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PREPARATION OF
RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS
FOR NUCLEAR POWER PLANTS

A Guidance Manual for Users of Standard Technical Specifications

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TABLE OF CONTENTS

CHAPTER 1

INTRODUCTION

1.1 PURPOSE

The purpose of this manual is to describe methods found acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for the calculation of certain key values required in the preparation of proposed radiological effluent Technical Specifications using the Standard Technical Specifications for light-water-cooled nuclear power plants. This manual also provides guidance to applicants for operating licenses for nuclear power plants in the preparation of proposed radiological effluent Technical Specifications or in preparing requests for changes to existing radiological effluent Technical Specifications for operating licenses. The manual additionally describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment systems.

1.2 BACKGROUND

Section 50.36, "Technical Specifications," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities" (Ref. 1), requires that each nuclear power reactor operating license issued by the NRC contain Technical Specifications that set forth limits, operating conditions, and other regulatory requirements imposed on the facility operation for the protection of the health and safety of the public. Conditions and limitations corresponding to certain key values which are system-dependent and site-related are to be incorporated in these Technical Specifications for compliance with 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors." Under the provisions of Section 50.36, each applicant for an operating license is required to submit proposed Technical Specifications for his facility, including the supporting bases. These are reviewed by the NRC staff to assure that the proposed Technical Specifications contain such conditions and limitations as deemed appropriate and necessary; approved Technical Specifications are then included as Appendix A of the operating license.

Standard Technical Specifications have been developed by the staff for each appropriate nuclear steam supply system (NSSS) vendor to provide guidance to applicants for the preparation of proposed Technical Specifications. These are as follows:

NUREG-0212 - "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors"

NUREG-0103 - "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors"

NUREG-0452 - "Standard Technical Specifications for Westinghouse Pressurized Water Reactors"

NUREG-0123 - "Standard Technical Specifications for General Electric Boiling Water Reactors"

Current Standard Technical Specifications are available from the National Technical Information Service (NTIS), Springfield, Virginia, 22161. These Standard Technical Specifications will contain the radiological effluent Technical Specifications to be used by the applicant for an operating license. In the interim, model radioactive effluent Technical Specifications have been provided in NUREG-0472 (Ref. 2) for pressurized water reactors, and NUREG-0473 (Ref. 3) for boiling water reactors. Table 1.1 provides a summary of those applicable sections in the Standard Technical Specifications which contain conditions and limitations relative to the radiological effluent Technical Specifications to be discussed in this manual.

The Standard Technical Specifications contain the limiting conditions for operation necessary for complying with the Commission's regulations and are in a format that is acceptable to the NRC staff. The reporting requirements reflect the guidelines provided in Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications," Revision 4, August 1975 (Ref. 4) and Regulatory Guide 10.1, "Compilation of Reporting Requirements for Persons Subject to NRC Regulations," Revision 3, May 1977 (Ref. 5).

The methodology discussed in this manual and used to implement the requirements of 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Practicable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents" (Ref. 1), is consistent with the Regulatory Guides used in the staff's safety evaluations pursuant to 10 CFR 50.34a(c). These guides are as follows:

Regulatory Guide 1.109 - "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (Revision 1), October 1977. (Ref. 6)

Regulatory Guide 1.110 - "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," March 1976. (Ref. 7)

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. (Ref. 8)

Regulatory Guide 1.112 - "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," April 1976. (Ref. 9)

TABLE 1.1

SUMMARY OF APPLICABLE SECTIONS IN THE STANDARD TECHNICAL SPECIFICATIONS CONSIDERED IN THIS MANUAL

PLANT TYPE	Standard Technical Specification or Model Technical Specification in NUREG-0472 (Ref. 2) or NUREG-0473 (Ref. 3) SECTION	TABLES	BASES	TITLE	SECTION REFERENCE IN THIS MANUAL
PWR BWR	1.0	Yes	No	Definitions	2.0
PWR BWR	3/4.3.3.8	Yes	Yes	Radioactive Liquid Effluent Instrumentation	4.1
PWR BWR	3/4.3.3.9	Yes	Yes	Radioactive Gaseous Effluent Instrumentation	5.1
PWR BWR	3/4.11.1.1	Yes	Yes	Liquid Effluents, Concentration	4.2
PWR BWR	3/4.11.1.2	No	Yes	Liquid Effluents, Dose	4.3
PWR BWR	3/4.11.1.3	No	Yes	Liquid Effluents, Liquid Waste Treatment	4.5
PWR BWR	3/4.11.1.4	No	Yes	Liquid Effluents, Liquid Holdup Tanks	4.4
PWR BWR	3/4.11.2.1	Yes	Yes	Gaseous Effluents, Dose Rate	5.2
PWR BWR	3/4.11.2.2	No	Yes	Dose, Noble Gases	5.3
PWR BWR	3/4.11.2.3	No	Yes	Dose Radioiodines, Radioactive Material in Particulate Form and Radionuclides Other than Noble Gases	5.3
PWR BWR	3/4.11.2.4	No	Yes	Gaseous Effluents, Gaseous Waste Treatment	5.4
PWR BWR	3/4.11.2.5	No	Yes	Gaseous Effluents, Dose	3.8
PWR BWR	3/4.11.2.6A	No	Yes	Explosive Gas Mixtures (Systems designed to withstand a hydrogen explosion)	5.5
PWR BWR	3/4.11.2.6B	No	Yes	Explosive Gas Mixtures (Systems not designed to withstand a hydrogen explosion)	5.5
PWR	3/4.11.2.7	No	Yes	Gaseous Effluents, Gas Storage Tanks	5.6
BWR	3/4.11.2.7	No	Yes	Gaseous Effluents, Main Condenser	5.6
BWR	3/4.11.3.8	No	Yes	Gaseous Effluents, Mark I or II Containment (Optional)	3.0
PWR BWR	3/4.11.3.1	No	Yes	Solid Radioactive Waste	3.0
PWR BWR	3/4.12.1	Yes	Yes	Monitoring Program	4.3, 5.3
PWR BWR	3/4.12.2	No	Yes	Land Use Census	4.3, 5.3
PWR BWR	Fig. 3.11-1	No	No	Unrestricted Area Boundary for Liquid Effluents	2.0
PWR BWR	Fig. 5.1-1	No	No	Unrestricted Area Boundary for Gaseous Effluents	2.0
PWR BWR	6.9	No	No	Reporting Requirements	General

Regulatory Guide 1.113 - "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I" (Revision 1), April 1977. (Ref. 10)

Computer codes used to determine certain values and parameters used with the Regulatory Guides listed above are described in the following documents:

NUREG-0017 - "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1976 (Ref. 11)

NUREG-0016 - "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)," April 1976 (Ref. 12)

NUREG-0324 - "XOQDOQ, Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations," September 1977 (Ref. 13)

These guides and computer codes provide acceptable methods of complying with the Commission's regulations; however, conformance with the staff's guidelines on Standard Technical Specifications, Regulatory Guides, and dose calculation methodology is not required. If the proposed Technical Specifications include mathematical models and parameters that differ from the methodology used by the staff to calculate set points and release rates or estimate doses due to releases of radioactive materials in effluents, the parameters and calculations used shall be substantiated in the Offsite Dose Calculation Manual.

Based on the findings of the Environmental and Safety Hearings, the Commission may impose certain additional or alternative limiting conditions for operation based on values that are more restrictive than those determined using this manual or provided in the Standard Technical Specifications. In such cases, the limiting conditions will be based on the decisions of the Atomic Safety and Licensing Board rather than the limiting values determined by this manual or included in the Standard Technical Specifications.

CHAPTER 2

DEFINITIONS AND STAFF POSITIONS

2.1 DEFINITIONS

Section 1.0, DEFINITIONS, of the Standard Technical Specifications provides standard definitions of terms and phrases which appear capitalized throughout the specifications. Standard definitions are provided to assure licensing consistency. When a term or phrase is used in a limited subject area of the Standard Technical Specifications, it is defined in the limited subject area and referenced by Specification number, table, figure or footnote.

2.2 STAFF POSITIONS AUGMENTING STANDARD DEFINITIONS

In certain circumstances, terms used in the Standard Technical Specifications are defined or specified in applicable regulations, such as 10 CFR Part 50 (Ref. 1). Staff positions clarifying certain of these definitions are as discussed below.

LIMITING CONDITIONS FOR OPERATION (LCO) is defined in 10 CFR 50.36(c)(2) as "the lowest functional capability or performance levels of equipment required for safe operation of the facility." This definition is applicable to the components of the radioactive waste management systems during normal reactor operations, including anticipated operational occurrences. When an LCO for a nuclear power reactor is not met, the licensee shall either shut down the reactor or follow any remedial action permitted by the Technical Specifications until the LCO can be met. Remedial action by the licensee may include processing by normal or alternate modes of operation for the control of radioactive effluents using such existing equipment as may be installed in the radioactive waste management systems.

MAINTENANCE AND USE of the equipment installed in the radioactive waste management systems is required in 10 CFR 50.34a(c) and in 10 CFR 50.36a(a)(1). The term, MAINTENANCE AND USE, is applicable to the installed components of the liquid, gaseous, and solid radioactive waste management systems and to instrumentation installed for the monitoring and control of potentially radioactive effluents. MAINTENANCE AND USE does not require the installation of fully redundant systems; however, prudent management procedures, such as scheduled standby and maintenance periods should be employed. The Standard Technical Specifications specify levels or values above which equipment installed in the radioactive waste management systems shall be used to enable the licensee to show that he is exerting his best efforts to maintain levels of radioactive effluents "as low as is reasonably achievable," in accordance with 10 CFR 50.36a.

UNRESTRICTED AREA is defined in 10 CFR 20.3(a)(17) (Ref. 14), as "any area access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential quarters."

For purposes of implementation, the definition of UNRESTRICTED AREA has been expanded as follows: "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, and any area within the site boundary used for residential quarters or industrial, commercial, institutional and recreational facilities". The UNRESTRICTED AREA boundary may coincide with the exclusion (fenced) area boundary, as defined in 10 CFR 100.3(a) (Ref. 15), may include land areas owned by the licensee, provided that occupancy is controlled by the licensee for the purposes of meeting the requirements of 10 CFR Part 20, but does not include areas over water bodies.

To assure that the UNRESTRICTED AREA boundary is defined in each license, the Standard Technical Specifications require two maps, one for liquid effluents (Figure 3.11-1)* and one for gaseous effluents (Figure 5.1-1)* locating all points surrounding the facility at which the licensee shall comply with the expanded definition, given above. These boundaries shall be consistent with those established in the Safety Analysis Report, or the Final Hazard Summary for the facility. The UNRESTRICTED AREAS, established at or beyond these boundaries, are also considered in the LIMITING CONDITIONS FOR OPERATION to keep levels of radioactive materials in effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

RELEASE RATE used in this manual is defined as, "the discharge of radioactive materials in liquid or gaseous effluents per unit time." The "second" is used as the practical reporting time unit for establishing release rates to show compliance with the requirements of 10 CFR Part 20 and for establishing instantaneous limitations based on potentially radioactive releases. The "hour" is used as the practical reporting time unit in establishing average release rates to show conformance with the requirements of 10 CFR Part 50 for noble gas releases, for gaseous radioactive effluents other than noble gases (radioiodines and particulates) and for radioactive materials released in liquid effluents. Liquid releases are further subdivided into batch and continuous releases. Gaseous releases are subdivided into short- and long-term releases. These gaseous release subdivisions classify cumulative releases as being either less than or greater than 500 hrs/year, respectively, for gaseous effluents. Further discussion is provided in Sections 3.3, 4.2 and 5.2 of this manual.

RADIOACTIVE WASTE MANAGEMENT SYSTEMS are defined as all process and control equipment provided to reduce the amount or concentration of radioactive materials (in any form) released from the facility. The overall systems may be divided into subsystems to handle the radioactive materials contained in liquid and gaseous streams and in solid waste. The Standard Technical Specifications have adopted nomenclature for systems and components which are in common use in the industry. In preparing proposed Technical Specifications, the system and component names may be changed to correspond to the terminology used in the Final Safety Analysis Report (FSAR) or the Final Hazard Summary, if applicable. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be radioactive waste management system components and, therefore, are not within the scope of this manual.

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*This Figure is to be included in proposed Technical Specifications.

CHAPTER 3

SPECIAL CONSIDERATIONS

3.1 *MULTI-UNIT SITES WITH SHARED RADIOACTIVE WASTE MANAGEMENT SYSTEMS*

The Standard Technical Specifications are written on a "per unit" basis, since this is the format in which operating licenses are issued. When shared radioactive waste management systems are used by more than one reactor unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific reactor unit. The licensee should estimate the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing reactors sharing the treatment system. For determining conformance to LCOs, these allocations from shared treatment systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

It is preferred that discharge lines leading to common release points be separately controlled and monitored, and the licensee should consider instrumentation which will provide volumetric recording and radiological effluent monitoring, sampling and analyses on a unit basis. This has been accomplished for some units by continuous representative sampling and analysis of the streams prior to mixing and continuously monitored and controlled at the common release point. Multi-unit sites with common release points, but without shared radioactive waste management systems, are also required to determine the alarm/trip setpoints in Sections 4.1 and 5.1 of this manual to assure compliance with the requirements of 10 CFR Part 20 (Ref. 14) at each release point.

In preparing Technical Specifications for units at adjacent sites (multi-unit stations with a common boundary), the sites should be considered as a multi-unit site.

3.2 POPULATION EXPOSURE

The Standard Technical Specifications 3.11.1.3 and 3.11.2.4 require the use of equipment installed in the radioactive waste management system to meet the requirements of 10 CFR 50.36a (Ref. 1).

To assure that the Technical Specifications consider the population radiation doses, the use of the installed radioactive waste management system is required when the projected cumulative doses exceed an appropriate fraction of the individual dose limitations. This method of establishing use of the radwaste equipment assures that the staff's Appendix I cost-benefit analysis in the safety evaluation is not invalidated. Sections 4.5 and 5.4 of this manual provide these values for the Standard Technical Specifications. Guidance on reporting population exposures is provided in Regulatory Guide 1.21 (Ref. 17).

3.3 METEOROLOGICAL DATA

The Standard Technical Specifications consider the historical annual average atmospheric dispersion condition rather than real time dispersion conditions in determining the LCO for radioactive materials in gaseous effluents.

Releases are characterized as "long" or "short" term, depending on the frequency and duration of the releases. This characterization permits the matching of the releases to more appropriate atmospheric diffusion, dispersion and decay conditions.

"Long-term" refers to releases that are generally continuous and stable in release rate with some anticipated variation (i.e., <50%, based on a running monthly average) in release rate, such as is experienced in normal ventilation system effluents at nuclear power plants. Determination of doses due to long-term releases should use the historical annual average relative concentration (χ/Q) based on meteorological data summarized, as recommended in Regulatory Guide 1.111 (Ref. 8).

"Short-term" refers to releases that are intermittent in radionuclide concentrations or flow, such as releases from PWR gas storage tanks, PWR containment ventings and purges, BWR drywell purges (See Standard Technical Specification 3.11.3.8), BWR mechanical vacuum pump exhausts, and systems or components with infrequent use. Short-term releases may be due to operational variations which result in radioactive releases greater than 50% of the releases normally considered as long-term. Short-term releases from these sources during normal operation, including anticipated operational occurrences, are defined as those which occur for a total of 500 hours or less in a calendar year but not more than 150 hours in any quarter. Determination of doses due to short-term releases can use the annual average relative concentration (long-term) if it can be demonstrated that past short-term releases were sufficiently random in both time of day and duration (e.g., the short-term release periods were not dependent solely on atmospheric conditions or time of day) to be represented by the annual average dispersion conditions. Otherwise, the short-term relative concentration value should be calculated in accordance with the guidelines provided in NUREG-0324 (Ref. 13) for short-term release.

Even though "annual average" atmospheric dispersion conditions are used as basis for the Standard Technical Specifications, "real time" meteorological data should be summarized hour-by-hour and coupled with the corresponding releases, and the summary should be included in the SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT.

3.4 NATIONAL INTERIM PRIMARY DRINKING WATER REGULATIONS

Operators of nuclear power plants located on fresh water bodies which are used as sources of water for drinking water supply systems, are required to make a special report concerning the impact on the water supply system due to liquid effluent releases into the water bodies which are above the value(s) permitted in Specification 3.11.1.2 of the Standard Technical Specifications. The NRC has no legal responsibility to implement 40 CFR Part 141, "National Interim Primary Drinking Water Regulations" (Ref. 16), or to assure routine conformance to the Act since this is the responsibility of the Environmental Protection Agency and the water plant operator. This special report is intended for public information and as a tool to assure awareness by the licensee of the impact of radioactive liquid releases on the community's water supply system. The impact within the water supply system is dependent on treatment given to the water taken into the system. The water plant operator is responsible for providing appropriate treatment to assure that 40 CFR Part 141 requirements are met. While the operator of the nuclear power plant is not responsible for meeting the requirements of 40 CFR Part 141 in the water supply system, his success in meeting the requirements of Specification 3.11.1.2 will assure an environmentally acceptable impact on the water supply system. The non-radiological impact is separately considered in the Appendix B Technical Specifications.

3.5 SOLID WASTE MANAGEMENT SYSTEM - PROCESS CONTROL PROGRAM

Standard Technical Specification 3.11.3.1 requires the operator of each nuclear power plant to establish a PROCESS CONTROL PROGRAM for the solid radioactive waste management system. The purpose of the PROCESS CONTROL PROGRAM is to provide reasonable assurance of the complete SOLIDIFICATION of processed wastes and of the absence of free water in the processed waste. At the time the applicant submits proposed Technical Specifications, he should submit the PROCESS CONTROL PROGRAM for NRC review and approval prior to implementation. The PROCESS CONTROL PROGRAM should consist of the processing steps and a set of established process parameters, which include but are not limited to pH, oil content, ratio of solidification agent to influent waste, water content, and ratio of solidification agent to chemical additive for each type of anticipated waste (filter sludges, spent resins, evaporator bottoms, boric acid solutions, sodium sulfate solutions and filter media). The surveillance requirements in the Standard Technical Specifications provide the steps to be taken to assure that operation is within the parameters established by the PROCESS CONTROL PROGRAM. Packaging procedures should demonstrate conformance with Specification 3.11.3.1. The PROCESS CONTROL PROGRAM required by the Standard Technical Specifications is to be documented in the operating procedures for each reactor and available for review by the NRC inspector. A summary of changes to the PROCESS CONTROL PROGRAM shall be provided in the SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT.

3.6 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Standard Technical Specifications 3.3.3.8 and 3.3.3.9 require the operator of each nuclear power plant to establish alarm and trip action setpoints for each radioactive liquid and gaseous effluent release point in maintained, auditable records, determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM). The ODCM shall contain the methodology and parameters to be used in the calculation of offsite doses due to radioactive liquid and gaseous effluents pursuant to Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3, and the established limits of Specifications 3.11.1.1 and 3.11.2.1. The ODCM shall be submitted to the NRC with the proposed Technical Specifications for review and approval by the NRC. Changes to the ODCM shall be provided in the SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT.

3.7 IDENTIFICATION OF RADIONUCLIDES IN EFFLUENTS

In order to determine the radiological impact associated with the release of radioactive materials in liquid and gaseous effluents, the principal radionuclides contributing to the dose must be identified. Tables 4.11-1 and 4.11-2** of the Standard Technical Specifications contain the sampling and analysis programs required for identifying principal radionuclides in effluents. These tables were compiled using the guidelines of Regulatory Guide 1.21 (Ref. 17) and reflect current radiochemical analytical methods. Other methods may be necessary to enhance identification and analysis, as provided by the footnotes to Tables 4.11-1 and 4.11-2. In lieu of sample-analysis, if the applicant does not consider that the collection, radiochemical separation, and analytical methods are technically feasible or practical at the specified LLD, then the dose limitations in Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3 should be proportionally reduced by assuming the continued presence and release concentrations of those radionuclides as determined by the source term (GALE Code, Ref. 11 or 12). For example, the dose LCO may be reduced based on predicted radioactive materials in gaseous effuents from PWR turbine buildings if sampling is not provided. For BWRs and PWRs it may be reduced if carbon-14 analysis is not provided.

3.8 ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR NUCLEAR POWER OPERATIONS

Standard Technical Specification 3.11.2.5 specifies in the Action that when the calculated doses associated with the effluent releases exceed twice* the limits of any one of the Specifications 3.11.1.2, 3.11.2.2 or 3.11.2.3, the licensee shall prepare and submit a Special Report to the Commission and limit subsequent releases such that the dose or dose commitment to a real individual from all uranium fuel cycle sources is limited to ≤ 25 mrem to the total body or any organ (except the thyroid, which is limited to ≤ 75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures to all real individuals from all uranium fuel cycle sources (including all liquid and gaseous effluent pathways and direct radiation) are less than the standards in 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operations (Ref. 18). If analysis indicates that releases resulting in doses that exceed the 40 CFR 190 Standard could occur, obtain a variance from the Commission to permit such releases. The Standard Technical Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3 consider doses to a real individual and apply to each reactor but do not include any other portion of the uranium fuel cycle or direct shine from the reactor.

The "Uranium fuel cycle" is defined in 40 CFR Part 190.02(b) as:

"Uranium fuel cycle means the operations of milling of uranium ore, chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electrical power for public use utilizing nuclear energy, but excludes mining operations, operations at waste disposal sites, transportation of any radioactive material in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle."

The following general guidelines are presented for preparation of the Special Report:

1) determine which uranium fuel cycle facilities or operations, in addition to the nuclear power reactor units at the site, contribute to the annual dose to the maximum exposed member of the public. The maximum exposed member of the public for

*This value may be reduced for multi-unit sites depending on staff analysis.

**These Tables are to be included in proposed Technical Specifications.

this evaluation may or may not correspond to the individual considered in the Technical Specification;

2) determine the total annual dose to this person from all existing pathways and sources of radioactivity and radiation using the methodologies described in this NUREG document and applicable references. Where additional information on pathways and nuclides is needed, the best available information should be used and documented;

3) include direct radiation from the site in the dose determination. An acceptable method for calculating radiation from the N-16 component of direct radiation is: SKYSHINE, A Computer Procedure for Evaluating Effects of Structure Design on N-16 Gamma-Ray Dose Rates, Radiation Research Associates, Inc. Report RRA-T7209, November 1972 (Ref. 19).

   In addition to N-16, all direct radiation from the plant and storage facilities should be considered in the dose determination. The direct dose component (including N-16) may be determined by calculation or actual measurement (e.g., high pressure ionization chamber). The calculation or actual measurement must be documented in this Special Report.

The 25 mrem and 75 mrem dose standards are effective December 1, 1979, except for doses arising from operations associated with the milling of uranium ore which is effective December 1, 1980.

Further information on the method of implementation of 40 CFR Part 190 is being developed by the NRC staff.

CHAPTER 4

LIQUID EFFLUENTS

4.1 INSTRUMENTATION

Standard Technical Specification 3.3.3.8 requires that:

"The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-11* shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The setpoints shall be determined in accordance with procedures as described in the ODCM, and shall be recorded in the station log."

Table 3.3-11* provides a list of radioactive liquid effluent monitoring instrumentation needed to comply with the requirements of General Design Criteria (GDC) 60, 63, and 64 of Appendix A to 10 CFR Part 50 (Ref. 1). The list includes instrumentation such as radioactivity monitoring and sampling devices, automatic control devices, and essential flow and level devices which are components of the monitoring channels. The list uses common nomenclature for the effluent streams; however, the names may be revised, as necessary, to conform with a particular plant's nomenclature. Deletion of any item listed should be justified. Clarification of proposed Technical Specifications should be provided by a simple drawing or sketch showing stream intersections, instrumentation, and control features. Duplicate instrumentation (i.e., instruments that measure different sensor parameters or ranges) should be listed separately. The channel logic should assure that the alarmed trip action is not negated by switching.

The plant procedures should contain a quality assurance program for instruments as recommended in Regulatory Guide 4.15 (Ref. 20).

4.1.1 Setpoint Determination to be Provided in the ODCM

The alarm and trip setpoint(s) for each instrument channel listed in Table 3.3-11* should be provided and 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. If the alarm and the automatic control trip are separate devices, the alarm/trip setpoint in the ODCM should list the separate trip setpoints. The alarm/trip setpoint in the ODCM should list the alarm setpoint where any trip actions are by manual initiation. The method for calculating fixed and adjustable setpoints shall be provided in the ODCM and auditable records shall be maintained indicating the actual setpoints used at all times. For setpoint calculations, see the Addendum to this manual.

*This Table is to be included in proposed Technical Specifications.

The alarm/trip setpoint for a liquid effluent radiation monitor should be determined based on the instantaneous concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 2, and are to be applied at the point of discharge for that stream, pipe or conduit into the unrestricted area, as defined by Figure 3.1-1.* The alarm/trip setpoint should not consider dilution, dispersion or decay in the unrestricted area beyond the release point. An isolation control valve or system shall be provided on the discharge line prior to the release point to permit termination of radioactive releases prior to exceeding the instantaneous concentration limits of 10 CFR Part 20. The isolation and radiation monitoring points are to be located upstream of the release point far enough to assure that the time lag between the established alarm and isolation of the release will not permit a release exceeding these limits. If the stream is diluted by non-radioactive effluents, and the stream dilution and effluent isolation control system is in the exclusion area, the monitor's alarm/trip setpoint may be determined by considering the known in-plant dilution. In-plant dilution is the ratio of the total release rate at the release point into the unrestricted area to the release rate of the undiluted stream, and should be based on continuous measurement of these liquid flows. In such cases, alarm/trip setpoints should also be provided on the flow or level instrumentation with indication in the main control room. The minimum or actual instantaneous in-plant dilution ratio on which the liquid effluent radiation monitor alarm/trip setpoint has been based, should be continuously measured to aid prompt corrective action to satisfy Specification 3.11.1.1.

Conservative assumptions may be included in establishing setpoints to account for such system variables as the control and measurement system efficiency and detection capabilities during normal and anticipated operating conditions, the effects of multiple release points with common or shared in-plant dilution, variability of dilution flow and principal radionuclide composition, and the time lag between alarm/trip action and final isolation of the radioactive effluent. A record of analyses showing current spectra of radionuclides used to calibrate radiation monitors should be maintained in the plant records.

The instruments listed in Table 3.3-11** should also be included in Table 4.3-11** to provide the instrument surveillance requirements, such as calibration, source checking, functional testing and channel checking.

4.2 REQUIREMENT FOR IMPLEMENTING 10 CFR PART 20

In preparing proposed Technical Specifications, Figure 3.11-1* should consist of a map of the site area, showing the unrestricted area boundary for liquid effluents, as defined in 10 CFR 20.3(a)(17). Guidelines for preparing the figure are contained in Section 2.1.1 of Regulatory Guide 1.70 (Ref. 21).

Standard Technical Specification 3.11.1.1 specifies that:

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*This Figure is to be included in proposed Technical Specifications.

**See footnote on page 12.

"The concentration of radioactive material released from the site to unrestricted areas (see Figure 3.11-1)* shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than noble gases and 2×10^{-4} μCi/ml total activity concentration for all dissolved or entrained noble gases."

The concentration limits provided in 10 CFR Part 20, Appendix B, Table II, Column 2, do not include an MPC for noble gases dissolved or entrained in liquid effluents. An MPC of 2×10^{-4} μCi/ml has been established, based on the assumption that xenon-135 is the controlling radionuclide; the Xe-135 MPC in air (submersion) was converted to an equivalent concentration in water using the method described in International Commission on Radiological Protection (ICRP) Publication 2 (Ref. 22). The value of 2×10^{-4} μCi/ml shall be used for a mixture of dissolved or entrained noble gases, not otherwise identified in liquid releases.

To demonstrate that the Specifications are being met, the surveillance requirements specify that a sampling and analysis program be implemented according to Table 4.11-1.** There are two general types of releases: batch and continuous. A batch release is the discharge of liquid waste of a discrete volume. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume or system that has an input flow during the continuous release. For example, releases from sample monitor tanks are batch, and releases from steam generator blowdown are continuous. The sampling and analysis frequency and the type of analysis required by the Standard Technical Specifications is given in Table 4.11-1 for each type of release. The lower limit of detection is also specified in Table 4.11-1 for typical in-plant radiochemical analysis equipment. This program meets the requirements of 10 CFR Part 50, GDC 64, and conforms to the guidelines given in Regulatory Guides 1.21 (Ref. 17) and 4.15 (Ref. 20).

4.3 REQUIREMENT FOR IMPLEMENTING 10 CFR PART 50

The Standard Technical Specification 3.11.1.2 requires that the cumulative dose contributions be determined in accordance with the ODCM at least once per 31 days. The cumulative dose contributions should consider the dose contributions from the maximum exposed individual's consumption of fish, invertebrates and potable water, as appropriate. Normally, the adult is the maximum exposed individual. All of these pathways should be considered in the calculation unless demonstrated not to be present. For many plant sites, the dose calculations may be performed assuming conservative dilution factors and receptor locations to show compliance with the Technical Specification rather than a more rigorous determination. The relationships presented below are acceptable for inclusion in the ODCM. If other methods are selected to implement this Specification, it is expected that the alternate method will include the same general features considered below.

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*See footnote on page 13.

**See footnote on page 12.

The dose contributions for the total time period $\sum_{\ell=1}^{m} \Delta t_\ell$ should be determined by calculation at least once per 31 days and a cumulative summation of these total body and any organ doses should be maintained for each calendar quarter. These dose contributions should be calculated for all radionuclides identified in liquid effluents released to unrestricted areas using the following expression:

$$D_\tau = \sum_i [A_{i\tau} \sum_{\ell=1}^{m} \Delta t_\ell \quad C_{i\ell} \; F_\ell]$$

where:

D_τ = the cumulative dose commitment to the total body or any organ, τ, from the liquid effluents for the total time period $\sum_{\ell=1}^{m} \Delta t_\ell$, in mrem.

Δt_ℓ = the length of the ℓth time period over which $C_{i\ell}$ and F_ℓ are averaged for all liquid releases, in hours.

$C_{i\ell}$ = the average concentration of radionuclide, i, in undiluted liquid effluent during time period Δt_ℓ from any liquid release, in μCi/ml.

$A_{i\tau}$ = the site related ingestion dose commitment factor to the total body or any organ τ for each identified principal gamma and beta emitter listed in Table 4.11-1,* in mrem-ml per hr-μCi.

F_ℓ = the near field average dilution factor for $C_{i\ell}$ during any liquid effluent release. Defined as the ratio of the maximum undiluted liquid waste flow during release to the product of the average flow from the site discharge structure to unrestricted receiving waters times ___. (___ is the site specific applicable factor for the mixing effect of the discharge structure.)

The term $C_{i\ell}$ is the composite undiluted concentration of radioactive material in liquid waste at the common release point determined from the Radioactive Liquid Waste Sampling and Analysis Program, Table 4.11-1* in the Standard Technical Specifications. All dilution factors beyond the sample point(s) are to be included in the F_ℓ and $A_{i\tau}$ terms.

The term F_ℓ is a near field average dilution factor, considering the combined liquid releases from each unit even if there is more than one release point to the unrestricted area per unit within one-quarter mile of each other. As described in Section 3.1 of this manual, multi-unit sites with shared radioactive waste management systems should calculate the total continuous and batch liquid release concentrations for each reactor. The value of the term F_ℓ should be determined as:

*See footnote on page 12.

$$F_{\ell} = \frac{\text{liquid radioactive waste flow per unit}}{\text{discharge structure exit flow per unit x applicable factor}}$$

The liquid radioactive waste flow is the maximum flow from all continuous and batch radioactive effluent releases specified in Table 4.11-1, from all liquid radioactive waste management systems, per unit. The discharge structure exit flow is the average flow during disposal from the discharge structure release point into the receiving water body (in an unrestricted area) per unit. The definition of F_{ℓ} also requires a value to be included in Specification 3.11.1.2 for the dilution as a result of mixing effects in the near field of the discharge structure. For plants with once through cooling, the applicable factor is set equal to one, i.e., no additional dilution is considered. For plants with cooling towers, onsite ponds, or lagoons, the factor shall be a number such that the product of the average blowdown flow to the receiving water body, in cfs and the applicable factor, is 1000 cfs or less. The 1000 cfs figure was selected to correspond to a typical flow for a unit with once-through cooling water and agrees with the staff method for determining compliance with Appendix I (Ref. 1) at the OL stage. The value of this applicable factor is to be included in the blank provided for the term F_{ℓ}. The actual dilution factor value is dependent upon the dilution available in the near field of the receiving water body; however, the applicable factor is limited, as stated above.

4.3.1 Dose Factor Related to Liquid Effluents

The above equation for calculating the dose contributions requires the use of a dose factor $A_{i\tau}$ for each nuclide, i, which embodies the dose factors, pathway transfer factors (e.g., bioaccumulation factors), pathway usage factors, and dilution factors for the points of pathway origin. The adult total body dose factor and the maximum adult organ dose factor for each radionuclide will be used from Table E-11 of Regulatory Guide 1.109 (Ref. 6); thus the list should contain critical organ dose factors for various organs. The dose factor may be written:

$$A_{i\tau} = k_o(U_w/D_w + U_F BF_i + U_I BI_i)DF_i$$

where

$A_{i\tau}$ = composite dose parameter for the total body or critical organ of an adult for nuclide, i, for all appropriate pathways, mrem/hr per µCi/ml.

k_o = units conversion factor, $1.14\text{x}10^5 = 10^6$ pCi/µCi x 10^3ml/kg ÷ 8760 hr/yr.

U_w = 730 kg/yr, adult water consumption (fresh water site only).

U_F = 21 kg/yr, adult fish consumption (all sites).

U_I = 5 kg/yr, adult invertebrate consumption (salt water site only).

BF_i = Bioaccumulation factor for nuclide, i, in fish (fresh or salt water site, as applicable), pCi/kg per pCi/l, from Table A-1 of Regulatory Guide 1.109 (Ref. 4).

BI_i = Bioaccumulation factor for nuclide, i, in invertebrates (salt water only), pCi/kg per pCi/l, from Table A-1 of Regulatory Guide 1.109 (Ref. 4).

DF_i = Dose conversion factor for nuclide, i, for adults in pre-selected organ, τ, in mrem/pCi, from Table E-11 of Regulatory Guide 1.109 (Ref. 4).

D_w = Dilution factor from the near field area within one-quarter mile of the release point(s) to the potable water intake for the adult water consumption (fresh water site only).

Inserting the usage factors of Regulatory Guide 1.109 (Ref. 6) as appropriate into the equation gives the following expressions:

For Fresh Water sites: $A_{i\tau} = 1.14\times10^5(730/D_w + 21BF_i)DF_i$

For Salt Water sites: $A_{i\tau} = 1.14\times10^5(21BF_i + 5BI_i)DF_i$

As noted, all the factors required to calculate the values of $A_{i\tau}$ should be contained in the ODCM. The staff's method of calculating dilution factors for aquatic dispersion is provided in Regulatory Guide 1.113 (Ref. 10). The ODCM should include a detailed presentation of the calculation model and a tabulation of all values assigned to the parameters in expressions used to implement the Specification 3.11.1.2.

4.3.2 Special Reports

The Standard Technical Specifications 3.11.1.2, 3.11.1.3, 3.11.2.2, 3.11.2.3, 3.11.2.4 and 3.11.3.1 require that action be taken when the Specification cannot be met. The action is in the form of special reports, in lieu of licensee event reports, indicating the corrective action to be taken to reduce the dose impact due to the release of radioactive materials in liquid effluents.

These special reports should be prepared using the methodology provided in this manual to determine the dose impact. Such information in the special reports will be used by the staff in determining if the corrective action proposed by the licensee is adequate to bring the releases within the design objectives of Appendix I to 10 CFR Part 50. These special reports may also require submitting additional information as described in Section 3.4 of this manual.

4.4 SPECIFICATION ON RADIOACTIVITY CONTENTS IN LIQUID-CONTAINING TANKS

Standard Technical Specification 3.11.1.4 and Tables 3.3-11* and 4.3-11* list liquid-containing tanks outside containment that are to be analyzed periodically to verify that the radioactivity content (in curies, excluding tritium and dissolved or entrained noble gases) is below the specified value. Tanks included in this Specification are those that are not surrounded by liners, dikes or walls capable of holding the tank contents and do not have tank overflow and drains connected to the liquid radioactive waste management system. Indoor tanks are not included unless an analysis based on design basis fission product leakage from the fuel results in radionuclide concentrations in excess of the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, where leaked fluid is capable of affecting the nearest existing or known future water supply** in an unrestricted area.

* See footnote on page 12.

** "Supply" means a well or surface water intake that is used as a water source for direct human consumption or indirectly through animals, crops or food processing. "Known future" water supply means potential wells or surface water intakes which are identified, or may be reasonably deduced from available information.

For those tanks that are determined to be included in Specification 3.11.1.4 and Tables 3.3-11 and 4.3-11, a curie limit should be determined based on the methodology presented in Appendices A or B of this manual, using the PWR-RATAFR Computer Code for pressurized water reactor plants, or the BWR-RATAFR Computer Code for boiling water reactor plants, respectively. The methodology is based on the calculated radionuclide inventory in the tank at 80% capacity using a design basis fission product source term of (1) 1% of the operating fission product inventory in the core being released to the primary coolant for a PWR, or (2) a fission product release consistent with a noble gas release rate of 100 μCi/MWt-sec at 30 minutes decay for a BWR. These Computer Codes determine the radionuclide inventory in a tank that would result in concentrations equal to the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at (1) the nearest potable water supply and (2) the nearest surface water supply in an unrestricted area.

By excluding tritium and dissolved or entrained noble gases from the surveillance analyses, since these can be estimated for any licensee event report, Specification 3.11.1.4 should include the lowest curie quantity of activation and mixed fission products determined for any tank listed in Specification 3.11.1.4 as the curie limit for all tanks included in that Specification.

Most operating reactors have required the use of temporary process and storage tanks during maintenance and service periods, or when temporary solidification equipment is used at the facility; therefore, the Specification 3.11.1.4 should indicate a "temporary tank." The curie limit for a temporary tank may be calculated by the above method, but should be limited to $\leq$ 10 curies, excluding tritium and dissolved or entrained gases. If the temporary tank is mobile and not used for more than a calendar quarter, it need not be included in Tables 3.3-11* and 4.3-11.*

4.5 SPECIFICATION ON THE USE OF LIQUID RADIOACTIVE WASTE MANAGEMENT SYSTEM

Standard Technical Specification 3.11.1.3 specifies that:

> "The liquid radwaste treatment system shall be OPERABLE. The appropriate subsystems shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent releases to unrestricted areas (see Figure 3.11-1)* when averaged over 31 days, exceeds 0.06 mrem to the total body or 0.2 mrem to any organ."

The operability of the liquid radioactive waste management system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The term "liquid radwaste treatment system" involves all of the installed and available liquid radioactive waste management system equipment, as well as their controls, power instrumentation, and services that make the system functional. Equipment that is considered standby or redundant is also included, since their function is to assure operability. The specification also permits alternate treatment paths using alternate subsystems and equipment to be used in the event that the normal treatment is inoperable.

*See footnote on page 13.

This Specification requires maintenance and use of the liquid radioactive waste management system for conformance to 10 CFR Part 50.36a. Maintenance and use of the radioactive waste management system components are discussed in Section 2.2 of this manual.

To determine if use of the installed equipment is necessary, the licensee must project the cumulative liquid effluent releases over the ensuing 31 days. These releases should include all plant effluents from all liquid radioactive waste management and liquid waste disposal system components that are planned to be operated at the projected capacity and performance of each component used during the specified time. These releases should include a margin, based on operating data, for anticipated and unplanned operational occurrences and should use the methodology discussed in Section 4.3 of this manual. The impact from this projected cumulative release is to be compared to 0.06 mrem for the total body or 0.2 mrem for any organ. If the projection indicates these values will be exceeded, then the installed liquid radioactive waste management system components that will reduce those radioactive materials in liquid effluents and the projected impact, must be used.

The values for the projected impact, given above, correspond to approximately one forty-eighth of the design objective values of Appendix I, Section II.A of 10 CFR Part 50 in a month, and if continued at this rate for a year, they would correspond to less than one-fourth the values limited by Specification 3.11.1.2.b. The calculations of projected cumulative dose impact that could result from the proposed operation should use the methodology provided in Section 4.3 of this manual.

CHAPTER 5

GASEOUS EFFLUENTS

5.1 INSTRUMENTATION

Standard Technical Specification 3.3.3.9 requires that:

"The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.3-12* shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with procedures as described in the ODCM, and shall be recorded in the station log."

Table 3.3-12* provides a list of the radioactive gaseous effluent monitoring instrumentation needed to comply with the requirements of General Design Criteria (GDC) 60, 63, and 64 of Appendix A to 10 CFR Part 50. The list includes instrumentation such as radioactivity monitoring and sampling devices, automatic control devices and essential flow and level devices that are components of the monitoring systems. The list uses common nomenclature for the effluent streams; however, the names may be revised as necessary, to conform with a particular plant's nomenclature. The list may include effluent streams which are not applicable to a given plant, or which may be duplicated and, therefore, should be tailored for the proposed Technical Specifications. Clarification of proposed Technical Specifications should be provided by a simple drawing or sketch, showing stream intersections, instrumentation and control features. Duplicate instrumentation (i.e., instruments that measure different sensor parameters or ranges) should be listed separately in Tables 3.3-12* and 4.3-12.* The channel logic should assure that the alarmed trip action is not negated by switching.

The plant procedures should contain a quality assurance program for instruments as recommended in Regulatory Guide 4.15 (Ref. 20).

5.1.1 Setpoint Determination to be Provided in the ODCM

The alarm/trip setpoint or automatic control trip setpoint for each instrument channel listed in Table 3.3-12* should be provided and should correspond to a value(s) which represents a safe margin of assurance that the instantaneous gaseous release limit of Specification 3.11.2.1(a) will not be exceeded. For channels with separate alarm and automatic control trips, the setpoint for the automatic control trip should be the established value referenced above; the corresponding setpoint for alarm/trip should be established such that an alarm trip will occur either in advance of the automatic control trip or simultaneously with the automatic control trip. For channels with alarm trips only, the setpoint for the alarm/trip should be the established value referenced above, provided that the manual or procedural response to the alarm represents a safe margin of assurance that the instantaneous gaseous release limit of 10 CFR Part 20 will not be exceeded. The alarm/trip setpoint in the ODCM should list the alarm setpoint for those channels where any trip actions are by

*This Table is to be included in the proposed Technical Specifications.

manual initiation. The method for calculating fixed or adjustable setpoints shall be provided in the ODCM and auditable records shall be maintained indicating the actual setpoints used at all times.

The alarm/trip setpoint for any gaseous effluent radiation monitor should be determined based on the instantaneous (see RELEASE RATE fundamental time units in Section 2.2 of this manual) concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 1, and are to be applied at the point at which the discharge leaves the restricted area boundary into an unrestricted area, as defined by Figure 5.1-1.** The bases for each setpoint should consider the type of release at the monitor's location, e.g., long-term releases using the long-term atmospheric dispersion conditions or, if applicable (see Section 3.3 of this manual), short-term releases using the short-term atmospheric dispersion conditions. An isolation control valve or system shall be provided on the discharge line upstream of the release point to permit isolation prior to exceeding the specified release limits.

If the alarm/trip setpoints are based on predetermined factors accounting for atmospheric conditions, the elevation of the release point should be considered. The symbols used in the equations in this manual use a subscript (s) for a free-standing stack for elevated releases (or vents that take the plume out of the building wake) and a subscript (v) for vent for releases that are not completely elevated. Guidance on the staff's method for estimating atmospheric transport and dispersion of gaseous effluents in routine releases is provided in Regulatory Guide 1.111 (Ref. 8). The radiation monitor alarm/trip setpoints for each release point should be based on the radioactive noble gases in gaseous effluents. It is not considered to be practicable to apply instantaneous alarm/trip setpoints to integrating radiation monitors sensitive to radioiodines, radioactive materials in particulate form and radionuclides other than noble gases. Alarm/trip setpoints should also be provided in the main control room for flow measurement devices which are part of continuous monitoring or sampling systems and should alarm on loss-of-flow or departure from an established flow range. In all cases, conservative assumptions may be necessary in establishing these setpoints to account for system variables, such as the control and measurement system efficiency and detection capabilities during normal, anticipated, and unusual operating conditions, the variability in release flow and principal radionuclides, and the time lag between alarm/trip action and the final isolation of the radioactive effluent. The current spectrum of radionuclides used to calibrate radiation monitors should be maintained in the plant records. The instruments listed in Table 3.3-12* should also be included in Table 4.3-12,* to provide the instrument surveillance requirements, such as calibration, source checking, functional testing, and channel checking.

5.2 DOSE LIMIT FOR IMPLEMENTING 10 CFR PART 20

In preparing proposed Technical Specifications, Figure 5.1-1** should consist of a map of the site area showing the exclusion boundary, as defined in 10 CFR 100.3(a) (Ref. 15) and the unrestricted area boundary, as defined in 10 CFR 20.3(a)(17) (Ref. 14). Guidelines for this figure are contained in Section 2.1.1 of Regulatory Guide 1.70 (Ref. 21). Details on the release point locations and significant elevations should be given in Figure 5.1-1.**

*See footnote on page 20.

**This Figure is to be included in proposed Technical Specifications.

5.2.1 Implementation of 10 CFR Part 20 - Airborne Releases

The Standard Technical Specification 3.11.2.1 implements 10 CFR Part 20 as follows:

"The instantaneous dose rate in unrestricted areas (see Figure 5.1-1)** due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be < 500 mrem/yr to the total body and < 3000 mrem/yr to the skin, and

b. The dose rate limit for all radioiodines and for all radioactive materials in particulate form and radionuclides other than noble gases with half lives greater than 8 days shall be < 1500 mrem/yr to any organ."

The ODCM should provide the mathematical relationships used to implement the above specification. The relationships presented below are acceptable for inclusion in the ODCM. If other methods are selected to implement the specification, it is expected that the alternative method will include the same general features considered below.

a. Release rate limit for noble gases:

$$\sum_i [V_i \dot{Q}_{is} + K_i ((\overline{\chi/Q})_v \dot{Q}_{iv})] < 500 \text{ mrem/yr, and}$$

$$\sum_i [(L_i (\overline{\chi/Q})_s + 1.1 B_i) \dot{Q}_{is} + (L_i + 1.1 M_i)((\overline{\chi/Q})_v \dot{Q}_{iv})] < 3000 \text{ mrem/yr}$$

where the terms are defined below.

b. Release rate limit for all radionuclides and radioactive materials in particulate form and radionuclides other than noble gases:

$$\sum_i P_i [W_s \dot{Q}_{is} + W_v \dot{Q}_{iv}] < 1500 \text{ mrem/yr}$$

where:

K_i = The total body dose factor due to gamma emissions for each identified noble gas radionuclide, in mrem/yr per $\mu Ci/m^3$.

L_i = The skin dose factor due to beta emissions for each identified noble gas radionuclide, in mrem/yr per $\mu Ci/m^3$.

M_i = The air dose factor due to gamma emissions for each identified noble gas radionuclide, in mrad/yr per $\mu Ci/m^3$ (unit conversion constant of 1.1 mrem/mrad converts air dose to skin dose).

P_i = The dose parameter for radionuclides other than noble gases for the inhalation pathway, in mrem/yr per $\mu Ci/m^3$ and for food and ground plane pathways, in m^2(mrem/yr per $\mu Ci/sec$). The dose factors are based on the critical individual organ and most restrictive age group (infant).

V_i = The constant for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume, derived in accordance with the dose methodology in Regulatory Guide 1.109, Appendix B, Section 1, in mrem/yr per $\mu Ci/sec$.

B_i = The constant for long-term releases (greater than 500 hrs/yr) for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume, derived in accordance with the dose methodology in Regulatory Guide 1.109, Appendix B, Section 1, in mrad/yr per $\mu Ci/sec$.

**See footnote on page 21.

$\dot{Q}_{is}$ = The release rate of radionuclides, i, in gaseous effluents from free-standing stack, in µCi/sec (per unit, unless otherwise specified.)

$\dot{Q}_{iv}$ = The release rate of radionuclides, i, in gaseous effluent from all vent releases, in µCi/sec (per unit, unless otherwise specified).

$(\overline{\chi/Q})_s$ = _____ sec/m³. For free-standing stack releases. The highest calculated annual average relative concentration for any area at or beyond the unrestricted area boundary.

$(\overline{\chi/Q})_v$ = _____ sec/m³. For all vent releases. The highest calculated annual average relative concentration for any area at or beyond the unrestricted area boundary.

W_v = The highest calculated annual average dispersion parameter for estimating the dose to an individual at the controlling location due to all vent releases:

W_v = _____ sec/m³, for the inhalation pathway. The location is the unrestricted area boundary in the ___ sector.

W_v = ____ meters^{-2}, for the food and ground plane pathways. The location is the unrestricted area boundary in the ___ sector.

W_s = The highest calculated annual average dispersion parameter for estimating the dose to an individual at the controlling location due to free-standing stack releases:

W_s = _____ sec/m³, for the inhalation pathway. The location is the unrestricted area boundary in the ___ sector.

W_s = ____ meters^{-2}, for the food and ground plane pathways. The location is the unrestricted area boundary in the ___ sector."

SPECIAL NOTES: (1) If there are no free-standing stacks, the factors denoted by the subscript, s, need not be considered. (2) In all cases, the tritium releases use the first W parameter, based on relative concentration (sec/m³). (3) All radioiodines are assumed to be released in elemental form. If analysis includes the capability of determining elemental and nonelemental forms in all releases, the food pathway parameters may be adjusted accordingly.

The Specification is applicable to the location (unrestricted area boundary or beyond), characterized by the values of the parameters V_i, B_i or $(\overline{\chi/Q})$ which result in the maximum total body or skin dose commitment. In the event that the analyses indicates a different location for the total body and skin dose limitations, the location selected for consideration shall be that which minimizes the allowable release values.

The factors K_i, L_i, and M_i relate the radionuclide airborne concentrations to various dose rates assuming a semi-infinite cloud. These factors may be taken directly from Table B-1 of the Regulatory Guide 1.109 (Ref. 6), if the values therein are multiplied by 10^6 to convert picocuries^{-1} to microcuries^{-1} as used in the above equations. A tabulation of these factors should be included in the ODCM.

The B_i and V_i factors for the radionuclides are based on the finite plume model of Regulatory Guide 1.109 (Ref. 6). From Equation 6 of Section 6.2 of this Regulatory Guide, B_i can be expressed as:

$$B_i = \frac{K}{r_d} \sum_{jk\ell}\sum\sum \frac{f_{jk}\, A_{\ell i}\, \mu_a\, E_\ell I}{u_j} \left(\frac{\text{mrad/yr}}{\mu\text{Ci/sec}}\right)$$

Where:

I = the results of numerical integration over the plume spatial distribution of the airborne activity as defined by the meteorological condition of wind speed (u_j) and atmospheric stability class k for a particular wind direction.

K = a numerical constant representing unit conversions, presented below.

r_d = the distance from the release point to the receptor location, in meters.

u_j = the mean wind speed assigned to the jth wind speed class, in meters/sec.

f_{jk} = the joint frequency of occurrence of the jth wind speed class and kth stability class (dimensionless).

$A_{\ell i}$ = the number of photons of energy corresponding to the ℓth energy group emitted per transformation of the ith radionuclide, in number/transformation.

E_ℓ = the energy assigned to the ℓth energy group, in Mev.

μ_a = the energy absorption coefficient in air for photon energy E_ℓ, in meters^{-1}.

The constant K follows from Equation 6 of Section C.2.a of Regulatory Guide 1.109 (Ref. 6), as:

$$K = \frac{260 \text{ mrad(radians)(m}^3\text{)(transformation)}}{\text{sec(Mev)(Ci)}} \left(\frac{16 \text{ sectors}}{2\pi \text{ radians}}\right)\left(10^{-6}\ \frac{\text{Ci}}{\mu\text{Ci}}\right)\left(3.15\text{x}10^{7}\ \frac{\text{sec}}{\text{yr}}\right)$$

$$= 2.1\text{x}10^{4} \text{ mrad (m}^3\text{)(transformation)/yr(Mev)(}\mu\text{Ci)}.$$

The V_i factor is computed with conversion from air dose to tissue depth dose, thus:

$$V_i = 1.1\,\frac{K}{r_d} \sum_{jk\ell}\sum\sum \frac{f_{jk} A_{\ell i} \mu_a\ E_\ell\ I}{u_j}\left[e^{-\mu_T T_d}\right] \left(\frac{\text{mrem/yr}}{\mu\text{Ci/sec}}\right)$$

where:

μ_T = the tissue energy absorption coefficient for photons of energy E_ℓ, in cm^2/gm.

T_d = the tissue density thickness taken to represent the total body dose ($5 gm/cm^2$).

1.1 = the ratio of the tissue to air absorption coefficients over the energy range of photons of interest. This ratio converts dose (mrad) to dose equivalent (mrem).

The parameter, P_i, contained in the radioiodine and particulates Specification 3.11.2.1.b includes pathway transport parameters of the ith radionuclide, the receptor's usage of the pathway media and the dosimetry of the exposure. Pathway usage rates and the internal dosimetry are functions of the receptor's age; however, the youngest age group, the infant, will always receive the maximum dose under the exposure conditions for Specification 3.11.2.1. For the infant exposure, separate values of P_i may be calculated using the PARTS computer program given in Appendix C of this manual for the inhalation pathway which uses a W parameter based on $(\overline{\chi/Q})$, and the food (milk) and ground pathway which uses a W parameter normally based on $(\overline{D/Q})$, except for tritium, for application in the ODCM. The following sections provide detail on calculating these P_i values for inclusion in the ODCM.

The values of P_i are independent of vent and stack release elevation. In the case of tritium, $(\overline{\chi/Q})$ is the W parameter for the food (milk) pathway as well as the inhalation pathway. As tritium is a weak beta emitter, the ground plane contribution is zero for tritium. (NOTE: The value for the P_i (food) for tritium is 2.4×10^3 mrem/yr per $\mu Ci/m^3$.) If the controlling locations for vent and stack releases are different, the controlling location for vent releases should be used in Specification 3.11.2.1.

Omitting the subscripts for vent and stack releases, the dose rate from the ith radionuclide (except tritium) is:

$$P_i \text{ (inhalation)}(\overline{\chi/Q})\, Q + [P_i\text{(food)} + P_i\text{(ground plane)}]\, (\overline{D/Q})Q \text{ (mrem/yr)}$$

and for tritium, is:

$$P_i \text{ (inhalation)}(\overline{\chi/Q})Q + P_i\text{(food)}(\overline{\chi/Q})Q = 3.0 \times 10^3 (\overline{\chi/Q})Q \text{ (mrem/yr)}$$

5.2.1.1 Calculation of P_i (Inhalation)

$$P_i = K'(BR)\ DFA_i \text{ (mrem/yr per } \mu Ci/m^3)$$

where:

K' = a constant of unit conversion, 10^6 pCi/μCi.

BR = the breathing rate of the infant age group, in m^3/yr.

DFA_i = the maximum organ inhalation dose factor for the infant age group for the ith radionuclide, in mrem/pCi. The total body is considered as an organ in the selection of DFA_i.

The age group considered is the infant group. The infant's breathing rate is taken as 1400 m^3/yr from Table E-5 of Regulatory Guide 1.109 (Ref. 6). The inhalation dose factors for the infant, DFA_i are presented in Table E-10 of Regulatory Guide 1.109, in units of mrem/pCi.

Resolution of the units yields:

$$P_i \text{ (inhalation)} = 1.4 \times 10^9\ DFA_i$$

5.2.1.2 Calculation of P_i (Ground Plane)

$$P_i = K'K''DFG_i \; (1-e^{-\lambda_i t})/\lambda_i \quad (m^2\cdot mrem/yr \text{ per } \mu Ci/sec)$$

Where:

K' = a constant of unit conversion, 10^6 pCi/μCi.

K'' = a constant of unit conversion, 8760 hr/year.

λ_i = the decay constant for the ith radionuclide, sec^{-1}.

t = the exposure period, 3.15×10^7 sec (1 year).

DFG_i = the ground plane dose conversion factor for the ith radionuclide (mrem/hr per pCi/m^2).

The deposition rate onto the ground plane results in a ground plane concentration that is assumed to persist over a year with radiological decay the only operating removal mechanism for each radionuclide. The ground plane dose conversion factors for the ith radionuclide, DFG_i, are presented in Table E-6 of Regulatory Guide 1.109 (Ref. 6), in units of mrem/hr per pCi/m^2.

Resolution of the units yields:

$$P_i \text{ (Ground)} = 8.76 \times 10^9 \; DFG_i \; (1-e^{-\lambda_i t})/\lambda_i.$$

5.2.1.3 Calculation of P_i (Food)

$$P_i = K'r \frac{Q_F(U_{ap})}{Y_p(\lambda_i+\lambda_w)} F_m \; DFL_i \; [e^{-\lambda_i t_f}] \; (m^2\cdot mrem/yr \text{ per } \mu Ci/sec)$$

where:

K' = a constant of unit conversion, 10^6 pCi/μCi.

Q_F = the cow's consumption rate, in kg/day (wet weight).

U_{ap} = the infant's milk consumption rate, in liters/yr.

Y_p = the agricultural productivity by unit area, in kg/m^2

F_m = the stable element transfer coefficients, in days/liter.

r = fraction of deposited activity retained on cow's feed grass.

DFL_i = the maximum organ ingestion dose factor for the ith radionuclide, in mrem/pCi.

λ_i = the decay constant for the ith radionuclide, in sec^{-1}.

λ_w = the decay constant for removal of activity on leaf and plant surfaces by weathering, 5.73×10^{-7} sec^{-1} (corresponding to a 14 day half-time).

t_f = the transport time from pasture to cow, to milk, to infant, in sec.

A fraction of the airborne deposition is captured by the ground plant vegetation cover. The captured material is removed from the vegetation (grass) by both radiological decay and weathering processes.

The values of Q_F, U_{ap}, and Y_p are provided in Regulatory Guide 1.109 (Ref. 6), Tables E-3, E-5, and E-15, as 50 kg/day, 330 liters/day and 0.7 kg/m^2, respectively. The value t_f is provided in Regulatory Guide 1.109 (Ref. 6), Table E-15, as 2 days (1.73×10^5 seconds). The fraction, r, has a value of 1.0 for radioiodines and 0.2 for particulates, as presented in Regulatory Guide 1.109 (Ref. 6), Table E-15.

Table E-1 of Regulatory Guide 1.109 (Ref. 4) provides the stable element transfer coefficients, F_m, and Table E-14 provides the ingestion dose factors, DFL_i, for the infant's organs. The organ with the maximum value of DFL_i is to be used.

Resolution of the units yields:

$$P_i \text{ (food)} = 2.4 \times 10^{10} \frac{rF_m}{\lambda_i + \lambda_w} DFL_i \left[e^{-\lambda_i t_f}\right] \text{ (m}^2\cdot\text{mrem/yr per } \mu\text{Ci/sec)}$$

for all radionuclides, except tritium.

The concentration of tritium in milk is based on its airborne concentration rather than the deposition rate.

$$P_i = K'K'''F_m Q_F U_{ap} DFL_i \left[0.75(0.5/H)\right] \text{ (mrem/yr per } \mu\text{Ci/m}^3\text{)}$$

where:

K''' = a constant of unit conversion, 10^3 gm/kg.

H = absolute humidity of the atmosphere, in gm/m^3.

0.75 = the fraction of total feed that is water.

0.5 = the ratio of the specific activity of the feed grass water to the atmospheric water.

From Table E-1 and E-14 of Regulatory Guide 1.109 (Ref. 6), the values of F_m and DFL_i for tritium are 1.0×10^{-2} day/liter and 3.08×10^{-7} mrem per pCi, respectively. Assuming an average absolute humidity of 8 grams/meter^3, the resolution of units yields:

$$P_i \text{ (food)} = 2.4 \times 10^3 \text{ mrem/yr per } \mu\text{Ci/m}^3$$

for tritium, only.

5.3 DOSE LIMIT FOR IMPLEMENTING 10 CFR PART 50

In preparing proposed Technical Specifications, Figure 5.1-1* should consist of a map of the site area showing the unrestricted area boundary for gaseous effluents as defined in Section 5.2 of this manual. Guidelines for this figure are contained in Regulatory Guide 1.70, Section 2.1.1 (Ref. 21).

*See footnote on page 21.

5.3.1 REQUIREMENTS FOR IMPLEMENTING 10 CFR PART 50

The Standard Technical Specifications 3.11.2.2 and 3.11.2.3 implement 10 CFR Part 50, Appendix I, as follows:

"The air dose in unrestricted areas (see Figure 5.1-1)* due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter, to $\leq$ 5 mrad for gamma radiation and $\leq$ 10 mrad for beta radiation;

b. During any calendar year, to $\leq$ 10 mrad for gamma radiation and $\leq$ 20 mrad for beta radiation;

(The dose design objectives may be reduced based on expected public occupancy of areas, e.g., beaches and visitor centers within the unrestricted area boundary. (For PWRs only) the dose design objectives may be reduced based on predicted noble gas releases from the turbine building, if effluent sampling is not provided.)"

"The dose to an individual from radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days in gaseous effluents released to unrestricted areas (see Figure 5.1-1)* shall be limited to the following:

a. During any calendar quarter to $\leq$ 7.5 mrem; and

b. During any calendar year to $\leq$ 15 mrem."

The ODCM should provide the mathematical relationships used to implement the Specifications. The relationships presented below are acceptable for inclusion in the ODCM. If other methods are selected to implement these Specifications, it is expected that the alternative method will include the same general features considered below.

The air dose in unrestricted area (see Figure 5.1-1)* due to noble gases released in gaseous effluents should be determined by the following expressions:

a. During any calendar quarter, for gamma radiation:

$$3.17 \times 10^{-8} \sum_i \left[M_i \, [(\overline{\chi/Q})_v \tilde{Q}_{iv} + (\overline{\chi/q})_v \tilde{q}_{iv}] + [B_i \tilde{Q}_{is} + b_i \tilde{q}_{is}] \right] \leq 5 \text{ mrad, and}$$

During any calendar quarter, for beta radiation:

$$3.17 \times 10^{-8} \sum_i \; N_i \, [(\overline{\chi/Q})_v \tilde{Q}_{iv} + (\overline{\chi/q})_v \tilde{q}_{iv} + (\overline{\chi/Q})_s \tilde{Q}_{is} + (\overline{\chi/Q})_s \tilde{q}_{is}] \leq 10 \text{ mrad, and}$$

b. During any calendar year, for gamma radiation:

$$3.17 \times 10^{-8} \sum_i \left[M_i \, [(\overline{\chi/Q})_v \tilde{Q}_{iv} + (\overline{\chi/q})_v \tilde{q}_{iv}] + [B_i \tilde{Q}_{is} + b_i \tilde{q}_{is}] \right] \leq 10 \text{ mrad, and}$$

During any calendar year, for beta radiation:

$$3.17 \times 10^{-8} \sum_i \; N_i \, [(\overline{\chi/Q})_v \tilde{Q}_{iv} + (\overline{\chi/q})_v \tilde{q}_{iy} + (\overline{\chi/Q})_s \tilde{Q}_{is} + (\overline{\chi/q})_s \tilde{q}_{is}] \leq 20 \text{ mrad}$$

where:

M_i = The air dose factor due to gamma emmissions for each identified noble gas radionculide, in mrad/yr per $\mu Ci/m^3$.

*See footnote on page 21.

N_i = The air dose factor due to beta emissions for each identified noble gas radionuclide, in mrad/yr per $\mu Ci/m^3$.

$(\overline{\chi/Q})_v$ = ________ sec/m^3. For vent releases. The highest calculated annual average relative concentration for area at or beyond the unrestricted area boundary for long term releases (greater than 500 hrs/year).

$(\overline{\chi/q})_v$ = ________ sec/m^3. For vent releases. The relative concentration for areas at or beyond the unrestricted area boundary for short term releases (equal to or less than 500 hrs/year).

$(\overline{\chi/Q})_s$ = ________ sec/m^3. For free-standing stack releases. The highest calculated annual average relative concentration for areas at or beyond the unrestricted area boundary for long term releases (greater than 500 hrs/year).

$(\overline{\chi/q})_s$ = ________ sec/m^3. For free-standing stack releases. The relative concentration for areas at or beyond the unrestricted area boundary for short term releases (equal to or less than 500 hrs/year).

$\tilde{q}_{is}$ = The average release of noble gas radionuclides in gaseous effluents, i, for short term releases (equal to or less than 500 hrs/year) from the free-standing stack, in µCi. Releases shall be cumulative over the calendar quarter or year as appropriate.

$\tilde{q}_{iv}$ = The average release of noble gas radionuclides in gaseous effluents, i, for short term releases (equal to or less than 500 hrs/year) from all vents, in µCi. Releases shall be cumulative over the calendar quarter or year as appropriate.

$\tilde{Q}_{is}$ = The average release of noble gas radionuclides in gaseous releases, i, for long term releases (greater than 500 hrs/year) from the free-standing stack, in µCi. Release shall be cumulative over the calendar quarter or year as appropriate.

$\tilde{Q}_{iv}$ = The average release of noble gas radionuclides in gaseous effluents, i, for long term releases (greater than 500 hrs/yr) from all vents, in µCi. Releases shall be cumulative over the calendar quarter or year as appropriate.

B_i = The constant for long term releases (greater than 500 hrs/yr) for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume, derived in accordance with the dose methodology in Regulatory Guide 1.109, Appendix B, Section 1, in mrad/yr per µCi/sec.

b_i = The constant for short term releases (equal to or less than 500 hrs/yr) for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume, derived in accordance with the dose methodology in Regulatory Guide 1.109, Appendix B, Section 1, in mrad/yr per µCi/sec.

3.17×10^{-8} = The inverse of the number of seconds in a year.

The dose to an individual from radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days in gaseous effluents released to unrestricted areas (see Figure 5.1-1)* should be determined by the following expressions:

a. During any calendar quarter:

$$3.17 \times 10^{-8} \sum_i R_i \left[W_s \tilde{Q}_{is} + w_s \tilde{q}_{is} + W_v \tilde{Q}_{iv} + w_v \tilde{q}_{iv}\right] \leq 7.5 \text{ mrem, and}$$

*See footnote on page 21.

b. During any calendar year:

$$3.17 \times 10^{-8} \sum_i R_i \, [W_s \tilde{Q}_{is} + w_s \tilde{q}_{is} + W_v \tilde{Q}_{iv} + w_v \tilde{q}_{iv}] \leq 15 \text{ mrem}$$

where:

$\tilde{Q}_i$ = The releases of radionuclides, radioactive materials in particulate form, and radionuclides other than noble gases in gaseous effluents, i, for long term releases greater than 500 hrs/yr, in μCi. Releases shall be cumulative over the calendar quarter or year as appropriate.

$\tilde{q}_i$ = The releases of radionuclides, radioactive materials in particulate form and radionuclides other than noble gases in gaseous effluents, i, for short term releases equal to or less than 500 hrs/yr, in μCi. Releases shall be cumulative over the calendar quarter or year as appropriate.

W = The dispersion parameter for estimating the dose to an individual at the controlling location for long term releases (greater than 500 hrs/yr):

W = $(\overline{\chi/Q})$ for the inhalation pathway, in sec/m^3.

W = $(\overline{D/Q})$ for the food and ground plane pathways in meters^{-2}.

w = The dispersion parameter for estimating the dose to an individual at the controlling location for short term releases (equal to or less than 500 hrs/yr):

w = $(\overline{\chi/q})$ for the inhalation pathway in sec/m^3.

w = $(\overline{D/q})$ for the food and ground plane pathway in meters^{-2}.

3.17×10^{-8} = The inverse of the number of seconds in a year.

R_i = The dose factor for each identified radionuclide, i, in m^2(mrem/yr) per μCi/sec or mrem/yr per μCi/m^3.

For the direction sectors with existing pathways within 5 miles from the unit, use the values of R_i for these pathways. If no real pathway exists within 5 miles from the center of the building complex, use the cow-milk R_i assuming that this pathway exists at the 4.5 to 5.0 mile distance in the worst sector. If the R_i for an existing pathway within 5 miles is less than a cow-milk R_i at 4.5 to 5.0 miles, then use the value of the cow-milk R_i at 4.5 to 5.0 miles. The pathway values used for calculating dose contributions shall be consistent with the results of the land use census performed pursuant to Specification 3.12.2. The controlling value of R_i for each radionuclide shall be determined and provided in tabular form in the ODCM. The parameters W and w shall correspond to the applicable pathway location.

SPECIAL NOTES: (1) If there is no free-standing stack, the factors denoted by the subscript, s, need not be considered. (2) In all cases, the tritium releases use the first W or w parameter, based on relative concentration (sec/m^3). (3) All radioiodines are assumed to be released in elemental form. If analysis includes the capability of determining the elemental and non-elemental forms in all releases, the food pathway parameters may be adjusted accordingly.

The following information is provided to further clarify the application of these Specifications and provide more information regarding the individual factors. The ODCM should include a detailed presentation of the calculational model and a complete tabulation of all values assigned to each parameter.

The noble gas Specification 3.11.2.2 is to be evaluated at the location in the unrestricted area where analyses of annual average air doses were found to be maximum. In the event that the analyses indicate different locations for the beta and gamma limitations, the location selected for consideration shall be that which minimizes the allowable release values due to gamma radiation.

The radioiodine and particulate Specification 3.11.2.3 is applicable to the location in the unrestricted area where the combination of existing pathways and receptor age groups indicates the maximum potential exposures. The inhalation and ground plane exposure pathways shall be considered to exist at all locations. The grass-cow-milk, grass-cow-meat, and vegetation pathways are considered based on their existence at the various locations. Of the various age groups, the infant or child receives the largest dose; thus, only these two age groups will be discussed. It is the intent, however, that a licensee undertake annual surveys of the age groups and the land use at the various locations about the site, reports these results according to Specification 3.12.2 and determines the applicable parameters of R_i for tabulation in the ODCM. The new parameters shall be submitted to the NRC for review and approval prior to implementation.

The M_i and N_i factors of the noble gas relationship relate the airborne concentration of the noble gas to the air dose rates assuming a semi-infinite cloud. These factors may be taken directly from Table B-1 of the Regulatory Guide 1.109 (Ref. 6), and the values therein have been multiplied by 10^6 to convert picocuries^{-1} to microcuries^{-1}.

The factor, B_i, is defined in Section 5.2.1 of this manual. The corresponding short-term factor, b_i is computed following the same procedure replacing the meteorological variables, j, k, and l, for long-term releases with variables for short-term releases using the methodology provided in Regulatory Guide 1.111 (Ref. 8), NUREG-0324 (Ref. 13), and NUREG-75/087 (Ref. 23). Such information should be provided in tabular form in the ODCM.

In developing the R_i values, separate expressions are written for each of the potential pathways. These expressions are similar to those developed in Section 5.2.1 of this manual for P_i, and are denoted by $R_i^G[D/Q]$, $R_i^I[\chi/Q]$, $R_i^C[D/Q]$, $R_i^M[D/Q]$ and $R_i^V[D/Q]$, where the superscripts G, I, C, M, and V refer to ground plane, inhalation, cow's milk, meat and vegetation, respectively. The 'argument' notation, [ ], indicates the appropriate dispersion parameter, W, to be applied with the R_i factor. Note that the argument is not included in the following expressions. In the case of tritium, the dispersion parameter, W, is always taken as $(\overline{\chi/Q})$. The R_i parameter is independent of long-term or short-term releases and should be provided in tabular form in the ODCM.

5.3.1.1 Inhalation Pathway Factor, $R_i^I[\chi/Q]$

$$R_i^I[\chi/Q] = K'(BR)_a\ (DFA_i)_a \text{ (mrem/yr per } \mu Ci/m^3)$$

where:

K' = a constant of unit conversion, 10^6 pCi/µCi.

$(BR)_a$ = the breathing rate of the receptor of age group (a), in m^3/yr.

$(DFA_i)_a$ = the maximum organ inhalation dose factor for the receptor of age group (a) for the ith radionuclide, in mrem/pCi. The total body is considered as an organ in the selection of $(DFA_i)_a$.

The breathing rates $(BR)_a$ for the various age groups are tabulated below, as given in Table E-5 of the Regulatory Guide 1.109 (Ref. 6).

Age Group (a)	Breathing Rate (m^3/yr)
Infant	1400
Child	3700
Teen	8000
Adult	8000

Inhalation dose factors $(DFA_i)_a$ for the various age groups are given in Tables E-7 through E-10 of Regulatory Guide 1.109 (Ref. 6).

5.3.1.2 Ground Plane Pathway Factor, R_i^G [D/Q]

$$R_i^G[D/Q] = K'K''(SF)DFG_i[(1-e^{-\lambda_i t})/\lambda_i] \quad (m^2 \cdot mrem/yr \text{ per } \mu Ci/sec)$$

where:

K' = a constant of unit conversion, 10^6 pCi/μCi.

K" = a constant of unit conversion, 8760 hr/year.

λ_i = the decay constant for the ith radionuclide, sec^{-1}.

t = the exposure time, 4.73×10^8 sec (15 years).

DFG_i = the ground plane dose conversion factor for the ith radionuclide (mrem/hr per pCi/m^2).

SF = the shielding factor (dimensionless).

A shielding factor of 0.7 is suggested in Table E-15 of Regulatory Guide 1.109 (Ref. 6). A tabulation of DFG_i values is presented in Table E-6 of Regulatory Guide 1.109 (Ref. 6).

5.3.1.3 Grass-Cow-Milk Pathway Factor, R_i^C [D/Q]

$$R_i^C[D/Q] = K' \frac{Q_F(U_{ap})}{\lambda_i+\lambda_w} F_m(r)(DFL_i)_a \left[\frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s)e^{-\lambda_i t_h}}{Y_s} \right] e^{-\lambda_i t_f}$$

(m^2·mrem/yr per μCi/sec)

where:

K' = a constant of unit conversion, 10^6 pCi/μCi.

Q_F = the cow's consumption rate, in kg/day (wet weight).

U_{ap} = the receptor's milk consumption rate for age (a), in liters/yr.

Y_p = the agricultural productivity by unit area of pasture feed grass, in kg/m^2.

Y_s = the agricultural productivity by unit area of stored feed, in kg/m^2.

F_m = the stable element transfer coefficients, in days/liter.

r = fraction of deposited activity retained on cow's feed grass.

$(DFL_i)_a$ = the maximum organ ingestion dose factor for the ith radionuclide for the receptor in age group (a), in mrem/pCi.

λ_i = the decay constant for the ith radionuclide, in sec^{-1}.

λ_w = the decay constant for removal of activity on leaf and plant surfaces by weathering, 5.73×10^{-7} sec^{-1} (corresponding to a 14 day half-life).

t_f = the transport time from pasture to cow, to milk, to receptor, in sec.

t_h = the transport time from pasture, to harvest, to cow, to milk, to receptor, in sec.

f_p = fraction of the year that the cow is on pasture (dimensionless).

f_s = fraction of the cow feed that is pasture grass while the cow is on pasture (dimensionless).

SPECIAL NOTE: The above equation is applicable in the case that the milk animal is a goat.

Milk cattle are considered to be fed from two potential sources, pasture grass and stored feeds. Following the development in Regulatory Guide 1.109 (Ref. 6), the values of f_p and f_s will be considered unity, in lieu of site specific information provided in the annual land census report by the licensee.

Tabulated below are the appropriate parameter values and their reference to Regulatory Guide 1.109 (Ref. 6). In the case that the milk animal is a goat, rather than a cow, refer to Regulatory Guide 1.109 for the appropriate parameter values.

Parameter		Value	Table (Ref. 6)
r (dimensionless)		1.0 for radioiodine	E-15
		0.2 for particulates	E-15
F_m (days/liter)		Each stable element	E-1
U_{ap} (liters/yr)	- Infant	330	E-5
	- Child	330	E-5
	- Teen	400	E-5
	- Adult	310	E-5
$(DFL_i)_a$ (mrem/pCi)		Each radionuclide	E-11 to E-14
Y_p (kg/m^2)		0.7	E-15
Y_s (kg/m^2)		2.0	E-15
t_f (seconds)		1.73×10^5 (2 days)	E-15
t_h (seconds)		7.78×10^6 (90 days)	E-15
Q_F (kg/day)		50	E-3

The concentration of tritium in milk is based on the airborne concentration rather than the deposition. Therefore, the R_i^C is based on $[\chi/Q]$:

$$R_i^C[\chi/Q] = K'K'''F_m Q_F U_{ap} (DFL_i)_a \; [0.75(0.5/H)] \text{ (mrem/yr per } \mu Ci/m^3)$$

where:

K''' = a constant of unit conversion, 10^3 gm/kg.

H = absolute humidity of the atmosphere, in gm/m^3.

0.75 = the fraction of total feed that is water.

0.5 = the ratio of the specific activity of the feed grass water to the atmospheric water.

and other parameters and values are given above. The value of H may be considered as 8 grams/meter^3, in lieu of site specific information (Ref. 6).

5.3.1.4 Grass-Cow-Meat Pathway Factor, $R_i^M[D/Q]$

The integrated concentration in meat follows in a similar manner to the development for the milk pathway, therefore:

$$R_i^M[D/Q] = K' \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} F_f(r)(DFL_i)_a \left[\frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s)e^{-\lambda_i t_h}}{Y_s} \right] e^{-\lambda_i t_f}$$

$$(\text{m}^2 \cdot \text{mrem/yr per } \mu\text{Ci/sec})$$

where:

F_f = the stable element transfer coefficients, in days/kg.

U_{ap} = the receptor's meat consumption rate for age (a), in kg/yr.

t_f = the transport time from pasture to receptor, in sec.

t_h = the transport time from crop field to receptor, in sec.

Tabulated below are the appropriate parameter values and their reference to Regulatory Guide 1.109 (Ref. 6).

Parameter	Value	Table (Ref. 6)
r (dimensionless)	1.0 for radioiodine	E-15
	0.2 for particulates	E-15
F_f (days/kg)	Each stable element	E-1
U_{ap} (kg/yr) - Infant	0	E-5
- Child	41	E-5
- Teen	65	E-5
- Adult	110	E-5
$(DFL_i)_a$ (mrem/pCi)	Each radionuclide	E-11 to E-14
Y_p (kg/m2)	0.7	E-15

Parameter	Value	Table (Ref. 6)
Y_s (kg/m^2)	2.0	E-15
t_f (seconds)	1.73 X 10^6 (20 days)	E-15
t_h (seconds)	7.78 X 10^6 (90 days)	E-15
Q_F (kg/day)	50	E-3

The concentration of tritium in meat is based on its airborne concentration rather than the deposition. Therefore, the R_i^M is based on $[\chi/Q]$:

$$R_i^M[\chi/Q] = K'K'''F_f Q_F U_{ap} (DFL_i)_a \; [0.75(0.5/H)] \text{ (mrem/yr per } \mu\text{Ci/m}^3)$$

where all terms are defined above and Section 5.3.1.3 of this manual.

5.3.1.5 Vegetation Pathway Factor, $R_i^V[D/Q]$

The integrated concentration in vegetation consumed by man follows the expression developed in the derivation of the milk factor. Man is considered to consume two types of vegetation (fresh and stored) that differ only in the time period between harvest and consumption, therefore:

$$R_i^V[D/Q] = K' \left[\frac{(r)}{Y_v(\lambda_i + \lambda_w)} \right] (DFL_i)_a \left[U_a^L f_L e^{-\lambda_i t_L} + U_a^S f_g e^{-\lambda_i t_h} \right]$$

(m^2·mrem/yr per μCi/sec)

where:

K' = a constant of unit conversion, 10^6 pCi/μCi.

U_a^L = the consumption rate of fresh leafy vegetation by the receptor in age group (a), in kg/yr.

U_a^S = the consumption rate of stored vegetation by the receptor in age group (a), in kg/yr.

f_L = the fraction of the annual intake of fresh leafy vegetation grown locally.

f_g = the fraction of the annual intake of stored vegetation grown locally.

t_L = the average time between harvest of leafy vegetation and its consumption, in seconds.

t_h = the average time between harvest of stored vegetation and its consumption, in seconds.

Y_v = the vegetation areal density, in kg/m^2.

and all other factors are defined in Section 5.3.1.3 of this manual.

Tabulated below are the appropriate parameter values and their reference to Regulatory Guide 1.109 (Ref. 6).

Parameter	Value	Table (Ref. 6)
r (dimensionless)	1.0 for radioiodines	E-1
	0.2 for particulates	E-1
$(DFL_i)_a$ (mrem/pCi)	Each radionuclide	E-11 to E-14
U_a^L (kg/yr) - Infant	0	E-5
- Child	26	E-5
- Teen	42	E-5
- Adult	64	E-5
U_a^S (kg/yr) - Infant	0	E-5
- Child	520	E-5
- Teen	630	E-5
- Adult	520	E-5
f_L (dimensionless)	site specific (default = 1.0)	
f_g (dimensionless)	site specific (default = 0.76) (see Ref. 6, page 28)	
t_L (seconds)	8.6×10^4 (1 day)	E-15
t_h (seconds)	5.18×10^6 (60 days)	E-15
Y_v (kg/m^2)	2.0	E-15

The concentration of tritium in vegetation is based on the airborne concentration rather than the deposition. Therefore, the R_i^V is based on [x/Q]:

$$R_i^V[x/Q] = K'K''' \quad U_a^L f_L + U_a^S f_g \quad (DFL_i)_a \quad [0.75(0.5/H)] \text{ (mrem/yr per } \mu\text{Ci/m}^3).$$

where all terms have been defined above and in Section 5.3.1.3 of this manual.

The staff has developed a computer code PARTS for calculating the R_i parameters, which is described in Appendix D of this manual.

5.4 SPECIFICATION ON THE USE OF GASEOUS RADIOACTIVE WASTE MANAGEMENT SYSTEM

Standard Technical Specification 3.11.2.4 specifies that:

"The gaseous radwaste treatment system shall be OPERABLE. The appropriate subsystems shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent releases from all release points to unrestricted areas (see Figure 5.1-1)* would result in a dose in any period of 31 days that exceeds 0.2 mrad for gamma radiation, 0.4 mrad for beta radiation, or 0.3 mrem to any organ for that same 31 day period."

The operability of the gaseous radioactive waste management system ensures that this system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The term "gaseous radwaste treatment system" includes all of the installed and available gaseous radioactive waste management system equipment, as well as their controls, power, instrumentation and services that make the system functional. Equipment that is considered standby or redundant is also included, since the function is to assure operability. The action also permits alternate treatment paths using alternate subsystems and equipment to be used in the event that the normal treatment is inoperable.

*See footnote on page 21.

This Specification provides impetus to maintain and use the gaseous radioactive waste management system as required in 10 CFR 50.36a. Maintenance and use of the gaseous radioactive waste management system components are discussed in Section 2.2 of this manual. Since features; such as, free-standing stacks or elevated vents are not considered to be treatment systems, the NRC staff considers that BWR offgas systems must be used without bypassing normal treatment during reactor operation.

To determine if use of the installed equipment, other than in BWR offgas systems, is necessary, the licensee should calculate the expected dose to an individual in the unrestricted area by projecting the plant's cumulative gaseous effluent release over a 31-day period. These releases should include all potentially radioactive plant gaseous effluents from all gaseous radioactive waste management systems and ventilation exhaust treatment systems. Calculations should include a margin, based on operating data, for anticipated operational occurrences and should use the dose calculation models discussed in Section 5.3 of this manual. The dose impact from the projected 31-day release should be compared to 0.2 mrad for gamma radiation, 0.4 mrad for beta radiation, or 0.3 mrem to any organ. If the projection indicates these values will be exceeded, then installed radwaste treatment system components, which are capable of reducing the quantities or concentrations of radioactive materials in gaseous effluents and which are capable of reducing the projected impact to less than the values specified above, must be used. The values for the projected impact, given above, corresponds to approximately one forty-eighth of the annual design dose objective values of Appendix I, Section II.B and II.C of 10 CFR Part 50 in a month, and if continued for a year, these values would correspond to less than one-fourth the values limited by Specifications 3.11.2.2.b and 3.11.2.3.b. The calculation of projected cumulative dose impact that could result from the proposed operation may use the methodology provided in Section 5.3 of this manual.

5.5 SPECIFICATION ON EXPLOSIVE GAS MIXTURE LIMITATION

The Standard Technical Specifications for BWRs and PWRs contain Specification 3.11.2.6A and alternate Specification 3.11.2.6B for the limiting conditions for operation for systems designed to treat and store radioactive gases which also contain quantities of uncombined hydrogen and oxygen. Specification 3.11.2.6A applies to a system designed to withstand a hydrogen explosion. If all components of the system, from containment to the release point, are designed and tested to 20 times the normal operating pressure, the system is considered to be designed to withstand a hydrogen explosion. Alternate Specification 3.11.2.6B applies to a system not considered to be designed to withstand a hydrogen explosion.

The functional name, "waste gas holdup system," has been used in this manual to include the various system designs found in BWRs and PWRs which serve the same basic purpose, i.e., to remove radioactive waste gases from the reactor coolant, to treat and hold gases for radioactive decay, and to monitor and control the radioactive materials in the gaseous waste prior to final release.

The potentially explosive components of the waste gas holdup system may be effectively inerted by nitrogen or steam, treated and re-used in the plant or stored and released after delay. The treatment may involve hydrogen-oxygen recombiners, filters, holdup tanks, decay pipes, charcoal adsorbers, and cryogenic stills. The Specification is provided to ensure that the concentrations of potentially explosive gases contained in the system are maintained outside the explosive envelope for hydrogen and oxygen (i.e., less than 4% H_2 by volume or less than 4% O_2 by volume). The alternate Specification 3.11.2.6B provides an additional setpoint limitation to ensure that the automatic dilution, inerting or recombiner control is functioning to maintain the relative concentration of components of potentially explosive gas mixtures outside one-half the above flammability limits (i.e., 2% H_2 and/or O_2). Based on the design, the licensee should specify the gas to be measured: hydrogen, oxygen or both hydrogen and oxygen.

5.6 SPECIFICATIONS UNIQUE TO LWR DESIGN FEATURES

The Standard Technical Specifications contain several Specifications unique to certain design features of PWRs and BWRs; in general, these Specifications contain limiting conditions for operation. The following Sections describe these limitations and the method for determining the limiting values.

5.6.1 PWR Gas Storage Tank Specification 3.11.2.7

Specification 3.11.2.7 requires that the quantity of radioactive gas in each gas storage tank at a PWR be limited to a predetermined curie content. It is not applicable to PWRs that use adsorption units for gas holdup, but is applicable for compressed gas storage and for cryogenic storage systems. The purpose of this Specification is to assure that, in the event of an uncontrolled release of the tank contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem.

Determination of the curie limit should consider the following expression;

$$\sum Q_{iT} \leq \frac{500 \text{ mrem } (3.15 \times 10^7 \text{ sec/year})}{10^6 \ \mu\text{Ci/Ci} \ \sum_i K_i (\overline{\chi/Q})_{DBA} \ (\text{mrem}\cdot\text{sec}/\mu\text{Ci}\cdot\text{yr})}$$

where:

$\sum Q_{iT}$ = The sum quantity of all noble gas nuclides (i) in a gas storage tank based on a gas mixture resulting from gaseous wastes, in curies.

K_i = The total body dose factor due to gamma emissions for each identified noble gas radionuclide, in mrem/yr per $\mu Ci/m^3$. (See Section 5.2.1 of this manual.)

$(\overline{\chi/Q})_{DBA}$ = The relative concentration at the exclusion area boundary used for evaluation of design basis accidents for ground release conditions, in sec/m^3. Guidelines are provided in Standard Review Plan 2.3.4 (Ref. 23).

Normally the major radioactive nuclide constituent in PWR waste gas storage tanks is Xe-133. Radiation monitoring and sampling of these tanks should consider the Xe-133 (equivalent) concentration. Plant procedures shall not permit operation with communication between tanks.

Alternate Specification 3.11.2.7, "PWR Waste Gas Processing System," should be used for plants that use adsorption units for gas holdup prior to release. This Specification requires that the gross radioactivity in noble gases removed from the waste gas system by means of steam jet air ejectors (or other devices) and as measured prior to entering the adsorption systems at PWR plants shall be limited by a release rate alarm setpoint with indication in the main control room. The purpose of this pretreatment continuous radiation monitor setpoint is to provide reasonable assurance that the potential accident total body dose to an individual at the exclusion area boundary will not exceed a small fraction of the limits specified in 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. Guidelines for determining the release rate limit are provided in Standard Review Plan 15.7.1 (Ref. 24), and guidance on the radiation monitoring instrumentation is given in Regulatory Guide 1.97 (Ref. 25).

5.6.2 BWR Main Condenser Evacuation System Specification 3.11.2.7

This Specification requires that the gross radioactivity in noble gases removed from the main condenser by means of steam jet air ejectors (or other devices) and as measured prior to entering the treatment, adsorption and delay systems at BWR plants shall be limited by a release rate alarm setpoint with indication in the main control room. The purpose of this pretreatment continuous radiation monitor setpoint is to provide reasonable assurance that the potential total body accident dose to an individual at the exclusion area boundary will not exceed a small fraction of the limits specified in 10 CFR Part 100 in the event this effluent including the radioactivity accumulated in the treatment system is inadvertently discharged directly to the environment without treatment.

The method for determining the specified rate limit, in μCi/sec, should use the following assumptions:

1. The release of radioactive material from the fuel is postulated to have an isotopic composition of noble gases determined from noble gas source term distribution for a 3500 MWt reactor. These values should be scaled linearly for reactors of higher and lower powers.

2. The assumptions related to the release of radioactive material from the system following the accident are:

   a. For systems which are not fully detonation-resistant or for those systems which are equipped with rupture discs that have not been isolated or loop seals which do not vent back to the main condenser, release occurs from a break just downstream of the main condenser evacuation system. Release from the main condenser evacuation system is assumed to be at ground level and a delay of five minutes is assumed to account for radioactive decay during transit from the release point to the exclusion area boundary.

NOBLE GAS SOURCE TERM

Isotope	Approx. Half-Life	Source Term, μCi/sec, at the main condenser evacuation system (SJAE) 0 Decay
Xe-140	13.7 s	1.1×10^6
Kr-90	33 s	9.8×10^5
Xe-139	41.0 s	9.8×10^5
Kr-89	3.2 m	4.6×10^5
Xe-137	3.8 m	5.3×10^5
Xe-138	14.0 m	3.1×10^5
Xe-135m	15.6 m	9.1×10^4
Kr-87	76 m	7.0×10^4
Kr-83m	1.86 hr	1.2×10^4
Kr-88	2.8 hr	7.0×10^4
Kr-85m	4.4 hr	1.1×10^4
Xe-135	9.2 hr	7.7×10^4
Xe-133m	2.3 d	1.0×10^3
Xe-133	5.27 d	2.9×10^4
Xe-131m	11.9 d	5.2×10^1
Kr-85	10.76 y	7.0×10^1
	Total	$\sim 4.7 \times 10^6$

b. For systems which are detonation-resistant (i.e., rupture discs have been isolated and loop seals are vented back to the main condenser), release from the main condenser evacuation system exits via the normal release point.

c. Activity release into the system continues for one hour following the accident at the Technical Specification limit unless positive means (such as automatic isolation] are provided to limit the releases from this source.

d. Radioiodines and activation gases may be ignored.

e. No deposition during downwind transport occurs.

f. The total radioactive inventory (neglecting radioiodines and activation gases) in any delay lines and from the process equipment is released within a two-hour period with no decay.

g. The total noble gas content of all charcoal delay beds is released over a period of two hours. A fractional release of the particulate inventory on the charcoal beds should be assumed. The rate of release is equal to the rate of absorption using the information contained in NUREG-0016 (Ref. 12).

h. The main condenser air inleakage is 6 scfm for three shell main condensers.

3. The relative concentration, $(\overline{\chi/Q})_{DBA}$, to be used is described in Section 5.6.1 of this manual.

Based on these assumptions, the applicant should backcalculate from a whole body dose of 2.5 rem to an individual at the exclusion area boundary to obtain the main condenser evacuation system rate limit, in μCi/sec, having a noble gas isotopic distribution proportional to the noble gas source term, given above. For licensed BWR facilities that have a Technical Specification limit, in μCi/sec, based on a 5 rem consequence criteria, a reevaluation of the specified limit is not required. Guidance on the radiation monitoring instrumentation is given in Regulatory Guide 1.97 (Ref. 25).

5.6.3 PWR Monitoring of Steam Generator Blowdown Flask Tank Vent

Standard Technical Specification 3.3.3.9, including Tables 3.3-12* and 4.3-12,* requires that PWRs continuously monitor the steam generator blowdown tank vent for gross noble gas radioactivity and continuously sample for radioactive iodines and particulates. Many PWRs with U-tube steam generators direct their blowdown to a blowdown treatment system without venting, so that item f in Tables 3.3-12 and 4.3-12 is not applicable. Others vent their flash tank to tne main condenser, where the airborne radionuclides are either removed by condensing steam or drawn into the main condenser evacuation system, where they are then monitored prior to release to the atmosphere; therefore, this design provision makes item f not applicable. However, there are several operating PWR's that direct their blowdown to a flash tank which is vented directly to the atmosphere. Monitoring these releases presents serious difficulties, due to the presence of steam in the exhaust. In lieu of a flash tank vent radiation monitor, a determination of the release of radioiodine-131 via the flash tank vent can be made by calculating from a measured concentration in the secondary water by the following equation:

$$\dot{Q}_y = \bar{C}_y [R_{SGB}] f_{FT} (1 - SQ_{FTv})$$

where:

Q_y = The release rate of radioiodine-131, y, from the steam generator flash tank vent, in μCi/sec.

$\bar{C}_y$ = The concentration of radioiodine-131, y, in the secondary coolant water averaged over not more than one week, in μCi/ml.

R_{SGB} = The steam generator blowdown rate to the flash tank, in ml/sec.

f_{FT} = The fraction of blowdown flashed in the falsh tank determined from a heat balance taken around the flash tank at the applicable reactor power level.

SQ_{FTv} = The measured steam quality in the flash tank vent; or an assumed value of 0.85, based on NUREG-0017 (Ref. 11).

If this option is chosen, the applicant shall perform this calcuation every time measurements of secondary water radioiodine concentrations are required by Technical Specifications, and the calculated release shall be assumed at this calculated level until the next secondary water analysis is completed. These calculations shall be provided by the applicant in his semiannual effluent release report.

*See footnote on page 20.

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3. U.S. Nuclear Regulatory Commission, "Draft Radiological Effluent Technical Specifications for PWR's," USNRC NUREG-0473, Revision 1, Washington, D.C. 20555, October 1978.

4. Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications," Revision 4, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, August 1975.

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7. Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, March 1976.

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9. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, April 1976.

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12. U.S. Nuclear Regulatory Commission, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)," USNRC Report NUREG-0016, Washington, D.C. 20555, April 1976.

13. U.S. Nuclear Regulatory Commission, "XOQDOQ, Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations," USNRC Report NUREG-0324, Washington, D.C. 20555, September 1977.

14. Title 10, "Energy," Chaper I, Code of Federal Regulations; Part 20, pages 144 to 172, U.S. Government Printing Office, Washington, D.C. 20402, January 1, 1977.

15. Title 10, "Energy," Chapter I, Code of Federal Regulations; Part 100, pages 409 to 421, U.S. Government Printing Office, Washington, D.C. 20402, January 1, 1977.

16. Title 40, "Protection of Environment," Chapter I, Code of Federal Regulations, Part 141, pages 169 to 182, U.S. Government Printing Office, Washington, D.C. 20402, January 1, 1977.

17. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, June 1974.

18. Title 40, "Protection of Environment," Chapter I, Code of Federal Regulations, Part 190, Federal Register, Vol. 42, No. 9, pages 2858 to 2861, Washington, D.C. 20402, January 13, 1977.

19. "SKYSHINE, A Computer Program for Evaluating Effects of Structure Design on N-16 Gamma-Ray Dose Rates," *Radiation Research Associates*, Inc., Report RRA-T7209, Fort Worth, Texas, 76107, November 1972.

20. Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment," U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, December 1977.

21. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," Revision 2, USNRC Report NUREG-75/094, Washington, D.C. 20555, September 1975.

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ADDENDUM

Setpoint Calculations

The radiological effluent Technical Specifications require alarm/trip setpoints for radiation monitors and flow measurement devices for each effluent line. Setpoint values are to be calculated to assure that alarm and trip actions occur prior to exceeding the limits of 10 CFR 20 at the release point to the unrestricted area. The calculated alarm and trip action setpoints to be specified in the ODCM for each radioactive liquid effluent line monitor and flow measurement device must satisfy the following equation:

$$\frac{cf}{F+f} \leq C$$

where:

C = the effluent concentration limit (Specification 3.11.1) implementing 10 CFR 20 for the site, in μCi/ml

c = the setpoint, in μCi/ml, of the radioactivity monitor measuring the radioactivity concentration in the effluent line prior to dilution and subsequent release; the setpoint, which is proportional to the volumetric flow of the effluent line and inversely proportional to the volumetric flow of the dilution stream plus the effluent stream, represents a value which, if exceeded, would result in concentrations exceeding the limits of 10 CFR 20 in the unrestricted area

f = the flow setpoint as measured at the radiation monitor location, in volume per unit time, but in the same units as F, below

F = the dilution water flow setpoint as measured prior to the release point, in volume per unit time.

[Note that if no dilution is provided, $c \leq C$. Also, note that when (F) is large compared to (f), then $F+f \cong F$.]

The equation is satisfied when the following alarm/trip setpoints are provided for each effluent line in the ODCM:

$$f \leq \frac{CF}{c} \quad \text{(in ml/sec; for example).}$$

$$F \leq \frac{cf}{C} \quad \text{(in ml/sec; for example).}$$

$$c \leq \frac{CF}{f} \quad \text{(in μCi/ml; for example).}$$

Some plants may be operated using a fixed value for one or more of these three variables, c, f or F.

Example 1

By using a constant capacity radwaste system discharge pump (on the undiluted stream) the value of (f) is fixed; therefore, the setpoints to be given in the ODCM are:

f = ______________________ ml/sec (fixed)

$F >$ ______________________ ml/sec = cf/C

c = ______________________ xF μCi/ml $\leq$ CF/f

If $C = 3x10^{-8}$ μCi/ml, $f = 4000$ ml/sec and $F > 4x10^6$ ml/sec, the radiation monitor setpoint is calculated as follows:

$$c \leq CF/f$$

$$= \frac{(3x10^{-8})F}{4000} = 7.5x10^{-12}F \text{ μCi/ml.}$$

If F is measured at some value in excess of the limiting value (the limiting value is $4x10^6$ ml/sec in this example), then c may be established proportionately. If $F = 8x10^6$ ml/sec, the alarm setpoint is:

$$c = 7.5x10^{-12}F \text{ (μCi/ml per ml/sec)(ml/sec)}$$

$$= 7.5x10^{-12}(8x106) = 6x10^{-5} \text{ μCi/ml.}$$

In this case, the alarm setpoint for the radioactive liquid effluent line monitor can be established at $6x10^{-5}$ μCi/ml, provided that an automatic isolation/control trip action occurs to satisfy the condition:

$$c/F < 7x5x10^{-12} \text{ μCi/ml per ml/sec.}$$

Example 2

By using a constant capacity dilution pump (on the dilution stream prior to a mixing box), the value of (F) is fixed; therefore, the setpoints to be given in the ODCM are:

$f <$ ______________________ ml/sec = CF/c

F = ______________________ ml/sec (fixed)

c = ______________________ x(1/f)μCi/ml $\leq$ CF/f

If $C = 3x10^{-8}$ μCi/ml, $F = 4x10^6$ ml/sec and $f < 4000$ ml/sec, the radiation monitor setpoint is calculated as follows:

$$c \leq CF/f$$

$$= \frac{(3x10^{-8} \text{ X } 4x10^6)}{f} = 0.12(1/f) \text{ μCi/ml.}$$

If f is measured at some value less than the limiting value (the limiting value is 4000 ml/sec in this example), then c may be established proportionately. If f = 1000 ml/sec, the alarm setpoint is:

$$c = 0.12(1/f)\text{(μCi/sec)(sec/ml)}$$

$$= \frac{0.12}{1000} = 1.2x10^{-4} \text{ μCi/ml.}$$

In this case, the alarm setpoint for the radioactive liquid effluent line monitor can be established at 1.2×10^{-4} μCi/ml, provided that an automatic isolation/control trip action occurs to satisfy the condition:

$$cf > 0.12 \ \mu Ci/sec.$$

Value of c

A detailed description of the method to be used to obtain the value of (c) should be provided in the ODCM. Since (c) is dependent on the radionuclide distribution, yields, calibration and the monitor's parameters, each of these variables should be considered and the fixed or adjustable setpoint method of determination described in the ODCM for each effluent monitor. This may be accomplished by tabulation. Changes to the ODCM shall be provided in the SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT.