

ACTION 29 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at the **LOWER LIMIT OF DETECTION** defined in Table 4.11-1:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcurie/gram **DOSE EQUIVALENT I-131**.

b. At least once per 48 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcurie/gram **DOSE EQUIVALENT I-131**.

ACTION 30 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

TABLE 4.3-12
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	P	R(2)	SA(1)
b. Steam Generator Blowdown Effluent Line	D	P	R(2)	SA(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	NA	R	NA
b. Steam Generator Blowdown Effluent Line	D(3)	NA	R	NA

TABLE 4.3-12 (Continued)

TABLE NOTATION

(1) The **CHANNEL FUNCTIONAL TEST** shall also demonstrate that automatic isolation of this pathway and/or Control Room alarm annunciation occur if the appropriate following condition(s) exists:

1. Instrument indicates measured levels above the alarm or fixed setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The initial **CHANNEL 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 **CHANNEL CALIBRATION**, sources that have been related to the initial calibration can be used.

(3) **CHANNEL CHECK** shall consist of verifying indication of flow during periods of release. **CHANNEL CHECK** shall be made at least once per 24 hours during periods in which effluent releases are made.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS:

Concentration

CONTROLS

3.11.1.1 The concentration of radioactive material released in liquid effluents to **UNRESTRICTED AREAS** shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to **UNRESTRICTED AREAS** exceeding the above limits, without delay restore the concentration to within the above limits.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the **ODCM** to assure that the concentrations at the point of release are maintained within the limits of Control 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	MINIMUM SAMPLING FREQUENCY	ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (mCi/ml)
A. Batch Waste Releases[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
			Mo-99, Ce-144	$2x10^{-6}$
	P Each Batch	M Composite[d]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	$5x10^{-8}$
B. Turbine Building Sump	M	M	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
			Mo-99, Ce-144	$2x10^{-6}$

TABLE 4.11-1 (Continued)

TABLE NOTATION

a The LLD is defined, for purposes of these 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:

$$LLD = \frac{4.66\, s_b}{E \bullet V \bullet 2.22 \times 10^6 \bullet Y \bullet \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" **LOWER LIMIT OF DETECTION** as defined above, as microcuries per unit mass or volume,

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

E is the counting efficiency, as counts per disintegration,

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

2.22×10^6 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 radionuclide, and

Δt is the elapsed time between sample collection, or end of the sample collection period, and time of counting.

Typical values of E, V, Y, and Δt 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.

TABLE 4.11-1 (Continued)

TABLE NOTATION

b Prior to sampling Reactor Coolant Waste and Miscellaneous Waste for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

c The principal gamma emitters for which the LLD control applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137 and Ce-141. 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 Annual Radioactive Effluent Release Report pursuant to Technical Specification 5.6.3.

d A **COMPOSITE SAMPLE** is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged in which the method of sampling employed results in a specimen that is representative of the liquids released.

LIQUID EFFLUENTS:

Dose

CONTROLS

3.11.1.2 The dose or dose commitment to a **MEMBER OF THE PUBLIC** from radioactive materials in liquid effluents released to **UNRESTRICTED AREAS** shall be limited:

a. During any calendar quarter to less than or equal to 3.0 mrems to the total body and to less than or equal to 10 mrems to any organ, and

b. During any calendar year to less than or equal to 6 mrems to the total body and to less than or equal to 20 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective **ACTIONS** that have been taken to reduce the releases and the proposed corrective **ACTIONS** to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

LIQUID EFFLUENTS:

Liquid Radwaste Processing System

CONTROLS

3.11.1.3 The **LIQUID RADWASTE PROCESSING SYSTEM** shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the calculated doses due to the liquid effluent to **UNRESTRICTED AREAS** exceeds 0.36 mrem to the total body or 1.20 mrem to any organ in a 92 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without processing and in excess of the above limits, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days a Special Report that includes the following information:

1. Explanation of why liquid radwaste was being discharged without processing, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

2. **ACTION**(s) taken to restore the inoperable equipment to **OPERABLE** status, and

3. Summary description of **ACTION**(s) taken to prevent a recurrence.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3 Doses over a 92-day period due to liquid releases to **UNRESTRICTED AREAS** shall be calculated at least once per 31 days in accordance with the methodology and parameters in the **ODCM**.

GASEOUS EFFLUENTS:

Dose Rate

CONTROLS

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) shall be limited to the following:

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

b. For iodine-131 and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the **ODCM**.

4.11.2.1.2 The dose rate due to iodine-131 and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the **ODCM** by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] µCi/ml
A. Waste Gas Storage[f] Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[b] (Gaseous Emissions Only)	1×10^{-4}
B. Containment Purge and Vent	P Each Batch Grab Sample	P Each Batch	Principal Gamma Emitters[b] (Gaseous Emissions Only)	1×10^{-4}
C. Main Vent	M[c] Grab Sample	M[C]	Principal Gamma Emitters[b] (Gaseous Emissions Only)	1×10^{-4}
	Continuous[d]	M	H-3	1×10^{-6}
	Continuous[d]	W Charcoal Sample[e]	I-131	1×10^{-12}
	Continuous[d]	W Particulate Sample[e]	Principal Gamma Emitters[b] (I-131, others)	1×10^{-11}
	Continuous[d]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[d]	Q Composite Particulate Sample	Sr-89, Sr90	1×10^{-11}
	Continuous[d]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}
D. Incinerated Oil[g]	P Each Batch[h]	P Each Batch[h]	Principal Gamma Emitters	5×10^{-7}

Jan 2011

TABLE 4.11-2 (Continued)

TABLE NOTATION

a The LLD is defined, for purposes of these 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:

$$LLD = \frac{4.66\, s_b}{E \bullet V \bullet 2.22 \times 10^6 \bullet Y \bullet \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" **LOWER LIMIT OF DETECTION** as defined above, as microcuries per unit mass or volume,

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

E is the counting efficiency, as counts per disintegration,

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

2.22×10^6 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 radionuclide, and

Δt for plant effluents is the elapsed time between sample collection or end of the sample collection period, and time of counting.

Typical values of E, V, Y, and Δt 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.

__TABLE 4.11-2__ (Continued)

TABLE NOTATION

b The principal gamma emitters for which the LLD control applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report pursuant to Technical Specification 5.6.3.

c Sampling and analysis shall also be performed following shutdown, **STARTUP**, or a **THERMAL POWER** change exceeding 15 percent of **RATED THERMAL POWER** within one hour unless (1) analysis shows that the **DOSE EQUIVALENT I-131** concentration in the primary coolant has not increased more than a factor of 5, and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 5.

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 3.11.2.1, 3.11.2.2 and 3.11.2.3.

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. When sample collection time is less than seven days, the corresponding LLDs may be increased by a proportional factor.

f Collect sample and analyze daily for total Curie content per Technical Requirements Manual 15.11.2.1 when the Reactor Coolant System specific activity of Xe-133 is greater than 150 uCi/ml.

g Incinerated oil may be discharged via points other than the main vent (e.g., Auxiliary Boiler). Releases shall be accounted for based on pre-release grab sample data.

h Samples for incinerated oil releases shall be collected from and representative of filtered oil in liquid form.

GASEOUS EFFLUENTS:

Dose - Noble Gases

CONTROLS

3.11.2.2 The air dose due to noble gases released in gaseous effluents to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation and;

b. During any calendar year: Less than or equal to 20 mrads for gamma radiation and less than or equal to 40 mrads for beta radiation.

APPLICABILITY: **At all times.**

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective **ACTIONS** that have been taken to reduce the releases and the proposed corrective **ACTIONS** to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the **ODCM** at least once per 31 days.

GASEOUS EFFLUENTS:

Dose - Iodine-131 & Radionuclides in Particulate Form

CONTROLS

3.11.2.3 The dose to a **MEMBER OF THE PUBLIC** from iodine-131 and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 15 mrems to any organ and,

b. During any calendar year: Less than or equal to 30 mrems to any organ.

c. Less than 0.1% of the limits of 3.11.2.3(a) and (b) as a result of burning contaminated oil.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131 and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective **ACTIONS** that have been taken to reduce the releases and the proposed corrective **ACTIONS** to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and the current calendar year for iodine-131 and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

GASEOUS EFFLUENTS:

Gaseous Radwaste Processing System

CONTROLS

3.11.2.4 The **GASEOUS RADWASTE PROCESSING SYSTEM** and the **VENTILATION EXHAUST TREATMENT SYSTEM** shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the gaseous effluent air doses due to gaseous effluent releases, to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) exceeds 1.20 mrad for gamma radiation and 2.4 mrad for beta radiation in a 92 day period. The **VENTILATION EXHAUST PROCESSING SYSTEM** shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the calculated doses due to gaseous effluent releases, to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) exceeds 1.8 mrem to any organ in a 92 day period.

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without processing and in excess of the above limits, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days a Special Report that includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without processing, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. **ACTION**(s) taken to restore the inoperable equipment to **OPERABLE** status, and

   3. Summary description of **ACTION**(s) taken to prevent a recurrence.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses over a 92-day period due to gaseous releases shall be calculated at least once per 31 days in accordance with the methodology and parameters in the **ODCM**.

TOTAL DOSE

CONTROLS

3.11.4 The annual (calendar year) dose or dose commitment to any **MEMBER OF THE PUBLIC** due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: **At all times.**

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Control 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations shall be made including direct radiation contributions from the reactor units and outside storage tanks to determine whether the above limits of Control 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days a Special Report that defines the corrective **ACTION** to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a **MEMBER OF THE PUBLIC** from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff **ACTION** on the request is complete.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Controls 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the **ODCM**.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and outside storage tanks shall be determined in accordance with the methodology and parameters in the **ODCM**. This requirement is applicable only under conditions set forth in Control 3.11.4.a.

RADIOLOGICAL ENVIRONMENTAL MONITORING

PROGRAM

CONTROLS

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Technical Specification 5.6.2, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sample at a specified location exceeding the reporting levels of Table 3.12-2, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days after receiving the sample analysis a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective **ACTIONS** to be taken 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 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sample this report shall be submitted if:

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

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to a **MEMBER OF THE PUBLIC** is equal to or greater than the calendar year limits of Controls 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

---

* The methodology and parameters used to estimate the potential annual dose to a **MEMBER OF THE PUBLIC** shall be indicated in this report.

c. With fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Technical Specification 5.6.2, identify the cause of the unavailability of samples and identify the new location(s) for obtaining the replacement samples in the next Annual Radiological Environmental Operating Report.

d. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the **ODCM**, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL 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]	23 routine monitoring stations (DR1 - DR23) either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the **SITE BOUNDARY** (DR1 - DR09);[f] an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site (DR10 - DR18); the remaining stations (DR19 - DR23) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 area to serve as a control station.	At least Quarterly	Gamma dose at least quarterly

TABLE 3.12-1 (Continued)

RADIOLOGICAL 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	Samples from 5 locations (A1-A5): 3 samples (A1-A3) from close to the 3 **SITE BOUNDARY** locations, in different sectors of the highest calculated annual average ground-level D/Q.[f] 1 sample (A4) from the vicinity of a community having the highest calculated annual average ground-level D/Q. 1 sample (A5) from a control location, as for example 15-30 km distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Canister I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[c] Gamma isotopic analysis[d] of composite (by location) quarterly

TABLE 3.12-1 (Continued)

RADIOLOGICAL 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 (Wa1) 1 sample at discharge area (Wa2)	**COMPOSITE SAMPLE** over 1-month period[e]	**GAMMA ISOTOPIC ANALYSIS**[d] monthly. Composite for tritium analysis quarterly
b. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value (Wb1)	Semiannually	**GAMMA ISOTOPIC ANALYSIS**[d] semiannually

TABLE 3.12-1 (Continued)

RADIOLOGICAL 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. Fish and Invertebrates	3 samples of commercially and/or recreationally important species (2 fish species and 1 invertebrate species) in vicinity of plant discharge area (Ia1-Ia3).	Sample in season, or semiannually if they are not seasonal	**GAMMA ISOTOPIC ANALYSIS**[d] on edible portions
	3 samples of same species in areas not influenced by plant discharge (Ia4-Ia6).		
b. Food Products	Samples of 3 different kinds of broad leaf vegetation grown near the **SITE BOUNDARY** at 2 different locations of highest predicted annual average ground level D/Q (Ib1-Ib6).[f]	Monthly during growing season	Gamma isotopic[d] and I-131 analysis
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction (Ib7-Ib9).	Monthly during growing season	Gamma isotopic[d] and I-131 analysis

TABLE 3.12-1 (Continued)

TABLE NOTATION

a. The code in parenthesis, e.g., DR1, A1, defines generic sample locations in this control that can be used to identify the specific locations in the map(s) and table in the **ODCM**. Specific parameters of distance and direction sector from the central point between the two containment buildings and additional description where pertinent, is provided for each sample location in Table 3.12-1, and in a table and figure(s) in the **ODCM**. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, October 1978", and 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. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective **ACTION** prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. Pursuant to Technical Specification 5.6.2, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining samples in the next Annual Radiological Environmental Operating Report and also include in the report a revised figure(s) and table for the **ODCM** reflecting the new location(s).

TABLE 3.12-1 (Continued)

TABLE NOTATION

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 as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. 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. Due to the geographical limitations, 9 sectors are monitored around the Calvert Cliffs Nuclear Power Plant.

c. Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, **GAMMA ISOTOPIC ANALYSIS** shall be performed on the individual samples.

d. **GAMMA ISOTOPIC ANALYSIS** means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

e. A **COMPOSITE SAMPLE** is one in which the quantity (aliquot) of liquid sampled 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 obtaining a representative sample.

f. Exception to these locations is in the South Sector where DR7, A1, 1b4, 1b5, and 1b6 are located approximately 0.7 km from the release point. This location is conservative with respect to the site boundary, which is located approximately 2.1 km from the release point.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/M³)	FISH & INVERTEBRATES (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000#				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

For drinking water samples. This is a 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a,b]

LOWER LIMIT OF DETECTION (LLD)[c]

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/M^3)	FISH & INVERTEBRATES (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg,dry)
Gross Beta	4	0.01				
H-3	2,000[#]					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

[#] If no drinking water pathway exists, a value of 3000 pCi/l may be used.

TABLE 4.12-1 (Continued)

TABLE NOTATION

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 pursuant to Technical Specification 5.6.2.

b

Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

c

The LLD is defined, for purposes of these 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:

$$LLD = \frac{4.66\, s_b}{E \bullet V \bullet 2.22 \bullet Y \bullet \exp(-\lambda \Delta t)}$$

Where:

LLD is the "A priori" **LOWER LIMIT OF DETECTION** as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

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

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

TABLE 4.12-1 (Continued)

TABLE NOTATION

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting.

Typical values of E, V, Y and Δt 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. 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 pursuant to Technical Specification 5.6.2.

d LLD for drinking water samples. If no drinking water pathway exists, the LLD of **GAMMA ISOTOPIC ANALYSIS** may be used.

RADIOLOGICAL ENVIRONMENTAL MONITORING:

LAND USE CENSUS

CONTROLS

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles), the location in each of the 9 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the locations in each of the 9 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation).

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Control 4.11.2.3, identify the new location(s) in the next Annual Radiological Environmental Operating Report, pursuant to Technical Specification 5.6.2.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Control 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Technical Specification 5.6.2, identify the new location(s) in the next Annual Radiological Environmental Operating Report and also include in the report a revised figure(s) and table for the **ODCM** reflecting the new location(s).

c. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2.

---

* 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. Specifications for broad leaf vegetation sampling in Table 3.12-1.4b shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING:

INTERLABORATORY COMPARISON PROGRAM

CONTROLS

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective **ACTIONS** taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the **ODCM**. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2.

PART 5.0 : CALCULATIONAL METHODOLOGIES

RADIOACTIVE LIQUID EFFLUENTS

RELEASE PATHWAYS [B527]

1. Introduction

   a) Radioactive liquid waste generated as a result of operating the Calvert Cliffs Nuclear Power Plant (CCNPP) may be released to the Chesapeake Bay[1].

   b) There are three pathways by which waste water, non-radioactive and radioactive, may be discharged from the site to the bay:

      (1) Outfall 001,

      (2) Outfall 003,

      (3) Outfall 004.

   c) A diagram which shows the location of each outfall is included as Attachment 1.

2. Outfall 001

   a) Water from the Chesapeake Bay is pumped through the condensers and is discharged to the Chesapeake Bay through the circulating water discharge conduits, known as outfall 001.

   b) The liquid radioactive waste is mixed with and diluted by the circulating water prior to exiting the discharge conduit.

   c) The circulating water discharge conduit extends 850 feet into the Chesapeake bay.

   d) The circulating water discharge conduit may accept liquid radioactive waste from various sources. Sources which may contribute radioactive material to the circulating water discharge conduit are tabulated in Attachment 2.

3. Outfalls 003 and 004

   a) There are two other potential pathways for the release of radioactive liquids to the bay. These pathways are designated outfall outfall 003 and outfall 004.

   b) Sources which could potentially contribute radioactive material to each of these outfalls are tabulated in Attachment 2.

   c) Outfall 002 was modified in 2001 to be discharged via outfall 001.

4. Unmonitored release paths not shown on Attachment 3 should be evaluated and added to the ODCM as necessary.

---

1 The federal controls and administrative limits associated with the release of radioactive materials from CCNPP are discussed elsewhere in this document.

5. Safety evaluations have been conducted for operating the following systems after they have become contaminated:

a) component cooling water system

(1) In accordance with applicable safety evaluations (Ref. 22), continued operation of this system is allowed as long as the concentration of radionuclides in the component cooling water system is less than 3,000 MPCs.

b) plant heating system

(1) In accordance with applicable safety evaluations (Ref. 24 and Ref. 28), continued operation of this system is allowed as long as the concentration of radionuclides in the plant heating system is less than 0.3 MPCs.

c) auxiliary boiler system

(1) In accordance with applicable safety evaluations (Ref. 27 and Ref. 30) continued operation of this system is allowed as long as the concentration of radionuclides in the auxiliary boiler steam drum is less than 96 MPCs.

d) demineralized water system

(1) In accordance with applicable safety evaluations (Ref. 25 and Ref. 31), continued operation of this system is allowed as long as the concentration of radionuclides in the demineralized water system is less than 0.3 MPCs.

e) service water system

(1) In accordance with applicable safety evaluations (Ref. 26 and Ref. 32), continued operation of this system is allowed as long as the concentration of radionuclides in the service water system is less than 0.1 MPCs.

f) condensate storage tank

(1) In accordance with applicable safety evaluations (Ref. 33), continued operation of this system is allowed as long as the concentration of radionuclides in the condensate storage tank is less than 1.0 MPCs.

TYPES OF LIQUID RELEASES

1. Liquid radwaste discharges have been classified as **CONTINUOUS** or **BATCH** as shown on Attachment 2.

PROCESSING EQUIPMENT

1. Simplified Flow Diagram

   a) An overview of the liquid waste processing system, including major equipment and (normal) flow paths, is outlined on Attachment 4.

   b) There is no processing equipment for wastes discharged through outfalls 003 and 004; however, the waste shall be analyzed for radioactivity in accordance with the analysis frequencies contained in approved CHEMISTRY SECTION procedures.

2. Modifications

   a) Licensee initiated major changes to the Liquid Radioactive Waste System shall be reported to the Commission in the Radioactive Effluent Release Report for the period in which the modification to the waste system is completed. The discussion of each change shall contain:

      (1) A description of the equipment, components and processes involved; and

      (2) Documentation of the fact that the change, including the safety analysis, was reviewed and found acceptable by the onsite review function.

      The report shall also include changes to the ODCM, in accordance with Technical Specification 5.6.3.

   b) A "major" change or modification includes, but is not limited, to the removal or permanent bypass of any of the following:

      (1) degassifier

      (2) reactor coolant waste receiver tank

      (3) reactor coolant waste monitor tank

      (4) letdown filter

      (5) reactor coolant waste ion exchanger

      (6) miscellaneous waste ion exchanger

      (7) miscellaneous waste filter

      (8) miscellaneous waste receiver tank

      (9) miscellaneous waste monitor tank

      (10) evaporator

3. Detailed Description

a) A detailed description of the liquid waste processing system is beyond the scope of the ODCM.

b) For more information on the Miscellaneous Liquid Waste Processing System see UFSAR and System Description SD-071A, "Miscellaneous Liquid Waste Processing System Description."

c) For more information on the Reactor Coolant Waste Processing System see System Description SD-071B, "Reactor Coolant Waste Processing System Description."

d) For more information see the Updated Final Safety Analysis Report, Chapter 11, "Waste Processing And Radiation Protection."

Jan 2011

LIQUID EFFLUENT RADIATION MONITOR ALARM AND FIXED SETPOINTS

1. Liquid Waste Discharge Radiation Monitor (0-RE-2201)

a) General description

(1) number of radiation elements: one

(2) type of radiation element: in-line scintillation detector

(3) output: analog

(4) radiation indicator: 0-RI-2201

(5) units for radiation indicator: counts per minute

(6) supplier: Westinghouse

b) Functions of 0-RE-2201

(1) continuously measure the activity contained in liquid waste discharge line (Control 3.3.3.10)

(2) continuously indicate (via 0-RI-2201) the activity of liquids contained in liquid waste discharge line (Control 3.3.3.10)

(3) alarm (via 0-RI-2201) prior to exceeding 10 CFR 20, Appendix B, Table II, Column 2 limits (per Control 3.11.1.1)

(4) automatically terminate discharges from the liquid waste processing system by closing the discharge isolation valves (MWS-2201-CV, and MWS-2202-CV) whenever the radiation indicator (0-RI-2201) exceeds the fixed setpoint.

c) **OPERABILITY** of 0-RE-2201

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s)

(2) The functions of 0-RE-2201 are listed in paragraph (b) above.

d) Monitors equivalent to 0-RE-2201

(1) There are no equivalent monitors for 0-RE-2201.

e) Radiological effluent controls for 0-RE-2201

(1) Liquid waste discharges via this pathway may continue if any one of the following two conditions are satisfied (per Control 3.3.3.10).

(a) The liquid waste discharge monitor, 0-RE-2201, is operable and the alarm-with-automatic-termination setpoint for this monitor is set to ensure the concentrations of radioactive materials released in liquid effluents to **UNRESTRICTED AREAS** are less than the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2, or

(b) two independent samples are analyzed in accordance with Control 4.11.1.1.1; AND at least two technically qualified members of the Facility Staff independently verify the release rate calculations; AND two qualified operators verify the discharge valve line up.

f) Surveillances for 0-RE-2201

(1) Control 4.3.3.10 requires demonstrating the **OPERABILITY** of 0-RE-2201 by satisfying the checks, calibrations, and tests listed below:

(a) **CHANNEL CHECK** within the past 24 hours

(b) **SOURCE CHECK** prior to each release

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

g) Setpoints for 0-RI-2201

(1) There are three radiation alarm setpoints associated with, or otherwise related to, the liquid waste discharge monitor.

(a) 0-RI-2201 fixed high radiation alarm and automatic termination setpoint

(b) 0-RI-2201 adjustable plant computer high radiation alarm and manual termination setpoint

(c) 0-RI-2201 low radiation alarm setpoint

(2) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows:

(a) The 0-RI-2201 fixed high radiation alarm and automatic termination setpoint will be referred to as the fixed setpoint.

(b) The 0-RI-2201 adjustable plant computer high radiation alarm and manual termination setpoint will be referred to as the adjustable setpoint.

(c) The 0-RI-2201 low radiation alarm setpoint will be referred to as the low setpoint.

(3) Each of these alarm setpoints is described below.

h) Fixed setpoint for 0-RI-2201

(1) General information

(a) This setpoint is not adjusted for each release.

(b) Whenever the fixed setpoint is exceeded, discharges from the liquid waste processing system will be automatically suspended.

(c) The fixed setpoint corresponds to the maximum concentration of radionuclides allowed in liquid waste discharged from the liquid waste processing system.

(d) The current value for the fixed setpoint is specified in the CCNPP Alarm Manual.

(e) The CCNPP Alarm Manual refers to this alarm and fixed setpoint as the 0-RI-2201 Liquid Waste Discharge High Alarm.

(f) The fixed setpoint is integral to the liquid release discharge monitor, as purchased from the supplier.

(g) The fixed setpoint is administratively controlled by EN-1-100.

(h) The fixed setpoint shall be calculated as described below[1].

(2) Calculating the fixed setpoint for 0-RI-2201

(a) The fixed setpoint for 0-RI-2201 shall be calculated as follows:

---

1 Per Control 3.3.3.10.

FIXED SETPOINT FOR 0-RI-2201

$$S_{fix} \leq K_{sf}\{ (F_d / F_u) \Sigma [(A_{iLn}) (e_i)] + Bkg \} \quad \text{Eq. 1L}^{1}$$

S_{fix} = the fixed setpoint for 0-RI-2201 (cpm)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the 10 CFR 20 MPC limit (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed setpoint is always less than or equal to the limits of 10 CFR 20.

A safety factor of 1.00 will yield a fixed setpoint which corresponds to 1.0 MPC.

A safety factor of 0.500 will yield a fixed setpoint which corresponds to 0.5 MPCs.

It is recommended that a safety factor of 0.5 be used for calculating the fixed setpoint, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133, section 4.1.1, which states, "The alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous liquid release limit of 10 CFR Part 20 is not exceeded."

This safety margin will prevent minor fluctuations in the nominal circulating water flow rate (and other statistical aberrations) from adversely impacting the calculated fixed setpoint.

F_d = the dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The dilution water flow rate is actually the sum of the minimum circulating water flow rate, the minimum salt water flow rate, and the maximum undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

---

1 This formula has been derived from NUREG-0133, Addendum, page AA-1.

DILUTION WATER FLOW RATE

$F_d = F_{cw} + F_{sw} + F_u$		**Eq. 2L**

F_{cw} = the minimum circulating water system flow rate necessary to conduct liquid releases

A minimum of two circulating water pumps (one circulating water pump per conduit accepting radwaste) shall be operable when discharging liquid radwaste -- via this monitor -- to outfall 001.

Additional circulating water pumps may be required as specified in approved Chemistry Procedures.

If a release of liquid radwaste is to be conducted using less than minimum circulating water flow rate necessary to conduct liquid releases, then the release shall not be allowed until a new setpoint has been calculated and entered into 0-RI-2201 (unless the existing setpoint is conservative), or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_{sw} = the minimum salt water system flow rate necessary to release liquid radwaste

If the minimum salt water flow rate available for liquid releases is unknown, difficult to predict, or may decrease during a liquid release, a minimum salt water flow rate of 0 gpm should be used when calculating the fixed setpoint.

If a release of liquid radwaste is to be conducted using less than the minimum salt water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 0-RI-2201 (unless the existing setpoint is conservative), or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_u = maximum undiluted radwaste flow rate (gpm)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 2.

Since the flow rate of undiluted liquid waste (120 gpm maximum) is insignificant relative to the circulating water flow rate (200,000 gpm per circulating water pump), the flow rate of undiluted liquid waste need not be considered when calculating the dilution water flow rate.

Since the maximum undiluted radwaste flow rate is used for calculating the fixed setpoint, a flow setpoint is not required--for the flow measuring device (0-FE-2199) in the effluent line--to verify compliance with Control 3.3.3.10.

A_{iLn} = specific activity limit for the release of radionuclide, i, to **UNRESTRICTED AREAS** (calculated in accordance with 10 CFR 20, Appendix B, Table II, Note 1 as described below; microcuries per milliliter).

In order to calculate a meaningful and accurate fixed setpoint, the specific radionuclides, i, chosen for calculating the fixed setpoint should correspond to those radionuclides found in **TYPICAL RADWASTE RELEASES** from CCNPP.

Attachment 5 provides further guidance for determining the identity of those radionuclides, i, to be used to calculate a fixed setpoint.

Values for A_{iLn} shall be calculated, as described below, for each radionuclide.

SPECIFIC ACTIVITY LIMIT FOR RADIONUCLIDE i

$A_{ILn} = (f_i)(A_{TLn})$		**Eq. 3L**

f_i = a fraction which represents the relative activity contribution of nuclide i to the average total effluent activity (unitless)

The average total effluent activity does not include tritium or dissolved and entrained noble gases.

This value may be obtained using the guidance provided on Attachment 5.

A_{TLn} = the sum of the total specific activities of all radionuclides found in a TYPICAL LIQUID RADWASTE RELEASE (microcuries/cm^3)

This value corresponds to 1 MPC.

A_{TLn} shall be calculated as shown below.

TOTAL SPECIFIC ACTIVITY EQUIVALENT TO 1 MPC

$\Sigma[(f_i)(A_{TLn})/A_{iLt}] = 1$	**Eq. 4L**[1]

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 2 (microcuries/cm^3)

For all the radionuclides found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 2.

1 = the MPC limit (MPCs) for **UNRESTRICTED AREAS**

This value is based on the MPC limit from 10 CFR 20, Table II, Note 1.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background prior to initiating the liquid release (cpm)

Instead of using an approximation of the detector background, a value of 0 cpm may be used as the detector background if so desired.

(3) Documenting the fixed setpoint for 0-RI-2201

(a) Whenever the fixed setpoint is calculated, the specific values chosen for each of the parameters shall be documented in accordance with EN-1-100.

(4) Changing the fixed setpoint for 0-RI-2201

[1] This formula has been derived from 10 CFR 20, Table II, Notes 1, 2, and 3.

(a) If the fixed setpoint calculated in accordance with equation 1L exceeds the maximum range of the monitor, the fixed setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) The fixed setpoint for 0-RI-2201 may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in equation 1L have changed.

(d) The fixed setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL LIQUID EFFLUENTS, f_i, has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the minimum dilution water flow rate is not available for a liquid release,

iii) the values listed in 10 CFR 20, Table II, column 2 have changed,

iv) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

v) the monitor is not conservative in its function (see "Functions of 0-RE-2201" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving the fixed setpoint.

---

1 As determined in accordance with Attachment 5.

i) The adjustable setpoint for 0-RI-2201

(1) General information

(a) The adjustable setpoint is calculated and adjusted prior to each release from the liquid waste processing system.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1).

(c) Whenever the adjustable setpoint is exceeded, discharges from the liquid waste processing system will be manually suspended.

(d) See OI-17C, "Reactor Coolant Waste Processing System", or OI-17D, "Miscellaneous Waste Processing System," for a full list of operator **ACTIONS** taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected in discharges from the liquid waste processing system.

(f) The value for the adjustable setpoint is recorded on the liquid release permit in accordance with CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the liquid release discharge monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer, which monitors output from 0-RI-2201, and provides an alarm to plant operators when the adjustable setpoint has been exceeded.

(i) A value for the adjustable setpoint shall be calculated prior to each release as shown below.

(2) Calculating the adjustable setpoint for 0-RI-2201

(a) The adjustable setpoint is based on the specific activities of the radionuclides in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1), and shall be calculated as shown below.

ADJUSTABLE SETPOINT FOR 0-RI-2201

$$S_{adj} \leq 1.50 \left[\Sigma (A_{iu}) (e_i) + Bkg \right] \qquad \text{Eq. 5L}$$

S_{adj} = the adjustable setpoint for 0-RI-2201 (cpm)

1.50 = a constant, actually a safety factor, which allows for fluctuations in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, or (5) errors associated with monitor calibrations.[1]

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream (microcuries per milliliter)

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

---

(3) Documenting the adjustable setpoint for 0-RI-2201

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures.

(4) Changing the adjustable setpoint for 0-RI-2201

(a) In all cases, the adjustable setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(b) If the adjustable setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(c) CHEMISTRY SECTION procedures contain administrative controls associated with calculating and approving an adjustable setpoint.

(d) Whenever 0-RI-2201 is operable, the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release from the liquid waste processing system.

---

1 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

(e) Plant Operating Instructions contain administrative controls associated with entering the adjustable setpoint into the plant computer.

j) The low setpoint for 0-RI-2201

(1) This setpoint is integral to the liquid release discharge monitor, as purchased from the supplier.

(2) The current value for the low setpoint is specified in the CCNPP Alarm Manual.

(3) The low setpoint may be used to determine the **OPERABILITY** of this monitor (per Control 4.3.3.10, **CHANNEL FUNCTIONAL TEST**).

(4) The alarm generated by the low setpoint may be used to terminate a release in the event 0-RE-2201 fails (i.e., downscale failure or circuit failure) in accordance with Control 4.3.3.10.

(5) The low setpoint calculations are not described in the ODCM.

(6) Changes to the low setpoint are controlled by EN-1-100.

2. Steam Generator Blowdown Effluent Radiation Monitors (1/2-RE-4095)

a) General description of 1/2-RE-4095

(1) number of radiation elements: one per unit

(2) Type of radiation elements: in-line scintillation detectors

(3) output: analog

(4) Designations for radiation indicators

(a) 1-RI-4095

(b) 2-RI-4095

(5) units for radiation indicator: counts per minute

(6) supplier: Westinghouse

b) Functions of 1/2-RE-4095

(1) continuously measure the activity contained in steam generator blowdown effluent line (Control 3.3.3.10)

(2) continuously indicate (via 1/2-RI-4095) the activity of liquids contained in the steam generator blowdown effluent line (Control 3.3.3.10)

(3) alarm (via 1/2-RI-4095) prior to exceeding 10 CFR 20, Appendix B, Table II, Column 2 limits (per Control 3.11.1.1).

(4) automatically terminate steam generator blowdown releases to **UNRESTRICTED AREAS** when the radiation indicator (1/2-RI-4095) exceeds the fixed setpoint

c) **OPERABILITY** of 1/2-RE-4095

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s).

(2) The functions of this monitor are listed in section (b) above.

(3) It should be noted that if the steam generator blowdown processing system heat exchangers are bypassed, it is possible for blowdown flow to bypass 1/2-RE-4095. If blowdown flow is allowed to bypass 1/2-RE-4095, the minimum channels **OPERABILITY** requirement of Control 3.3.3.10 may not be satisfied.

d) Monitors equivalent to 1/2-RE-4095

(1) 1/2-RE-4014 is normally considered the PRIMARY MONITOR for measuring activity released via the steam generator blowdown processing system, and 1/2-RE-4095 is normally considered the BACKUP MONITOR for measuring activity released via the steam generator blowdown processing system.

(2) In the event PRIMARY MONITOR (1/2-RE-4014) is inoperable or otherwise unavailable, the BACKUP MONITOR (1/2-RE-4095) may fulfill the measuring, indicating, and alarming functions normally provided by the PRIMARY MONITOR.

(3) 1/2-RE-4014, Steam Generator Blowdown Tank Radiation Monitor is considered to be "equivalent" monitor to 1/2-RE-4095 as specified below.

(4) 1/2-RE-4014, Steam Generator Blowdown Tank Radiation Monitor, may perform measurement, indication, alarm, and isolation functions (see "Functions of 1/2-RE-4014" earlier in this section) which limit the concentration of radioactive materials released to **UNRESTRICTED AREAS** in accordance with Control 3.11.1.1 as long as the following conditions are satisfied:

(a) the **OPERABILITY** of 1/2-RE-4014 must be demonstrated in accordance with Control 4.3.3.10, Table 3.3-13(1b), and

(b) the blowdown ion exchangers are isolated (or the decontamination factors for all radionuclides are verified to be greater than or equal to one for the duration of the release), and

(c) the specific activities of radionuclides in the blowdown tank radiation monitor are representative of the activities of the radionuclides in the blowdown effluent line.

e) Radiological effluent controls for 1/2-RE-4095

(1) Steam generator blowdown releases via this pathway may continue if any one of the following two conditions are satisfied (per Control 3.3.3.10):

(a) A steam generator blowdown monitor (either 1/2-RE-4095 or 1/2-RE-4014) is **OPERABLE** (see "**OPERABILITY** of 1/2-RE-4095"; and "**OPERABILITY** of 1/2-RE-4014" earlier in this section) AND the setpoint for this monitor is set to ensure the concentrations of radioactive materials released in liquid effluents to **UNRESTRICTED AREAS** are less than the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2, or

(b) a grab sample is collected AND analyzed in accordance with the provisions described below:

i) analysis shall determine EITHER the gross gamma activity or gross beta activity of the sample

ii) sampling and analysis shall be completed at least once per 12 hours if the specific activity of the steam generator blowdown is greater than 0.01 microcuries per gram IODINE DOSE EQUIVALENT

iii) sampling and analysis shall be completed at least once per 48 hours if the specific activity of the steam generator blowdown is less than or equal to 0.01 microcuries per gram IODINE DOSE EQUIVALENT

iv) the LOWER LIMITS OF DETECTION (LLDs) for the gross gamma and/or gross beta analyses are in accordance with the LLDs of Control 4.11.1.1.1

f) Surveillances for 1/2-RE-4095

(1) Control 4.3.3.10 requires demonstrating the **OPERABILITY** of 1/2-RE-4095 by satisfying the checks, calibrations, and tests listed below

(a) **CHANNEL CHECK** within the past 24 hours

(b) **SOURCE CHECK** prior to each release

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

g) Setpoints for 1/2-RI-4095

(1) There are three radiation alarm setpoints associated with the Steam Generator Blowdown Effluent Radiation Monitor.

(a) 1/2-RI-4095 fixed high radiation alarm and automatic termination setpoint

(b) 1/2-RI-4095 adjustable plant computer high radiation alarm and manual termination setpoint

(c) 1/2-RI-4095 low radiation alarm setpoint

(2) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows:

(a) The 1/2-RI-4095 fixed high radiation alarm and automatic termination setpoint will be referred to as the fixed setpoint.

(b) The 1/2-RI-4095 adjustable plant computer high radiation alarm and manual termination setpoint will be referred to as the adjustable setpoint.

(c) The 1/2-RI-4095 low radiation alarm setpoint will be referred to as the low setpoint.

(3) Each of these alarm setpoints are described below.

h) The fixed setpoint for 1/2-RI-4095

(1) General information

(a) The fixed setpoint is not adjusted for each release.

(b) Whenever the fixed setpoint is exceeded, discharges from the steam generator blowdown processing system to **UNRESTRICTED AREAS** will be automatically suspended.

(c) The fixed setpoint corresponds to the maximum concentration of radionuclides allowed in liquid waste discharged from the steam generator blowdown processing system.

(d) The current value for the fixed setpoint is specified in the CCNPP Alarm Manual.

(e) The CCNPP Alarm Manual refers to this setpoint as the 1/2-RI-4095 Steam Generator Blowdown Recovery High Alarm.

(f) This setpoint is integral to the steam generator blowdown discharge monitor, as purchased from the manufacturer.

(g) The fixed setpoint is administratively controlled by EN-1-100.

(h) The fixed setpoint shall be calculated as described below[1].

(2) Calculating the fixed setpoint for 1/2-RI-4095

(a) The fixed setpoint for 1/2-RI-4095 shall be calculated in accordance with equation 1L.

FIXED SETPOINT FOR 1/2-RI-4095

$$S_{fix} \leq K_{sf} \{ (F_d/F_u) \Sigma [(A_{iLn}) (e_i)] + Bkg \} \qquad \text{Eq. 1L}^{2}$$

S_{fix} = the fixed setpoint for 1/2-RI-4095 (cpm)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the 10 CFR 20 MPC limit (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed setpoint is always less than or equal to the limits of 10 CFR 20.

A safety factor of 1.00 will yield a fixed setpoint which corresponds to 1.0 MPC.

A safety factor of 0.500 will yield a fixed setpoint which corresponds to 0.5 MPCs.

It is recommended that a safety factor of 0.5 be used for calculating the fixed setpoint, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133, section 4.1.1, which states, "The alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous liquid release limit of 10 CFR Part 20 is not exceeded."

1 Per Control 3.3.3.10.

2 Equation 1L has been derived from NUREG-0133, Addendum, page AA-1.

This safety margin will prevent minor fluctuations in the nominal circulating water flow rate (and other statistical aberrations) from adversely impacting the calculated fixed setpoint.

F_d = the dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

the dilution water flow rate is actually the sum of the minimum circulating water flow rate, the minimum salt water flow rate, and the maximum undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \quad \textbf{Eq. 2L}$$

F_{cw} = the minimum circulating water system flow rate necessary to conduct liquid releases

A minimum of two circulating water pumps (one circulating water pump per conduit accepting radwaste) shall be operable when discharging liquid radwaste -- via this monitor -- to outfall 001.

The flow rate of each circulating water pump is 200,000 gallons per minute.

Additional circulating water pumps may be required as specified in approved CHEMISTRY SECTION procedures.

If a release of liquid radwaste from the steam generator blowdown system is to be conducted using less than minimum circulating water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 1/2-RI-4095 (unless the existing setpoint is conservative), or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_{sw} = the minimum salt water system flow rate necessary to release liquid radwaste

If the minimum salt water flow rate available for liquid releases is unknown, difficult to predict, or may decrease during a liquid release, a minimum salt water flow rate of 0 gpm should be used when calculating the fixed setpoint.

The maximum flow rate for one salt water pump is listed on Attachment 2.

If a release of liquid radwaste is to be conducted using less than the minimum salt water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 1/2-RI-4095 (unless the existing setpoint is conservative), or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_u = maximum undiluted radwaste flow rate (gpm)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 2.

Since the steam generator blowdown flow rate (225 gpm maximum) is insignificant relative to the circulating water flow rate (200,000 gpm per circulating water pump), the steam generator blowdown flow rate need not be considered when calculating the dilution water flow rate.

Since the maximum undiluted radwaste flow rate is used for calculating the fixed setpoint, a flow setpoint is not required--for the flow measuring device in the effluent line--to verify compliance with Control 3.3.3.10.

A_{iLn} = specific activity limit for the release of radionuclide, i, to **UNRESTRICTED AREAS** (calculated in accordance with 10 CFR 20, Appendix B, Table II, Note 1 as described below; microcuries per milliliter)

In order to calculate a meaningful and accurate fixed setpoint, the specific radionuclides, i, chosen for calculating the fixed setpoint should correspond to those radionuclides found in **TYPICAL RADWASTE RELEASES** from CCNPP.

Attachment 5 provides further guidance for determining which radionuclides, i, should be used to calculate a fixed setpoint.

Values for A_{iLn} shall be calculated, as described below, for each radionuclide.

SPECIFIC ACTIVITY LIMIT FOR EACH RADIONUCLIDE i

$$A_{iLn} = (f_i)(A_{TLn}) \qquad \textbf{Eq. 3L}$$

f_i = a fraction which represents the relative activity contribution of nuclide i to the average total effluent activity (unitless)

The average total effluent activity does not include tritium or dissolved and entrained noble gases.

This value may be obtained using the guidance provided on Attachment 5.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

A_{TLn} = the sum of the total specific activities of all radionuclides found in a TYPICAL LIQUID RADWASTE RELEASE (microcuries/cm^3)

This value corresponds to 1 MPC.

A_{TLn} shall be calculated as shown below.

TOTAL SPECIFIC ACTIVITY EQUIVALENT TO 1 MPC

$$\Sigma [(f_i) (A_{TLn}) / A_{iLt}] = 1 \qquad \textbf{Eq. 4L}^1$$

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 2 (microcuries/cm^3)

For all the radionuclides found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 2.

1 = the MPC limit (MPCs) for **UNRESTRICTED AREAS**

This value is based on the MPC limit from 10 CFR 20, Table II, Note 1.

Bkg = an approximation of the detector background (cpm)

Instead of using an approximation of the detector background, a value of 0 cpm may be used as the detector background if so desired.

---

(3) Documenting the fixed setpoint for 1/2-RI-4095

(a) Whenever the fixed setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with EN-1-100.

(4) Changing the fixed setpoint for 1/2-RI-4095

(a) If the fixed setpoint calculated in accordance with equation 1L exceeds the maximum range of the monitor, the fixed setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) The fixed setpoint may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in equation 1L have changed.

---

1 This formula has been derived from 10 CFR 20, Table II, Notes 1, 2, and 3.

(d) The fixed setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL LIQUID EFFLUENTS has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the minimum dilution water flow rate is not available for a liquid release,

iii) the values listed in 10 CFR 20, Table II, column 2 have changed,

iv) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

v) the monitor is not conservative in its function (see "Functions of 1/2-RE-4095" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving the fixed setpoint.

i) The adjustable setpoint for 1/2-RI-4095

(1) General information

(a) The adjustable setpoint is calculated and adjusted prior to each release from the steam generator blowdown processing system.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1).

(c) Whenever the adjustable setpoint is exceeded, discharges from the steam generator blowdown processing system will be manually suspended.

(d) See OI-8A for a full list of operator actions taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected in discharges from the steam generator blowdown processing system.

(f) The value for the adjustable setpoint is recorded on the liquid release permit in accordance with CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the steam generator blowdown effluent monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 1/2-RI-4095, and provides an alarm to plant operators when the adjustable setpoint has been exceeded.

(i) A value for the adjustable setpoint shall be calculated prior to each release as shown below.

(2) Calculating the adjustable setpoint for 1/2-RI-4095

(a) The adjustable setpoint is based on the specific activities of radionuclides in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1), and shall be calculated as shown below.

ADJUSTABLE SETPOINT FOR 1/2-RI-4095

$$S_{adj} \leq 2.50 \left[\Sigma (A_{iu}) (e_i) + Bkg \right] \quad \text{Eq. 5L}^{2}$$

Where,

S_{adj} = the adjustable setpoint for 1/2-RI-4095 (cpm)

2.50 = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, or (5) errors associated with monitor calibrations.[3]

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream (microcuries per milliliter)

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background prior to initiating the liquid release (cpm)

(3) Documenting the adjustable setpoint, 1/2-RI-4095

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures.

1 As determined in accordance with Attachment 5.
2 This formula may be derived from NUREG-0133, Addendum, page AA-1.
3 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

(4) Changing the adjustable setpoint for 1/2-RI-4095

(a) If the adjustable setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) In all cases, the adjustable setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(c) Chemistry procedures contain administrative controls associated with calculating and approving an adjustable setpoint.

(d) Whenever 1/2-RI-4095 is operable, the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release from the steam generator blowdown processing system.

(e) Plant Operating Instructions (e.g., OI-8A) contain administrative controls associated with entering the adjustable setpoint in the plant computer.

j) The low setpoint for 1/2-RI-4095

(1) This alarm is integral to the steam generator blowdown effluent monitor, as purchased from the manufacturer.

(2) The current value for the low alarm setpoint is specified in the CCNPP Alarm Manual.

(3) The low alarm setpoint may be used to determine **OPERABILITY** of this monitor (per Control 4.3.3.10, **CHANNEL FUNCTIONAL TEST**).

(4) The low alarm setpoint may be used to terminate a release in the event 1/2-RE-4095 fails (i.e., downscale failure or circuit failure).

(5) Changes to the low alarm setpoint are controlled by EN-1-100.

(6) The ODCM does not address the calculations associated with the low alarm setpoint.

3. Steam Generator Blowdown Tank Radiation Monitors (1/2-RE-4014)

a) General description of 1/2-RE-4014

(1) number of radiation elements: one per unit

(2) Type of radiation element: off-line scintillation detector

(3) output: analog

(4) Designations for radiation indicators

(a) 1-RI-4014

(b) 2-RI-4014

(5) units for radiation indicator: counts per minute

(6) supplier: Westinghouse

(7) A 3 gallon per minute sample is cooled, passed through the detector, and pumped back into the steam generator blowdown tank.

b) Functions of 1/2-RE-4014

(1) continuously measure the activity contained in an off-line sample of the steam generator blowdown tank (Control 3.3.3.10)

(2) continuously indicate (via 1/2-RI-4014) the specific activity in an off-line sample of the steam generator blowdown tank (Control 3.3.3.10)

(3) alarm (via 1/2-RI-4014) prior to exceeding the 10 CFR 20, Appendix B, Table II, Column 2 limits (per Control 3.11.1.1)

(4) automatically terminate steam generator blowdown releases to **UNRESTRICTED AREAS** when the radiation indicator (1/2-RI-4014) exceeds the fixed alarm setpoint.

c) **OPERABILITY** of 1/2-RE-4014

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s)

(2) The functions of this monitor are listed in section (b) above.

d) Monitors equivalent to 1/2-RE-4014

(1) 1/2-RE-4014 is normally considered the PRIMARY MONITOR for measuring activity released via the steam generator blowdown processing system, and 1/2-RE-4095 is normally considered the BACKUP MONITOR for measuring activity released via the steam generator blowdown processing system.

(2) In the event PRIMARY MONITOR (1/2-RE-4014) is inoperable or otherwise unavailable, the BACKUP MONITOR (1/2-RE-4095) may fulfill the measuring, indicating, and alarming functions normally provided by the PRIMARY MONITOR.

(3) 1/2-RE-4095, Steam Generator Blowdown Effluent Radiation Monitor, performs measurement, indication, alarm, and isolation functions equivalent to 1/2-RE-4014 (see "Function of 1/2-RE-4014" earlier in this section), unless 1/2-RE-4095 is bypassed as described below.

(4) It should be noted that if the steam generator blowdown processing system heat exchangers are bypassed, it is possible for blowdown flow to bypass 1/2-RE-4095 (the steam generator blowdown effluent radiation monitor). If blowdown flow is allowed to bypass 1/2-RE-4095, the minimum channels **OPERABILITY** requirement of Control 3.3.3.10 may not be satisfied (unless 1/2-RE-4014 is operable).

e) Radiological effluent controls for 1/2-RE-4014

(1) Steam generator blowdown releases via this pathway may continue if any one of the following two conditions are satisfied (per Control 3.3.3.10):

(a) A steam generator blowdown monitor (either 1/2-RE-4095 or 1/2-RE-4014) is **OPERABLE** (see "**OPERABILITY** of 1/2-RE-4095"; and "**OPERABILITY** of 1/2-RE-4014" earlier in this section) AND the alarm setpoint for this monitor is set to ensure the concentrations of radioactive materials released in liquid effluents to **UNRESTRICTED AREAS** are less than the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2, or

(b) a grab sample is collected AND analyzed in accordance with the provisions described below:

i) analysis shall determine EITHER the gross gamma activity or gross beta activity of the sample

ii) sampling and analysis shall be completed at least once per 12 hours if the specific activity of the steam generator blowdown is greater than 0.01 microcuries per gram IODINE DOSE EQUIVALENT

iii) sampling and analysis shall be completed at least once per 48 hours if the specific activity of the steam generator blowdown is less than or equal to 0.01 microcuries per gram IODINE DOSE EQUIVALENT

iv) the LOWER LIMITS OF DETECTION for the gross gamma and/or gross beta analyses are in accordance with the LLDs of Control 4.11.1.1.1

f) Surveillances for 1/2-RE-4014

(1) Control 4.3.3.10 requires demonstrating the **OPERABILITY** of 1/2-RE-4014 by satisfying the checks, calibrations, and tests listed below

(a) **CHANNEL CHECK** within the past 24 hours

(b) **SOURCE CHECK** prior to each release

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

g) Setpoints for 1/2-RI-4014

(1) There are three radiation alarm setpoints associated with the Steam Generator Blowdown Tank Radiation Monitor

(a) 1/2-RI-4014 fixed high radiation alarm and automatic termination setpoint

(b) 1/2-RI-4014 adjustable plant computer high radiation alarm and manual termination setpoint

(c) 1/2-RI-4014 low radiation alarm setpoint

(2) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows:

(a) The 1/2-RI-4014 fixed high radiation alarm and automatic termination setpoint will be referred to as the fixed setpoint.

(b) The 1/2-RI-4014 adjustable plant computer high radiation alarm and manual termination setpoint will be referred to as the adjustable setpoint.

(c) The 1/2-RI-4014 low radiation alarm setpoint will be referred to as the low setpoint.

(3) Each of these alarms is described below.

h) The fixed setpoint for 1/2-RI-4014

(1) General Information

(a) The fixed setpoint is not adjusted for each release.

(b) Whenever the fixed setpoint is exceeded, discharges from the steam generator blowdown processing system to **UNRESTRICTED AREAS** will be automatically suspended.

(c) The fixed setpoint corresponds to the maximum concentration of radionuclides allowed in liquid waste discharged from the steam generator blowdown processing system.

(d) The current value for the fixed setpoint is specified in the CCNPP Alarm Manual.

(e) The CCNPP Alarm Manual refers to the fixed setpoint as the 1/2-RI-4014 Steam Generator Blowdown High Alarm.

(f) The fixed setpoint is integral to the liquid release discharge monitor, as purchased from the manufacturer.

(g) The fixed setpoint is administratively controlled by EN-1-100.

(h) The fixed setpoint shall be calculated as described below[1].

(2) Calculating the fixed setpoint for 1/2-RI-4014

(a) The fixed setpoint for 1/2-RI-4014 shall be calculated as follows:

---

1 Per Control 3.3.3.10.

FIXED SETPOINT FOR 1/2-RI-4014

$S_{fix} \leq K_{sf} \{ (F_d / F_u) \Sigma [(A_{iLn}) (e_i)] + Bkg \}$		Eq. 1L[1]

Where,

S_{fix} = the fixed setpoint for 1/2-RI-4014 (cpm)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the 10 CFR 20 MPC limit (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed setpoint is always less than or equal to the limits of 10 CFR 20.

A safety factor of 1.00 will yield a fixed setpoint which corresponds to 1.0 MPC.

A safety factor of 0.500 will yield a fixed setpoint which corresponds to 0.5 MPCs.

It is recommended that a safety factor of 0.5 be used for calculating the fixed setpoint, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133, SECTION 4.1.1, which states, "The alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous liquid release limit of 10 CFR Part 20 is not exceeded."

This safety margin will prevent minor fluctuations in the nominal circulating water flow rate (and other statistical aberrations) from adversely impacting the calculated fixed setpoint.

F_d = the dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The dilution water flow rate is actually the sum of the minimum circulating water flow rate, the minimum salt water flow rate, and the maximum undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

---

1 Equation 1L has been derived from NUREG-0133, Addendum, page AA-1.

DILUTION WATER FLOW RATE

$F_d = F_{cw} + F_{sw} + F_u$		**Eq. 2L**

F_{cw} = the minimum circulating water system flow rate necessary to conduct liquid releases

A minimum of two circulating water pumps (one circulating water pump per conduit accepting radwaste) shall be operable when discharging liquid radwaste -- via this monitor -- to outfall 001.

The flow rate of each circulating water pump is 200,000 gallons per minute.

Additional circulating water pumps may be required as specified in approved CHEMISTRY SECTION procedures.

If a release of liquid radwaste from the steam generator blowdown system is to be conducted using less than minimum circulating water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 1/2-RI-4014 (unless the existing setpoint is conservative), or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_{sw} = the minimum salt water system flow rate necessary to release liquid radwaste

If the minimum salt water flow rate available for liquid releases is unknown, difficult to predict, or may decrease during a liquid release, a minimum salt water flow rate of 0 gpm should be used when calculating the fixed setpoint.

The maximum flow rate for one salt water pump is listed on Attachment 2.

If a release of liquid radwaste is to be conducted using less than the minimum salt water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 1/2-RI-4014 (unless the existing setpoint is conservative), or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_u = maximum undiluted radwaste flow rate (gpm)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 2.

Since the steam generator blowdown flow rate (225 gpm maximum) is insignificant relative to the circulating water flow rate (200,000 gpm per circulating water pump), the steam generator blowdown flow rate need not be considered when calculating the dilution water flow rate.

Since the maximum undiluted radwaste flow rate is used for calculating the fixed setpoint, a flow setpoint is not required--for the flow measuring device in the effluent line--to verify compliance with Control 3.3.3.10.

A_{iLn} = specific activity limit for the release of radionuclide, i, to **UNRESTRICTED AREAS** (calculated in accordance with 10 CFR 20, Appendix B, Table II, Note 1 as described below; microcuries per milliliter)

In order to calculate a meaningful and accurate fixed setpoint, the specific radionuclides, i, chosen for calculating the fixed setpoint should correspond to those radionuclides found in **TYPICAL RADWASTE RELEASES** from CCNPP.

Attachment 5 provides further guidance for determining which radionuclides, i, should be used to calculate a fixed setpoint. An acceptable alternative is to assume an isotopic mix which results in a more conservative setpoint.

Values for A_{iLn} shall be calculated, as described below, for each radionuclide.

SPECIFIC ACTIVITY LIMIT FOR RADIONUCLIDE, i

$$A_{iLn} = (f_i)(A_{TLn}) \qquad \textbf{Eq. 3L}$$

f_i = a fraction which represents the relative activity contribution of nuclide i to the average total effluent activity (unitless)

The average total effluent activity does not include tritium or dissolved and entrained noble gases.

This value may be obtained using the guidance provided on Attachment 5.

A_{TLn} = the sum of the total specific activities of all radionuclides found in a TYPICAL LIQUID RADWASTE RELEASE (microcuries/cm^3)

This value corresponds to 1 MPC.

Calculate the value of A_{TLn} as shown below.

TOTAL SPECIFIC ACTIVITY CORRESPONDING TO 1 MPC AT THE **SITE BOUNDARY**

$$\Sigma [(f_i)(A_{TLn}) / A_{iLt}] = 1 \qquad \textbf{Eq. 4L}^{1}$$

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 2 (microcuries/cm^3)

For all the radionuclides found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 2.

1 This formula has been derived from 10 CFR 20, Table II, Notes 1, 2, and 3.

1 = the MPC limit (MPCs) for **UNRESTRICTED AREAS**

This value is based on the MPC limit from 10 CFR 20, Table II, Note 1.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

Instead of using an approximation of the detector background, a value of 0 cpm may be used as the detector background if so desired.

---

(3) Documenting the fixed setpoint for 1/2-RI-4014

(a) Whenever the fixed setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with EN-1-100.

(4) Changing the fixed setpoint for 1/2-RI-4014

(a) If the fixed setpoint calculated in accordance with equation 1L exceeds the maximum range of the monitor, the fixed setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) The fixed setpoint may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in equation 1L (identified in this section of the ODCM) have changed.

(d) The fixed setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL LIQUID EFFLUENTS has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the minimum dilution water flow rate is not available for a liquid release,

iii) the values listed in 10 CFR 20, Table II, column 2 have changed,

---

[1] As determined in accordance with Attachment 5.

iv) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

v) the monitor is not conservative in its function (see "Functions of 1/2-RE-4014" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving the fixed setpoint.

i) The adjustable setpoint for 1/2-RI-4014

(1) General information

(a) The adjustable setpoint is calculated and adjusted prior to each release from the steam generator blowdown processing system.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1).

(c) Whenever the adjustable setpoint is exceeded, discharges from the steam generator blowdown processing system will be manually suspended.

(d) See OI-8A for a full list of operator **ACTIONS** taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected in discharges from the steam generator blowdown processing system.

(f) The value for the adjustable setpoint is recorded on the liquid release permit in accordance with CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the steam generator blowdown tank radiation monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 1/2-RI-4014, and provides an alarm to plant operators when the 1/2-RI-4014 adjustable setpoint has been exceeded.

(i) A value for the adjustable setpoint shall be calculated prior to each release as shown below.

(2) Calculating the adjustable setpoint for 1/2-RI-4014

(a) The adjustable setpoint is based on the specific activity of the radionuclides in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1), and is calculated as shown below.

ADJUSTABLE SETPOINT FOR 1/2-RI-4014

$$S_{adj} \leq 2.50 \left[\Sigma \left(A_{iu} \right) \left(e_i \right) + Bkg \right] \qquad \text{Eq. 5L}^{1}$$

S_{adj} = the adjustable setpoint for 1/2-RI-4014 (cpm)

2.50 = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) uncertainties associated with sample analysis, and (5) uncertainties associated with monitor calibrations.[2]

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream (microcuries per milliliter)

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

(3) Documenting the adjustable setpoint for 1/2-RI-4014

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures.

(4) Changing the adjustable setpoint for 1/2-RI-4014

(a) If the adjustable setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) In all cases, the adjustable setpoint shall be set to a value which is less than or equal to the fixed setpoint.

[1] Equation 5L has been derived from NUREG-0133, Addendum, page AA-1.
[2] The "analysis uncertainties" and "calibration uncertainties" refer to uncertainties which are within established quality assurance and quality control limits.

(c) Chemistry procedures contain administrative controls associated with calculating and approving an adjustable setpoint.

(d) Whenever 1/2-RI-4014 is operable, the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release from the steam generator blowdown and processing system.

(e) Plant Operating Instructions (e.g., OI-8A) contain administrative controls associated with entering the adjustable setpoint in the plant computer.

j) The low alarm setpoint for 1/2-RI-4014

(1) This alarm is integral to the steam generator blowdown tank monitor, as purchased from the manufacturer.

(2) The current value for the low alarm setpoint is specified in the CCNPP Alarm Manual.

(3) Changes to the low alarm setpoint are controlled by EN-1-100.

(4) The ODCM does not address the calculations associated with the low alarm setpoint.

LIMITS ON RADIONUCLIDE CONCENTRATIONS IN LIQUID EFFLUENTS

1. Introduction

a) 10 CFR 20, Appendix B, specifies concentration limits associated with the release of radioactive materials to **UNRESTRICTED AREAS**.

b) Radiological effluent controls have been established to implement the requirements of 10 CFR 20.

c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

a) The concentration of radioactive material released in liquid effluents to **UNRESTRICTED AREAS** shall be limited to the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases (per Control 3.11.1.1).

b) It should be noted that NUREG-0133 specifies that the concentration of radioactive materials in liquid effluents to **UNRESTRICTED AREAS** shall be limited to 2 E-4 microcuries per milliliter for dissolved or entrained noble gases, this control has not been incorporated into the CCNPP Technical Specifications, and as a result, the ODCM does not include calculations for same.

c) The routine surveillances which are performed to verify compliance with these radiological effluent controls are described below.

3. Surveillance Requirements

a) Routine surveillances for **BATCH RELEASES**

(1) Sample each "BATCH" of radioactive liquid waste prior to its release (per Control 4.11.1.1.1).

(2) Determine the concentrations of principle gamma emitters (including, but not limited to, Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, Ce-141, I-131, Mo-99, and Ce-144) in the "BATCH" sample, prior to the "BATCH" release (per Control 4.11.1.1.1).

(3) Determine the concentration of tritium contained in a monthly **COMPOSITE SAMPLE** at least once per month (per Control 4.11.1.1.1).

(4) Determine the concentrations of Sr-89 and Sr-90 contained in a quarterly **COMPOSITE SAMPLE** at least once per quarter (per Control 4.11.1.1.1).

(5) Calculate the concentrations of radionuclides in the receiving waters at the point the liquid radioactive waste is released to **UNRESTRICTED AREAS** (per Control 4.11.1.1.2).

b) Routine surveillances for **CONTINUOUS RELEASES**

(1) Sample **CONTINUOUS RELEASES** of radioactive liquid waste at least once per month (per Control 4.11.1.1.1).

(2) Determine the concentrations of principle gamma emitters (including, but not limited to, Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, Ce-141, I-131, Mo-99, and Ce-144) in the undiluted, CONTINUOUS waste stream sample (per Control 4.11.1.1.1).

(3) Calculate the concentrations of radionuclides in the receiving waters at the point the liquid radioactive waste is released to **UNRESTRICTED AREAS** (per Control 4.11.1.1.2).

4. Responsible Plant Organizations

a) The CHEMISTRY SECTION is responsible for performing the sampling, analysis, and calculations required by the above surveillances.

b) The conditions which initiate the required surveillances are contained in the following section.

5. Initiating Conditions

a) The surveillances for **BATCH RELEASES** shall be performed prior to each **BATCH RELEASE**[1] and may be performed more often as specified in approved CHEMISTRY SECTION procedures.

b) The surveillances for **CONTINUOUS RELEASES** shall be performed at least monthly[2] (until the **CONTINUOUS RELEASE** has been terminated) and, if activity is identified[3] in the CONTINUOUS waste stream, the surveillances may be performed more often as specified in approved CHEMISTRY SECTION procedures.

6. Calculation Methodology

a) At CCNPP, there are two methodologies for calculating the concentrations of radionuclides in the receiving waters, at the point of release to **UNRESTRICTED AREAS**, resulting from the discharge of liquid waste.

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

b) Rigorous method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, ensure the concentrations of radionuclides in **UNRESTRICTED AREAS** are less than one MPC by verifying the following inequality is true:

---

1 Per Control 4.11.1.1.1.
2 Per Control 4.11.1.1.1.
3 The LOWER LIMITS OF DETECTION shall conform to Control 4.11.1.1.1, Table 4.11-1.

LIMIT ON CONCENTRATIONS OF RADIONUCLIDES IN **UNRESTRICTED AREAS** (RIGOROUS METHOD)

$(F_u / F_d) \Sigma (A_{iu} / A_{iLt}) \leq 1$	**Eq. 6L**[1]

F_d = the actual dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$F_d = F_{cw} + F_{sw} + F_u$	**Eq. 2L**

F_{cw} = the actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = the salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available, a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 2.

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 2 (microcuries/cm^3)

A_{iu} = the specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

1 = the MPC limit (MPCs) for **UNRESTRICTED AREAS**

This value is based on the MPC limit from 10 CFR 20, Table II, Note 1.

---

c) Simplified method

[1] Equation 6L has been derived from 10 CFR 20, Appendix B, Table II, Note 1.

(1) Whenever a computer system and appropriate software are unavailable to perform the rigorous calculations described in the previous section, ensure the concentrations of radionuclides in **UNRESTRICTED AREAS** are less than one MPC by verifying the following inequality is true.

LIMIT ON CONCENTRATIONS OF RADIONUCLIDES IN **UNRESTRICTED AREAS** (SIMPLIFIED METHOD)

$$\{ F_u / [(F_d) (A_{I131Lt}) (K_{sf})] \} \sum A_{iu} \leq 1 \qquad \textbf{Eq. 7L}^{1}$$

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 2.

F_d = the actual dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \qquad \textbf{Eq. 2L}$$

F_{cw} = the actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = the actual salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

A_{I131Lt} = the specific activity limit for I-131 corresponding to the limiting concentration specified in 10 CFR 20, Appendix B, Table II, Column 2

This value is 3E-7 microcuries per milliliter.

A_{iu} = the specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the activity limit of 10 CFR 20, Appendix B, Table II, Column 2, (unitless) (per Control 3.11.1.1)

The safety factor chosen shall be less than or equal to 1.00. This ensures the activity is always less than or equal to the activity limit of 10 CFR 20, Appendix B, Table II, Column 2.

---

1 This equation has been derived from 10 CFR 20, Appendix B, Table II, Note 1.

A safety factor of 1.00 will yield a activity which corresponds to the 10 CFR 20, Appendix B, Table II, Column 2 activity limit.

A safety factor of 0.500 will yield an activity which corresponds to one-half the activity limit of 10 CFR 20, Appendix B, Table II, Column 2.

It is recommended that a safety factor of 1.0 be used for calculating the activity, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive liquid release permits. This administrative control is designed to minimize the possibility of violating 10 CFR 20, Appendix B, Table II, Column 2 limits whenever simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 7L to account for any potential nonconservatism associated with applying the I-131 MPC limit to all radionuclides identified in the liquid release. Such nonconservatism could conceivable be present whenever radionuclides which have an MPC value higher than the I-131 MPC limit are present in a liquid release.

---

d) Once the rigorous or simplified calculations have been completed, the calculation results are compared to the applicable limits and corrective **ACTIONS** are initiated as described below.

7. Corrective **ACTIONS**

a) CHEMISTRY SECTION surveillance procedures shall contain and/or reference administrative and/or Technical Specification limits for concentration of radionuclides in liquid effluents and shall specify corrective **ACTIONS** to be initiated when these limits are exceeded.

b) Refer to Control 3.11.1.1 for **ACTIONS** to be taken in the event the concentrations of radionuclides in **UNRESTRICTED AREAS** exceed one MPC.

LIMITS ON CUMULATIVE TOTAL BODY DOSES AND CUMULATIVE ORGAN DOSES FOR LIQUID EFFLUENTS

1. Introduction

   a) Appendix I to 10 CFR 50 specifies total body dose limits and organ dose limits associated with the release of radioactive liquids to **UNRESTRICTED AREAS**.

   b) Radiological effluent controls have been established to implement the requirements of 10 CFR 50, Appendix I.

   c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

   a) The total body dose to **MEMBERS OF THE PUBLIC**, from liquid waste discharged to **UNRESTRICTED AREAS**, shall be less than 3 mrems per calendar quarter and 6 mrems per calendar year (Control 3.11.1.2).

   b) The organ doses to **MEMBERS OF THE PUBLIC**, from liquid waste discharged to **UNRESTRICTED AREAS**, shall be less than 10 mrems per calendar quarter and 20 mrems per calendar year (Control 3.11.1.2).

   c) The routine surveillances which are performed to verify compliance with these radiological effluent controls are described below.

3. Surveillance Requirement(s)

   a) Cumulative total body doses to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--for the current calendar month, the calendar quarter, the current calendar year, and the previous 92 days--shall be calculated, in accordance with equation 8L, at least once per 31 days (per Control 4.11.1.2 and 4.11.1.3).

   b) Cumulative organ doses to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--for the current calendar month, the current calendar quarter, the current calendar year, and the previous 92 days--shall be calculated, in accordance with equation 8L, at least once per 31 days (per Control 4.11.1.2 and 4.11.1.3).

4. Responsible Plant Organizations

   a) The CHEMISTRY SECTION is responsible for performing the surveillances, listed above, whenever the appropriate initiating conditions, listed below, are present.

5. Initiating Conditions

a) Perform the surveillances, listed above, at least once per 31 days (per Control 4.11.1.2).

b) For **BATCH RELEASES**, perform the surveillances, listed above, prior to each **BATCH RELEASE** of radioactive liquid waste in accordance with approved CHEMISTRY SECTION procedures.

c) For **CONTINUOUS RELEASES**, perform the surveillances, listed above, at least monthly (until the **CONTINUOUS RELEASE** has been terminated) and, if activity is identified[1] in the CONTINUOUS waste stream, the surveillances may be performed more often as specified in approved CHEMISTRY SECTION procedures.

d) Whenever the correct initiating conditions are present, the calculations required by the above mentioned surveillance(s) shall be completed in accordance with the methodology listed in the next section.

6. Calculation Methodology

a) The cumulative total body dose and the cumulative organ doses (for the current calendar month, current calendar quarter, previous 92 days, and current calendar year) shall be calculated as follows:

CUMULATIVE TOTAL BODY OR ORGAN, o, DOSE FROM LIQUID RELEASES, r

$$D_{ToL} = \Sigma D_{or} \quad \textbf{Eq. 8L}$$

D_{ToL} = the sum total of the total body or organ, o, dose for all liquid releases discharged during the applicable time interval

D_{or} = the organ or total body dose resulting from release, r

Calculate the values of D_{or} for each liquid release as described below.

b) At CCNPP, two methods exist for calculating D_{or} (i.e., the total body and organ doses resulting from any single release of radioactive liquid to an **UNRESTRICTED AREA**).

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

[1] The LOWER LIMITS OF DETECTION shall conform to Control 4.11.1.1.1, Table 4.11-1.

c) Rigorous method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, the dose commitments due to each release of radioactive liquid to an **UNRESTRICTED AREA** shall be calculated in accordance with the following equation:

DOSE TO THE TOTAL BODY OR ORGAN, o, FROM A LIQUID RELEASE (RIGOROUS EQUATION)

$$D_o = \{ V_u / [(60)(F_d)] \} \Sigma \{ (A_{iu})(C_{io}) \} \qquad \textbf{Eq. 9L}^{1}$$

Where,

F_d = actual dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \qquad \textbf{Eq. 2L}$$

F_{cw} = actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = actual salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 2.

V_u = volume of undiluted radwaste (gallons)

---

1 Equation 9L has been derived from NUREG-0133, 4.3.

A_{iu} = specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

C_{io} = liquid release dose factor for nuclide, i, and organ, o (mrem/hr per microcurie/ml)

The liquid release dose factors for principal gamma and beta emitters are listed in Attachment 6.

The liquid release dose factors for principal gamma and beta emitters were obtained in accordance with the methodology of NUREG-0133, section 4.3.1 (for salt water sites).

60 = a constant, the number of minutes per hour

(3) In the event a computer system is unavailable, a simplified equation may be used to calculate the total body and organ dose commitments due to individual liquid releases.

(4) The simplified method is presented below.

d) Simplified methods

(1) Whenever a computer system is unavailable to perform the rigorous total body dose calculations described in the previous section, the total body dose commitments--due to each release of radioactive liquid to an **UNRESTRICTED AREA**--may be calculated in accordance with equation 10L.

(2) Whenever a computer system is unavailable to perform the rigorous organ dose calculations described in the previous section, the dose commitments--to the maximum exposed organ, due to each release of radioactive liquid to an **UNRESTRICTED AREA**--may be calculated in accordance with equation 11L.

TOTAL BODY DOSE FROM A LIQUID RELEASE (SIMPLIFIED EQUATION)

$$D_{tb} = \{ [(58.6)(V_u)] / [(K_{sf})(F_d)] \} \Sigma A_{iu} \qquad \textbf{Eq. 10L}^1$$

D_{tb} = dose commitment, to the total body, due to each release of radioactive liquid to an **UNRESTRICTED AREA**

58.6 = a conversion constant which includes:

1) 1.33E4 mrem/hr per microcurie/ml (the total body dose factor for Cs-134)
2) 1.000 hr/60.00 min
3) 1.000 gallons/3.785 liters

V_u = volume of undiluted radwaste (liters)

---

1 Equations 10L has been derived from NUREG-0133, 4.3.

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP total body dose limit to the total body dose limit of Control 3.11.1.2, (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the total body dose is always less than or equal to the total body dose limit of Control 3.11.1.2.

A safety factor of 1.00 will yield a total body dose which corresponds to the total body dose limit of Control 3.11.1.2.

A safety factor of 0.500 will yield an total body dose which corresponds to one-half the total body dose limit of Control 3.11.1.2.

It is recommended that a safety factor of 1.0 be used for calculating the total body dose, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive liquid release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.1.2 when simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 10L to account for any potential nonconservatism associated with applying the Cs-134 total body dose conversion factor to all radionuclides identified in the liquid release. Such nonconservatism could conceivable be present whenever radionuclides having a dose conversion factor greater than that of Cs-134 are present in a liquid release.

F_d = actual dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \qquad \text{Eq. 2L}$$

F_{cw} = actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = actual salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available, a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 2.

A_{iu} = specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

DOSE TO THE ORGAN, o, FROM A LIQUID RELEASE (SIMPLIFIED EQUATION)

$$D_o = \{ [(449)(V_u)] / [(K_{sf})(F_d)] \} \Sigma A_{iu} \qquad \text{Eq. 11L}^{1}$$

D_o = the dose commitment to the maximum exposed ORGAN due to each release of radioactive liquid waste to an **UNRESTRICTED AREA**

449 = a conversion constant which includes:

1) 1.02E5 mrem/hr per microcurie/ml (the dose factor for I-131 to the thyroid)
2) 1.000 hr/60.00 min
3) 1.000 gallons/3.785 liters

V_u = volume of undiluted radwaste (liters)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP organ dose limit to the organ dose limit of Control 3.11.1.2, (unitless)

---

1 Equations 11L has been derived from NUREG-0133, 4.3.

The safety factor chosen shall be less than or equal to 1.00. This ensures the organ dose is always less than or equal to the organ dose limit of Control 3.11.1.2.

A safety factor of 1.00 will yield a organ dose which corresponds to the organ dose limit of Control 3.11.1.2.

A safety factor of 0.500 will yield an organ dose which corresponds to one-half the organ dose limit of Control 3.11.1.2.

It is recommended that a safety factor of 1.0 be used for calculating the organ dose, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive liquid release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.1.2 when simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 11L to account for any potential nonconservatism associated with applying the I-131 thyroid dose conversion factor to all radionuclides identified in the liquid release. Such nonconservatism could conceivable be present whenever radionuclides having a dose conversion factor greater than that of I-131 are present in a liquid release.

F_d = dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \qquad \textbf{Eq. 2L}$$

F_{cw} = actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available, a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 2.

A_{iu} = specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

---

7. Corrective **ACTIONS**

   a) CHEMISTRY SECTION surveillance procedures shall contain and/or reference administrative and/or Control limits for cumulative total body doses or cumulative organ doses resulting from liquid effluents and shall specify corrective actions to be initiated when these limits are exceeded.

   b) Refer to Control 3.11.1.2 for actions to be taken in the event the calculated cumulative total body dose exceeds 3 mrem per calendar quarter or 6 mrem per calendar year.

   c) Refer to Control 3.11.1.2 for actions to be taken in the event the calculated cumulative organ dose--for any organ--exceeds 10 mrem per calendar quarter or 20 mrem per calendar year.

LIMITS FOR THE LIQUID WASTE PROCESSING SYSTEM

1. Introduction

   a) 10 CFR 50.36a requires licensees to maintain and use the equipment installed in the liquid waste processing system for the purpose of controlling effluents to the environment.

   b) Radiological effluent controls have been established to implement the requirements of 10 CFR 50.36a.

   c) Different documents (e.g., Technical Specifications or NUREG-0472) may reference this effluent control using a variety of synonyms. Examples of some of these synonyms include terms such as the PROJECTED DOSE, the 92-DAY PROJECTED DOSE, the 92-DAY CUMULATIVE DOSE, the DOSE IN A 92-DAY PERIOD, or the DOSE FOR THE PREVIOUS 92 DAYS. The ODCM methodology for calculating this dose – by whatever name – is based on a simple sum of the dose contributions for the previous 92 days. As result, all the terms listed above are considered as equivalent quantities for purposes of this effluent control at CCNPP. For simplicity, the ODCM generally uses the phrase “dose for the previous 92 days” when referencing this effluent control.

d) These radiological effluent controls are described below.

2. Radiological Effluent Controls

a) The liquid radwaste processing system shall be used to reduce the quantity of radioactive materials in liquid waste released to the environment whenever the total body dose to **MEMBERS OF THE PUBLIC**, from liquid waste discharged to **UNRESTRICTED AREAS**, is greater than 0.36 mrem for the previous 92 days (per Control 3.11.1.3).

b) The liquid radwaste processing system shall be used to reduce the quantity of radioactive materials in liquid waste released to the environment whenever the organ doses to **MEMBERS OF THE PUBLIC**, from liquid waste discharged to **UNRESTRICTED AREAS**, is greater than 1.20 mrem for the previous 92 days (per Control 3.11.1.3).

c) The routine surveillances which are performed to verify compliance with this radiological effluent controls are described below.

3. Surveillance Requirement(s)

a) The previous 92-day, cumulative, total body dose--to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--shall be calculated, as described in the ODCM, at least once per 31 days (per Control 4.11.1.3).

b) The previous 92-day, cumulative, organ doses--to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--shall be calculated, as described in the ODCM, at least once per 31 days (per Control 4.11.1.3).

c) The CCNPP organization(s) responsible for performing these surveillances are identified in the next section.

4. Responsible Plant Organization(s)

a) The CHEMISTRY SECTION is responsible for performing the surveillances, listed above, whenever the appropriate initiating conditions, listed below, are present.

5. Initiating Conditions

a) Perform the surveillances, listed above, at least once per 31 days (per Control 4.11.1.3).

b) For **BATCH RELEASES**, perform the surveillances, listed above, prior to each **BATCH RELEASE** of radioactive liquid waste in accordance with approved CHEMISTRY SECTION procedures.

c) For **CONTINUOUS RELEASES**, perform the surveillances, listed above, at least monthly (until the **CONTINUOUS RELEASE** has been terminated) and, if activity is identified[1] in the CONTINUOUS waste stream, the surveillances may be performed more often as specified in approved CHEMISTRY SECTION procedures.

d) Whenever the correct initiating conditions are present, the calculations required by the above mentioned surveillance(s) shall be completed in accordance with the methodology contained in the following section.

6. Calculation Methodology

a) The cumulative total body dose and the cumulative organ doses for the previous 92 days shall be calculated in accordance with equation 8L found in the section, "Limits On Cumulative Total Body Doses And Cumulative Organ Doses For Liquid Effluents", found elsewhere in the ODCM.

7. Corrective **ACTIONS**

a) CHEMISTRY SECTION surveillance procedures shall contain/and or reference administrative and/or Control limits for cumulative total body doses or cumulative organ doses resulting from liquid effluents and shall specify corrective **ACTIONS** to be initiated when these limits are exceeded.

b) Refer to Control 3.11.1.3 for **ACTIONS** to be taken in the event the calculated cumulative total body dose exceeds 0.36 mrem for the previous 92 days.

c) Refer to Control 3.11.1.3 for **ACTIONS** to be taken in the event the calculated cumulative organ dose--for any organ--exceeds 1.2 mrem for the previous 92 days.

[1] The LOWER LIMITS OF DETECTION shall conform to Control 4.11.1.1.1, Table 4.11-1.

RADIOACTIVE GASEOUS EFFLUENTS

RELEASE PATHWAYS [B527]

1. Introduction

   a) Radioactive gaseous waste generated from operation of CCNPP may be released to the atmosphere.

   b) By design (i.e., in the absence of primary-to-secondary leaks), there are 2 pathways by which waste gas from the site may be discharged to the atmosphere. These pathways are listed below. General information related to each of these potential release pathways is contained on Attachment 7.

      (1) Unit 1 main vent stack

      (2) Unit 2 main vent stack

   c) Depending on plant conditions, (e.g., primary-to-secondary leaks) a potential exists for the release of radioactive materials from other pathways. Examples of these pathways are listed below. General information related to each of these potential release pathways is contained on Attachment 8.

      (1) auxiliary boiler deaerator

      (2) steam generator atmospheric steam dump system

      (3) plant nitrogen system

      (4) turbine building ventilation exhaust

      (5) emergency air lock

      (6) plant compressed air

      (7) main steam line penetrations

      (8) containment equipment hatch

      (9) auxiliary feed pumps

      (10) gland steam exhaust condenser

   d) All of these pathways are described below.

2. Unit 1 Main Vent Stack

   a) Dilution air and radioactive gaseous waste are discharged to the atmosphere through the Unit 1 main vent stack.

   b) The radioactive gaseous waste is mixed with and diluted by the outside air and building exhausts prior to exiting the Unit 1 main vent stack.

c) The Unit 1 main vent stack is secured to the Unit 1 reactor containment building.

d) The top of the Unit 1 main vent stack is at elevation 203.5 feet (mean sea level, MSL), and as such is 10.1 feet above the top of the reactor containment building dome. As a result, the Unit 1 main vent stack does not qualify as a "free-standing" stack greater than 80 meters tall[1].

e) The Unit 1 main vent stack is designed to accept gaseous radioactive waste from various sources. Sources which may contribute radioactive material to the Unit 1 main vent stack are tabulated in Attachment 7.

3. Unit 2 Main Vent Stack

a) Dilution air and radioactive gaseous waste are discharged to the atmosphere through the Unit 2 main vent stack.

b) *The Unit 2 main vent stack is designed to accept radioactive gaseous waste from* various sources.

c) The radioactive gaseous waste is mixed with and diluted by the outside air and building exhausts prior to exiting the Unit 2 main vent stack.

d) The Unit 2 main vent stack is secured to the Unit 2 reactor containment building.

e) The top of the Unit 2 main vent stack is at elevation 203.5 feet (MSL), and as such is 10.1 feet above the top of the reactor containment building dome. As a result, the Unit 2 main vent stack does not qualify as a "free-standing" stack greater than 80 meters tall.[1]

f) The Unit 2 main vent stack is designed to accept gaseous radioactive waste from various sources. Sources which may contribute radioactive material to the Unit 2 main vent stack are tabulated in Attachment 7.

4. Auxiliary Boiler Deaerator

a) Radioactive gases may be vented from the auxiliary boiler deaerator during periods of primary to secondary leakage.

b) Steam from the Moisture Separator Reheater (MSR) may be used in the deaerator. In the event of a primary to secondary leak, the MSR steam could become contaminated. Therefore, a potential exists for the release of radioactive gases in steam discharged from the auxiliary boiler deaerator.

c) The discharge of steam is accomplished via a relief vent, 0-VBV-1891, which allows excess pressure to be vented to atmosphere.

d) In the event the auxiliary boiler deaerator were to become contaminated, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known:

(1) the MSR steam activity obtained from a sample,

---

1 As defined by Regulatory Guide 1.111

(2) the duration of the discharge,

(3) the estimated steam discharge flow rate, and

(4) the measured or average annual meteorological conditions.

e) In accordance with applicable safety evaluations[1], continued operation of this system is allowed as long as the concentration of radionuclides in the auxiliary boiler steam drum is less than 96 MPCs.

5. Steam Generator Atmospheric Steam Dump System

a) Radioactive gases are not normally vented from this pathway.

b) Radioactive gases may be vented from the steam generator atmospheric steam dump system during periods of primary to secondary leakage.

c) If a primary to secondary leak is present and the steam dump valves are opened, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known (per UFSAR, 10.1.2.2):

(1) the specific activity of a main steam sample as determined by **GAMMA ISOTOPIC ANALYSIS**,

(2) the duration of discharge,

(3) the estimated steam discharge flow rate, and

(4) the measured or average annual meteorological conditions.

d) The total capacity of the atmospheric steam dump valve is 5 percent of steam flow with the reactor at full power (per UFSAR, 10.1.2.2).

6. Plant Nitrogen System

a) Radioactive gases are not normally vented from this pathway.

b) Nitrogen is supplied to various components which contain radioactive materials (e.g., VCT).

c) In the event the plant nitrogen system were to become contaminated, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known:

(1) the specific activity of the gas in the plant nitrogen system as determined by **GAMMA ISOTOPIC ANALYSIS**,

(2) the pressure of the nitrogen system,

(3) the volume of the nitrogen system, and

---

[1] See 50.59 Log No. 90-0-027-037-R1.

(4) the measured or average annual meteorological conditions.

d) It should be noted that the amount of radioactivity released could be estimated based on knowledge of other related parameters.

e) In accordance with applicable safety evaluations[1], continued operation of this system is allowed as long as the concentration of radionuclides is less than 13,400 MPCs.

7. Turbine Building Exhaust

a) Radioactive gases are not normally vented from this pathway.

b) In the event radioactive gases were to be released through the turbine building exhaust, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known:

(1) the specific activity of the turbine building air,

(2) the duration of the discharge,

(3) the estimated flow rate during the discharge, and

(4) the measured or average annual meteorological conditions.

8. Emergency Air Lock

a) Radioactive gases are not normally vented from this pathway.

b) In the event radioactive gases were to be released through the emergency air lock, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known:

(1) the containment air activity obtained from a sample,

(2) the volume of the air lock (9.558 cubic meters),

(3) the measured or average annual meteorological conditions.

9. Plant Compressed Air

a) Radioactive gases are not normally vented from this pathway.

b) In the event the plant compressed air system were to become contaminated, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known:

(1) the specific activity of the compressed air system,

(2) the pressure of the compressed air system,

(3) the volume of the compressed air system, and

---

1 See 50.59 Log No. 90-0-074-011-R1.

(4) the measured or average annual meteorological conditions.

10. Main Steam Line Penetrations

a) Radioactive gases are not normally vented from this pathway.

b) This penetration is cooled by outside air.

c) Gases may be released to the atmosphere through safety vents to the roof at elevation 91.5 feet.

d) See UFSAR 9.8.2.3.

11. Steam Driven Auxiliary Feed Pumps

a) Radioactive gases are not normally vented from this pathway.

b) In the event radioactive gases were to be released through the auxiliary feed pumps, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known:

(1) the activity in the steam,

(2) the volume of steam released.

12. Containment Equipment Hatch

a) Radioactive gases are not normally vented from this pathway.

b) In the event radioactive gases were to be released through the containment equipment hatch, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known:

(1) the containment air activity obtained from a sample,

(2) the volume of the air released,

(3) the measured or average annual meteorological conditions.

13. Gland Steam Exhaust Condenser

a) Radioactive gases are not normally vented from this pathway.

b) In the event radioactive gases were to be released through the gland steam exhaust condenser, the amount of radioactivity released and the resulting doses/dose rates at the **SITE BOUNDARY** can be estimated if the following parameters are known:

(1) activity in the gland steam exhaust condenser,

(2) the flow rate through the gland steam exhaust condenser,

(3) the duration of the leak, and

(4) the measured or average annual meteorological conditions.

14. Other unmonitored release paths should be evaluated and added to the ODCM as necessary.

TYPES OF GASEOUS RELEASES

1. All gaseous radwaste releases are classified as either **BATCH RELEASES** or **CONTINUOUS RELEASES**.

2. The definition of **BATCH RELEASE** is included in the definitions section of the ODCM.

3. The definition of **CONTINUOUS RELEASE** is included in the definitions section of the ODCM.

4. Gaseous radwaste discharges have been classified as CONTINUOUS or BATCH as shown on Attachments 7 and 8.

PROCESSING EQUIPMENT

1. Simplified Flow Diagram

   a) An overview of the gaseous waste processing system, including major equipment and (normal) flow paths, is outlined on Attachment 9.

2. Modifications

   a) Licensed initiated major changes to the gaseous waste processing system shall be reported to the Commission in the Radioactive Effluent Release Report for the period in which the modification to the waste system was completed (per Technical Specification 5.6.3). The discussion of each change shall contain:

      (1) A description of the equipment, components and processes involved; and

      (2) Documentation of the fact that the change, including the safety analysis, was reviewed and found acceptable by the onsite review function.

   b) A "major" change or modification includes, but is not limited to, the removal or permanent bypass of any of the following:

      (1) waste gas decay tank

      (2) waste gas surge tank

      (3) degassifier

      (4) HEPA filter

      (5) charcoal filter

3. Detailed Description

   a) A detailed description of the gaseous waste processing system is beyond the scope of the ODCM.

b) For more information on the Waste Gas System, see the CCNPP System Description Number 14A, "Waste Gas System."

c) For more information on the Waste Gas System, see the CCNPP Updated Final Safety Analysis Report, Chapter 11, "Waste Processing And Radiation Protection."

GASEOUS EFFLUENT RADIATION MONITORS AND SETPOINTS

1. Wide Range Gas Monitor (1-RE-5416)

a) General description

(1) The Wide Range Gas Monitor (WRGM) contains 3 radiation elements

(a) low-range noble gas detector

i) Designation of radiation element: 1-RE-5416

ii) type of radiation element: Off-line scintillation

iii) output: digital

iv) Radiation indicator: 1-RIC-5415

v) units for radiation indicator are user programmable and are normally set to microcuries per cubic centimeter or microcuries per second

vi) supplier: Sorrento Electronics (formerly General Atomics)

(b) mid-range, noble gas detector

i) Designation of radiation element: 1-RE-5417

ii) type of radiation element: Solid state

iii) This noble gas monitor is used to measure the release of radioactivity from unit 1 main vent in the event of an accident. (UFSAR, 11.2.3.2.12)

iv) setpoints for the mid-range detector will not be addressed in the ODCM

(c) high-range, noble gas detector

i) Designation of radiation element: 1-RE-5418

ii) type of radiation element: Solid state

iii) This detector is used to measure the release of radioactivity from unit 1 main vent in the event of an accident. (UFSAR, 11.2.3.2.12)

iv) setpoints for the high-range detector will not be addressed in the ODCM

(2) The low range detector will be the only detector addressed further in the ODCM.

b) Functions of 1-RE-5416

(1) continuously measure the release rate of noble gases emanating from the Unit 1 main vent stack (Control 4.11.2.1.1 or 4.11.2.1.2, Table 4.11-2)

(2) continuously indicate (via 1-RIC-5415) the release rate of noble gases emanating from the Unit 1 main vent stack (Control 4.11.2.1.1 or Control 4.11.2.1.2, Table 4.11-2)

(3) alarm (via 1-RIC-5415) prior to exceeding the site-boundary, noble-gas, total-body-dose-rate limit of 500 mr/yr (per Control 3.11.2.1.a)

(4) alarm (via 1-RIC-5415) prior to exceeding the site-boundary, noble-gas, skin-dose-rate limit of 3000 mr/yr (per Control 3.11.2.1.a)

c) **OPERABILITY** of 1-RE-5416

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s).

(2) The functions of this monitor are listed in section (b) above.

d) Monitors equivalent to 1-RE-5416

(1) 1-RE-5415 [the "Westinghouse Plant Vent Stack Monitor"] has the capability of providing the measurement and alarm functions of 1-RE-5416 during times when 1-RE-5416 is declared inoperable.

(2) 1-RE-5415 provides redundant monitoring [for 1-RE-5416] at the low end of the concentration ranges (UFSAR 11.2.3.2.12).

(3) In the event 1-RE-5415 is inoperable or otherwise unavailable, 1-RE-5416 may fulfill the measuring, indicating, and alarming functions normally provided by 1-RE-5415.

e) Radiological effluent controls for 1-RE-5416

(1) Control 3.3.3.9 states that releases via the plant vent stack may continue if any one of the following three conditions are satisfied:

(a) 1-RE-5415 is operable AND the alarm setpoint for 1-RI-5415 is set to ensure the annual dose rates due to noble gases at the **SITE BOUNDARY** are less than 500 mr/yr to the total body and are less than 3000 mr/yr to the skin (per Control 3.11.2.1.a), or

(b) an "equivalent monitor" is operable AND the alarm trip setpoint for the "equivalent monitor" is set to ensure annual dose rates due to noble gases at the **SITE BOUNDARY** are less than 500 mr/yr to the total body and are less than 3000 mr/yr to the skin (per Control 3.11.2.1.a), or

(c) grab samples are obtained and analyzed for gross activity at least once per 24 hours in accordance with Controls 3.11.2.1.a, 4.11.2.1.1, and 4.11.2.1.2 (per Control 4.3.3.9, Table 3.3-12, **ACTION** 37).

f) Surveillances for 1-RE-5416

(1) Control 4.3.3.9 requires demonstrating the **OPERABILITY** of 1-RE-5416 by satisfying the checks, calibrations, and tests listed below:

(a) **CHANNEL CHECK** within the past 24 hours

(b) **SOURCE CHECK** within the past 31 days

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

g) Setpoints for 1-RIC-5415

(1) Requirements and commitments

(a) The alarm and fixed setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9)

(b) The method for calculating fixed or adjustable setpoints shall be provided in the ODCM (per NUREG-0133, 5.1.1).

(2) There are four radiation alarm setpoints associated with, or otherwise related to, the WRGM.

(a) 1-RIC-5415 fixed high-high radiation alarm setpoint

(b) 1-RIC-5415 fixed high radiation alarm setpoint

(c) 1-RIC-5415 adjustable plant computer high radiation alarm setpoint

(d) 1-RIC-5415 adjustable plant computer alert setpoint.

(3) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows

(a) The 1-RIC-5415 fixed high-high radiation alarm setpoint will be referred to as the fixed high alarm setpoint

(b) The 1-RIC-5415 fixed high radiation alarm setpoint will be referred to as the fixed alert setpoint

(c) The 1-RIC-5415 adjustable plant computer high radiation alarm setpoint will be referred to as the adjustable setpoint

(d) The 1-RIC-5415 adjustable plant computer alert setpoint will be referred to as the alert setpoint.

(4) Each of these alarm setpoints are described below.

h) Fixed high alarm setpoint for 1-RIC-5415

(1) General information

(a) The fixed high alarm setpoint is not adjusted for each release.

(b) Whenever the fixed high alarm setpoint is exceeded, an alarm will be generated.

(c) The current value for the fixed high alarm setpoint is specified in the CCNPP Alarm Manual.

(d) The CCNPP Alarm Manual[1] refers to the fixed high alarm setpoint as the Unit 1 Wide Range Noble Gas Radiation Monitor high alarm setpoint.

(e) The fixed high alarm setpoint is integral to the WRGM, as purchased from the supplier.

(f) The fixed high alarm setpoint is administratively controlled by EN-1-100.

(g) The fixed high alarm setpoint shall be calculated as described below.

(2) Calculating the fixed high alarm setpoint for 1-RIC-5415

(a) The fixed high alarm setpoint for 1-RIC-5415 (WRGM) shall be calculated in accordance with equation 1G.[2]

FIXED HIGH ALARM SETPOINT FOR 1-RIC-5415

$$S_{fixhh} \leq [c' / (x/Q)] [F_{dx} / (F_{d1} + F_{d2})] [A_{TLn}] \qquad \text{Eq. 1G}^3$$

Where,

S_{fixhh} = the fixed high alarm setpoint for 1-RIC-5415 (microcuries per second)

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

[1] The CCNPP Alarm Manual is controlled by NO-1-201.
[2] The alarm and trip setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9).
[3] Equation 1G has been derived from NUREG-0133, 5.2.1.

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

Unit 1 main vent releases are considered "long-term" releases[1], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the setpoint calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases)

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

F_{dx} = the estimated main vent stack (diluted gaseous radwaste) flow rate for unit, x, (cubic meters per second)

The estimated main vent stack flow rates for Unit 1 and Unit 2 are defined below.

F_{d1} = the estimated main vent stack flow rate for Unit 1 (cubic meters per second)

Since the main vent stack flow rate will vary depending on the configuration of air dampers and the input gas streams, nominal main vent stack flow rate is used to calculate the fixed high alarm setpoint.

Use the nominal Unit 1 main vent stack flow rate listed on Attachment 7.

The main vent stack flow rate shall be determined, in accordance with approved procedures, at least once per 6 months ($\pm$ 25%). The Test and Equipment Unit shall be responsible for performing this test. The results of the main vent flow rate test shall be evaluated to ensure the main vent flow rates used in the ODCM are an accurate reflection of the true main vent flow rates. The Radiological Effluent Technical Specifications (RETS) Program Manager is responsible for modifying the (main vent flow rates used in the) ODCM in the event the main vent flow rate for either Unit 1 or Unit 2 has increased to a value which is greater than the maximum discharge flow rates listed on Attachment 7.

F_{d2} = the estimated main vent stack (diluted gaseous radwaste) flow rate for unit 2 (cubic meters per second)

Since the main vent stack flow rate will vary depending on the configuration of air dampers and the input gas streams, nominal main vent stack flow rate is used to calculate the fixed high alarm setpoint.

Use the nominal Unit 2 main vent stack flow rate listed on Attachment 7.

The main vent stack flow rate shall be determined, in accordance with approved procedures, at least once per 6 months ($\pm$ 25%). The Test and Equipment Unit shall be responsible for performing this test. The results of the main vent flow rate test shall be evaluated to ensure the main vent flow rates used in the ODCM are an accurate reflection of the true main vent flow rates. The RETS Program Manager is responsible for modifying the (main vent flow rates used in the) ODCM in the event the main vent flow rate for either Unit 1 or Unit 2 has increased to a value which is greater than the maximum discharge flow rates listed on Attachment 7.

---

1 NUREG-0133, 3.3

A_{TLn} = the sum of the total specific activities of all radionuclides found in TYPICAL GASEOUS RADWASTE RELEASES (microcuries/cm^3)

Calculate A_{TLn} in accordance with equation 2G.

SPECIFIC ACTIVITY CORRESPONDING TO THE **SITE BOUNDARY** LIMIT

$$\Sigma [(f_i) (A_{TLn})] / A_{iLt} \leq L_{MPC}$$ **Eq. 2G**[1]

Where,

f_i = a fraction which represents the relative activity contribution of noble gas radionuclide i to the total noble gas activity for TYPICAL GASEOUS EFFLUENTS (unitless)

This value may be obtained using the guidance provided on Attachment 5.

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 1 (microcuries/cm^3)

For all the radionuclides found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 1. An acceptable alternative is to assume an isotopic mix which results in a more conservative setpoint.

L_{MPC} = the site MPC limit (MPCs) for **UNRESTRICTED AREAS**

The value chosen for L_{MPC} in this equation is 2. The basis for this limit is 10 CFR 50.72.

It has been shown[2] that, for the radionuclides present in TYPICAL GASEOUS EFFLUENTS from CCNPP, the 2 MPC limit is more restrictive than the limits of Control 3.11.2.1(a).

It should be noted that by using "2" as the MPC limit (10 CFR 50.72), instead of using the limits of Control 3.11.2.1(a), a safety factor has been incorporated into equation 2G.

An alarm setpoint corresponding to 2 MPCs serves to initiate a determination of whether the "4-hour NRC notification" (specified in 10 CFR 50.72) is required.

(3) Documenting the fixed high alarm setpoint

(a) Whenever the fixed high alarm setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with EN-1-100.

1 This equation has been derived from 10 CFR 20, Appendix B, Table II, Note 1.

2 Addendum To Setpoint Calculations For WRGM Monitors 1-RIC-5415 and 2-RIC-5415, R.L. Conatser, December 10, 1991.

(4) Changing the fixed high alarm setpoint for 1-RIC-5415

(a) If the fixed high alarm setpoint calculated in accordance with equation 1G exceeds the maximum range of the monitor, the fixed high setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) The fixed high alarm setpoint may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in the setpoint calculation equations (identified in this section of the ODCM) have changed.

(d) The fixed high alarm setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL GASEOUS EFFLUENTS has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the historical maximum annual average atmospheric dispersion factor has changed,

iii) the MPC limit at the **SITE BOUNDARY**, (L_{MPC}) has changed,

iv) the Unit 1 or Unit 2 main vent stack flow rate has changed by greater than or equal to 10%[2],

v) the values listed in 10 CFR 20, Table II, column 1 have changed,

vi) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

vii) the monitor is not conservative in its function (see "Functions of 1-RE-5416" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving fixed high alarm setpoint.

---

1 As determined in accordance with Attachment 5.

2 As determined by surveillance test results (e.g., STP-M-462-1 or STP-M-462-2).

i) Fixed alert setpoint for 1-RIC-5415

(1) General information

(a) The fixed alert setpoint is not adjusted for each release.

(b) Whenever the fixed alert setpoint is exceeded, an alarm will be generated.

(c) The CCNPP Alarm Manual does not reference this setpoint.

(d) The fixed alert setpoint is integral to the WRGM, as purchased from the supplier.

(e) The current value for the fixed alert setpoint is specified in the CCNPP Setpoint File.

(f) The fixed alert setpoint is administratively controlled by EN-1-100.

(g) The fixed alert setpoint shall be calculated as described below[1].

(2) Calculating the fixed alert setpoint for 1-RIC-5415

(a) The fixed alert setpoint for 1-RIC-5415 shall be calculated as described below:

FIXED ALERT SETPOINT FOR 1-RIC-5415

$$S_{fixh} \leq K_{sf} [S_{fixhh}] \qquad \textbf{Eq. 3G}$$

Where,

S_{fixh} = the fixed alert setpoint for 1-RIC-5415 (microcuries per second)

S_{fixhh} = the fixed high alarm setpoint for 1-RIC-5415 (microcuries per second)

K_{sf} = a constant, actually a safety factor, which is the fraction of the fixed high setpoint (unitless).

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed alert setpoint is always less than or equal to the fixed high alarm setpoint.

A safety factor of 1.00 will yield a fixed alert setpoint which corresponds to the fixed high alarm setpoint.

A safety factor of 0.100 will yield a fixed alert setpoint which corresponds to one-tenth the fixed high alarm setpoint.

---

1 The alarm and trip setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9).

It is recommended that a safety factor of 0.1 be used for calculating the fixed alert setpoint, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a value less than 1.00 does provide plant personnel with adequate time to respond to changing plant conditions and to initiate corrective **ACTIONS** so as to minimize the possibility of violating either the 10 CFR 50.72 limit or the Control 3.3.3.9 limits.

The use of the safety factor is consistent with ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133 which states that "... the alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous gaseous release limit of Control 3.11.2.1(a) will not be exceeded." (per NUREG-0133, 5.1.1).

This safety margin will prevent minor fluctuations in the nominal plant vent stack flow rates, errors in monitor efficiencies, and other statistical aberrations from adversely impacting the calculated fixed alert setpoint.

---

(3) Documenting the fixed alert setpoint

(a) Whenever the fixed alert setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with EN-1-100.

(4) Changing the fixed alert setpoint for 1-RIC-5415

(a) A setpoint change should be initiated whenever any of the parameters identified in equation 3G have changed.

(b) The fixed alert setpoint should be changed whenever the fixed high setpoint is changed.

(c) The fixed alert setpoint should be changed if the value of the safety factor is changed.

(d) See EN-1-100 for a description of activities associated with setpoint changes and setpoint approvals.

j) Adjustable alarm setpoint for 1-RIC-5415

(1) General information

(a) Whenever this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9), the adjustable setpoint is calculated and adjusted prior to each release of a WGDT, each containment vent, and each containment purge discharged via the main vent.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in either the WGDT or the containment building, whichever is applicable. (The radionuclide concentrations are determined by radiochemical analysis in accordance with applicable CHEMISTRY SECTION procedures as required by Control 4.11.2.1.2).

(c) Whenever the adjustable setpoint is exceeded, the WGDT, PURGE, or vent discharge via the main vent will be manually suspended.

(d) Refer to the Alarm Manual for a full list of operator ACTIONS taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected when discharging a WGDT, a containment vent, or a containment purge via the main vent. For containment purges during outages, system evolutions may cause containment atmosphere activity to increase above what is normally expected for short periods of time.

(f) The value for the adjustable setpoint is recorded on the gaseous release permit in accordance with applicable CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the main vent radiation monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 1/2-RIC-5415, and provides an alarm to plant operators when the 1/2-RIC-5415 adjustable setpoint has been exceeded.

(i) When this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9), a value for the adjustable alarm setpoint shall be calculated prior to each release of a WGDT, each containment vent, and each containment purge as shown below.

(2) Calculating the adjustable setpoint for 1/2-RIC-5415

(a) The adjustable setpoint is based on the specific activity of the radionuclides in the undiluted gaseous waste (as determined by radiochemical analysis per Control 4.11.2.1.2), and the alarm setpoint is calculated as shown below.

ADJUSTABLE SETPOINT FOR 1/2-RIC-5415

$\mathbf{S_{adj} \leq (K_{sf})\ [\ (c'\ F_u \sum A_{iu}\ e_i\ ) + Bkg\ ]}$	**Eq. 29G**[1]

S_{adj} = the adjustable setpoint for 1/2-RIC-5415 (microcuries per second)

K_{sf} = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, (5) errors associated with monitor calibrations[2], and (6) anticipated short term variations in containment activity (applicable to containment purges only).

It is recommended that a safety factor of 10 for containment purge releases be used for calculating the adjustable setpoint. However, other values for purge releases -- not to exceed 10 -- may be used as directed by the General Supervisor Chemistry. A safety factor of 1.5 shall be used for all other gaseous releases.

The particular value selected for the safety factor is somewhat arbitrary, however a value less than or equal to 10 does provide plant personnel with adequate time to respond to changing plant conditions and to initiate corrective **ACTIONS** so as to minimize the possibility of violating either the 10 CFR 50.72 limit or the Control 3.3.3.9 limits.

The use of the safety factor is consistent with ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133 which states that ". . . the alarm and trip setpoints . . . should correspond to a value(s) which represents a safe margin of assurance that the instantaneous gaseous release limit of Control 3.11.2.1(a) will not be exceeded."
(per NUREG-0133, 5.1.1).

---

1 Equation 29G has been derived from NUREG-0133, Addendum, page AA-1.

2 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

This safety margin will prevent minor fluctuations in the nominal plant vent stack flow rates, errors in monitor efficiencies, and other statistical aberrations from adversely impacting the calculated adjustable setpoint. Additionally for a special case of containment purges during outages, the safety factor allows for short term variations in activity created as a result of system evolutions in containment.

F_u = maximum undiluted radwaste flow rate (cubic meters per second)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 7.

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream, either the WGDT or the containment building (microcuries per cubic centimeter)

e_i = absolute detector efficiency for nuclide, i (microcuries Xe-133 equivalent per microcuries nuclide i)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (microcuries per second)

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

(3) Documenting the adjustable setpoint for 1/2-RIC-5415

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures (e.g., CP-604).

(4) Changing the adjustable setpoint for 1/2-RIC-5415

(a) In all cases, the adjustable setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(b) If the adjustable setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(c) CHEMISTRY SECTION procedures (e.g., CP-604) contain administrative controls associated with calculating and approving an adjustable setpoint.

(d) Whenever this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9) the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release of a WGDT, a containment vent, or a containment purge via the main vent.

k) Alert setpoint for 1-RIC-5415

(1) General information

(a) The alert setpoint is applicable to containment purges only.

(b) Whenever this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9), the alert setpoint is calculated and adjusted prior to each containment purge discharged via the main vent.

(c) The alert setpoint is based on the specific activities of the radionuclides present in the containment building. (The radionuclide concentrations are determined by radiochemical analysis in accordance with applicable CHEMISTRY SECTION procedures as required by Control 4.11.2.1.2).

(d) Whenever the alert setpoint is exceeded the PURGE via the main vent may continue.

(e) The alert setpoint corresponds to a level of activity which indicates additional source term(s) may be present, and as a result, additional notifications and/or actions are required to identify the source and to accurately account for the activity discharged.

(f) The value for the alert setpoint is recorded on the gaseous release permit in accordance with applicable CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the main vent radiation monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 1/2-RIC-5415, and provides an alarm to plant operators when the 1/2-RIC-5415 alert setpoint has been exceeded.

(i) When this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9), a value for the alert setpoint shall be calculated prior to each containment purge via the main vent as shown below.

(2) Calculating the alert setpoint for 1/2-RIC-5415

(a) The alert setpoint is based on the specific activity of the radionuclides in the undiluted gaseous waste (as determined by radiochemical analysis per Control 4.11.2.1.2), and the setpoint is calculated as shown below.

ALERT SETPOINT FOR 1/2-RIC-5415

$$S_{alert} \leq (1.50) \; [\, (c' \, F_u \, \Sigma \, A_{iu} \, e_i \,) + Bkg \,] \qquad \textbf{Eq. 29G}[1]$$

S_{alert} = the alert setpoint for 1/2-RIC-5415 (microcuries per second)

1.50 = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, and (5) errors associated with monitor calibrations[2].

---

1 Equation 29G has been derived from NUREG-0133, Addendum, page AA-1.

2 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

The use of the safety factor is consistent with ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133 which states that "... the alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous gaseous release limit of Control 3.11.2.1(a) will not be exceeded." (per NUREG-0133, 5.1.1).

This safety margin will prevent minor fluctuations in the nominal plant vent stack flow rates, errors in monitor efficiencies, and other statistical aberrations from adversely impacting the calculated alert setpoint.

F_u = maximum undiluted radwaste flow rate (cubic meters per second)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 7.

A_{iu} = specific activity of radionuclide, i, in the containment building (microcuries per cubic centimeter)

e_i = absolute detector efficiency for nuclide, i (microcuries Xe-133 equivalent per microcuries nuclide i)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (microcuries per second)

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

---

(3) Documenting the alert setpoint for 1/2-RIC-5415

(a) Whenever the alert setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures (e.g., CP-604).

(4) Changing the alert setpoint for 1/2-RIC-5415

(a) In all cases, the alert setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(b) If the alert setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(c) CHEMISTRY SECTION procedures (e.g., CP-604) contain administrative controls associated with calculating and approving an alert setpoint.

(d) Whenever this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9) the calculated value for the alert setpoint shall be entered into the plant computer prior to each containment purge via the main vent.

2. Wide Range Gas Monitor (2-RE-5416)

a) all information related to 1-RE-5416 is applicable to the Unit 2 WRGM with the following exceptions(s)

b) Monitors equivalent to 2-RE-5416

(1) 2-RE-5415 [the "Westinghouse Plant Vent Stack Monitor"] has the capability of providing the measurement and alarm functions of 2-RE-5416 during times when 2-RE-5416 is declared inoperable

(2) 2-RE-5415 provides redundant monitoring [for 2-RE-5416] at the low end of the concentration ranges (UFSAR 11.2.3.2.12)

3. Westinghouse Plant Vent Stack Monitor (1-RE-5415)

a) The Westinghouse Plant Vent Stack Monitor contains 2 radiation elements

(1) 1-RE-5414

(a) particulate detector

(b) off-line scintillation detector

(c) analog output

(d) supplies signals to radiation indicator 1/2-RI-5414

(e) values displayed by 1/2-RI-5414 are in units of counts per minute

(f) the detector manufacturer is Westinghouse

(2) 1-RE-5415

(a) noble gas detector

(b) off-line GM Tube

(c) analog output

(d) supplies signals to radiation indicator 1/2-RI-5415

(e) values displayed by 1/2-RI-5415 are in units of counts per minute

(f) the detector manufacturer is Westinghouse

b) Functions of 1-RE-5414

(1) The functions of 1-RE-5414 are mentioned here only as a basis for excluding this radiation element from the setpoint controls of Control 3.3.3.9.

(2) This monitor (the particulate monitor) was retired in place.

c) Functions of 1-RE-5415[1]

(1) continuously measure the activity (cpm) of noble gases emanating from the Unit 1 main vent stack (Control 4.11.2.1.2, Table 4.11-2)

(2) continuously indicate (via 1-RI-5415) the activity (cpm) of noble gases emanating from the Unit 1 main vent stack (Control 4.11.2.1.2, Table 4.11-2)

(3) alarm (via 1-RI-5415) prior to exceeding the site-boundary, noble-gas, total-body-dose-rate limit of 500 mr/yr (per Control 3.11.2.1.a)

(4) alarm (via 1-RIC-5415) prior to exceeding the site-boundary, noble-gas, skin-dose-rate limit of 3000 mr/yr (per Control 3.11.2.1.a)

---

1 This (radiation element) monitors noble gases. Other radiation elements monitor particulates in this waste stream.

d) **OPERABILITY** of 1-RE-5415

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s).

(2) The functions of 1-RE-5415 are listed in section (c) above.

e) Monitors equivalent to 1-RE-5415

(1) The Wide Range Gas Monitor (i.e., 1-RE-5416) has the capability of providing the measurement and alarm functions of 1-RE-5415 during times when 1-RE-5415 is declared inoperable.

(2) 1-RE-5415 provides redundant monitoring [for 1-RE-5416] at the low end of the concentration ranges (UFSAR 11.2.3.2.12).

(3) In the event 1-RE-5415 is inoperable or otherwise unavailable, 1-RE-5416 may fulfill the measuring, indicating, and alarming functions normally provided by 1-RE-5415.

(4) The absence of a radiation element dedicated to measuring the particulate activity in the Wide Range Gas Monitor does not preclude the use of 1-RE-5416 as a backup for 1-RE-5415. This is mentioned only as a basis for excluding 1/2-RE-5414 from the setpoint controls of Control 3.3.3.9 (see "Functions of 1-RE-5414" earlier in this section).

f) Radiological effluent controls for 1-RE-5415

(1) Control 3.3.3.9 states that releases via the plant vent stack may continue if any one of the following three conditions are satisfied

(a) 1-RE-5415 is operable AND the alarm setpoint for 1-RI-5415 is set to ensure the annual dose rates due to noble gases at the **SITE BOUNDARY** are less than 500 mr/yr to the total body and are less than 3000 mr/yr to the skin (per Control 3.11.2.1.a), or

(b) an "equivalent monitor" (see section (e) above) is operable AND the alarm setpoint for the "equivalent monitor" is set to ensure annual dose rates due to noble gases at the **SITE BOUNDARY** are less than 500 mr/yr to the total body and are less than 3000 mr/yr to the skin (per Control 3.11.2.1.a), or

(c) grab samples are obtained and analyzed for gross activity at least once per 24 hours in accordance with Controls 3.11.2.1.a, 4.11.2.1.1, and 4.11.2.1.2 (per Control 4.3.3.9, Table 3.3-12, **ACTION** 37).

(2) Control 3.11.2.1.b (i.e., dose rates due to iodines and particulates at the **SITE BOUNDARY**) is not applicable to noble gas detector or to the setpoints related to the noble gas detector 1-RE-5415. As a result, the 1500 my/yr organ dose limit is not included as a radiological effluent control in this section of the ODCM.

g) Surveillances for 1-RE-5415

(1) Control 4.3.3.9 requires demonstrating the **OPERABILITY** of 1-RE-5415 by satisfying the checks, calibrations, and tests listed below:

(a) **CHANNEL CHECK** within the past 24 hours

(b) **SOURCE CHECK** within the past 31 days

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

h) Setpoints for 1-RI-5415

(1) Requirements and commitments

(a) The alarm and fixed setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9)

(b) The method for calculating fixed or adjustable setpoints shall be provided in the ODCM. (NUREG-0133, 5.1.1)

(2) There are four alarms associated with, or otherwise related to, 1-RE-5415.

(a) 1-RI-5415 fixed high radiation alarm setpoint

(b) 1-RI-5415 adjustable plant computer high radiation alarm setpoint

(c) 1-RI-5415 low radiation alarm setpoint

(d) 1-RI-5415 adjustable plant computer alert setpoint.

(3) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows

(a) The 1-RI-5415 fixed high radiation alarm setpoint will be referred to as the fixed setpoint.

(b) The 1-RI-5415 adjustable plant computer high radiation alarm setpoint will be referred to as the adjustable setpoint.

(c) The 1-RI-5415 low radiation alarm setpoint will be referred to as the low setpoint.

(d) The 1-RI-5415 adjustable plant computer alert setpoint will be referred to as the alert setpoint.

(4) Each of these alarm setpoints are described below.

i) The fixed setpoint for 1-RI-5415

(1) General information

(a) The fixed setpoint is not adjusted for each release.

(b) Whenever the fixed setpoint is exceeded, an alarm will be generated.

(c) The current value for the fixed setpoint is specified in the CCNPP Alarm Manual.

(d) The CCNPP Alarm Manual refers to this setpoint as the 1-RI-5415 High Alarm Setpoint.

(e) The fixed setpoint is integral to the Main Vent (Westinghouse) RMS as purchased from the supplier.

(f) The fixed setpoint is administratively controlled by EN-1-100.

(g) The fixed setpoint shall be calculated as described below[1].

(2) Calculating the fixed setpoint for 1-RI-5415

(a) The fixed setpoint for 1-RI-5415 (plant vent stack monitor) shall be calculated in accordance with equation 4G.

THE FIXED SETPOINT FOR 1-RI-5415

$$S_{fix} \leq \{ K_{sf} / [(x/Q) (F_{d1} + F_{d2})] \} \Sigma [(e_i) (A_{iLn})] \qquad \text{Eq. 4G}^2$$

S_{fix} = the fixed setpoint for 1-RI-5415 (counts per minute)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the MPC limit, L_{MPC}, used in equation 2G (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed setpoint is always less than or equal to the MPC limit, L_{MPC}, used in equation 2G.

---

1 The alarm and trip setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9).

2 Equation 4G has been derived from NUREG-0133, 5.2.1, (the 500 mr/yr equation).

A safety factor of 1.00 will yield a fixed setpoint which corresponds to the MPC limit, L_{MPC}, in equation 2G.

A safety factor of 0.500 will yield a fixed setpoint which corresponds to one-half the MPC limit, L_{MPC}, in equation 2G.

It is recommended that a safety factor of 1.0 be used for calculating the fixed setpoint, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with adequate time to respond to changing plant conditions and to initiate corrective **ACTIONS** so as to minimize the possibility of violating either the 10 CFR 50.72 limit or the Control 3.3.3.9 limits.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133, section 5.1.1, which states that "... the alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous gaseous release limit of Control 3.11.2.1(a) will not be exceeded."

This safety margin will prevent minor fluctuations in the nominal plant vent stack flow rates, errors in detector efficiencies, and other statistical aberrations from adversely impacting the calculated fixed setpoint.

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

Unit 1 and Unit 2 main vent releases are considered "long-term" releases[1], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the setpoint calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases).

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

---

1 NUREG-0133, 3.3

F_{d1} = the estimated main vent stack flow rate for Unit 1 (cubic meters per second)

Since the main vent stack flow rate will vary depending on the configuration of air dampers and the input gas streams, nominal main vent stack flow rate is used to calculate the fixed setpoint.

Use the nominal Unit 1 main vent stack flow rate listed on Attachment 7.

The main vent stack flow rate shall be determined, in accordance with approved procedures, at least once per 6 months ($\pm$ 25%). The Test and Equipment Unit shall be responsible for performing this test. The results of the main vent flow rate test shall be evaluated to ensure the main vent flow rates used in the ODCM are an accurate reflection of the true main vent flow rates. The RETS Program Manager is responsible for modifying the (main vent flow rates used in the) ODCM in the event the main vent flow rate for either Unit 1 or Unit 2 has increased to a value which is greater than the maximum discharge flow rates listed on Attachment 7.

F_{d2} = the estimated main vent stack flow rate for unit 2 (cubic meters per second)

Since the main vent stack flow rate will vary depending on the configuration of air dampers and the input gas streams, nominal main vent stack flow rate is used to calculate the fixed setpoint.

Use the nominal Unit 2 main vent stack flow rate listed on Attachment 7.

The main vent stack flow rate shall be determined, in accordance with approved procedures, at least once per 6 months ($\pm$ 25%). The Test and Equipment Unit shall be responsible for performing this test. The results of the main vent flow rate test shall be evaluated to ensure the main vent flow rates used in the ODCM are an accurate reflection of the true main vent flow rates. The RETS Program Manager is responsible for modifying the (main vent flow rates used in the) ODCM in the event the main vent flow rate for either Unit 1 or Unit 2 has increased to a value which is greater than the maximum discharge flow rates listed on Attachment 7.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

A_{iLn} = the specific activities of radionuclide, i, found in TYPICAL GASEOUS RADWASTE RELEASES (calculated in accordance with 10 CFR 20, Appendix B, Table II, Note 1 as described below; microcuries per milliliter)

Calculate A_{iLn} in accordance with equation 5G.

SPECIFIC ACTIVITY LIMIT FOR NUCLIDE I IN A RADIONUCLIDE MIXTURE

$$A_{iLn} = (f_i)(A_{TLn}) \qquad \text{Eq. 5G}$$

f_i = a fraction which represents the relative activity contribution of noble gas radionuclide i to the total noble gas activity for TYPICAL GASEOUS EFFLUENTS (unitless)

This value may be obtained using the guidance provided on Attachment 5.

A_{TLn} = the sum of the total specific activities of all noble gas radionuclides found in TYPICAL GASEOUS RADWASTE RELEASES (microcuries/cm^3)

Calculate A_{TLn} in accordance with equation 2G.

SPECIFIC ACTIVITY CORRESPONDING TO THE **SITE BOUNDARY** LIMIT

$$\Sigma [(f_i)(A_{TLn})] / A_{iLt} \leq L_{MPC} \qquad \text{Eq. 2G}^{1}$$

L_{MPC} = the MPC limit

The value chosen for L_{MPC} in this equation is 2. The basis for this limit is 10 CFR 50.72.

It has been shown[2] that, for the radionuclides present in TYPICAL GASEOUS EFFLUENTS from CCNPP, the 2 MPC limit is more restrictive than the limits of Control 3.3.3.9.

It should be noted that by using "2" as the MPC limit (10 CFR 50.72), instead of using the limits of Control 3.11.2.1(a), a safety factor has been incorporated into equation 2G.

The use of 2 MPCs as a safety margin is consistent with the provisions of NUREG-0133, section 5.1.1, which states that, "... in all cases, conservative assumptions may be necessary in establishing these setpoints to account for system variables, ... the variability in release flow, ... and the time lag between alarm and final isolation of radioactive effluents."

An alarm setpoint corresponding to 2 MPCs serves to initiate a determination of whether the "4-hour NRC notification" (specified in 10 CFR 50.72) is required.

The use of a limiting specific activity equivalent to 2 MPCs is consistent with the provisions of 10 CFR 20.

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 1 (microcuries/cm^3)

For all the radionuclides found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 1. An acceptable alternative is to assume an isotopic mix which results in a more conservative setpoint.

---

1 Equation 2G has been derived from 10 CFR 20, Appendix B, Table II, Note 1.

2 Addendum To Setpoint Calculations For WRGM Monitors 1-RIC-5415 and 2-RIC-5415, R.L. Conatser, December 10, 1991.

(3) The low setpoint for 1-RI-5415

(a) The ODCM does not address the calculations associated with the low setpoint.

(b) The low setpoint is specified in the CCNPP Alarm Manual.

(c) The low setpoint may be used to determine **OPERABILITY** of this monitor (in accordance with the provisions of Control 4.3.3.9, Table 4.3-11, Note 2).

(4) Adjusting the fixed setpoint for 1-RI-5415

(a) If the fixed setpoint calculated in accordance with equation 4G exceeds the maximum range of the monitor, the fixed setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) The fixed setpoint may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in equation 4G have changed.

(d) The fixed setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL GASEOUS EFFLUENTS has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the historical maximum annual average atmospheric dispersion factor has changed,

iii) the MPC limit at the **SITE BOUNDARY**, (presently 2 MPCs) has changed,

iv) the estimated Unit 1 main vent stack flow rate or Unit 2 main vent stack flow rate has changed by greater than or equal to 10%[2],

v) the values listed in 10 CFR 20, Table II, column 1 have changed,

vi) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

---

1 As determined in accordance with Attachment 5.

2 As determined by surveillance test results (e.g., STP-M-462-1, STP-M-462-2).

vii) the monitor is not conservative in its function (see section "Functions of 1/2-RE-5415" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving the fixed setpoint.

j) Adjustable setpoint for 1/2-RI-5415

(1) General information

(a) Whenever this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9), the adjustable setpoint is calculated and adjusted prior to each release of a WGDT, each containment vent, and each containment purge discharged via the main vent.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in either the WGDT or the containment building, whichever is applicable. (The radionuclide concentrations are determined by radiochemical analysis in accordance with applicable CHEMISTRY SECTION procedures as required by Control 4.11.2.1.2).

(c) Whenever the adjustable setpoint is exceeded, the WGDT, **PURGE**, or vent discharge via the main vent will be manually suspended.

(d) Refer to the Alarm Manual for a full list of operator **ACTIONS** taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected when discharging a WGDT, a containment vent, or a containment purge via the main vent. For containment purges during outages, system evolutions may cause containment atmosphere activity to increase above what is normally expected for short periods of time.

(f) The value for the adjustable setpoint is recorded on the gaseous release permit in accordance with applicable CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the main vent radiation monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 1/2-RI-5415, and provides an alarm to plant operators when the 1/2-RI-5415 adjustable setpoint has been exceeded.

(i) When this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9), a value for the adjustable setpoint shall be calculated prior to each release of a WGDT, each containment vent, and each containment purge as shown below.

(2) Calculating the adjustable setpoint for 1/2-RI-5415

(a) The adjustable alarm setpoint is based on the specific activity of the radionuclides in the undiluted gaseous waste (as determined by radiochemical analysis per Control 4.11.2.1.2), and the alarm setpoint is calculated as shown below.

ADJUSTABLE SETPOINT FOR 1/2-RI-5415

$$S_{adj} \leq (K_{sf}) [(F_u / F_{dx}) [\Sigma (A_{iu}) (e_i)] + Bkg] \qquad \text{Eq. 27G}^{1}$$

S_{adj} = the adjustable setpoint for 1/2-RI-5415 (cpm)

K_{sf} = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, (5) errors associated with monitor calibrations[2], and (6) anticipated short term variations in activity (this applicable to containment purges only).

It is recommended that a safety factor of 10 for containment purge releases be used for calculating the adjustable setpoint. However, other values for purge releases -- not to exceed 10 -- may be used as directed by the General Supervisor Chemistry. A safety factor of 1.5 shall be used for all other gaseous releases.

The particular value selected for the safety factor is somewhat arbitrary, however a value less than or equal to 10 does provide plant personnel with adequate time to respond to changing plant conditions and to initiate corrective **ACTIONS** so as to minimize the possibility of violating either the 10 CFR 50.72 limit or the Control 3.3.3.9 limits.

The use of the safety factor is consistent with ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133 which states that "... the alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous gaseous release limit of Control 3.11.2.1(a) will not be exceeded." (per NUREG-0133, 5.1.1).

This safety margin will prevent minor fluctuations in the nominal plant vent stack flow rates, errors in monitor efficiencies, and other statistical aberrations from adversely impacting the calculated adjustable setpoint. Additionally for a special case of containment purges during outages, the safety factor allows for short term variations in activity created as a result of system evolutions in containment.

---

1 Equation 27G has been derived from NUREG-0133, Addendum, page AA-1.

2 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

F_u = maximum undiluted radwaste flow rate (cubic meters per second)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 7.

F_{dx} = the estimated main vent stack (diluted gaseous radwaste) flow rate for unit x (cubic meters per second)

Since the main vent stack flow rate will vary depending on the reactor unit, the configuration of air dampers, and the input gas streams, nominal main vent stack flow rate is used to calculate the adjustable setpoint.

Use the nominal main vent stack flow rate, for the appropriate unit, listed on Attachment 7.

The main vent stack flow rate shall be determined, in accordance with approved procedures, at least once per 6 months ($\pm$ 25%). The Test and Equipment Unit shall be responsible for performing this test. The results of the main vent flow rate test shall be evaluated to ensure the main vent flow rates used in the ODCM are an accurate reflection of the true main vent flow rates. The RETS Program Manager is responsible for modifying the (main vent flow rates used in the) ODCM in the event the main vent flow rate for either Unit 1 or Unit 2 has increased to a value which is greater than the maximum discharge flow rates listed on Attachment 7.

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream, either the WGDT or containment building as applicable (microcuries per milliliter)

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

---

(3) Documenting the adjustable setpoint for 1/2-RI-5415

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures (e.g., CP-604).

(4) Changing the adjustable setpoint for 1/2-RI-5415

(a) In all cases, the adjustable alarm setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(b) If the adjustable setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(c) CHEMISTRY SECTION procedures (e.g., CP-604) contain administrative controls associated with calculating and approving an adjustable setpoint.

(d) Whenever this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9) the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release of a WGDT, a containment vent, or a containment purge via the main vent.

k) Alert setpoint for 1/2-RI-5415

(1) General information

(a) The alert setpoint is applicable to containment purges only.

(b) Whenever this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9), the alert setpoint is calculated and adjusted prior to each containment purge discharged via the main vent.

(c) The alert setpoint is based on the specific activities of the radionuclides present in the containment building. (The radionuclide concentrations are determined by radiochemical analysis in accordance with applicable CHEMISTRY SECTION procedures as required by Control 4.11.2.1.2).

(d) Whenever the alert setpoint is exceeded, the **PURGE** via the main vent may continue.

(e) The alert setpoint corresponds to a level of activity which indicates additional source term(s) may be present, and as a result, additional notifications and/or actions are required to identify the source and to accurately account for the activity discharged.

(f) The value for the alert setpoint is recorded on the gaseous release permit in accordance with applicable CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the main vent radiation monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 1/2-RI-5415, and provides an alarm to plant operators when the 1/2-RI-5415 alert setpoint has been exceeded.

(i) When this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9), a value for the alert setpoint shall be calculated prior to each containment purge as shown below.

(2) Calculating the alert setpoint for 1/2-RI-5415

(a) The alert setpoint is based on the specific activity of the radionuclides in the undiluted gaseous waste (as determined by radiochemical analysis per Control 4.11.2.1.2), and the setpoint is calculated as shown below.

ALERT SETPOINT FOR 1/2-RI-5415

$S_{alert} \leq 1.50\ [(F_u / F_{dx}) [\Sigma (A_{iu}) (e_i)] + Bkg]$	Eq. 27G[1]

S_{alert} = the alert setpoint for 1/2-RI-5415 (cpm)

1.50 = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless).

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, and (5) errors associated with monitor calibrations.[2]

The use of the safety factor is consistent with ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133 which states that "... the alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous gaseous release limit of Control 3.11.2.1(a) will not be exceeded." (per NUREG-0133, 5.1.1).

This safety margin will prevent minor fluctuations in the nominal plant vent stack flow rates, errors in monitor efficiencies, and other statistical aberrations from adversely impacting the calculated alert setpoint.

F_u = maximum undiluted radwaste flow rate (cubic meters per second)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachment 7.

F_{dx} = the estimated main vent stack (diluted gaseous radwaste) flow rate for unit x (cubic meters per second)

Since the main vent stack flow rate will vary depending on the reactor unit, the configuration of air dampers, and the input gas streams, nominal main vent stack flow rate is used to calculate the alert setpoint.

Use the nominal main vent stack flow rate, for the appropriate unit, listed on Attachment 7.

The main vent stack flow rate shall be determined, in accordance with approved procedures, at least once per 6 months ($\pm$ 25%). The Test and Equipment Unit shall be responsible for performing this test. The results of the main vent flow rate test shall be evaluated to ensure the main vent flow rates used in the ODCM are an accurate reflection of the true main vent flow rates. The RETS Program Manager is responsible for modifying the (main vent flow rates used in the) ODCM in the event the main vent flow rate for either Unit 1 or Unit 2 has increased to a value which is greater than the maximum discharge flow rates listed on Attachment 7.

A_{iu} = specific activity of radionuclide, i, in the containment building (microcuries per milliliter)

---

1 Equation 27G has been derived from NUREG-0133, Addendum, page AA-1.

2 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

---

(3) Documenting the alert setpoint for 1/2-RI-5415

(a) Whenever the alert setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures (e.g., CP-604).

(4) Changing the alert setpoint for 1/2-RI-5415

(a) In all cases, the alert setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(b) If the alert setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(c) CHEMISTRY SECTION procedures (e.g., CP-604) contain administrative controls associated with calculating and approving an alert setpoint.

(d) Whenever this monitor is satisfying the minimum channels operable requirement (per Control 3.3.3.9) the calculated value for the alert setpoint shall be entered into the plant computer prior to each containment purge via the main vent.

l) The low setpoint for 1/2-RI-5415

(1) This alarm is integral to the main vent monitor, as purchased from the supplier.

(2) The current value for the low alarm setpoint is specified in the CCNPP Alarm Manual.

(3) The low setpoint may be used to determine the **OPERABILITY** of this monitor (per Control 4.3.3.9, **CHANNEL FUNCTIONAL TEST**).

(4) The alarm generated by the low setpoint may be used to terminate a release in the event 1/2-RI-5415 fails (i.e., downscale failure or circuit failure) in accordance with Control 4.3.3.9.

(5) The low setpoint calculations are not described in the ODCM.

(6) Changes to the low setpoint are controlled by EN-1-100.

4. Westinghouse Plant Vent Stack Monitor (2-RE-5415)

a) All information related to 1-RE-5415 is applicable to the Unit 2 plant vent stack monitor with the following exception(s):

b) Monitors equivalent to 2-RE-5415

(1) 2-RE-5416 [the "WRNGM"] has the capability of providing the measurement and alarm functions of 2-RE-5415 during times when 2-RE-5415 is declared inoperable.

(2) 2-RE-5415 provides redundant monitoring [for 2-RE-5416] at the low end of the concentration ranges (UFSAR 11.2.3.2.12).

5. **Gaseous Radwaste Processing System** Radiation Monitor (0-RE-2191)

a) General description

(1) The **GASEOUS RADWASTE PROCESSING SYSTEM** Radiation Monitor (Waste Gas Decay Tank Radiation Monitor) contains 1 radiation element.

(2) It is a noble gas detector.

(3) The detector is an in-line GM tube (UFSAR, Table 11-10).

(4) The radiation element is designated 0-RE-2191.

(5) The radiation indicators designated 0-RI-2191.

(6) The units for the radiation indicator are counts per minute.

(7) The monitor was manufactured by Westinghouse.

b) Functions of 0-RE-2191

(1) continuously measure the release rate of noble gases emanating from the waste gas decay tank discharge header (Control 4.11.2.1.2, Table 4.11-2)

(2) continuously indicate (via 0-RI-2191) the activity (cpm) of noble gases emanating from the waste gas decay tank discharge header (Control 3.3.3.9 **OPERABILITY** requirement)

(3) alarm (via 1-RI-2191) prior to exceeding the site-boundary, noble-gas, total-body-dose-rate limit of 500 mr/yr (per Control 3.11.2.1.a)

(4) alarm (via 1-RI-2191) prior to exceeding the site-boundary, noble-gas, skin-dose-rate limit of 3000 mr/yr (per Control 3.11.2.1.a)

c) **OPERABILITY** of 0-RE-2191

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s).

(2) For more information on the function(s) of this monitor, see "Functions of 0-RE-2191" elsewhere in this section of the ODCM.

d) Monitors equivalent to 0-RE-2191

(1) There are no equivalent monitors associated with 0-RE-2191 since there are no other radiation monitors permanently installed in the waste gas discharge header, however, Control 3.3.3.9 defines the plant vent stack monitor as a "BACKUP MONITOR."

(2) 0-RE-2191 is designated the PRIMARY MONITOR for measuring noble gas activity released via the **GASEOUS RADWASTE PROCESSING SYSTEM**.

(3) 1-RE-5415 (or 1-RE-5416) is designated the BACKUP MONITOR if the WGDT is discharged via the Unit 1 main vent.

(4) 2-RE-5415 (or 2-RE-5416) is designated the BACKUP MONITOR if the WGDT is discharged via the Unit 2 main vent.

(5) WGDTs may be discharged through either the Unit 1 or Unit 2 main vent stack.

(6) The BACKUP MONITOR has the capability of ensuring the noble gas activity released from the **GASEOUS RADWASTE PROCESSING SYSTEM**--to the plant vent stack--does not exceed Control 3.11.2.1(a) at the **SITE BOUNDARY** (Control 3.3.3.9).

(7) In the event PRIMARY MONITOR (0-RE-2191) is inoperable or otherwise unavailable, the designated BACKUP MONITOR (either 1-RE-5415, 1-RE-5416, 2-RE-5415, or 2-RE-5416) may fulfill the measuring, indicating, and alarming functions normally provided by the PRIMARY MONITOR as long as plant operators record the BACKUP MONITOR readings every 15 minutes (Control 3.3.3.9, Table 3.3-12, **ACTION** 35a).

e) Radiological effluent controls for 0-RE-2191

(1) Control 3.3.3.9 states that releases via the **GASEOUS RADWASTE PROCESSING SYSTEM** may continue if ANY ONE of the following three conditions are satisfied:

(a) 0-RE-2191 is operable AND the alarm setpoint for 0-RI-2191 is set to ensure the annual dose rates due to noble gases at the **SITE BOUNDARY** are less than 500 mr/yr to the total body and are less than 3000 mr/yr to the skin (per Control 3.11.2.1.a), or

(b) One "BACKUP MONITOR" (see section (e) above) is operable; AND the "BACKUP MONITOR" readings are recorded every 15 minutes during the release; AND the alarm setpoint for the "BACKUP MONITOR" is set to ensure the annual dose rates due to noble gases at the **SITE BOUNDARY** are less than 500 mr/yr to the total body and are less than 3000 mr/yr to the skin (per Control 3.11.2.1.a), or

(c) All three activities described below are completed prior to the release:

i) at least two independent samples of the waste gas decay tank's contents are analyzed, and

ii) at least two technically qualified members of the Facility Staff independently verify the release rate calculations, and

iii) two qualified operators verify the discharge valve lineup.

f) Surveillances for 0-RE-2191

(1) Control 4.3.3.9 requires demonstrating the **OPERABILITY** of 0-RE-2191 by satisfying the checks, calibrations, and tests listed below

(a) **CHANNEL CHECK** prior to each release

(b) **SOURCE CHECK** prior to each release

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

g) Setpoints for 0-RI-2191

(1) Requirements and commitments

(a) The alarm and fixed setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9)

(b) The method for calculating fixed or adjustable setpoints shall be provided in the ODCM. (NUREG-0133, 5.1.1)

(2) There are three radiation alarm setpoints associated with, or otherwise related to, 0-RE-2191.

(a) 0-RE-2191 fixed high radiation alarm and automatic termination setpoint

(b) 0-RE-2191 adjustable plant computer high radiation alarm and manual termination setpoint

(c) 0-RE-2191 low radiation alarm setpoint

(3) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows

(a) The 0-RE-2191 fixed high radiation alarm and automatic termination setpoint will be referred to as the fixed setpoint.

(b) The 0-RE-2191 adjustable plant computer high radiation alarm and manual termination setpoint will be referred to as the adjustable setpoint.

(c) The 0-RE-2191 low radiation alarm setpoint will be referred to as the low setpoint.

(4) Each of these alarm setpoints are described below.

h) Fixed setpoint for 0-RI-2191

(1) General information

(a) The fixed setpoint is not adjusted for each release.

(b) The fixed setpoint is an alarm and termination setpoint.

(c) Whenever the fixed setpoint is exceeded, an alarm will be generated, and the WGDT release will be automatically suspended.

(d) The fixed setpoint corresponds to the maximum concentration of radionuclides allowed (by equation 6G) in gaseous waste discharged from the gaseous radwaste processing system.

(e) The current value for the fixed setpoint is specified in the CCNPP Alarm Manual.

(f) The CCNPP Alarm Manual refers to this setpoint as the 0-RI-2191 High Radiation Alarm Setpoint.

(g) The fixed setpoint is integral to the waste gas discharge monitor, as purchased from the supplier.

(h) The fixed setpoint is administratively controlled by EN-1-100.

(i) The fixed setpoint shall be calculated as described below[1].

---

1 The alarm and trip setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9).

(2) Calculating the fixed setpoint for 0-RI-2191

(a) The fixed setpoint for 0-RI-2191 (waste gas discharge monitor) shall be calculated as described below:

FIXED SETPOINT FOR 0-RI-2191

$$S_{fix} \leq K_{sf} \{ \{ 1 / [(x/Q) (F_u)] \} \Sigma [(e_i) (A_{iLn})] + Bkg \} \qquad \text{Eq. 6G}^{1}$$

Where,

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the MPC limit, L_{MPC}, used in equation 2G (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed setpoint is always less than or equal to the MPC limit, L_{MPC}, used in equation 2G.

A safety factor of 1.00 is used for calculating the fixed setpoint.

By setting the safety factor to 1, the safety factor is disabled.

Although it may appear that if this safety factor is set to 1.0, no safety margin exists, in actuality, another margin of safety has been incorporated into equation 2G (see definition of L_{MPC}).

A safety factor of 1.00 will yield a fixed setpoint which corresponds to the MPC limit, L_{MPC}, in equation 2G.

A safety factor of 0.500 will yield a fixed setpoint which corresponds to one-half the MPC limit, L_{MPC}, in equation 2G.

Other values of safety factors--not to exceed 1.00--may be used for calculating the fixed setpoint as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a value less than or equal to 1.0 does provide plant personnel with adequate time to respond to changing plant conditions and to initiate corrective **ACTIONS** so as to minimize the possibility of violating either the 10 CFR 50.72 limit or the Control 3.3.3.9 limits.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133 which states that "... the alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous gaseous release limit of Control 3.11.2.1(a) will not be exceeded." (per NUREG-0133, 5.1.1).

This safety margin will prevent minor fluctuations in the nominal WGDT discharge flow rates, errors in detector efficiencies, and other statistical aberrations from adversely impacting the calculated fixed setpoint.

S_{fix} = the fixed setpoint for 0-RI-2191 (cpm)

---

1 Equation 6G has been derived from NUREG-0133, 5.2.1, (the 500 mr/yr equation).

x/Q = the highest calculated historical annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

A waste gas decay tank release via the Unit 1 or Unit 2 main vent is considered a "long-term" release[1], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the setpoint calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases).

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

F_u = the estimated maximum flow rate of undiluted gases through the waste gas discharge header (cubic meters per second)

Since WGDT pressure is the motive force for discharge of a WGDT, the waste gas flow rate will continually decrease as the release progresses (i.e., as tank pressure is decreased).

Use the estimated maximum WGDT discharge flow rate, listed on Attachment 7, to calculate the fixed setpoint.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

A_{iLn} = the specific activities of radionuclide, i, found in TYPICAL GASEOUS RADWASTE RELEASES (calculated in accordance with 10 CFR 20, Appendix B, Table II, Note 1 as described below; microcuries per milliliter)

Calculate A_{iLn} in accordance with equation 5G.

SPECIFIC ACTIVITY LIMIT FOR NUCLIDE I IN A RADIONUCLIDE MIXTURE

$$A_{iLn} = (f_i)(A_{TLn}) \quad \textbf{Eq. 5G}$$

f_i = a fraction which represents the relative activity contribution of noble gas radionuclide i to the total noble gas activity for TYPICAL GASEOUS EFFLUENTS (unitless)

This value may be obtained using the guidance provided on Attachment 5.

A_{TLn} = the sum of the total specific activities of all noble gas radionuclides found in TYPICAL GASEOUS RADWASTE RELEASES (microcuries/cm^3).

Calculate A_{TLn} in accordance with equation 2G.

---

1 NUREG-0133, 3.3

SPECIFIC ACTIVITY OF NOBLE GASES AT THE **SITE BOUNDARY**

$$\Sigma [(f_i) (A_{TLn})] / A_{iLt} \leq L_{MPC} \quad \text{Eq. 2G}^{1}$$

Where,

L_{MPC} = the MPC limit

The value chosen for L_{MPC} in this equation is 2. The basis for this limit is 10 CFR 50.72.

It has been shown[2] that, for the radionuclides present in TYPICAL GASEOUS EFFLUENTS from CCNPP, the 2 MPC limit is more restrictive than the limits of Control 3.3.3.9.

It should be noted that by using "2" as the MPC limit (10 CFR 50.72), instead of using the limits of Control 3.11.2.1(a), a safety factor has been incorporated into equation 2G.

The use of 2 MPCs as a safety margin is consistent with the provisions of NUREG-0133, section 5.1.1, which states that, "... in all cases, conservative assumptions may be necessary in establishing these setpoints to account for system variables, ... the variability in release flow, ... and the time lag between alarm and final isolation of radioactive effluents." (NUREG-0133, 5.1.1)

An alarm setpoint corresponding to 2 MPCs serves to initiate a determination of whether the "4-hour NRC notification" (specified in 10 CFR 50.72) is required.

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 1 (microcuries/cm^3)

For all the radionuclides found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 1. An acceptable alternative is to ensure an isotopic mix which results in a more conservative setpoint.

Bkg = an approximation of the detector background prior to initiating the gaseous release (cpm)

Instead of using an approximation of the detector background, a value of 0 cpm may be used as the detector background if so desired.

---

(3) Documenting the fixed setpoint for 0-RI-2191

(a) Whenever the fixed setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with EN-1-100.

(4) Changing the fixed setpoint for 0-RI-2191

(a) If the fixed setpoint calculated in accordance with equation 6G exceeds the maximum range of the monitor, the fixed setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

---

1 Equation 2G has been derived from 10 CFR 20, Appendix B, Table II, Note 1.

2 Addendum To Setpoint Calculations For WRGM Monitors 1-RIC-5415 and 2-RIC-5415, R.L. Conatser, December 10, 1991.

(b) The fixed setpoint may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in equations 2G, 5G, or 6G have changed.

(d) The fixed setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL GASEOUS EFFLUENTS has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the historical maximum annual average atmospheric dispersion factor has changed,

iii) the MPC limit at the **SITE BOUNDARY**, (presently 2 MPCs) has changed,

iv) values listed in 10 CFR 20, Table II, column 1 have changed,

v) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

vi) the monitor is not conservative in its function (see section "Functions of 0-RE-2191" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving fixed setpoint.

---

1 As determined in accordance with Attachment 5.

i) Adjustable setpoint for 0-RI-2191

(1) General information

(a) Whenever this radiation monitor is operable, the adjustable setpoint is calculated and adjusted prior to each release of a WGDT.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in the WGDT. (The radionuclide concentrations are determined by radiochemical analysis in accordance with applicable CHEMISTRY SECTION procedures as required by Control 4.11.2.1.2).

(c) Whenever the adjustable setpoint is exceeded, the WGDT discharge will be manually suspended.

(d) Refer to the radwaste Alarm Manual for a full list of operator **ACTIONS** taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected when discharging a WGDT.

(f) The value for the adjustable setpoint is recorded on the gaseous release permit in accordance with applicable CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the **GASEOUS RADWASTE PROCESSING SYSTEM** radiation monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 0-RI-2191, and provides an alarm to plant operators when the 0-RI-2191 adjustable setpoint has been exceeded.

(i) When this monitor is operable, a value for the adjustable setpoint shall be calculated prior to each release of a WGDT as shown below.

(2) Calculating the adjustable setpoint for 0-RI-2191

(a) The adjustable setpoint is based on the specific activity of the radionuclides in the undiluted gaseous waste (as determined by radiochemical analysis per Control 4.11.2.1.2), and is calculated as shown below.

ADJUSTABLE SETPOINT FOR 0-RI-2191

$$S_{adj} \leq 1.50 [\Sigma (A_{iu}) (e_i) + Bkg] \qquad \textbf{Eq. 28G}^1$$

S_{adj} = the adjustable setpoint for 0-RI-2191 (cpm)

1.50 = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, and (5) errors associated with monitor calibrations.[2]

F_u = maximum allowed undiluted radwaste flow rate (cubic meters per second)

The maximum allowed undiluted radwaste flow rate for a WGDT is tabulated in Attachment 7.

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream (microcuries per milliliter)

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

---

(3) Documenting the adjustable setpoint for 0-RI-2191

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures (e.g., CP-604).

(4) Changing the adjustable setpoint for 0-RI-2191

(a) In all cases, the adjustable setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(b) If the adjustable setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(c) CHEMISTRY SECTION procedures (e.g., CP-604) contain administrative controls associated with calculating and approving an adjustable setpoint.

(d) Whenever this monitor is operable, the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release of a WGDT via the main vent.

---

1 Equation 28G has been derived from NUREG-0133, Addendum, page AA-1.

2 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

j) The low setpoint for 0-RI-2191

(1) This alarm is integral to the main vent monitor, as purchased from the supplier.

(2) The current value for the low setpoint is specified in the CCNPP Alarm Manual.

(3) The low setpoint may be used to determine the **OPERABILITY** of this monitor (per Control 4.3.3.9, **CHANNEL FUNCTIONAL TEST**).

(4) The alarm generated by the low setpoint may be used to terminate a release in the event 0-RI-2191 fails (i.e., downscale failure or circuit failure) in accordance with Control 4.3.3.9.

(5) The low setpoint calculations are not described in the ODCM.

(6) Changes to the low setpoint are controlled by EN-1-100.

ANNUAL TOTAL BODY DOSE RATE DUE TO NOBLE GASES IN GASEOUS EFFLUENTS

1. Introduction

a) 10 CFR 20.1301 specifies dose rate limits associated with the release of radioactive materials to **UNRESTRICTED AREAS**.

b) Radiological effluent controls have been established to implement the requirements of 10 CFR 20.1301. The 10 CFR 50 Appendix I, Design Objectives for ALARA Radioactive Effluents, upon which these calculations are based, are more restrictive than the public dose limits of 10 CFR 20.1301.

c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

a) The annual total body dose rate, due to noble gases in gaseous waste discharged to **UNRESTRICTED AREAS**, shall be less than 500 mr/yr (per Control 3.11.2.1).

b) The routine surveillances which are performed to verify compliance with this radiological effluent control is described below.

3. Surveillance Requirement

a) The annual total body dose rate, due to noble gases in all gaseous effluents discharged from the site, shall be determined in accordance with equation 7G (per Control 4.11.2.1.1).

b) The results of the radioactive gaseous waste sampling and analysis program (required by Control 4.11.2.1.2, and implemented by various CCNPP CHEMISTRY SECTION procedures) are used to calculate the annual total body dose rate due to noble gases in gaseous effluents.

c) The plant group(s) responsible for performing the required surveillances are identified below.

4. Responsible Plant Organization(s)

a) The CHEMISTRY SECTION is responsible for calculating the annual total body dose rate due to noble gases in gaseous effluents.

b) The CCNPP CHEMISTRY SECTION calculates the annual total body dose rate whenever the appropriate initiating conditions are present.

c) These initiating conditions are contained in the following section.

5. Initiating Conditions

a) The annual total body dose rate due to noble gases in gaseous effluents is calculated for each release of a WGDT.

b) The annual total body dose rate due to noble gases in gaseous effluents is calculated for each vent of a containment building.

c) The annual total body dose rate due to noble gases in gaseous effluents is calculated for each **PURGE** of a containment building.

d) The annual total body dose rate due to noble gases in gaseous effluents is calculated at least weekly[1] for CONTINUOUS discharges from plant vent stacks.

e) The annual total body dose rate due to noble gases in gaseous effluents is calculated for each discharge of combustion products resulting from the burning of contaminated oil.

f) The annual total body dose rate due to noble gases in gaseous effluents is calculated for each **ABNORMAL AND/OR UNANTICIPATED RADIOACTIVE GAS RELEASE**.

g) Whenever the correct initiating conditions are present, the annual total body dose rates shall be calculated as described below.

6. Calculation Methodology

a) The annual total body dose rate, at the **SITE BOUNDARY**, due to noble gases in gaseous effluents released to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 7G.[2]

---

1 The frequency is controlled by the implementing procedure, and is based on plant conditions. Under no conditions shall the frequency be less than once per month (Control 4.11.2.1.1 or 4.11.2.1.2, Table 4.11-2).

2 The alarm and trip setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9).

ANNUAL TOTAL BODY DOSE RATE DUE TO NOBLE GASES IN ALL GAS RELEASES

$$D_{t0} = \Sigma D_{tr} \qquad \text{Eq. 7G}$$

D_{t0} = the site-boundary annual total body dose rate due to noble gases in all gaseous effluents discharged (simultaneously) from the site (mrem/year)

D_{tr} = the site-boundary annual total body dose rate due to noble gases in release, r (mrem/year)

Sum for all releases, r, which are discharged simultaneously.

An example of a SIMULTANEOUS RELEASE would include the release of noble gas radionuclides from the Unit 1 plant vent stack while also discharging noble gases from the Unit 2 plant vent stack.

An example of a SIMULTANEOUS RELEASE would include the release of noble gas radionuclides from the Unit 1 plant vent stack while also discharging a waste gas decay tank.

Calculate the values of D_{tr} for each SIMULTANEOUS RELEASE as shown below.

---

b) At CCNPP, two methods exist for calculating D_{tr} (i.e., annual total body dose rate at the **SITE BOUNDARY** due to noble gases contained in a gaseous radwaste release, r, discharged from the site).

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

c) Rigorous method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, the annual total body dose rate due to noble gases in gaseous effluents discharged from the site to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 8G.

ANNUAL TOTAL BODY DOSE RATE DUE TO NOBLE GASES IN GAS RELEASE, r (RIGOROUS METHOD)

$$D_{tr} = (x/Q)\,[\Sigma\,(K_i)\,(Q_{ir})\,] \qquad \textbf{Eq. 8G}^{1}$$

Where,

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose rate calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

K_i = the total body dose factor due to gamma emissions for each identified noble gas radionuclide, i (mrem/yr per microcurie/cubic meter)

The total-body dose factors for gamma rays from noble gas radionuclides were obtained from Regulatory Guide 1.109, Appendix B, Table B-1.

The total-body dose factors for various noble gas radionuclides are tabulated in Attachment 10.

Q_{ir} = the release rate of noble gas radionuclide, i, in (simultaneous) gaseous release, r (microcuries/second).

Calculate the values of Q_{ir} for each SIMULTANEOUS RELEASE as shown below.

1 Equations 8G has been derived from NUREG-0133, 5.2.1, and Regulatory Guide 1.109 (Appendix B, Equation B-8 and Section C.2.e).

2 NUREG-0133, 3.3

INSTANTANEOUS RELEASE RATE OF NOBLE GAS NUCLIDE i IN GASEOUS RELEASE r

$$Q_{ir} = (A_{ir})(F_r)(c') \qquad \text{Eq. 9G}$$

Where,

A_{ir} = the specific activity of noble gas radionuclide, i, in (simultaneous) release, r (microcuries/cubic centimeter)

F_r = the discharge flow rate for (simultaneous) release, r (cubic meters per second)

If the discharge flow rate is unknown (e.g., the release has not been conducted), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the annual total body dose rate.

Whenever possible, the actual discharge flow rate determined from actual release conditions (e.g., initial pressure, volume, and temperature of a WGDT along with final pressure and temperature) shall be used in equation 9G.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

---

d) Simplified method

(1) If a computer system and the appropriate software are NOT available, the annual total body dose rate due to noble gases in gaseous effluents discharged from the site to **UNRESTRICTED AREAS** may be calculated in accordance with equation 10G.

ANNUAL TOTAL BODY DOSE RATE DUE TO NOBLE GASES IN GAS RELEASE, r (SIMPLIFIED METHOD)

$$D_{tr} = [(x/Q)(K_{avg})/(K_{sf})] \Sigma Q_{ir} \qquad \text{Eq. 10G[1]}$$

Where,

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose rate calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

---

1 Equations 10G has been derived from NUREG-0133, 5.2.1, and historical, site-specific data.

2 NUREG-0133, 3.3

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

K_{avg} = the empirically derived, site specific, average, total body, dose factor due to gamma emissions from TYPICAL GASEOUS EFFLUENTS (mrem/yr per microcurie/cubic meter)

A site-specific, average, gamma total body dose factor for TYPICAL GASEOUS EFFLUENTS has been calculated from historical data.

The calculation of this site-specific, average, gamma air dose factor is presented on Attachment 11 (use section 3.4.5 of the old ODCM.)

Refer to the table on Attachment 11 for the current value for the empirically derived, site specific, average gamma total body dose factor.

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP annual total body dose rate limit to the annual total body dose rate limit of Control 3.11.2.1, (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the annual total body dose rate is always less than or equal to the annual total body dose rate limit of Control 3.11.2.1.

A safety factor of 1.00 will yield an annual total body dose rate which corresponds to the annual total body dose rate limit of Control 3.11.2.1.

A safety factor of 0.500 will yield an annual total body dose rate which corresponds to one-half the annual total body dose rate limit of Control 3.11.2.1.

It is recommended that a safety factor of 1.0 be used for calculating the annual total body dose rate, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive gaseous release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.2.1 when simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 10G to account for any potential nonconservatism associated with applying the empirically derived total body gamma dose factor, K_{avg}, to all radionuclides identified in the gaseous release. Such nonconservatism could conceivable be present whenever radionuclides having a total body gamma dose factor greater than K_{avg} are present in a gaseous release.

Q_{ir} = the release rate of noble gas radionuclide, i, in (simultaneous) gaseous release, r (microcuries/second)

Calculate the values of Q_{ir} for each SIMULTANEOUS RELEASE in accordance with equation 9G.

INSTANTANEOUS RELEASE RATE OF NOBLE GAS NUCLIDE i IN GASEOUS RELEASE r

$$Q_{ir} = (A_{ir})(F_r)(c') \qquad \textbf{Eq. 9G}$$

Where,

A_{ir} = the specific activity of noble gas radionuclide, i, in (simultaneous) release, r (microcuries/cubic centimeter)

F_r = the discharge flow rate for (simultaneous) release, r (cubic meters per second)

If the discharge flow rate is unknown (e.g., the release has not been conducted), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the annual total body dose rate.

Whenever possible, the actual discharge flow rate determined from actual release conditions (e.g., initial pressure, volume, and temperature of a WGDT along with final pressure and temperature) shall be used in equation 9G.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

---

e) Radiation monitoring system algorithms

(1) The plant vent stack radiation monitoring systems display values which are proportional to the annual total body dose rates due to noble gases emanating from the plant vent stacks.

(2) The values displayed by the plant vent stack radiation monitoring systems are not used for the purpose of effluent accountability per se, but the values displayed can provide a gross approximation of annual total body dose rate (see Control 3.3.3.9).

(3) The Westinghouse Main Vent Noble Gas Monitor, 1/2-RE-5415, is an analog system and does not employ instrument algorithm to determine noble gas release rates.

(a) It is possible to approximate the noble gas release rates for the Unit 1 and Unit 2 main vents based on output from 1/2-RI-5415.

(b) These calculations are described elsewhere in the ODCM. (See equation 4G in the section "Calculating the Fixed Setpoint for 1/2-RI-5415.")

(4) The Sorrento WRNGM, 1/2-RE-5416, is a digital radiation monitoring system which employs an instrument algorithm to determine noble release rates (microcuries per second).

(a) It is possible to approximate the noble gas release rates for the Unit 1 and Unit 2 main vents based on output from 1/2-RIC-5415.

(b) These calculations are described elsewhere in the ODCM. (See equation 1G in the section "Calculating the Fixed High-High Alarm Setpoint for 1/2-RIC-5415.")

(c) The instrument algorithms and the (data base) values accessed by the instrument algorithms are controlled by EN-1-100.

f) Once the calculations above have been completed, the calculation results are compared to the applicable limits and corrective **ACTIONS** are initiated as described below.

7. Corrective actions

a) CHEMISTRY SECTION surveillance procedures shall contain/and or reference administrative and/or Control limits for annual total body dose rates for gaseous effluents and shall specify corrective actions to be initiated when these limits are exceeded.

b) Refer to Control 3.11.2.1 for actions to be taken in the event the calculated annual total body dose rate due to noble gases in gaseous effluents exceeds 500 mr/yr.

ANNUAL SKIN DOSE RATE DUE TO NOBLE GASES IN GASEOUS EFFLUENTS

1. Introduction

   a) 10 CFR 20.1301 specifies dose rate limits associated with the release of radioactive materials to **UNRESTRICTED AREAS**.

   b) Radiological effluent controls have been established to implement the requirements of 10 CFR 20.1301.

   c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

   a) The annual skin dose rate, due to noble gases in gaseous waste discharged to **UNRESTRICTED AREAS**, shall be less than 3000 mr/yr (per Control 3.11.2.1).

   b) The routine surveillances which are performed to verify compliance with this radiological effluent controls are described below.

3. Surveillance Requirement

   a) The annual skin dose rate at the **SITE BOUNDARY**, due to noble gases in all gaseous effluents discharged from the site, shall be determined in accordance with equation 11G (per Control 4.11.2.1.1).

   b) The results of the radioactive gaseous waste sampling and analysis program (required by Control 4.11.2.1.2, and implemented by various CCNPP CHEMISTRY SECTION procedures) are used to calculate the annual skin dose rate due to noble gases in gaseous effluents.

   c) The plant group(s) responsible for performing the required surveillances are identified below.

4. Responsible Plant Organization(s)

   a) The CHEMISTRY SECTION is responsible for calculating the annual skin dose rate due to noble gases in gaseous effluents.

   b) The CCNPP CHEMISTRY SECTION calculates the annual skin dose rate whenever the appropriate initiating conditions are present.

   c) These initiating conditions are contained in the following section.

5. Initiating Conditions

   a) The annual skin dose rate due to noble gases in all gaseous effluents discharged from the site is calculated for each release of a WGDT.

   b) The annual skin dose rate due to noble gases in all gaseous effluents discharged from the site is calculated for each vent of a containment building.

c) The annual skin dose rate due to noble gases in all gaseous effluents discharged from the site is calculated for each **PURGE** of a containment building.

d) The annual skin dose rate due to noble gases in all gaseous effluents discharged from the site is calculated at least weekly[1] for CONTINUOUS discharges from plant vent stacks.

e) The annual skin dose rate due to noble gases in all gaseous effluents discharged from the site is calculated for each discharge of combustion products resulting from the burning of contaminated oil.

f) The annual skin dose rate due to noble gases in all gaseous effluents discharged from the site is calculated for each **ABNORMAL AND/OR UNANTICIPATED RADIOACTIVE GAS RELEASE**.

g) Whenever the correct initiating conditions are present, the annual skin dose rates shall be calculated as described below.

6. Calculation Methodology

a) The annual skin dose rate, at the **SITE BOUNDARY**, due to noble gases in all gaseous effluents discharged simultaneously from the site to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 11G.[2]

ANNUAL SKIN DOSE RATE DUE TO NOBLE GASES IN ALL SIMULTANEOUS GAS RELEASES, r

$$D_{s0} = \Sigma D_{sr} \qquad \textbf{Eq. 11G}$$

D_{s0} = the annual skin dose rate at the **SITE BOUNDARY** due to noble gases in all simultaneous discharges of gaseous radwaste from the site ("Unit 0")

D_{sr} = the annual skin dose rate at the **SITE BOUNDARY** due to noble gases in release, r

Sum for all releases, r, which are discharged simultaneously.

An example of a SIMULTANEOUS RELEASE would include the release of noble gas radionuclides from the Unit 1 plant vent stack while also discharging noble gases from the Unit 2 plant vent stack.

An example of a SIMULTANEOUS RELEASE would include the release of noble gas radionuclides from the Unit 1 plant vent stack while also discharging a waste gas decay tank.

Calculate the values of D_{sr} for each SIMULTANEOUS RELEASE as shown below.

---

1 The frequency is controlled by the implementing procedure and is based on plant conditions. Under no conditions shall the frequency be less than once per month (Control 4.11.2.1.1 or 4.11.2.1.2, Table 4.11-2).

2 The alarm and trip setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9).

b) At CCNPP, two methods exist for calculating D_{sr} (i.e., annual total skin dose rate at the **SITE BOUNDARY** due to noble gases contained in a gaseous radwaste release, r, discharged from the site).

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

c) Rigorous Method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, the annual skin dose rate due to noble gases in gaseous release, r, discharged from the site to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 12G.

ANUAL SKIN DOSE RATE DUE TO NOBLE GASES IN GAS RELEASE, r
(RIGOROUS METHOD)

$$D_{sr} = (x/Q)\,\Sigma\,\{\,[\,L_i + (1.1)\,(M_i)\,]\,(Q_{ir})\,\} \qquad \textbf{Eq. 12G}^1$$

Where,

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose rate calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

L_i = the skin dose factor due to beta emissions for each identified noble gas radionuclide, i (mrem/yr per microcurie/cubic meter)

The beta skin dose factors have been obtained from Regulatory Guide 1.109, Appendix B, Table B-1.

The beta skin dose factors for various noble gas radionuclides are tabulated in Attachment 10.

1 Equation 12G has been derived from NUREG-0133, 5.2.1, and Regulatory Guide 1.109 (Appendix B, Equation B-9 and Section C.2.f).

2 NUREG-0133, 3.3

M_i = the air dose factor due to gamma emissions for each identified noble gas radionuclide, i (mrad/yr per microcurie/cubic meter)

The gamma air dose factors have been obtained from Regulatory Guide 1.109, Appendix B, Table B-1.

The gamma air dose factors for various noble gas radionuclides are tabulated in Attachment 10.

1.1 = The conversion constant, 1.1 mrem/mrad, represents the skin dose (1.1 mrem) equivalent to air dose (1.0 mrad), and is used to convert air dose to skin dose.

Q_{ir} = the release rate of noble gas radionuclide, i, in (simultaneous) release, r (microcuries/second).

This value shall be calculated in accordance with equation 9G.

INSTANTANEOUS RELEASE RATE OF NOBLE GAS NUCLIDE i IN GASEOUS RELEASE r

$$Q_{ir} = (A_{ir})(F_r)(c') \qquad \textbf{Eq. 9G}$$

A_{ir} = the specific activity of noble gas radionuclide, i, in (simultaneous) release, r (microcuries/cubic centimeter)

F_r = the discharge flow rate for (simultaneous) release, r (cubic meters per second)

If the discharge flow rate is unknown (e.g., the release has not been conducted), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the annual skin dose rate.

Whenever possible, the actual discharge flow rate determined from actual release conditions (e.g., initial pressure, volume, and temperature of a WGDT along with final pressure and temperature) shall be used in equation 9G.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

d) Simplified method

(1) If a computer system and the appropriate software are NOT available, the annual skin dose rate due to noble gases in gaseous effluents discharged from the site to **UNRESTRICTED AREAS** may be calculated in accordance with equation 13G.

ANNUAL SKIN DOSE RATE DUE TO NOBLE GASES IN GAS RELEASE, r (SIMPLIFIED METHOD)

$$D_{sr} = [(x/Q) / (K_{sf})] [L_{avg} + (1.1) (M_{avg})] \Sigma Q_{ir} \quad \textbf{Eq. 13G}^{1}$$

Where,

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose rate calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP annual skin dose rate limit to the annual skin dose rate limit of Control 3.11.2.1, (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the annual skin dose rate is always less than or equal to the annual skin dose rate limit of Control 3.11.2.1.

A safety factor of 1.00 will yield an annual skin dose rate which corresponds to the annual skin dose rate limit of Control 3.11.2.1.

A safety factor of 0.500 will yield an annual skin dose rate which corresponds to one-half the annual skin dose rate limit of Control 3.11.2.1.

It is recommended that a safety factor of 1.0 be used for calculating the annual skin dose rate, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive gaseous release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.2.1 when simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 13G to account for any potential nonconservatism associated with applying the empirically derived skin beta dose factor, L_{avg}, to all radionuclides identified in the gaseous release. Such nonconservatism could conceivable be present whenever radionuclides having a skin beta dose factor greater than L_{avg} are present in a gaseous release.

---

1 Equation 13G has been derived from NUREG-0133, 5.2.1, and Regulatory Guide 1.109 (Appendix B, Equation B-9 and Section C.2.f).

2 NUREG-0133, 3.3

L_{avg} = the empirically derived, site specific, average, skin dose factor due to beta emissions from TYPICAL GASEOUS EFFLUENTS (mrem/yr per microcurie/cubic meter)

A site-specific, average, beta skin dose factor for TYPICAL GASEOUS EFFLUENTS has been calculated from historical data.

The calculation of this site-specific, average, beta skin dose factor is presented on Attachment 11.

Refer to the table on Attachment 11 for the current value for the empirically derived, site specific, average beta skin dose factor.

M_{avg} = the empirically derived, site specific, average, air dose factor due to gamma emissions from TYPICAL GASEOUS EFFLUENTS (mrad/yr per microcurie/cubic meter)

A site-specific, average, gamma air dose factor for TYPICAL GASEOUS EFFLUENTS has been calculated from historical data.

The calculation of this site-specific, average, gamma air dose factor is presented on Attachment 11.

Refer to the table on Attachment 11 for the current value for the empirically derived, site specific, average gamma air dose factor.

1.1 = The conversion constant, 1.1 mrem/mrad, represents the skin dose (1.1 mrem) equivalent to air dose (1.0 mrad), and is used to convert air dose to skin dose.

Q_{ir} = the release rate of noble gas radionuclide, i, in (simultaneous) release, r (microcuries/second)

This value shall be calculated in accordance with equation 9G.

INSTANTANEOUS RELEASE RATE OF NOBLE GAS NUCLIDE i IN GASEOUS RELEASE r

$$Q_{ir} = (A_{ir})(F_r)(c') \qquad \textbf{Eq. 9G}$$

A_{ir} = the specific activity of noble gas radionuclide, i, in (simultaneous) release, r (microcuries/cubic centimeter)

F_r = the discharge flow rate for (simultaneous) release, r (cubic meters per second)

If the discharge flow rate is unknown (e.g., the release has not been conducted), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the annual skin dose rate.

Whenever possible, the actual discharge flow rate determined from actual release conditions (e.g., initial pressure, volume, and temperature of a WGDT along with final pressure and temperature) shall be used in equation 9G.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

e) Radiation monitoring system algorithms

(1) The plant vent stack radiation monitoring systems display values which are proportional to the annual skin dose rate due to noble gases emanating from the plant vent stacks.

(2) The values displayed by the plant vent stack radiation monitoring systems are not used for the purpose of effluent accountability per se, but the values displayed can provide a gross approximation of annual skin dose rate (see Control 3.3.3.9).

(3) The Westinghouse Main Vent Stack Noble Gas Monitor, 1/2-RE-5415, is an analog system and does not employ instrument algorithm to determine noble release rates.

(a) It is possible to approximate the noble gas release rates for the Unit 1 and Unit 2 main vents based on output from 1/2-RI-5415.

(b) These calculations are described elsewhere in the ODCM. (See equation 4G in the section "Calculating the Fixed Setpoint for 1/2-RI-5415.")

(4) The Sorrento WRNGM, 1/2-RE-5416, is a digital radiation monitoring system which employs an instrument algorithm to determine noble release rates (microcuries per second).

(a) It is possible to approximate the noble gas release rates for the Unit 1 and Unit 2 main vents based on output from 1/2-RIC-5415.

(b) These calculations are described elsewhere in the ODCM. (See equation 1G in the section "Calculating the Fixed High-High Alarm Setpoint for 1/2-RIC-5415.")

(5) The instrument algorithms and the (data base) values accessed by the instrument algorithms are controlled by EN-1-100.

f) Once the calculations above have been completed, the calculation results are compared to the applicable limits and corrective **ACTIONS** are initiated as described below.

7. Corrective actions

a) CHEMISTRY SECTION surveillance procedures shall contain/and or reference administrative and/or Control limits for annual skin dose rate for gaseous effluents and shall specify corrective actions to be initiated when these limits are exceeded.

b) Refer to Control 3.11.2.1 for actions to be taken in the event the calculated annual skin dose rate exceeds 3000 mr/yr.

ANNUAL ORGAN DOSE RATES DUE TO IODINES AND PARTICULATES IN GASEOUS EFFLUENTS

1. Introduction

   a) 10 CFR 20.1301 specifies dose rate limits associated with the release of radioactive materials to **UNRESTRICTED AREAS**.

   b) Radiological effluent controls were originally established to implement the requirements of 10 CFR 20.1301. The 10 CFR 50 Appendix I, Design Objectives for ALARA Radioactive Effluents, upon which these calculations are based, are more restrictive than the public dose limits of 10 CFR 20.1301.

   c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

   a) The annual organ dose rates, due to iodines and particulates in gaseous waste discharged to **UNRESTRICTED AREAS**, shall be less than 1500 mr/yr (per Control 3.11.2.1).

   b) The routine surveillances which are performed to verify compliance with this radiological effluent controls are described below.

3. Surveillance Requirements

   a) The CHEMISTRY Sections sampling and analysis procedure(s) shall describe the CCNPP radioactive gaseous waste sampling and analysis program (required by Control 4.11.2.1.2).

   b) The results of the radioactive gaseous waste sampling and analysis program are used to calculate the annual organ dose rates due to iodines and particulates in gaseous effluents.

   c) The plant group(s) responsible for performing the required surveillances are identified below.

4. Responsible Plant Organization(s)

   a) The CHEMISTRY SECTION is responsible for calculating the annual organ dose rates due to iodines and particulates in gaseous effluents.

   b) The CHEMISTRY SECTION calculates the annual organ dose rates whenever the appropriate initiating conditions are present.

   c) These initiating conditions are contained in the following section.

5. Initiating Conditions

a) The annual organ dose rate--for each organ and at the **SITE BOUNDARY**--due to iodines and particulates in gaseous effluents is calculated at least weekly[1] for CONTINUOUS discharges from plant vent stacks.

b) The annual organ dose rate--for each organ and at the **SITE BOUNDARY**--due to iodines and particulates in gaseous effluents is calculated for each discharge of combustion products resulting from the burning of contaminated oil.

c) The annual organ dose rate--for each organ and at the **SITE BOUNDARY**--due to iodines and particulates in gaseous effluents is calculated for each **ABNORMAL AND/OR UNANTICIPATED RADIOACTIVE GAS RELEASE**[2].

d) Whenever the correct initiating conditions are present, the annual organ dose rates shall be calculated as described below.

6. Calculation Methodology

a) The annual organ dose rate, at the **SITE BOUNDARY**, due to iodine and particulate radionuclides in gaseous effluents released to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 14G.[3]

ANNUAL ORGAN, o , DOSE RATE DUE TO IODINES AND PARTICULATES IN ALL SIMULTANEOUS GASEOUS RELEASES, r FROM THE SITE, 0

$$\mathbf{D_{o0} = \Sigma D_{or}} \qquad \textbf{Eq. 14G}$$

D_{o0} = the site-boundary annual organ dose rate due to iodine and particulate radionuclides in all gaseous effluents discharged simultaneously from the site ("Unit 0")

D_{or} = the site-boundary annual organ dose rate due to iodine and particulate radionuclides in release, r

Sum for all releases, r, which are discharged simultaneously.

An example of a SIMULTANEOUS RELEASE would include the release of iodines and particulate radionuclides from the Unit 1 plant vent stack while also discharging iodines and particulate radionuclides from the Unit 2 plant vent stack.

An example of a SIMULTANEOUS RELEASE would include the release of iodine and particulate radionuclides from the Unit 1 plant vent stack while also discharging a waste gas decay tank.

1 The frequency is controlled by the implementing procedure and is based on plant conditions. Under no conditions shall the frequency be less than once per month (Control 4.11.2.1.1 or 4.11.2.1.2, Table 4.11-2).

2 See the definition of ABNORMAL/UNANTICIPATED GAS RELEASE in the DEFINITIONS section of the ODCM.

3 The alarm and fixed setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9).

Calculate the values of D_{or} for each SIMULTANEOUS RELEASE as shown below.

b) At CCNPP, two methods exist for calculating D_{or} (i.e., the annual organ dose rates due to iodine and particulate radionuclides in gaseous effluents released to **UNRESTRICTED AREAS**).

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

c) Rigorous Method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, the annual organ dose rates due to iodines and particulates in gaseous effluents released to an **UNRESTRICTED AREA** shall be calculated in accordance with equation 15G.

ANNUAL ORGAN, o, DOSE RATE DUE TO IODINES AND PARTICULATES IN GASEOUS RELEASE, r (RIGOROUS METHOD)

$$D_{or} = (x/Q)\,\Sigma\,(P_i)\,(Q_{ir}) \qquad \textbf{Eq. 15G}^{1}$$

Where,

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose rate calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (UFSAR, 2.3.6.3).

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (UFSAR, 2.3.6.3).

P_i = the maximum organ inhalation pathway dose parameter for iodine and particulate radionuclides, i, for the most restrictive (i.e., child) age group (mrem/year per microcurie/cubic meter)

The inhalation pathway dose parameters have been obtained in accordance with NUREG-0133, 5.2.1.1.

---

1 Equation 15G has been derived from NUREG-0133, 5.2.1.
2 NUREG-0133, 3.3

The pathway dose factor specified in NUREG-0133, 5.2.1.b, specifies calculating the exposure to the "INFANT" age group, where the exposure is due to a combination of three separate pathways.

1) inhalation,

2) ground plane, and

3) food.

The latest NRC guidance has deleted the requirement to include the ground plane and food dose contributions when calculating maximum organ doses, therefore no pathway dose factors are calculated for the ground plane or food pathways.

The latest NRC guidance has changed the critical receptor age group from "infant" to "child."

The child, inhalation pathway dose parameters for various radionuclides, sorted by critical organ, are tabulated in Attachment 12.

It should be noted that the dose parameters, P_i, (listed in Attachment 12) calculated in accordance with NUREG-0133, section 5.2.1.1 and the latest NRC guidance are numerically equal to the "Inhalation Pathway Factors," K_i, calculated in accordance with NUREG-0133, section 5.3.1.1. As a result the ODCM does not contain two separate tables for values of P_i and K_i.

Q_{ir} = the release rate of iodine or particulate radionuclide, i, in (simultaneous) gaseous release, r (microcuries/second).

Calculate the values of Q_{ir} for each SIMULTANEOUS RELEASE in accordance with equation 9G.

INSTANTANEOUS RELEASE RATE OF IODINE OR PARTICULATE NUCLIDE i IN GASEOUS RELEASE r

$$Q_{ir} = (A_{ir})(F_r)(c') \qquad \textbf{Eq. 9G}$$

A_{ir} = the specific activity of iodine or particulate radionuclide, i, in (simultaneous) release, r (microcuries/cubic centimeter)

F_r = the discharge flow rate for (simultaneous) release, r (cubic meters per second)

If the discharge flow rate is unknown (e.g., the release has not been conducted), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the annual organ dose rate.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

d) simplified method

(1) If a computer system and the appropriate software are NOT available, the annual organ dose rate due to iodines and particulates in gaseous effluents discharged from the site to **UNRESTRICTED AREAS** may be calculated in accordance with equation 16G.

ANNUAL ORGAN, o, DOSE RATE DUE TO IODINES AND PARTICULATES IN GASEOUS RELEASE, r (SIMPLIFIED METHOD)

$$D_{or} = (1/K_{sf})(x/Q)(P_{max})\Sigma Q_{ir} \qquad \textbf{Eq. 16G}^{1}$$

Where,

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP organ dose rate limit to the organ dose rate limit of Control 3.11.2.1, (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the organ dose rate is always less than or equal to the organ dose rate limit of Control 3.11.2.1.

A safety factor of 1.00 will yield an organ dose rate which corresponds to the organ dose rate limit of Control 3.11.2.1.

A safety factor of 0.500 will yield an organ dose which corresponds to one-half the organ dose rate limit of Control 3.11.2.1.

It is recommended that a safety factor of 1.0 be used for calculating the organ dose rate, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive gaseous release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.2.1 when simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 16G to account for any potential nonconservatism associated with applying the dose parameter, P_{max}, to all radionuclides identified in the gaseous release. Such nonconservatism could conceivable be present whenever radionuclides having a dose parameter greater than P_{max} are present in a gaseous release.

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose rate calculations.

---

1 Equation 16G has been derived from NUREG-0133, 5.2.1.

2 NUREG-0133, 3.3

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

P_{max} = the most restrictive dose parameter which would be reasonably anticipated for the inhalation pathway, child age group, thyroid organ, and I-131 radionuclide (mrem/year per microcurie/cubic meter)

The inhalation pathway dose parameters have been obtained in accordance with NUREG-0133, 5.2.1.1.

The pathway dose factor specified in NUREG-0133, 5.2.1.b, specifies calculating the exposure to the "INFANT" age group, where the exposure is due to a combination of three separate pathways.

1) inhalation,

2) ground plane, and

3) food.

The latest NRC guidance has deleted the requirement to include the ground plane and food dose contributions when calculating maximum organ doses, therefore no pathway dose factors are calculated for the ground plane or food pathways.

The latest NRC guidance has changed the critical receptor age group from "infant" to "child."

The child, inhalation pathway dose parameters for various radionuclides, sorted by critical organ, are tabulated in Attachment 12.

It should be noted that the dose parameters, P_i, (listed in Attachment 12) calculated in accordance with NUREG-0133, section 5.2.1.1 and the latest NRC guidance are numerically equal to the "Inhalation Pathway Factors," K_i, calculated in accordance with NUREG-0133, section 5.3.1.1. As a result the ODCM does not contain two separate tables for values of P_i and K_i.

Q_{ir} = the release rate of iodine or particulate radionuclide, i, in (simultaneous) gaseous release, r (microcuries/second).

Calculate the values of Q_{ir} for each SIMULTANEOUS RELEASE in accordance with equation 9G.

INSTANTANEOUS RELEASE RATE OF IODINE OR PARTICULATE NUCLIDE i IN GASEOUS RELEASE r

$$Q_{ir} = (A_{ir})(F_r)(c') \qquad \text{Eq. 9G}$$

A_{ir} = the specific activity of iodine or particulate radionuclide, i, in (simultaneous) release, r (microcuries/cubic centimeter)

F_r = the discharge flow rate for (simultaneous) release, r (cubic meters per second)

If the discharge flow rate is unknown (e.g., the release has not been conducted), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the annual organ dose rate.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

c' = a conversion constant (1E6 cubic centimeters per cubic meter)

---

e) Once the calculations above have been completed, the calculation results are compared to the applicable limits and corrective actions are initiated as described below.

7. Corrective actions

a) CHEMISTRY SECTION surveillance procedures shall contain/and/or reference administrative and/or Control limits for annual organ dose rates for gaseous effluents and shall specify corrective actions to be initiated when these limits are exceeded.

b) Refer to Control 3.11.2.1 for actions to be taken in the event the calculated annual organ dose rate to any organ exceeds 1500 mr/yr.

CUMULATIVE GAMMA AIR DOSES DUE TO NOBLE GASES IN GASEOUS EFFLUENTS

1. Introduction

a) Appendix I to 10 CFR 50 specifies cumulative gamma air dose limits associated with the release of radioactive materials to **UNRESTRICTED AREAS**.

b) Radiological effluent controls have been established to implement the requirements of 10 CFR 50, Appendix I.

c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

   a) The cumulative gamma air dose, due to noble gases in gaseous effluents released to **UNRESTRICTED AREAS**, shall be less than 10 mrads in any calendar quarter, and shall be less than 20 mrads in any calendar year (per Control 3.11.2.2)

   b) The routine surveillances which are performed to verify compliance with these radiological effluent controls are described below.

3. Surveillance Requirement(s)

   a) The cumulative gamma air doses, for the current calendar month, the calendar quarter, and the current calendar year, due to noble gases in gaseous effluents, shall be determined at least once every 31 days (Control 4.11.2.2).

   b) The plant group(s) responsible for performing the required surveillance(s) are identified below.

4. Responsible Plant Organizations

   a) The CHEMISTRY SECTION is responsible for calculating the cumulative gamma air doses for the current calendar quarter and the current calendar year.

   b) The CHEMISTRY SECTION calculates the cumulative gamma air doses whenever the appropriate initiating conditions are present

   c) These initiating conditions are contained in the following section.

5. Initiating Conditions

   a) The cumulative gamma air doses due to noble gases in gaseous effluents shall be determined at least once per 31 days (Control 4.11.2.2).

   b) The cumulative gamma air doses due to noble gases in gaseous effluents shall be calculated for each release of a WGDT.

   c) The cumulative gamma air doses due to noble gases in gaseous effluents shall be calculated for each vent of a containment building.

   d) The cumulative gamma air doses due to noble gases in gaseous effluents shall be calculated for each **PURGE** of a containment building.

   e) The cumulative gamma air doses due to noble gases in gaseous effluents shall be calculated at least weekly[1] for CONTINUOUS discharges from plant vent stacks.

   f) The cumulative gamma air doses due to noble gases in gaseous effluents shall be calculated for each discharge of combustion products resulting from the burning of contaminated oil.

---

1 The frequency is controlled by the implementing procedure, and is based on plant conditions. Under no conditions shall the frequency be less than once per month (Control 4.11.2.1.1 or 4.11.2.1.2, Table 4.11-2).

g) The cumulative gamma air doses due to noble gases in gaseous effluents shall be calculated for each **ABNORMAL AND/OR UNANTICIPATED RADIOACTIVE GAS RELEASE**[1].

h) Whenever the correct initiating conditions are present, the cumulative gamma air doses shall be calculated as described below.

6. Calculation Methodology

a) The cumulative gamma air dose at the **SITE BOUNDARY** (e.g., for the current calendar month, current calendar quarter, current calendar year, or previous 92 days) due to noble gases in gaseous effluents shall be calculated using the following equation[2]:

---

1 The criteria used to define ABNORMAL AND UNANTICIPATED GAS RELEASES may be found in the implementing procedures.

2 The alarm and fixed setpoints ... shall be determined and adjusted in accordance with the methodology and parameters of the ODCM. (Control 3.3.3.9).

CUMULATIVE GAMMA, g, AIR DOSE FOR ALL GASEOUS RELEASES, r, DISCHARGED DURING TIME INTERVAL, t

$$D_{gt} = \Sigma D_{gr} \qquad \textbf{Eq. 17G}$$

Where,

D_{gt} = the cumulative gamma air dose (mrad) at the **SITE BOUNDARY** due to noble gas radionuclides contained in all gaseous radwaste discharged from the site during the time interval, t

D_{gr} = the cumulative gamma air dose (mrad) at the **SITE BOUNDARY** due to noble gas radionuclides contained in gaseous radwaste release, r, discharged from the site during the time interval of interest

Calculate the values of D_{gr} for each gaseous release as described below.

---

b) At CCNPP, two methods exist for calculating D_{gr} (i.e., the gamma air dose at the **SITE BOUNDARY** due to noble gas radionuclides contained in a gaseous radwaste release, r, discharged from the site during a specified time interval).

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

c) rigorous method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, the gamma air dose due to noble gases in gaseous effluents released to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 18G.

GAMMA AIR DOSE DUE TO NOBLE GASES IN GAS RELEASE, r (RIGOROUS EQUATION)

$$D_{gr} = (3.17E\text{-}8)(x/Q)\sum[(M_i)(Q'_{ir})] \qquad \textbf{Eq. 18G}[1]$$

Where,

3.17E-8 = The conversion constant, 3.17E-8, represents the inverse of the number of seconds in a year.

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

M_i = the air dose factor due to gamma emissions for each identified noble gas radionuclide, i (mrad/yr per microcurie/cubic meter)

The gamma air dose factors have been obtained from Regulatory Guide 1.109, Appendix B, Table B-1.

The gamma air dose factors for various noble gas radionuclides are tabulated in Attachment 10.

Q'_{ir} = the total (time averaged) activity of noble gas radionuclide, i, in gaseous release, r (microcuries).

At CCNPP, all releases are considered long term releases.

Calculate the values of Q_{ir} for each release in accordance with equation 19G.

---

1 Equation 18G has been derived from NUREG-0133, 5.3.1..

2 NUREG-0133, 3.3

TOTAL (TIME AVERAGED) ACTIVITY OF NOBLE GAS NUCLIDE i IN GASEOUS RELEASE r

$$Q'_{ir} = (A_{ir})(F_r)(t_{ir})(c') \qquad \textbf{Eq. 19G}$$

A_{ir} = the specific activity of noble gas radionuclide, i, in release, r, discharged during the time interval of interest (microcuries/cubic centimeter)

F_r = the discharge flow rate for release, r, discharged during the time interval of interest (cubic meters per second)

If the discharge flow rate is unknown (e.g., the gaseous radwaste has not been released), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the average activity for nuclide i.

Whenever possible, the actual discharge flow rate determined from actual release conditions (e.g., initial pressure, volume, and temperature of a WGDT along with final pressure and temperature) shall be used in equation 19G.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

t_{ir} = the duration of the gaseous radwaste release (seconds)

c' = a conversion constant, 1E6 cubic centimeters per cubic meter, which represents the number of cubic centimeters per cubic meter.

---

(3) In the event a computer system is unavailable, a simplified equation may be used to calculate the gamma air dose due to noble gases in gaseous effluents released to **UNRESTRICTED AREAS**.

(4) The simplified method is presented below.

d) simplified method

(1) If a computer system and appropriate software are NOT available to perform the rigorous gamma air dose calculation described in the previous section, the gamma air dose, due to noble gas radionuclides, in any single release of waste gases discharged to **UNRESTRICTED AREAS** may be calculated in accordance with equation 20G.

GAMMA AIR DOSE DUE TO NOBLE GASES IN GAS RELEASE, r (SIMPLIFIED EQUATION)

$$D_{gr} = [(3.17E\text{-}8) (x/Q) (M_{avg}) / K_{sf}] \Sigma Q'_{ir} \qquad \text{Eq. 20G}^{1}$$

3.17E-8 = The conversion constant, 3.17E-8, represents the inverse of the number of seconds in a year.

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

M_{avg} = the empirically derived, site specific, average gamma air dose factor for each identified noble gas radionuclide, i (mrad/yr per microcurie/cubic meter)

A site-specific, average, gamma air dose factor has been calculated from historical data.

The calculation of this site-specific, average, gamma air dose factor is presented on Attachment 11 (use section 3.4.5 of the old ODCM.)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP gamma air dose limit to the gamma air dose limit of Control 3.11.2.2, (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the gamma air dose is always less than or equal to the gamma air dose limit of Control 3.11.2.2.

A safety factor of 1.00 will yield an gamma air dose which corresponds to the gamma air dose limit of Control 3.11.2.2.

A safety factor of 0.500 will yield an gamma air dose which corresponds to one-half the gamma air dose limit of Control 3.11.2.2.

It is recommended that a safety factor of 1.0 be used for calculating the gamma air dose, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive gaseous release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.2.2 when simplifying assumptions are used.

1 Equation 20G has been derived from NUREG-0133, 5.3.1.
2 NUREG-0133, 3.3

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 20G to account for any potential nonconservatism associated with applying the empirically derived gamma air dose factor, M_{avg}, to all radionuclides identified in the gaseous release. Such nonconservatism could conceivable be present whenever radionuclides having a gamma air dose factor greater than M_{avg} are present in a gaseous release.

Q'_{ir} = the total (time averaged) activity of noble gas radionuclide, i, in gaseous release, r (microcuries)

At CCNPP, all releases are considered long term releases.

Calculate the values of Q_{ir} for each release in accordance with equation 19G.

TOTAL (TIME AVERAGED) ACTIVITY OF NOBLE GAS NUCLIDE i IN GASEOUS RELEASE r

$$Q'_{ir} = (A_{ir})(F_r)(t_{lr})(c') \qquad \textbf{Eq. 19G}$$

A_{ir} = the specific activity of noble gas radionuclide, i, in release, r, discharged during the time interval of interest (microcuries/cubic centimeter).

F_r = the discharge flow rate for release, r, discharged during the time interval of interest (cubic meters per second).

If the discharge flow rate is unknown (e.g., the gaseous radwaste has not been released), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the average activity for nuclide i.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

t_{lr} = the duration of the gaseous radwaste release (seconds).

c' = a conversion constant, 1E6 cubic centimeters per cubic meter, which represents the number of cubic centimeters per cubic meter.

e) Once the calculations above have been completed, the calculation results are compared to the applicable limits and corrective actions are initiated as described below.

7. Corrective actions

a) CHEMISTRY SECTION surveillance procedures shall contain/and or reference administrative and/or Control limits for quarterly and yearly gamma air doses for gaseous effluents and shall specify corrective actions to be initiated when these limits are exceeded.

b) Refer to Control 3.11.2.2 for actions to be taken in the event the calculated cumulative gamma air doses exceed 10 mrads per calendar quarter or 20 mrads per calendar year.

CUMULATIVE BETA AIR DOSES DUE TO NOBLE GASES IN GASEOUS EFFLUENTS

1. Introduction

a) Appendix I to 10 CFR 50 specifies cumulative beta air dose limits associated with the release of radioactive materials to **UNRESTRICTED AREAS**.

b) Radiological effluent controls have been established to implement the requirements of 10 CFR 50, Appendix I.

c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

a) The cumulative beta air dose, due to noble gases in gaseous effluents released to **UNRESTRICTED AREAS**, shall be less than 20 mrads in any calendar quarter, and shall be less than 40 mrads in any calendar year (per Control 3.11.2.2)

b) The routine surveillances which are performed to verify compliance with these radiological effluent controls are described below.

3. Surveillance Requirement(s)

a) The cumulative beta air doses, for the current calendar quarter and the current calendar year, due to noble gases in gaseous effluents, shall be determined at least once every 31 days (Control 4.11.2.2).

b) The plant group(s) responsible for performing the required surveillance(s) are identified below.

4. Responsible Plant Organizations

a) The CHEMISTRY SECTION is responsible for calculating the cumulative beta air doses for the current calendar quarter and the current calendar year.

b) The CHEMISTRY SECTION calculates the cumulative beta air doses whenever the appropriate initiating conditions are present

c) These initiating conditions are contained in the following section.

5. Initiating Conditions

a) The cumulative beta air doses due to noble gases in gaseous effluents shall be determined at least once per 31 days (Control 4.11.2.2).

b) The cumulative beta air doses due to noble gases in gaseous effluents are calculated for each release of a WGDT.

c) The cumulative beta air doses due to noble gases in gaseous effluents are calculated for each vent of a containment building.

d) The cumulative beta air doses due to noble gases in gaseous effluents are calculated for each **PURGE** of a containment building.

e) The cumulative beta air doses due to noble gases in gaseous effluents are calculated at least weekly[1] for CONTINUOUS discharges from plant vent stacks.

f) The cumulative beta air doses due to noble gases in gaseous effluents are calculated for each discharge of combustion products resulting from the burning of contaminated oil.

g) The cumulative beta air doses due to noble gases in gaseous effluents are calculated for each **ABNORMAL AND/OR UNANTICIPATED RADIOACTIVE GAS RELEASE**[2].

h) Whenever the correct initiating conditions are present, the cumulative beta air doses shall be calculated as described below.

6. Calculation Methodology

a) The cumulative beta air doses (e.g., for the current calendar month, current calendar quarter, current calendar year, or previous 92 days) due to noble gases in gaseous effluents shall be calculated in accordance with equation 21G.

---

1 The frequency is controlled by the implementing procedure, and is based on plant conditions. Under no conditions shall the frequency be less than once per month (Controls 4.11.2.1.1 or 4.11.2.1.2, Table 4.11-2).

2 The criteria used to define ABNORMAL AND UNANTICIPATED GAS RELEASES may be found in CP-612 or CP-604.

CUMULATIVE BETA AIR DOSE FOR ALL GASEOUS RELEASES, r, DISCHARGED DURING TIME INTERVAL, t

$$D_{\beta t} = \Sigma D_{\beta r} \qquad \text{Eq. 21G}$$

Where,

$D_{\beta t}$ = the cumulative beta air dose (mrad) at the **SITE BOUNDARY** due to noble gas radionuclides contained in all gaseous radwaste discharged from the site during the time interval, t

$D_{\beta r}$ = the beta air dose (mrad) due to noble gas radionuclides contained in gaseous radwaste release, r, discharged from the site during the time interval of interest

Calculate the values of $D_{\beta r}$ for each gaseous release as described below.

---

b) At CCNPP, two methods exist for calculating $D_{\beta r}$ (the beta air dose at the **SITE BOUNDARY** due to noble gas radionuclides contained in a gaseous radwaste release, r, discharged from the site).

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

c) Rigorous method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, the cumulative beta air dose due to noble gases in gaseous effluents released to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 22G.

BETA AIR DOSE DUE TO NOBLE GASES IN GAS RELEASE, r (RIGOROUS EQUATION)

$$D_{\beta r} = (3.17E\text{-}8)(x/Q)\sum[(N_i)(Q'_{ir})]$$ **Eq. 22G**[1]

Where,

$D_{\beta r}$ = the beta air dose due to noble gas radionuclides contained in gaseous radwaste release, r, discharged from the site during the time interval of interest

3.17E-8 = The conversion constant, 3.17E-8, represents the inverse of the number of seconds in a year.

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

N_i = the air dose factor due to beta emissions for each identified noble gas radionuclide, i (mrad/yr per microcurie/cubic meter)

The beta air dose factors have been obtained from Regulatory Guide 1.109, Appendix B, Table B-1.

The beta air dose factors for various noble gas radionuclides are tabulated in Attachment 10 (Attachment 1 of old ODCM).

Q'_{ir} = the total (time averaged) activity of noble gas radionuclide, i, in gaseous release, r (microcuries).

At CCNPP, all releases are considered long term releases.

Calculate the values of Q_{ir} for each release in accordance with equation 19G.

---

1 Equation 22G has been derived from NUREG-0133, 5.3.1.
2 NUREG-0133, 3.3

TOTAL (TIME AVERAGED) ACTIVITY OF NOBLE GAS NUCLIDE i IN GASEOUS RELEASE r

$$Q'_{ir} = (A_{ir})(F_r)(t_{ir})(c') \qquad \text{Eq. 19G}$$

A_{ir} = the specific activity of noble gas radionuclide, i, in release, r, discharged during the time interval of interest (microcuries/cubic centimeter).

F_r = the discharge flow rate for release, r, discharged during the time interval of interest (cubic meters per second).

If the discharge flow rate is unknown (e.g., the gaseous radwaste has not been released), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the average activity for nuclide i.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

t_{ir} = the duration of the gaseous radwaste release (seconds).

c' = a conversion constant, 1E6 cubic centimeters per cubic meter.

---

(3) In the event a computer system is unavailable, a simplified equation may be used to calculate the gamma air dose due to noble gases in gaseous effluents released to **UNRESTRICTED AREAS**.

(4) The simplified method is presented below.

d) Simplified method

(1) If a computer system and the appropriate software are NOT available to perform the rigorous beta air dose calculation described in the previous section, the beta air dose resulting from a single release of waste gases discharged to **UNRESTRICTED AREAS** may be calculated in accordance with equation 23G.

BETA AIR DOSE DUE TO NOBLE GASES IN GAS RELEASE, r (SIMPLIFIED EQUATION)

$D_{\beta r}$	=	$[(3.17E\text{-}8) (x/Q) (N_{avg}) / K_{sf}] \Sigma Q'_{ir}$	**Eq. 23G**[1]

3.17E-8 = The conversion constant, 3.17E-8, represents the inverse of the number of seconds in a year.

x/Q = the highest calculated annual average relative concentration for any area at or beyond the **UNRESTRICTED AREA** boundary (2.2E-6 seconds per cubic meter)

All releases are considered "long-term" releases[2], and as such, the highest historical annual average dispersion factor, (x/Q), is used in the dose calculations.

The highest annual average dispersion factor (x/Q) is 2.2E-6 (UFSAR, 2.3.6.3) for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

The maximum annual average on-shore concentrations occur in the southeast sector at a distance of 1300 meters for purposes of routine, long-term concentrations (e.g., routine noble gas releases) (UFSAR, 2.3.6.3).

N_{avg} = the empirically derived, site specific, average beta air dose factor for each identified noble gas radionuclide, i (mrad/yr per microcurie/cubic meter)

A site-specific, average, beta air dose factor has been calculated from historical data.

The calculation of this site-specific, average, beta air dose factor is presented on Attachment 11.

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP beta air dose limit to the beta air dose limit of Control 3.11.2.2, (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the beta air dose is always less than or equal to the beta air dose limit of Control 3.11.2.2.

A safety factor of 1.00 will yield an organ dose which corresponds to the beta air dose limit of Control 3.11.2.2.

A safety factor of 0.500 will yield an beta air dose which corresponds to one-half the beta air dose limit of Control 3.11.2.2.

It is recommended that a safety factor of 1.0 be used for calculating the beta air dose, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

1 Equation 23G has been derived from NUREG-0133, 5.3.1.
2 NUREG-0133, 3.3

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive gaseous release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.2.2 when simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 23G to account for any potential nonconservatism associated with applying the empirically derived beta air dose factor, N_{avg}, to all radionuclides identified in the gaseous release. Such nonconservatism could conceivable be present whenever radionuclides having a beta air dose factor greater than N_{avg} are present in a gaseous release.

Q'_{ir} = the total (time averaged) activity of noble gas radionuclide, i, in gaseous release, r (microcuries)

At CCNPP, all releases are considered long term releases.

Calculate the values of Q_{ir} for each release in accordance with equation 19G.

TOTAL (TIME AVERAGED) ACTIVITY OF NOBLE GAS NUCLIDE i IN GASEOUS RELEASE r

$$Q'_{ir} = (A_{ir})(F_r)(t_{ir})(c') \quad \textbf{Eq. 19G}$$

A_{ir} = the specific activity of noble gas radionuclide, i, in release, r, discharged during the time interval of interest (microcuries/cubic centimeter)

F_r = the discharge flow rate for release, r, discharged during the time interval of interest (cubic meters per second)

If the discharge flow rate is unknown (e.g., the gaseous radwaste has not been released), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the average activity for nuclide i.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

t_{ir} = the duration of the gaseous radwaste release (seconds).

c' = a conversion constant, 1E6 cubic centimeters per cubic meter.

---

e) Once the calculations above have been completed, the calculation results are compared to the applicable limits and corrective actions are initiated as described below.

7. Corrective actions

a) CHEMISTRY SECTION surveillance procedures shall contain/and or reference administrative and/or Control limits for quarterly and yearly beta air doses for gaseous effluents and shall specify corrective actions to be initiated when these limits are exceeded.

b) Refer to Control 3.11.2.2 for actions to be taken in the event the calculated cumulative beta air doses exceed 20 mrads per calendar quarter or 40 mrads per calendar year.

CUMULATIVE ORGAN DOSES DUE TO IODINES AND PARTICULATES IN GASEOUS EFFLUENTS

1. Introduction

a) Appendix I to 10 CFR 50 specifies cumulative organ dose limits associated with the release of radioactive materials to **UNRESTRICTED AREAS**.

b) Radiological effluent controls have been established to implement the requirements of 10 CFR 50, Appendix I.

c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

a) The cumulative organ dose due to iodines and particulates in gaseous effluents released to **UNRESTRICTED AREAS** shall be less than 15 mrems per calendar quarter, and shall be less than 30 mrems per calendar year (per Control 3.11.2.3).

b) The cumulative organ dose due to iodines and particulates in gaseous, contaminated oil combustion products released to **UNRESTRICTED AREAS** shall be less than 0.015 mrems per quarter, and shall be less than 0.030 mrems per year (per Control 3.11.2.3).

c) The routine surveillances which are performed to verify compliance with this radiological effluent controls are described below.

3. Surveillance Requirements

a) The cumulative organ doses (due to iodines and particulates in gaseous waste discharged to **UNRESTRICTED AREAS**), for the current calendar month, the current calendar quarter, and the current calendar year, shall be determined at least once every 31 days in accordance with the ODCM (per Control 4.11.2.3).

b) The plant group(s) responsible for performing the required surveillances are identified below.

4. Responsible Plant Organizations

a) The CHEMISTRY SECTION is responsible for implementing the surveillances required by Control 4.11.2.3.

b) The CCNPP CHEMISTRY SECTION calculates the cumulative organ doses whenever the appropriate initiating conditions are present

c) These initiating conditions are contained in the following section.

5. Initiating Conditions

a) The cumulative organ dose--for each organ--shall be determined at least once per 31 days (Control 4.11.2.2).

b) The cumulative organ dose--for each organ--due to iodines and particulates in gaseous effluents shall be calculated at least weekly[1] for CONTINUOUS discharges from plant vent stacks.

c) The cumulative organ dose--for each organ--due to iodines and particulates in gaseous effluents shall be calculated for each discharge of combustion products resulting from the burning of contaminated oil.

d) The cumulative organ dose--for each organ--due to iodines and particulates in gaseous effluents shall be calculated for each **ABNORMAL AND/OR UNANTICIPATED RADIOACTIVE GAS RELEASE**.

e) Whenever the correct initiating conditions are present, the annual cumulative organ doses shall be calculated as described below.

6. Calculation Methodology

a) The cumulative organ doses (for the calendar month, calendar quarter, previous 92 days, and calendar year) due to iodines and particulates in gaseous waste discharged to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 24G.

---

1 The frequency is controlled by the implementing procedure, and is based on plant conditions. Under no conditions shall the frequency be less than once per month (Controls 4.11.2.1.1 or 4.11.2.1.2, Table 4.11-2).

CUMULATIVE DOSE TO ORGAN, o, FROM ALL GASEOUS RELEASES, r, DISCHARGED DURING TIME INTERVAL, t

$$D_{ot} = \Sigma D_{or} \qquad \textbf{Eq. 24G}$$

Where,

D_{ot} = the cumulative dose (mrad) to organ, o, at the **SITE BOUNDARY**, due to iodine and particulate radionuclides contained in gaseous waste discharged from the site during the time interval, t

D_{or} = the dose (mrad) to organ, o, at the **SITE BOUNDARY** due to iodine and particulate radionuclides in gaseous release, r, discharged from the site during the time interval of interest

Calculate the values of D_{or} for each gaseous release as described below.

---

b) At CCNPP, two methods exist for calculating D_{or} (the organ doses due to iodines and particulates resulting from any single release of radioactive gases to an **UNRESTRICTED AREA**).

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

c) Rigorous method

(1) Application of the following equation may prove too rigorous for routine use unless a computer system and the appropriate software are available.

(2) If a computer system and the appropriate software are available, the organ doses due to iodines and particulates contained in any single release of radioactive gases to **UNRESTRICTED AREAS** shall be calculated in accordance with equation 25G.

DOSE TO ORGAN, o, DUE TO IODINES AND PARTICULATES IN GAS RELEASE, r (RIGOROUS EQUATION)

D_{or}	=	$(3.17E\text{-}8)(W_v)\sum(R_{ipao})(Q'_{ir})$	Eq. 25G[1]

Where,

D_{or} = the dose (mrem) to organ, o, at the **SITE BOUNDARY** due to iodine and particulate radionuclides in gaseous release, r, discharged from the site during the time interval of interest

3.17E-8 = The conversion constant, 3.17E-8, represents the inverse of the number of seconds in a year.

W_v = the dispersion parameter for estimating the dose to an individual at the controlling location for long term releases, and may assume one of two values as described below

W_v is x/Q for the inhalation pathway (2.2E-6 sec/cubic meter)

W_v is D/Q for the food and ground plane pathways (meters^{-2})

D/Q = the dispersion parameter at the controlling location for long term releases (meters^{-2})

The value for D/Q has been determined to be 8.63E-10 m^{-2}.[2]

The grass-cow-milk pathway is the controlling pathway.[3]

The controlling sector is the south-southwest sector.

The controlling location is at a distance of 4800 meters.[3]

R_{ipao} = the dose factor for each identified iodine or particulate radionuclide, i, exposure pathway, p, receptor age group, a, and organ, o (m^2 mrem/year per microcuries/second or mrem/year per microcuries/cubic meter)

dose factors have been derived for the following pathways

1) inhalation - see Attachment 12
2) ground plane - see Attachment 12
3) grass-cow-milk - see Attachment 12
4) grass-cow-meat - see Attachment 12
5) vegetation - see Attachment 12

The inhalation pathway dose factors were obtained using the formula from NUREG-0133, 5.3.1.1.

The ground plane dose factors were obtained using the formula from NUREG-0133, 5.3.1.2.

---

1 Equation 25G has been derived from NUREG-0133, 5.3.1.
2 See CP-607, Revision 2 section 3.4.3.
3 See the "Land Use Survey", 1990.

The grass-cow-milk pathway dose factors were obtained using the formula from NUREG-0133, 5.3.1.3.

The grass-cow-meat pathway dose factors were obtained using the formula from NUREG-0133, 5.3.1.4.

The vegetation pathway dose factors were obtained using the formula from NUREG-0133, 5.3.1.5.

Q'_{ir} = the total (time averaged) activity of iodine or particulate radionuclide, i, in gaseous release, r, discharged during the specified time interval (microcuries)

At CCNPP, all releases are considered long term releases.

Calculate the values of Q_{ir} for each release in accordance with equation 19G.

TOTAL (TIME AVERAGED) ACTIVITY OF IODINE OR PARTICULATE NUCLIDE i IN GASEOUS RELEASE r

$$Q'_{ir} = (A_{ir})(F_r)(t_{ir})(c') \qquad \textbf{Eq. 19G}$$

A_{ir} = the specific activity of iodine and particulate radionuclide, i, in release, r, discharged during the time interval of interest (microcuries/cubic centimeter)

F_r = the discharge flow rate for release, r, discharged during the time interval of interest (cubic meters per second)

If the discharge flow rate is unknown (e.g., the gaseous radwaste has not been released), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the average activity for nuclide i.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

t_{ir} = the duration of the gaseous radwaste release (seconds)

c' = a conversion constant, 1E6 cubic centimeters per cubic meter

---

(3) In the event a computer system and the appropriate software are unavailable, a simplified equation may be used to calculate the organ doses due to individual gaseous releases.

(4) The simplified method is presented below.

d) simplified method

(1) If a computer system and appropriate software are NOT available to perform the rigorous organ dose calculations described in the previous section, the organ doses due to iodines and particulates in a single release of radioactive gases discharged to an **UNRESTRICTED AREA** may be calculated in accordance with equation 26G.

DOSE TO ORGAN, o, FROM IODINES AND PARTICULATES IN GAS RELEASE, r (SIMPLIFIED EQUATION)

$D_{maxor} = [(3.17E\text{-}8) (W_v) (R_{I\text{-}131}) / K_{sf}] \Sigma (Q'_{ir})$	Eq. 26G[1]

D_{maxor} = the maximum dose to any organ, o, due to iodines and particulates contained in any single release, r, of radioactive gases to an **UNRESTRICTED AREA**

3.17E-8 = The conversion constant, 3.17E-8, represents the inverse of the number of seconds in a year.

D/Q = the dispersion parameter at the controlling location for long term releases (meters^{-2})

The value for D/Q has been determined to be 8.63E-10 m^{-2}.[2]

The grass-cow-milk pathway is the controlling pathway.[3]

The controlling sector is the south-southwest sector.

The controlling location is at a distance of 4800 meters.[3]

$R_{I\text{-}131}$ = the infant, thyroid, dose factor for I-131 via the grass-cow-milk pathway (m^2 mrem/year per microcuries/second)

This value is 1.05E12 and it is listed on Attachment 12.

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP organ dose limit to the organ dose limit of Control 3.11.2.3, (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the organ dose is always less than or equal to the organ dose limit of Control 3.11.2.3.

A safety factor of 1.00 will yield an organ dose which corresponds to the organ dose limit of Control 3.11.2.3.

A safety factor of 0.500 will yield an organ dose which corresponds to one-half the organ dose limit of Control 3.11.2.3.

It is recommended that a safety factor of 1.0 be used for calculating the organ dose, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive gaseous release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.2.3 when simplifying assumptions are used.

---

1 Equation 26G has been derived from NUREG-0133, 5.3.1.
2 See CP-607, Revision 2 section 3.4.3.
3 See the "Land Use Survey", 1990.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 26G to account for any potential nonconservatism associated with applying the infant, thyroid, grass-cow-milk dose factor, R_{I-131}, to all radionuclides identified in the gaseous release. Such nonconservatism could conceivable be present whenever radionuclides having a pathway dose factor greater than R_{I-131} are present in a gaseous release.

Q'_{ir} = the total (time averaged) activity of iodine or particulate radionuclide, i, in gaseous release, r (microcuries)

At CCNPP, all releases are considered long term releases.

This value shall be calculated in accordance with equation 19G.

TOTAL (TIME AVERAGED) ACTIVITY OF IODINE OR PARTICULATE NUCLIDE i IN GASEOUS RELEASE r

$$Q'_{ir} = (A_{ir})(F_r)(t_{Ir})(c') \qquad \textbf{Eq. 19G}$$

A_{ir} = the specific activity of iodine and particulate radionuclide, i, in release, r, discharged during the time interval of interest (microcuries/cubic centimeter)

F_r = the discharge flow rate for release, r, discharged during the time interval of interest (cubic meters per second)

If the discharge flow rate is unknown (e.g., the gaseous radwaste has not been released), the "Maximum Discharge Flow Rate" listed on Attachments 7 or 8 may be used to calculate the average activity for nuclide i.

Additional guidance for calculating discharge flow rates may be contained in approved CHEMISTRY SECTION procedures.

t_{Ir} = the duration of the gaseous radwaste release (seconds)

c' = a conversion constant, 1E6 cubic centimeters per cubic meter

e) Once the calculations above have been completed, the calculation results are compared to the applicable limits and corrective actions are initiated as described below.

7. Corrective actions

a) CHEMISTRY SECTION surveillance procedures shall contain/and or reference administrative and/or Control limits for cumulative organ dose for gaseous effluents and shall specify corrective actions to be initiated when these limits are exceeded.

b) Refer to Control 3.11.2.3 for actions to be taken in the event the calculated cumulative gamma air doses exceed any of the radiological effluent controls listed above.

LIMITS FOR THE GASEOUS RADWASTE PROCESSING SYSTEM

1. Introduction

   a) 10 CFR 50.36a requires licensees to maintain and use the equipment installed in the gaseous waste processing system for the purpose of controlling effluents to the environment.

   b) Radiological effluent controls have been established to implement the requirements of 10 CFR 50.36a.

   c) These radiological effluent controls are described below.

2. Radiological effluent controls

   a) The **GASEOUS RADWASTE PROCESSING SYSTEM** and the **VENTILATION EXHAUST PROCESSING SYSTEM** shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the gaseous effluent air dose, to areas at and beyond the **SITE BOUNDARY**, exceeds 1.20 mrads gamma radiation in a 92 day period (per Control 3.11.2.4).

   b) The **GASEOUS RADWASTE PROCESSING SYSTEM** and the **VENTILATION EXHAUST PROCESSING SYSTEM** shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the gaseous effluent air dose, to areas at and beyond the **SITE BOUNDARY**, exceeds 2.4 mrads beta radiation in a 92 day period (per Control 3.11.2.4).

   c) The **VENTILATION EXHAUST PROCESSING SYSTEM** shall be used to reduce the quantity of radioactive materials in gaseous waste prior to their discharge when the calculated doses due to gaseous effluent releases, to areas at and beyond the **SITE BOUNDARY** exceeds 1.80 mrem to any organ in a 92 day period (per Control 3.11.2.4).

   d) The routine surveillances which are performed to verify compliance with this radiological effluent controls are described below.

3. Surveillance Requirement(s)

   a) The cumulative gamma air dose, for the previous 92 days, due to noble gases in gaseous effluents, shall be determined at least once every 31 days (Control 4.11.2.2).

   b) The plant group(s) responsible for performing the required surveillance(s) are identified below.

4. Responsible Plant Organizations

   a) The CHEMISTRY SECTION is responsible for calculating the cumulative gamma air doses for the current calendar month, the previous 92 days, the current calendar quarter, and the current calendar year.

b) The cumulative gamma air dose for the previous 92 days is calculated whenever the appropriate initiating conditions are present

c) These initiating conditions are contained in the following section.

5. Initiating conditions

a) For a listing of initiating conditions associated with calculating gamma air doses, see "Initiating Conditions" in the section of the ODCM titled, "Cumulative Gamma Air Doses Due To Noble Gases In Gaseous Effluents."

b) For a listing of initiating conditions associated with calculating beta air doses, see "Initiating Conditions" in the section of the ODCM titled, "Cumulative Beta Air Doses Due To Noble Gases In Gaseous Effluents."

c) For a listing of initiating conditions associated with calculating cumulative organ doses, see "Initiating Conditions" in the section of the ODCM titled, "Cumulative Organ Doses Due To Iodines And Particulates In Gaseous Effluents."

6. Calculation methodology

a) Calculate the previous 92-day cumulative gamma air dose as described in the section "Cumulative Gamma Air Doses Due To Noble Gases In Gaseous Effluents."

b) Calculate the previous 92-day cumulative beta air dose as described in the section "Cumulative Beta Air Doses Due To Noble Gases In Gaseous Effluents."

c) Calculate the previous 92-day cumulative organ dose as described in the section "Cumulative Organ Doses Due To Iodines And Particulates In Gaseous Effluents."

7. Corrective actions

a) CHEMISTRY SECTION surveillance procedures shall contain/and or reference administrative and/or Control limits for 92-day cumulative gamma, beta, or organ doses for gaseous effluents and shall specify corrective actions to be initiated when these limits are exceeded.

b) Refer to Control 3.11.2.4 for actions to be taken in the event the calculated 92-day cumulative gamma air, beta air, or organ doses exceed any of the radiological effluent controls listed above.

LIMITS ON TOTAL ANNUAL DOSE -- GASES, LIQUIDS, AND URANIUM FUEL CYCLE SOURCES

1. Introduction

   a) 40 CFR 190 specifies annual dose limits for radionuclides released to the environment.

   b) Radiological effluent controls have been established to implement the requirements of 40 CFR 190.

   c) These radiological effluent controls are described below.

2. Radiological effluent controls

   a) The total body dose from exposure to the combination of liquid releases, gas releases, and uranium fuel cycle sources shall be less than 25 mrem for the current calendar year (per Control 3.11.4).

   b) The organ dose (for the maximum exposed organ, not including the thyroid) from exposure to the combination of liquid releases, gas releases, and uranium fuel cycle sources shall be less than 25 mrem for the current calendar year (per Control 3.11.4).

   c) The thyroid dose from exposure to the combination of liquid releases, gas releases, and uranium fuel cycle sources shall be less than 75 mrem for the current calendar year (per Control 3.11.4).

   d) The routine surveillances which are performed to verify compliance with these radiological effluent controls are described below.

3. Surveillance Requirements

   a) The cumulative gamma air doses, for current calendar month, the current calendar quarter, and the current calendar year, due to noble gases in gaseous effluents, shall be determined at least once every 31 days (Control 4.11.2.2).

   b) The cumulative organ doses (due to iodines and particulates in gaseous waste discharged to **UNRESTRICTED AREAS**), for the current calendar month, the current calendar quarter, and the current calendar year, shall be determined at least once every 31 days in accordance with the ODCM (per Control 4.11.2.3).

   c) Cumulative total body dose to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--for the current calendar month, the calendar quarter, and the current calendar year--shall be calculated at least once per 31 days (per Control 4.11.1.2).

   d) Cumulative organ doses to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--for the current calendar month, the current calendar quarter, and the current calendar year--shall be calculated at least once per 31 days (per Control 4.11.1.2).

   e) The direct radiation dose to **MEMBERS OF THE PUBLIC** exposed to uranium fuel cycle sources (i.e., reactor units and outside storage tanks) shall be determined IF THE APPROPRIATE INITIATING CONDITIONS ARE PRESENT.

4. Responsible Plant Organization(s)

a) The CHEMISTRY SECTION is responsible for implementing the effluent surveillances required by Control 4.11.4.1.

b) The CHEMISTRY SECTION is responsible for ensuring implementation of the direct radiation surveillances required by Control 4.11.4.2.

c) IT SHOULD BE NOTED THAT NO SURVEILLANCES NEED BE PERFORMED UNLESS THE APPROPRIATE INITIATING CONDITIONS ARE PRESENT.

d) These initiating conditions are contained in the following section.

5. Initiating conditions

a) The total dose from liquid releases, gas releases, and uranium fuel cycle sources shall be determined whenever the calculated doses from **liquid effluents** exceed any of the following (per Control 4.11.4.2):

(1) Six (6) mrem per quarter to the total body

(2) Twelve (12) mrem per calendar year to the total body

(3) Twenty (20) mrem per quarter to any organ

(4) Forty (40) mrem per calendar year to any organ

b) The total dose from liquid releases, gas releases, and uranium fuel cycle sources shall be determined whenever the calculated air doses from noble gasses in **gaseous effluents** exceed any of the following (per Control 4.11.4.2):

(1) Twenty (20) mrad gamma per quarter

(2) Forty (40) mrad gamma per calendar year

(3) Forty (40) mrad beta per quarter

(4) Eighty (80) mrad beta per calendar year

c) The total dose from liquid releases, gas releases, and uranium fuel cycle sources shall be determined whenever the calculated organ doses from iodines and particulates in **gaseous effluents** exceed any of the following (per Control 4.11.4.2):

(1) Thirty (30) mrem per quarter to any organ

(2) Sixty (60) mrem per calendar year to any organ

d) Whenever the correct initiating conditions are present, the total doses from liquid releases, gas releases, and uranium fuel cycle sources (for the calendar year) shall be calculated as shown below.

6. Calculation methodology

a) The total body dose and the organ doses from liquid releases, gas releases, and uranium fuel cycle sources (for the calendar year) shall be calculated in accordance with equation 1T and 2T respectively.

TOTAL, TOTAL BODY DOSE FROM LIQUID RELEASES, GAS RELEASES, AND URANIUM FUEL CYCLE SOURCES

$D_{tball} = D_{ToL} + D_{gt} + D_{tank}$	**Eq. 1T**

TOTAL ORGAN DOSES FROM LIQUID RELEASES, GAS RELEASES, AND URANIUM FUEL CYCLE SOURCES

$D_{oall} = D_{ToL} + D_{ot} + D_{tank}$	**Eq. 2T**

D_{tball} = the dose (mrem) to total body resulting from the combination of all gas releases, all liquid releases, and all uranium fuel cycle sources.

D_{oall} = the dose (mrem) to organ, o, resulting from the combination of all gas releases, all liquid releases, and all uranium fuel cycle sources.

Separate values shall be calculated for each of the organs listed below:

1. bone
2. liver
3. thyroid
4. kidney
5. lung
6. GI tract

D_{ToL} = the cumulative dose (mrem) to organ, o, for all liquid releases discharged in a given time interval

Calculate this value as specified by equation 8L.

D_{gt} = the site-boundary cumulative gamma air dose (mrad) due to noble gas radionuclides contained in all gaseous radwaste discharged from the site during the time interval, t

Calculate this value as specified by equation 17G, except substitute K_i for M_i (see Attachment 10).

D_{ot} = the site-boundary cumulative organ dose (mrem) resulting from the release of iodine and particulate radionuclides in gaseous releases from the site

Calculate this value as specified by equation 24G.

D_{tank} = the calendar-year cumulative dose (mrem) to the maximum exposed MEMBER OF THE PUBLIC due to direct radiation from the reactor units and outside storage tanks

This value shall be based on the results of direct radiation measurements from TLDs or continuous dose rate instruments placed near the **SITE BOUNDARY** (e.g., from radiological environmental monitoring sites DR1-DR9 described on Attachment 13 and shown on Attachment 18).

The CHEMISTRY SECTION, and the Radiation Safety Section are responsible for determining this value.

b) Compare the calculated values to the radiological effluent controls (listed in this section), and if any of the radiological effluent controls have been exceeded, perform the appropriate corrective actions listed below.

7. Corrective actions

a) CHEMISTRY SECTION surveillance procedures shall contain/and or reference administrative and/or Control limits for total dose for liquid releases, gaseous releases, and uranium fuel cycle sources and shall specify corrective actions to be initiated when these limits are exceeded.

b) Refer to Control 3.11.4 for actions to be taken in the event the total dose exceeds any of the radiological effluent controls listed above.

c) If any of the radiological effluent controls have been exceeded, refer to 40 CFR 302, Appendix B, and verify the quantities of radioactive materials released are less than the values specified. **[375B]**