U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE RG 1.21, REVISION 3

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MEASURING, EVALUATING, AND REPORTING RADIOACTIVE MATERIAL IN LIQUID AND GASEOUS EFFLUENTS AND SOLID WASTE

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for the following uses:

(1) measuring, evaluating, and reporting licensed (plant-related) radioactivity in effluents and solid radioactive waste shipments from nuclear power plants and spent fuel storage facilities, and

(2) assessing and reporting the public dose to demonstrate compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation" (Ref. 1), Title 40, (40 CFR) Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations" (Ref. 2), and nuclear power plant Technical Specifications.

This guide incorporates the risk-informed principles of the Reactor Oversight Process. A risk-informed, performance-based approach to regulatory decision making combines the "risk-informed" and "performance-based" elements discussed in the staff requirements memorandum to SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," dated February 24, 1999 (Ref. 3).

Applicability

This RG is a Division 1, "Power Reactors" RG, which applies to nuclear power plant licensees and applicants subject to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation," This RG is also applicable to specific and general licensees under 10 Part 72 for storage of spent fuel.

Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site in the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/,under Document Collections, in Regulatory Guides, at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html.

Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC's public Web site in the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/, under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/readingrm/adams.html, under ADAMS Accession Number (No.) ML21139A224. The regulatory analysis may be found in ADAMS under Accession No. ML20287A434. The associated draft guide DG-1377 may be found in ADAMS under Accession No. ML20287A423, and the staff responses to the public comments on DG-1377 may be found under ADAMS Accession No. ML21132A226.

This includes licenses issued under the following regulations:

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 4), applies to the licensing of production and utilization facilities.

- 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 5), applies to applicants and holders of combined licenses, standard design certifications, standard design approvals, and manufacturing licenses.

- 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste" (Ref. 6), applies to general licenses issued under Part 72 and to applicants for and holders of specific licenses under Part 72.

Applicable Regulations

The following regulations establish the regulatory basis for the radiological effluent control program:

- 10 CFR Part 20, "Standards for Protection Against Radiation"

    - 10 CFR 20.1003, "Definitions," defines terminology that is used in the regulations and in this regulatory guide.

    - 10 CFR 20.1301, "Dose limits for individual members of the public," establishes radiation dose limits for individual members of the public.

    - 10 CFR 20.1302, "Compliance with dose limits for individual members of the public," requires licensees to perform surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public.

    - 10 CFR 20.1402, "Radiological criteria for unrestricted use," establishes acceptance criteria for license termination to achieve the site's unrestricted use status after decommissioning.

    - 10 CFR 20.1501, "General," establishes requirements for performing radiological surveys.

    - 10 CFR 20.2001, "General requirements" (for waste disposal), establishes methods for disposing of licensed material.

    - 10 CFR 20.2103, "Records of surveys," requires licensees to maintain records of surveys and calibrations.

    - 10 CFR 20.2107, "Records of dose to individual members of the public," requires licensees to maintain records that demonstrate compliance with dose limits for members of the public.

- 10 CFR 20.2108, “Records of waste disposal,” requires licensees to maintain records of the disposal of licensed material.

- 10 CFR Part 20, Appendix B, “Annual Limits on Intakes (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewage,” establishes intake limits and airborne and liquid concentration limits for occupational exposure and member of the public exposure.

- 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”

  - 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors,” establishes numerical guides for design objectives and limiting conditions of operation to control radioactive effluents.

  - 10 CFR 50.36a, “Technical specifications on effluents from nuclear power reactors,” requires licensees to establish technical specifications with operating procedures and controls be established and followed and that the radioactive waste system be maintained and used.

  - 10 CFR 50.75, “Reporting and record keeping for decommissioning planning,” paragraph (g,) requires licensees to keep records of information important to decommissioning.

  - 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 60, “Control of Releases of Radioactive Materials to the Environment,” specifies that the nuclear power unit design shall include means to control suitably liquid and gaseous effluents and solid waste.

  - 10 CFR Part 50, Appendix A, GDC 64, “Monitoring Radioactivity Releases,” specifies that means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant fluids, effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, anticipated operational occurrences, and from postulated accidents.

  - 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low As Is Reasonably Achievable’ (ALARA) for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” establishes design objectives on a per reactor basis for meeting the requirements of 10 CFR 50.34a.

- 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants”

  - 10 CFR 52.0, “Scope,” requires Part 52 licensees to comply with all requirements in 10 CFR Chapter I that are applicable, which includes, for example, 10 CFR Part 20 as discussed above.

- 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste”

- 10 CFR 72.44(d) requires that each specific license must include technical specifications that establishes limits on the release of radioactive materials and the ALARA objectives for effluents and that require establishment of an environmental monitoring program to ensure compliance with those limits.

- 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," (Monitored Retrievable Storage Installation) establishes dose limits to any real individual (excluding occupational exposures) beyond the Part 72 controlled area (as defined in 10 CFR 72.3 and meeting the minimum size requirements in 72.106(b)).

- 10 CFR 72.126, "Criteria for radiological protection," requires radiation protection systems be provided with effluent and direct radiation monitoring systems and controls to limit releases to ALARA under normal conditions and control releases under accident conditions and ensure limits relating to releases to the general environment will not be exceeded.

- 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations"

  - 40 CFR 190.10, "Standards for normal operation," establishes standards for normal operations and annual dose equivalent standards and limits on the total quantity of radioactive materials entering the environment from the entire uranium fuel cycle.

  - 40 CFR 190.11, "Variances for unusual operations," establishes variances (allowances) for unusual operations where the standards in 40 CFR 190.10 may be exceeded.

- 40 CFR Part 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel and Transuranic Radioactive Wastes" (Ref. 7)

  - 40 CFR 191.03(a), "Standards," establishes standards for the management and storage of spent nuclear fuel or transuranic radioactive wastes.

Related Guidance

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants" (Ref. 8), provides guidance for an onsite meteorological measurements program.

- RG 1.97, Revisions 0, 1, 2 and 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," issued December 1975, August 1977, and December 1980, and May 1983, respectively (Ref. 9), provides guidance on instrumentation used to monitor plant variables and systems during and following an accident.

- RG 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," issued June 2006 (Ref. 10), endorses (with certain clarifying regulatory positions) the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations" (Ref. 11).

- RG 1.97, Revision 5, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” issued April 2019 (Ref. 12), endorses, with exceptions and clarifications, IEEE Std. 497-2016, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations” (Ref. 13).

- RG 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Demonstrating Compliance with 10 CFR Part 50, Appendix I” (Ref. 14), describes basic features of calculational models and assumptions used for the estimation of doses to the public.

- RG 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors” (Ref. 15), describes models and assumptions for the estimation of atmospheric dispersion of gaseous effluent releases.

- RG 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” (Ref. 16), provides acceptable methods for applicants to construct a nuclear power reactor to calculate realistic radioactive source terms for use in evaluating radioactive waste treatment systems to determine whether the design objectives of 10 CFR Part 50, Appendix I, are met, and to assess the environmental impact of radioactive effluents.

- RG 1.113, “Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I” (Ref. 17), describes general approaches for the analysis of releases of liquid effluents into surface water bodies.

- RG 1.184, "Decommissioning of Nuclear Power Reactors" (Ref. 18), provides guidance that during decommissioning, Technical Specifications require operational procedures for the control of effluent releases and submittal of annual effluent reports as specified by 10 CFR 50.36a.

- RG 1.185, “Standard Format and Content for Post-Shutdown Decommissioning Activities Report” (Ref. 19), identifies information licensees should provide to NRC and the public of the licensees expected decommissioning activities and schedule.

- RG 4.1, “Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants” (Ref. 20), describes acceptable programs for establishing and conducting an environmental monitoring program.

- RG 4.13, “Environmental Dosimetry—Performance Specifications, Testing, and Data Analysis” (Ref. 21), provides specifications for environmental dosimetry and methods of analyzing dosimetry to determine dose to members of the public.

- RG 4.15, “Quality Assurance for Radiological Monitoring Programs (Inception Through Normal Operations to License Termination)—Effluent Streams and the Environment” (Ref. 22), describes design and implementation programs to ensure the quality of the results of measurements of radioactive materials in the effluents from, and environment outside of, facilities that process, use, or store radioactive materials.

- RG 4.20, “Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors” (Ref. 23), provides guidance for meeting the constraint on airborne emissions of radioactive material as described in 10 CFR 20.1101(d).

- RG 4.25, “Assessment of Abnormal Radionuclide Discharges in Groundwater to the Unrestricted Area at Nuclear Power Plant Sites” (Ref. 24), describes an approach that is acceptable for use in assessing abnormal discharges of radionuclides in groundwater from the subsurface to the unrestricted area at nuclear power plant sites.

- Generic Letter (GL) 89-01, “Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of Technical Specifications and the Relocation of Procedural Details to the Offsite Dose Calculation Manual or Process Control Program,” dated January 31, 1989 (Ref. 25), provides guidance that the programmatic controls of the (former) Radiological Effluent Technical Specifications can be implemented in the Administrative Controls section of the TS and that the procedural details can be relocated to the licensee-controlled Offsite Dose Calculation Manual (ODCM) and Process Control Program (PCP) or equivalent documents.

- NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants" issued October 1978 (Ref. 26), is one of the bases documents for the Radioactive Effluent Controls Program in Standard Technical Specifications (section 5.5.4).

- NUREG-0016, Revision 1 and Revision 2, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling-Water Reactors (GALE-BWR 3.2 Code),” issued January 1979 and July 2020, respectively (Ref. 27), is a computerized mathematical model for calculating the release of radioactive materials in gaseous and liquid effluents from boiling-water reactors (BWRs).

- NUREG-0017, Revision 1 and Revision 2, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (GALE-PWR 3.2 Code),” issued April 1985 and July 2020, respectively (Ref. 28), is a computerized mathematical model for calculating the release of radioactive materials in gaseous and liquid effluents from pressurized-water reactors (PWRs).

- NUREG-0543, “Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (CFR Part 190),” issued January 1980 (Ref. 29), explains the rationale for using Appendix I to demonstrate compliance with 40 CFR 190 and methods for demonstrating compliance when radioactive effluents exceed Appendix I numerical guidance.

- NUREG-0737, “Clarification of TMI Action Plan Requirements,” issued November 1980 (Ref. 30), provides specific items that were approved by the NRC Commission following the accident at Three Mile Island Nuclear Station (TMI) for implementation at reactors.

- NUREG-1301, “Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors,” issued April 1991 (Ref. 31), provides the PWR effluent controls that may be removed from technical specifications and incorporated into the licensee’s ODCM (or equivalent).

- NUREG-1302, “Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors,” issued April 1991 (Ref. 32), provides the BWR effluent controls that may be removed from technical specifications and incorporated into the licensee’s ODCM (or equivalent).

- NUREG-1575, Rev. 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," issued August 2000 (Ref. 33), provides information on planning, conducting, evaluating, and documenting building surface and surface soil final status radiological surveys for demonstrating compliance with dose or risk-based regulations or standards.

- NUREG-1576, "Multi-Agency Radiological Laboratory Analytical Protocols Manual," issued July 2004 (Ref. 34), provides guidance for the planning, implementation, and assessment of projects that require the laboratory analysis of radionuclides.

- NUREG-1757, Volume 2, Revision 1, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," issued September 2006 (Ref. 35), provides guidance on compliance with 10 CFR Part 20, Subpart E – "Radiological Criteria for License Termination."

- NUREG-1940, "RASCAL 4: Description of Models and Methods," issued December 2012 (Ref. 36), provides a description of an emergency response consequence assessment tool including models and methods for source term calculations, atmospheric dispersion and deposition, and dose calculations.

- NUREG-1940, Supplement 1, "RASCAL 4.3: Description of Models and Methods," issued May 2015 (Ref. 37), describes the Radiological Assessment System for Consequence Analysis (RASCAL) models and methods for source term calculations, atmospheric dispersion and deposition, and dose calculations for accident analysis.

- NUREG/CR-6948, "Integrated Ground-Water Monitoring Strategy for NRC-Licensed Facilities and Sites," issued November 2007 (Ref. 38), presents a framework for assessing what, where, when, and how to monitor contamination in groundwater.

- NUREG/CR-6805, "A Comprehensive Strategy of Hydrogeology Modeling and Uncertainty Analysis for Nuclear Facilities and Sites," issued July 2003 (Ref. 39), describes a strategy for a systematic and comprehensive approach to hydrogeologic conceptualization, model development, and predictive uncertainty analysis.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. RGs are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 20, 50, 52, 72, that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), approval numbers 3150-0014, 3150-0011, 3150-0151, and 3150-0132, respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to

Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0014, 3150-0011, 3150-0151, and 3150-0132), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC20503; e- mail: oira_submission@omb.eop.gov.

Public Protection Notification

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B. DISCUSSION

Reason for Revision

This revision of RG 1.21 (Revision 3):

- Provides guidance and acceptable methods for calibration of accident-range radiation monitors,
- Revises guidance on recommendations for updating long-term, annual average χ/Q and D/Q values,
- Clarifies reporting requirements for low level radioactive waste (LLW) shipments, specifically that the report includes the waste shipped from the unit (plant site), and that waste classification does not need to be reported when shipped from the unit (plant site) to a waste processor,
- Clarifies the existing guidance in NUREG-1301 and NUREG-1302 that environmental monitoring for iodine (I) -131 in drinking water should be performed if a prospective dose evaluation of the annual thyroid dose from I-131 to a person in any age group from the drinking water route of exposure is greater than one mrem.
- Clarifies the existing process as currently described in Technical Specifications for making changes to effluent and environmental programs, and,
- Incorporates the existing Regulatory Issue Summary 2008-03, "Return/Reuse of Previously Discharged Radioactive Effluents" (Ref. 40).

Background

Six basic documents contain the primary regulatory guidance for implementing the 10 CFR Part 20 and 10 CFR Part 50 regulatory requirements and plant technical specifications related to monitoring and reporting of radioactive material in effluents and environmental media, solid radioactive waste shipments, and the public dose that results from licensed operation of a nuclear power plant:

(1) RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,"

(2) RG 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants,"

(3) RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Inception Through Normal Operations to License Termination)—Effluent Streams and the Environment,"

(4) RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Demonstrating Compliance with 10 CFR Part 50, Appendix I,"

(5) NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors," and

(6) NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors."

These documents, when used in an integrated manner, provide the basic guidance and implementation details for developing and maintaining effluent and environmental monitoring programs at nuclear power plants. RG 1.21, RG 4.1, RG 4.15, and RG 1.109 specify the guidance for radiological monitoring and the assessment of dose, and NUREG-1301 and NUREG-1302 provide specific implementation details for the effluent and environmental monitoring programs.

RG 1.21 addresses the measuring, evaluating, and reporting of effluent releases, solid radioactive waste shipments, and public dose from nuclear power plants. The guide describes the important concepts in planning and implementing an effluent and solid radioactive waste program. Concepts covered include meteorology, release points, monitoring methods, identification of principal radionuclides, unrestricted area boundaries, continuous and batch release methods, representative sampling, composite sampling, radioactivity measurements, decay corrections, quality assurance (QA), solid radioactive waste shipments, and public dose assessments. The dose to occupational workers, including contributions from activities associated with effluent programs (such as LLW processing, storage, and shipping, as well as dose from handling resins and filters for gaseous and liquid radioactive waste), is occupational dose associated with the licensed operation and is not included in RG 1.21.

RG 4.1 addresses the environmental monitoring program. The guide discusses principles and concepts important to environmental monitoring at nuclear power plants. The RG provides guidance on both the preoperational and operational Radiological Environmental Monitoring Programs (REMP) for the routinely monitored exposure pathways (inhalation, ingestion, and direct radiation). The guide defines the sampling media and sampling frequency, and the methods of comparing environmental measurements to effluent releases in the Annual Radiological Environmental Operating Report (AREOR).

RG 4.15 provides the basic principles of QA in all types of radiological monitoring programs for effluent streams and the environment. The guide provides principles for structuring organizational lines of communication and responsibility, using qualified personnel, implementing standard operating procedures, defining data quality objectives (DQOs), performing quality control (QC) checking for sampling and analysis, auditing the process, and taking corrective actions.

RG 1.109 provides the detailed implementation guidance for demonstrating that radioactive effluents conform to ALARA design objectives of 10 CFR Part 50, Appendix I. The RG describes calculational models and parameters for estimating dose from effluent releases, including the dispersion of the effluent in the atmosphere and surface water bodies.

NUREG-1301 and NUREG-1302 provide the detailed implementation guidance by describing effluent and environmental monitoring programs. These NUREGs provide guidance on meeting effluent monitoring and environmental sampling requirements, surveillance requirements for effluent monitors, types of monitors and samplers, sampling and analysis frequencies, types of analysis and radionuclides analyzed, lower limits of detection (LLDs), specific environmental media to be sampled, and reporting and program evaluation and revision.

Objectives of the Radiological Effluent Controls Program

The requirements for the radiological effluent control program are in 10 CFR Part 20 and the technical specifications that are part of a license, including limitations on dose conforming to 10 CFR Part 50, Appendix I. In addition, a facility's technical specifications describe specific regulatory requirements. Licensees can use these regulatory requirements and the RG 1.21 regulatory guidance as a basis for establishing the radiological effluent control program. The radiological effluent control program for a nuclear power plant has the following six basic objectives, which are also reflected in 10 CFR 50.36a and in site-specific Technical Specifications:

- ensure that effluent instrumentation has the functional capability to measure and analyze effluent discharges,
- ensure that effluent treatment systems are used to reduce effluent discharges to ALARA levels,
- establish instantaneous release-rate limitations on the concentrations of radioactive material,
- limit the annual and quarterly doses or dose commitment to members of the public in liquid and gaseous effluents to unrestricted areas,
- measure, evaluate, and report the quantities of radioactivity in gaseous effluents, liquid effluents, and solid radioactive waste shipments, and
- evaluate the dose to members of the public.

As required by technical specifications, Part 50 and Part 52 licensees must submit the Annual Radioactive Effluent Release Report (ARERR) before May 1 and the AREOR by May 15 of each year (unless a licensing basis exists for a different submittal date for one or both reports). Licensees use these reports to demonstrate compliance with the facility's technical specifications for the radioactive effluent control program. The reports demonstrate the following:

- effectiveness of effluent controls and measurement of the environmental impact of radioactive materials,
- compliance with the design objectives and limiting conditions for operation required to meet the ALARA criteria in 10 CFR Part 50, Appendix I,
- relationship between quantities of radioactive material discharged in effluents and resultant radiation dose to individuals,
- compliance with the radiation dose limits to members of the public established by the NRC and the U.S. Environmental Protection Agency (EPA), and
- compliance with the effluent reporting requirements of 10 CFR 50.36a[1].

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1 See Section C.9 of this regulatory guide for information regarding use of the ARERR or its format to also meet ISFSI effluent reporting requirements in 10 CFR 72.44(d) for specific licenses or imposed by certificate of compliance conditions for general licenses.

Consideration of International Standards[2]

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. These requirements and guides provide a system of Safety Standards Categories that reflect an international perspective on what constitutes a high level of safety. In developing or updating Regulatory Guides the NRC has considered IAEA Safety Requirements, Safety Guides, and other relevant reports in order to benefit from the international perspectives, pursuant to the Commission's International Policy Statement (Ref. 41) and NRC Management Directive and Handbook 6.6 (Ref. 42).

The following IAEA Safety Standards Series are consistent with the basic safety principles considered in developing this Regulatory Guide:

- IAEA General Safety Guide (GSG)-8, "Radiation Protection of the Public and the Environment," issued 2018 (Ref. 43)

- IAEA Specific Safety Guide NS-G-3.2, "Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants," issued 2002 (Ref. 44)

- IAEA GSG-9, "Regulatory Control of Radioactive Discharges to the Environment," issued 2018 (Ref. 45)

- IAEA GSG RS-G-1.8, "Environmental and Source Monitoring for Purposes of Radiation Protection," issued 2005 (Ref. 46)

- IAEA Nuclear Energy Series NP-T-3.16, "Accident Monitoring Systems for Nuclear Power Plants," issued 2015 (Ref. 47)

- IAEA-TECDOC-482, "Prevention and Mitigation of Groundwater Contamination from Radioactive Releases," Vienna, Austria, issued 1988 (Ref. 48)

- IAEA Safety Guide No. WS-G-3.1, "Remediation Process for Areas Affected by Past Activities and Accidents," Vienna, Austria, issued 2007 (Ref. 49)

- IAEA, "Management of Waste Containing Tritium and Carbon-14," Technical Report Series Number 421, Vienna, Austria, issued 2004 (Ref. 50)

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2 IAEA Safety Requirements and Guides may be found at https://www.iaea.org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria; telephone (+431) 2600-0; fax (+431) 2600-7; or e-mail Official.Mail@IAEA.Org. It should be noted that some of the international recommendations do not correspond to the NRC requirements which take precedence over the international guidance.

C. STAFF REGULATORY GUIDANCE

1. Effluent Monitoring

1.1 Effluent Monitoring Programs

Monitoring programs shall be established to identify and quantify principal radionuclides in effluents in accordance with 10 CFR 50.36a. NUREG-1301 (for PWRs) and NUREG-1302 (for BWRs) provide guidance on acceptable methods of generic controls and surveillance requirements, including frequency, duration, and methods of measurement. These NUREGs provide acceptable LLDs, guidance on batch releases and continuous releases, sampling frequencies, analysis frequencies and timelines, and composite sample guidance. Site-specific radiological effluent control programs that differ from the generic NUREG-1301 and NUREG-1302 guidance should be based on a documented evaluation or justification for such deviations as part of an ODCM authorized change, or, if submitted and approved as part of the original ODCM, in accordance with GL 89-01.

1.2 Release Points for Effluent Monitoring

The ODCM (or equivalent), as required by technical specifications, should identify the facility's significant release points (see definition in the glossary) used to quantify liquid and gaseous effluents discharged to the unrestricted area. For those release points containing contributions from two or more inputs (or systems), it is preferable to monitor each major input (or system) individually to avoid dilution effects, which may impede or prevent radionuclide identification. NUREG-1301 and NUREG-1302 contain detailed guidance for the content and format of a licensee's ODCM. For purposes of effluent and direct radiation monitoring, the ODCM should list and describe the following:

1. significant release points (see definition in Section 1.3 and in the glossary), which include stacks, vents, and liquid radioactive waste discharge points, among others;

2. less-significant release points (see definition in Section 1.4 and in the glossary) that are not normally classified as one of the significant release points but could become a significant release point based on expected operational occurrences (e.g., primary to secondary leakage for PWRs or failed fuel)[3];

3. the site environs map, which should show each of the following:

    a. significant release points,

    b. boundaries of the restricted area and the controlled area[4] (in accordance with 10 CFR Part 20 definitions),

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3 This list does not need to be exhaustive or all-inclusive but should demonstrate that the licensee has reasonably anticipated expected operational occurrences and their effects on radioactive discharges. Examples may include main steam line safety valves, steam-driven feedwater pumps, turbine building sumps, containment ice condensers, leachate seepage from unlined ponds, or evaporative releases from ponds in the restricted or controlled areas.

4 For ODCMs that also address Part 72 monitoring requirements, the boundaries of the Part 72 controlled area, as defined in 10 CFR 72.3 and meeting the minimum size requirements of 72.106 should be also be shown.

c. boundary of the unrestricted area[5] for liquid effluents (e.g., at the end of the pipe or entrance to a public waterway), and

d. boundary of the unrestricted area for gaseous effluents (e.g., the site boundary).

4. dose calculation methodologies for exposure pathways and routes of exposure that are identified in RG 1.109, if applicable; and

5. dose calculation methodologies for direct radiation if necessary (e.g., when assessing direct radiation from the facility)[6].

1.3 Monitoring a Significant Release Point

A significant release point is any location from which radioactive material is released that contributes greater than 1 percent of the activity discharged from all the release points for a particular type of effluent considered. RG 1.109 lists the three types of effluent as (1) liquid effluents, (2) noble gases released to the atmosphere, and (3) all other radionuclides discharged to the atmosphere.

The ODCM should list significant release points. Significant release points should be monitored in accordance with the ODCM. If a new significant release point is identified and is not listed in the ODCM, licensees should (1) establish an appropriate sampling interval (e.g., in site-specific procedures) and (2) update the ODCM within a reasonable timeframe (e.g., annually). Releases from a significant release point should be assessed based on an appropriate combination of actual sample analysis results, radiation monitor responses, flow rate indications, tank level indications, and system pressure indications as necessary to ensure that the amount of radioactive material released, and the corresponding doses, are not substantially underestimated (see 10 CFR Part 50, Appendix I, Section III, “Implementation”). If activity is detected when monitoring a significant release point, the radionuclides detected should be reported in the effluent totals (including those with half-lives less than 8 days) in the ARERR (i.e., in Table A-1 or Table A-2), provided that the amount discharged is significant to the three-digit exponential format required for the ARERR.

1.4 Monitoring a Less-Significant Release Point

NUREG-1301 and NUREG-1302 provide tables designating sampling and analysis frequencies for release points. Historically, these tables, together with the guidance from RG 1.21, Revision 1, issued June 1974 (Ref. 51) or RG 1.21, Revision 2, issued June 2009 (Ref. 52) provide sampling and analysis frequencies. Licensees may continue to use the guidance from NUREG-1301 or NUREG-1302 and/or Revision 1 or Revision 2 of RG 1.21 in accordance with their ODCMs. This method of assigning sample frequencies is simple to implement but, in certain cases, may entail an inappropriately large number of samples for less-significant release points with no—or extremely low—impact on the parameters reported in the ARERR. As a result, for less-significant release points, licensees may evaluate and assign more appropriate sampling frequencies. If a licensee wishes to deviate from the NUREG-1301 and NUREG-1302 sampling frequencies, the licensee’s evaluation must show that the changes (i.e., deviations from NUREG-1301 and NUREG-1302) maintain the levels of radioactive effluent control as stated in the technical specifications required by 10 CFR 20.1302; 40 CFR Part 190; 10 CFR 50.36a; and

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5 The boundaries of the unrestricted areas may be defined separately for liquid effluents, gaseous effluents, and if appropriate, for other radiological controls such as direct radiation.

6 The methodology should include background subtraction, and if appropriate, extrapolation of radiation measurements to points of interest (e.g., to the individual members of the public likely to receive the highest dose).

10 CFR Part 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations, and should be maintained in site documentation. Regardless of the surveillance frequencies, if activity is detected when monitoring a less-significant release point, the licensee must (in accordance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section III.A.1) report the cumulative activity in the effluent totals (i.e., in Table A-1 or Table A-2) in the ARERR (provided that the amount discharged is significant to the three-digit exponential format required for the ARERR).

Site documentation should identify less-significant release points, to the extent reasonable, but it is not necessary to list all possible release points in site documentation. Releases from a less-significant release point may be assessed (see Section 5.1) to the extent reasonable using assumptions and bounding calculations (in lieu of, or in addition to, sampling and analysis). When plant conditions change and such changes may reasonably affect the status of a less-significant release point (e.g., significant change in primary-to-secondary leakage in PWRs or substantial cross contamination between systems), the licensee should sample and analyze the affected less-significant release points. These sample results should be evaluated to (1) confirm the continued validity of the bounding calculations (if used) with regard to effluent accountability and (2) determine the impact (if any) on effluent accountability. The guidance in this RG on monitoring less-significant release points for purposes of accountability (through the ARERR) does not replace, supersede, or otherwise modify any responsibility for monitoring systems normally not contaminated, as outlined in NRC Inspection and Enforcement Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment," issued May 1980 (Ref. 53). A thoroughly designed and documented evaluation of a less-significant release point could also assist in the evaluation and characterization of abnormal releases and abnormal discharges (see Section 1.11 below).

1.5 Monitoring Leaks and Spills

An area where an unplanned release occurred in the onsite environs (e.g., a leak or spill) should be identified as an "impacted area," as defined in 10 CFR 50.2, "Definitions," for decommissioning purposes, and in accordance with NUREG-1757. A leak or spill should be assessed to obtain the necessary information for the ARERR, as specified in Section 9.5.1 of this RG.

Leaks or spills to the ground and/or subsurface will be diluted on contact with soil and water in the environment; therefore, samples of the undiluted liquid (from the source of the leak or spill) and samples of the affected soil (or surface water or subsurface groundwater) should be analyzed as soon as practical. In some instances, sampling, particularly soil sampling, may not be practical if the leak occurred in inaccessible areas or if there are extenuating considerations. In this respect, groundwater monitoring may be used as a surrogate for soil sampling. If sampling is not practical, the 10 CFR 50.75(g) records should describe why sampling was not conducted (e.g., the area was inaccessible or there were safety considerations). The licensee should ensure that the location and estimated volume of the leak or spill are recorded to identify the extent of the impacted area and predicted size or extent of the contaminant plume, both horizontally and vertically. If a spill is promptly and fully remediated (e.g., within 48 hours) and if subsequent surveys of the remediated area indicate no detectable residual radioactivity remaining in the soil or groundwater (see paragraph below), for purposes of reporting discharges in the ARERR, there was no liquid discharge to the unrestricted area, and the spill need not be reported in the ARERR. However, in accordance with 10 CFR 50.75(g), the decommissioning file should be updated to include a description of the leak or spill event. Licensees should review the decommissioning files before generating the ARERR to ensure that the ARERR includes the necessary information on leaks and spills.

When evaluating areas that have been remediated, the licensee should survey for residual radioactivity. There may be times when the licensee wants to verify that an area contains no residual radioactivity. There is existing regulatory guidance and information on analytical detection capabilities. Licensees should ensure that surveys are appropriate and reasonable, in accordance with 10 CFR 20.1501. Licensees should generally ensure that surveys are conducted using the appropriate sensitivity levels; e.g., refer to the environmental LLDs in NUREG-1301 and NUREG-1302, Table 4.12-1, "Detection Capabilities for Environmental Sample Analysis," or LLDs determined by using the methodology outlined in NUREG-1576. Additionally, licensees should apply plant-process-system knowledge when evaluating leaks and spills.

This RG provides guidance on information that licensees should provide in the ARERR. In that context, when leaks and spills of radioactive material are identified, prompt response and timely actions should be taken to the extent reasonable to (1) evaluate onsite radiological conditions and (2) ensure proper reporting of materials discharged off site. To realize these two goals, it may be necessary to isolate the leak or spill at the source, prevent the spread of the leak or spill, and remediate the affected area (if the licensee deems remediation to be reasonable and necessary).

For leaks and spills involving the discharge of radioactive material to an unrestricted area, licensees should follow RG 4.25 or equivalent methods to assess the amount of material discharged to the unrestricted area. The potential dose to members of the public from the leak or spill should be evaluated using realistic or bounding exposure scenarios. Attachment 6 to SECY-03-0069, "Results of the License Termination Rule Analysis," dated May 23, 2003 (Ref. 54), provides more information on the use of realistic scenarios.

For leaks and spills, licensees should perform surveys that are reasonable to evaluate the potential radiological hazard (as described in 10 CFR 20.1501). As a result, for leaks and spills, licensees may choose to use bounding assessments to estimate the potential hazard. For example, if a leak occurs on site and radioactive material is released at or below the ground surface, the licensee may choose to assess the potential hazard by estimating a conservatively large (e.g., bounding) volume of water as part of an assumed exposure pathway analysis (e.g., drinking water). Such assumptions would allow the licensee to assess the potential hazard to a hypothetical individual member of the public. A hazard assessment of this sort would be appropriate for inclusion in the supplemental information section of the ARERR. If there is no real exposure pathway to a member of the public, the licensee should indicate that the hazard assessment is a bounding estimate of the dose to a hypothetical individual member of the public, and no real individual member of the public received an actual exposure.

If licensees choose to notify local authorities of spills or leaks (e.g., because of local ordinances or local and State government agreements), the licensee should review the reporting requirements of 10 CFR 50.72(b)(xi) and information in NUREG-1022, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73," issued October 2000 (Ref. 55), for applicability. In such situations, licensees should ensure effective communication, using NUREG/BR-0308, "Effective Risk Communication," issued June 2004 (Ref. 56), especially when ensuring that the risk is described in the appropriate context. In general, licensees should notify the NRC when significant public concern is raised, in accordance with 10 CFR 50.72(b)(xi).

Although the licensee may choose to use its problem identification and resolution program (corrective action program) to document the evaluation of the spill or leak, appropriate documentation should be placed in, or cross referenced to, the decommissioning files, as required by 10 CFR 50.75(g).

Although prompt remediation is not a requirement (Ref. 57), remediation should be evaluated and implemented, as appropriate, based on licensee evaluations and risk-informed decisionmaking. The Electric Power Research Institute (EPRI) Report 1021104 "Groundwater and Soil Remediation Guidelines for Nuclear Power Plants," proprietary report issued December 2010 (Ref. 58) and EPRI

Report 1023464, "Groundwater and Soil Remediation Guidelines for Nuclear Power Plants," (Public Edition) Final Report, July 2011 (Ref. 59) may be useful in performing remediation evaluations.

Evaluation factors should include (1) the location and accessibility, (2) the concentrations of radionuclides and extent of the residual radioactivity, (3) the efficacy of monitored natural attenuation, (4) the volume of the release, (5) the mobility of the radionuclides, (6) the depth of the water table, and (7) whether "significant residual radioactivity" (see glossary) is expected at the time of decommissioning. Since the contaminants, concentrations, and extent of contamination are expected to vary over time or plant life (either increase based on anticipated future leaks and spills or decrease based on remediation or monitored natural attenuation), no one set of numerical values defines significant residual radioactivity. However, licensees may make remediation decisions based on their expectations of their ability to meet the decommissioning criteria of 10 CFR 20.1402 at the anticipated time of decommissioning.

Information that may be useful in this risk-informed decision making includes (1) NUREG-1757, Volume 1, Appendix H, "EPA/NRC Memorandum of Understanding," (2) NUREG-1757, Volume 2, Table H.1, "Acceptable License Termination Screening Values of Common Radionuclides for Building-Surface Contamination," and (3) the authorized derived concentration guideline levels for decommissioned nuclear power plants. For a more detailed analysis, licensees may use the computer codes described in NUREG/CR-6676, "Probabilistic Dose Analysis Parameter Distributions Developed for RESRAD and RESRAD-BUILD Codes," issued July 2000 (Ref. 60); NUREG/CR-6692, "Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes," issued November 2000 (Ref. 61); NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes," issued December 2000 (Ref. 62); and NUREG/CR-7267, "Default Parameter Values and Distribution in RESRAD-ONSITE V7.2, RESRAD-BUILD V3.5 and RESRAD-OFFSITE V4.0 Computer Codes (Ref. 63).

1.6 Monitoring Continuous Releases of Noble Gases

For continuous releases, gross radioactivity measurements are often the only practical means of continuous monitoring. These gross radioactivity measurements are typically used to actuate alarms and terminate (trip) effluent releases; by themselves, such measurements are generally not acceptable for demonstrating compliance with effluent discharge limits.

The use of continuously indicating radiation monitoring system results may be combined with sample analyses to more fully characterize and quantify a discharge. This technique may have particular applicability when (1) a short-term, rapid upscale indication of a process radiation monitor occurs during a release or (2) when there is a desire to verify whether a preliminary grab sample is representative. In these instances, the licensee should ensure that the radiation monitor responses (i.e., the radiation monitor efficiencies) for various radionuclides are well characterized.

Grab samples should be collected at scheduled frequencies in accordance with the ODCM (see NUREG-1301 and NUREG-1302 or as approved in GL 89-01 submittals) to quantify specific radionuclide concentrations and release rates. The frequency of sample collection and radionuclide analyses should be based on the degree of variance in (1) the magnitude of the discharge and (2) the relative radionuclide composition from an established norm. If the magnitude of the discharge and the relative nuclide composition of a continuous release vary significantly over the course of the discharge period, a combination of grab samples and continuous monitor readings can assist in accurately estimating the discharge. Continuous monitoring data (e.g., chart recorder data), as well as grab sample data, should be reviewed periodically and used to identify this variance from the established norm. Periodic evaluations should be made between gross radioactivity measurements and grab sample analyses of specific radionuclides. These evaluations should be used to verify (or modify) the conversion factors that correlate radiation monitor readings and concentrations of radionuclides in effluents.

NUREG-1301 and NUREG-1302 provide guidance on the Radiological Environmental Monitoring Program. Table 3.12-1 therein provides guidance on implementing the environmental monitoring program, including I-131 sampling and analysis on each composite of drinking water.

If a drinking water exposure pathway exists, a prospective dose evaluation should be performed based on I-131 in effluent discharges to determine the maximum likely annual I-131 thyroid dose to a person in any age group from the drinking water pathway. The purpose of the prospective dose evaluation is to determine the environmental sampling and analysis requirements for drinking water. Note: Freshwater fish ingestion is not included in the prospective dose evaluation of I-131 from the drinking water route of exposure.

If the likely dose from I-131 is greater than 1 mrem per year, a composite drinking water sample should be collected over a 2-week period and an I-131 analysis performed with an LLD of 1 pCi/liter. If the likely dose from I-131 is less than or equal to 1 mrem per year, a monthly composite sample should be collected, and an I-131 analysis performed with an LLD of 15 pCi/liter.

In addition, Standard Technical Specifications require determination of the projected dose contributions from radioactive effluents at least every 31 days, and determination of the cumulative dose contributions for the current calendar quarter and current calendar year.

1.7 Monitoring Batch Releases

For batch releases, measurements should be performed to identify principal radionuclides before a release. If an analysis of specific “hard-to-detect” radionuclides (such as strontium-89/90, nickel-63 and iron-55 in liquid releases) cannot be done before the batch release (see NUREG-1301 and NUREG-1302), the licensee should have collected representative samples for the purpose of subsequent composite analysis. The composite samples should be analyzed at the scheduled frequencies specified in NUREG-1301 and NUREG-1302 or at the revised frequencies specified by the licensee (with documented justification in accordance with ODCM change process specified in the technical specifications) (see Sections 1.3 and 1.4 of this RG).

Continuously indicating radiation monitoring system results may be combined with sample analyses to more fully characterize and quantify a discharge. This technique may have particular applicability when (1) a short-term, rapid upscale indication of a process radiation monitor occurs during a discharge or (2) when there is a desire to verify whether a preliminary grab sample is representative. In these instances, the licensee should ensure that radiation monitor responses (i.e., the radiation monitor efficiencies) for various radionuclides are well characterized.

1.8 Principal Radionuclides for Effluent Monitoring

This RG introduces the term “principal radionuclide” in a risk informed context. A licensee may evaluate the list of principal radionuclides for use at a particular site. The principal radionuclides may be determined based on their relative contribution to either (1) the public dose compared to the 10 CFR Part 50, Appendix I, design objective doses, or (2) the amount of activity discharged compared to other site radionuclides in the type of effluent being considered. Under this concept, radionuclides that have either a significant activity or a significant dose contribution should be monitored in accordance with a predetermined and appropriate analytical sensitivity level (LLD) outlined in a licensee’s ODCM. This implementation of “principal radionuclides” ensures that the ARERR appropriately includes both the

(1) radionuclides that are present in relatively large amounts but that contribute very little to dose and (2) radionuclides that are present in very small amounts but that have a relatively high contribution to dose.

If a risk-informed approach is used, principal radionuclides should be determined based on an evaluation over a time period that includes a refueling outage (e.g., one fuel cycle). A periodic reevaluation should be performed to determine whether the radionuclide mix has changed and to identify new principal radionuclides.

If a risk-informed approach is applied to the determination of principal radionuclides[7], the ODCM becomes the controlling document and specifies the list of principal radionuclides. If adopting this method, the licensee should update the ODCM with the list of principal radionuclides within 1 year of their identification. Licensees are allowed to revise the ODCM in accordance with the ODCM change process, as described in the plant's technical specifications (which includes documented evaluations of such changes).

If adopting a risk-informed approach, a radionuclide is considered a principal radionuclide if it contributes either (1) greater than 1 percent of the 10 CFR Part 50, Appendix I, design objective dose for all radionuclides in the type of effluent being considered or (2) greater than 1 percent of the activity of all radionuclides in the type of effluent being considered. RG 1.109 lists the three types of effluent as (1) liquid effluents, (2) noble gases released to the atmosphere, and (3) all other radionuclides released to the atmosphere. In this context, the term "principal radionuclide" has special significance for the required sensitivity levels (e.g., LLDs) for an analysis. The LLDs specified in NUREG-1301 and NUREG-1302 may be used, or LLDs may be determined based on the other methodologies (e.g., as outlined in NUREG-1576). Once principal radionuclides are identified, they should be monitored in accordance with the sensitivity levels (e.g., LLDs) listed in the ODCM.

During analysis of samples, licensees should apply the appropriate analytical sensitivities to ensure adequate surveys are conducted. NUREG-1301 and NUREG-1302 provide a list of "principal gamma emitters" for operating reactors for which an LLD control applies. Historically, this list and guidance from Revision 1 or Revision 2 provided the appropriate sensitivity levels for an analysis. Licensees may continue to use this historical guidance, which essentially classifies all radionuclides as principal radionuclides, and apply the analytical sensitivity levels (e.g., LLDs) directly from NUREG-1301 and NUREG-1302 and Revision 1 or 2 of RG 1.21. This method is simple to implement but, in certain cases, may entail inappropriately long count times or may involve alternate (or unnecessary) methods of analysis for low-activity radionuclides with no—or extremely low—dose significance.

Although the LLD list from NUREG-1301 and NUREG-1302 may be used to determine principal radionuclides, in reality, the principal radionuclides at a site will depend on site-specific factors, such as (1) the operating status of the facility (e.g., operating or in decommissioning), (2) the amount of failed fuel, (3) the extent of system leakage, (4) the sophistication of radioactive waste processing equipment, and (5) the level of expertise in operating radioactive waste processing systems. Since the principal radionuclides will vary from site to site, licensees that wish to deviate from the historical method of determining principal radionuclides (as described above) may adopt a risk-informed approach to identify principal radionuclides (and the associated sensitivity levels) at a site.

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7 With respect to principal radionuclides, "dose" is the measure of risk, whereas "activity" is not. For example, a relatively large amount of tritium released into a large body of water has little dose significance.

For radionuclides that are not identified as principal radionuclides, licensees may use their discretion with the sensitivity of analysis, provided the licensees determine that the changes maintain the levels of radioactive effluent controls required by the regulations in 10 CFR 20.1302; 40 CFR Part 190; 10 CFR 50.36a; and 10 CFR Part 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations. If licensees change their analytical sensitivities from those in their ODCM or equivalent, they must document the basis for the deviations. For example, DQOs and other concepts from RG 4.15 may be useful for determining risk-informed sensitivity levels for an analytical method.

The risk-informed concept of "principal radionuclides" does not reduce the requirement for reporting radionuclides detected in effluents. In addition to principal radionuclides, other radionuclides detected during routine monitoring of release points should be reported in the radioactive effluent release report and included in dose assessments to members of the public, consistent with site-specific technical specifications.

1.9 Carbon-14

Carbon (C)-14 is a naturally occurring isotope of carbon. Nuclear weapons testing in the 1950s and 1960s significantly increased the amount of C-14 in the atmosphere. Commercial nuclear reactors also produce C-14 but in much lower amounts than those produced naturally or from weapons testing. IAEA Report Number 421 provides relevant information on C-14 releases. The C-14 releases in PWRs occur primarily as a mix of organic carbon and carbon dioxide released from the waste gas system. In BWRs, C-14 releases occur mainly as carbon dioxide in gaseous waste.

Regulations in 10 CFR 50.36a require that operating procedures be developed for the control of effluents and that quantities of principal radionuclides be reported. The radioactive effluents from commercial nuclear power plants over time has decreased to the point that C-14 is likely to have become a principal radionuclide (as defined in this document) in gaseous effluents. Therefore, licensees must evaluate whether C-14 is a principal radionuclide for gaseous releases from their facility. Because the dose contribution of C-14 from liquid radioactive waste is much less than that contributed by gaseous radioactive waste, an evaluation of C-14 in liquid radioactive waste is not required.

The quantity of C-14 discharged can be estimated by use of a normalized C-14 source term and scaling factors based on power generation or estimated by use of the NUREG-0016 (GALE-BWR) and NUREG (GALE-PWR) computer codes. The National Council on Radiation Protection and Measurements Report No. 81, "Carbon-14 in the Environment," (Ref. 64) also provides information about the magnitude of C-14 in typical effluents from commercial nuclear power plants. These documents estimate that nominal annual releases of C-14 in gaseous effluents are approximately from 5 to 7.3 curies from PWRs and from 8 to 9.5 curies from BWRs.

The quantity of C-14 generated in BWR and PWR cores can also be estimated by a calculational method provided by the EPRI Report No. 1021106, "Estimation of Carbon-14 in Nuclear Power Plant Gaseous Effluents," issued December 2010 (Ref. 65) and EPRI Report No. 1024827 "Carbon-14 Dose Calculation Methods at Nuclear Power Plants," issued April 2012, (Ref. 66). If estimating C-14 based on scaling factors and fission rates, a precise and detailed evaluation of C-14 is not necessary. It is not necessary to calculate uncertainties for C-14 or to include C-14 uncertainty in any subsequent calculation of overall uncertainty.

Since the NRC published RG 1.21, Revision 1, in 1974, the analytical methods for determining C-14 have improved. Because the production of C-14 is expected to be relatively constant at a particular site, if sampling is performed for C-14 (instead of estimating C-14 discharges based on calculations), the sampling frequency may be adjusted to that interval that allows adequate measurement and reporting of effluents.

1.10 Return/Reuse of Previously Discharged Radioactive Effluents

Radioactive material properly released in gaseous or liquid effluents to the unrestricted area (excluding solid materials or soil) is not considered licensed material when returned to the facility as long as the concentration of radioactive material does not exceed 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," exempt concentration limits (otherwise a general or specific license is required). The water containing radioactive material returned from the environment can be used by the licensee and returned to the unrestricted area without being considered a new radioactive material effluent release. The basis for this determination is that the licensee has already accounted for this radioactive material when the effluent was originally discharged, provided that the subsequent use, possession, or release does not introduce a new significant dose pathway to a member of the public, as explained below.

Licensees are responsible for evaluating any new significant exposure pathway and the resultant radiological hazards associated with the return of radioactive material to the operating facility and its subsequent discharge to the environment. For purposes of estimating dose during operations or decommissioning, a new significant exposure pathway is any pathway that contributes dose that exceeds 10% of the dose criteria in 10 CFR 50 Appendix I, Section II (such that the dose from a new exposure pathway is unlikely to be substantially underestimated). Bounding dose assessments as described in Section 5.1 of this RG may be used in evaluating any new significant exposure pathway. Furthermore, before returning radioactive materials to the environment, licensees must demonstrate that these radioactive materials were previously disposed of in accordance with 10 CFR 20.2001(a)(3), or that the material is naturally occurring background radiation. Radioactive material previously not accounted for as an effluent that is entrained with returned/re-used water must be considered a new effluent disposal per 10 CFR 20.2001. See RIS 2008-03 for further details.

1.11 Abnormal Releases and Abnormal Discharges

In RG 1.21, Revision 1, the terms "release" and "discharge" were synonymous. In RG 1.21, Revision 2 and 3, the term "release" describes an effluent emitted from the plant to either the onsite or offsite environs, (regardless of where the effluent is located), and the term "discharge" describes that portion of an effluent that enters the offsite environs (e.g., the unrestricted area). Although the term "release" includes effluents to either (1) the onsite environs or (2) the offsite environs (e.g., the unrestricted area), this RG generally reserves use of the term "release" for the release of an effluent from the power plant into the onsite environs. The onsite environs in this context encompass locations outside of nuclear power plant systems, structures, and components, as described in the final safety analysis report or ODCM. This is a change in terminology with respect to the definition of "abnormal release" in RG 1.21, Revision 1, which defined abnormal releases to be "from the site boundary."

An "abnormal release" (see glossary) is an unplanned or uncontrolled release of licensed radioactive material into the onsite environs. Abnormal releases may be categorized as either batch or continuous, depending on the circumstances. By contrast, an "abnormal discharge" (see glossary) is an unplanned or uncontrolled discharge of licensed radioactive material to the unrestricted area. Abnormal discharges may also be categorized as either batch or continuous, depending on the circumstances. The distinction between the terms "abnormal release" and "abnormal discharge" is important for describing the staff position for measuring, evaluating, and reporting releases and discharges, especially where leaks and spills are involved.

That portion of an abnormal release discharged to the unrestricted area is reported as an abnormal discharge in the year in which the discharge to the unrestricted area occurred. The portion of an abnormal release that remains onsite is considered residual radioactivity (see 10 CFR Part 20) and is documented in accordance with 10 CFR 50.75(g).

Low-level radioactive system leakage resulting from minor equipment failures and component aging (wear and tear) may be expected to occur as an anticipated part of the plant operation. If such leakage is captured by, or directed to, a system designed to accept and handle radioactive material, including the subsequent planned and controlled discharge of the radioactive material (e.g., as described in the final safety analysis report or ODCM), that evolution is not considered an abnormal release. Normal system leakage captured by effluent ventilation control systems or sumps is not an abnormal release (provided that, before discharge of the radioactive material, the discharge is planned and controlled). (See also the definitions of "unplanned release" and "uncontrolled release" in the glossary.)

In certain circumstances, some subjectivity may be associated with the definitions of "unplanned release" and "uncontrolled release." In these situations, additional circumstances should be considered to determine whether an abnormal release occurred. A well-designed and documented evaluation of a release point can include an evaluation of the potential for an unplanned or uncontrolled release. The evaluation can establish bounding criteria that establish a threshold for an abnormal release based on planning and control. Generally, releases that may reasonably be categorized as both unplanned <u>and</u> uncontrolled should be considered abnormal releases.

For example, consider an underground pipe that carries radioactive liquid to an outside storage tank. If this pipe develops a leak, and licensed radioactive material escapes into the surrounding soil, it is considered an abnormal release if some portion or all of the radioactive material remains onsite. This type of leak should be reported as an abnormal release in the next ARERR. If the licensee predicts (e.g., based on its conceptual site model and subsequent groundwater monitoring results) that the radioactive material will enter the unrestricted area in 2 years, the resulting radioactive discharge (that would occur 2 years hence) will be considered an abnormal discharge. Therefore, the resulting radioactive discharge should be reported along with other data for the affected calendar year in a future ARERR (i.e., in this example, 3 years later). Both releases and discharges (either routine or abnormal) should be reported on a calendar-year basis for the year in which the release or discharge occurred.

Consider another example involving a volume of radioactive gas from the containment atmosphere that escapes the equipment hatch during a refueling outage (especially during the time interval when the containment purge exhaust fans are off). This would generally not be considered an abnormal discharge if (1) the duration was preplanned (e.g., for a "short" duration such as 12 hours), (2) the containment activity (gas, particulate, tritium, and iodine) was preplanned, known, and very low (e.g., such that a bounding estimate of the radioactive material discharged indicated there would be no measurable impact relative to typical discharges), (3) the containment activity was monitored (e.g., by sampling or radiation monitoring equipment), and (4) an evaluation was completed to identify a preplanned limiting (or "trigger") level of activity that would initiate remedial or mitigating action (e.g., close the equipment hatch to control gases escaping containment). In this example, the actions taken (i.e., preplanning and monitoring) before and during the evolution are sufficient to establish control of this discharge. As a result, this type of evolution should not be categorized as an abnormal discharge.

2. Effluent Sampling

2.1 Representative Sampling

NUREG-1301 and NUREG-1302 provide a typical schedule for radioactive effluent sample collection and analyses. Some licensees may have modified these sampling schedules (typically contained in the ODCM) as part of implementing GL 89-01, as approved by the NRC. Additional samples should be obtained as needed to characterize abnormal releases, abnormal discharges, or other significant operational evolutions. Samples should be representative of the overall effluent in the bulk stream, collection tank, or container. Licensees should ensure that representative samples were obtained from well-mixed streams or volumes of effluent at sampling points, using proper equipment and sampling procedures.

2.2 Sampling Liquid Radioactive Waste

Before sampling, large volumes of liquid waste should be mixed to ensure that sediments or particulate solids are distributed uniformly in the waste mixture. For example, a large tank may be mixed using a sparger system or recirculated three or more volumes to ensure that a representative sample can be obtained, as recommended by American Society for Testing and Materials (ASTM) D3370 - 18, "Standard Practices for Sampling Water from Flowing Process Streams (Ref. 67). If tank-mixing practices deviate from industry standards (i.e., those for recirculation or otherwise), the licensee should provide a technical evaluation or other justification. Sample points should be located where there is a minimum of disturbance of flow caused by fittings and other physical characteristics of the equipment and components. Sample nozzles should be inserted into the flow or liquid volume to ensure sampling of the bulk volume of pipes and tanks. Sample lines should be flushed for a sufficient period of time before sample extraction to remove sediment deposits and air and gas pockets. Generally, three sample line volumes should be purged as recommended by ASTM D3370 - 18, before withdrawing a sample, unless a technical evaluation or other justification is provided. A series of samples should be taken periodically during the interval of discharge to determine whether any differences exist as a function of time and to ensure that individual samples are indeed representative of the effluent mixture. In some instances, this may be accomplished by collecting one or more samples (either by "grab" or composite sampler) during the discharge and comparing with one or more samples taken before the discharge. If a series of samples is collected, these samples can be used to assess the amount of measurement uncertainty in obtaining representative samples.

2.3 Sampling Gaseous Radioactive Waste

Although all licensees may not be committed to RG 4.15, ANSI N42.18-2004, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents" (Ref. 68), ANSI N42.54-2018, "Instrumentation and Systems for Monitoring Radioactivity" (Ref. 69), and ANSI/Health Physics Society (HPS) N13.1-2011, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities" (Ref. 70), these documents provide general principles for designing and conducting monitoring programs for airborne effluents. The cited references also contain recommendations for obtaining valid samples of airborne radioactive material in effluents and the guidelines for sampling from ducts and stacks. Licensees should use the appropriate licensing documents to evaluate the validity of representative samples (e.g., evaluate the potential for inaccurate sampling of gaseous effluents that may bypass a particulate filter and collect on an iodine collection cartridge) and to identify any inaccurate sample analyses configurations or counting geometries.

2.4 Sampling Bias

Sampling and storage techniques that could bias quantitative results for effluent measurements should be evaluated and corrections applied as necessary. These biases include inaccurate measurement of sample volumes resulting from pressure drops in long sample lines and loss of particulates or iodine in sample lines resulting from deposition or plate-out. Samplers for gaseous waste should be evaluated for particulate deposition using ANSI/HPS N13.1-2011 or equivalent.

2.5 Composite Sampling

Composite samples should be representative of the average quantities and concentrations of radioactive materials discharged in liquid and gaseous effluents. Composite samples should be collected in proportion to the effluent flow rate or in proportion to the volume of each batch of effluent discharges.

2.6 Sample Preparation and Preservation

Sample preparation and storage methods should minimize the potential for loss of radioactive material (i.e., deposition of analyte on walls of the sample container or volatilization of analyte). Composite sample storage time should be as short as practical to preclude deposition on the storage container, or sample stabilization should be considered. Before quantitative radionuclide analyses for liquid effluent composites, licensees should ensure that samples are mixed thoroughly so that the sample is representative of the material discharged.

Procedures for handling, packaging, and storing samples should ensure that losses of radioactive materials or other factors causing sample deterioration do not invalidate the analysis. For example, filters should be stored carefully to prevent loss of radioactive material from the filter paper.

2.7 Short-Lived Radionuclides and Decay Corrections

In the analysis of short-lived radionuclides (e.g., short-lived noble gases), measurements should generally be made as soon as practical after collection to minimize loss by radioactive decay. In other cases, when needed to improve the detection of the longer-lived radionuclides, time should be allowed for the decay of short-lived, interfering radionuclides.

Some special considerations may be applicable when measuring short-lived radionuclides. In general, sample collection (or analysis frequencies) should take into account the half-lives of the radionuclides being measured. This may have special applicability for continuous samples or composite samples. It is generally best to select a compositing interval (and analysis frequency) appropriate for the effluent (radionuclide) being analyzed. In cases where the compositing interval is selected appropriately, analytical bias is minimized. One way to avoid analytical bias is to decrease the composite sampling interval (and analysis frequency).

To minimize bias in measurements, it may be necessary to decay correct analysis results for short-lived radionuclides. Licensees should be cognizant of those situations in which analytical bias may be introduced when analyzing short-lived radionuclides and should select appropriate methods to minimize such bias.

3. Effluent Dispersion (Meteorology and Hydrology)

3.1 Meteorological Data

Gaseous effluents discharged into the atmosphere are transported and diffused (or, in combination dispersed and, therefore diluted) as a function of (1) the atmospheric conditions in the local environment (including ambient meteorology and structural wake effects), (2) the topography of the region, and (3) the release characteristics of the effluents. In developing and implementing a monitoring program designed to collect site-specific meteorological data, licensees should, conform to the guidance consistent with their facility's current licensing basis but should also consider adopting the guidance in the current version of RG 1.23. The meteorological data do not need to be reported in the ARERR, but the data should be summarized and maintained as documentation (records). Licensees should prepare and maintain an annual meteorological summary report that provides the joint frequency distributions of wind direction and wind speed by atmospheric stability class (see RG 1.23, or, if applicable, Safety Guide 23, "Onsite Meteorological Programs," dated February 17, 1972 (Ref. 71)) on site for the life of the plant. In addition, the licensee should record hourly meteorological data (or shorter-term averages compatible with the appropriate dispersion models) and make the data available if needed for assessing abnormal gaseous releases.

3.2 Atmospheric Dispersion (Transport and Diffusion)

Site-specific meteorological data collected should be validated and used to generate gaseous effluent dispersion factors (χ/Q) and deposition factors (D/Q), in accordance with RG 1.111. The use of long-term annual-average meteorological conditions (based on 5 or more years of data) to determine χ/Q and D/Q is appropriate for continuous releases and for establishing instantaneous release rate set points. This practice may also be acceptable for calculating doses from intermittent releases if the releases occur randomly and with sufficient frequency to justify the use of annual-average meteorological conditions (see RG 1.111).

Personnel familiar with the equipment and typical site meteorological conditions should review the meteorological data. Data losses can be minimized by incorporating redundant sensors and equipment, and by maintaining an adequate inventory of spares, as part of the monitoring program design. Periodic data evaluation may include, but is not be limited to, promptly identifying and inspecting equipment failures and time to resolution, reviewing results of performance checks and calibrations, and confirming that measurements are within appropriate ranges (e.g., occurrence of excessive calm wind speeds, reasonable diurnal and seasonal variation of wind speed, wind direction, and temperature at each level and with height).

A change in χ/Q (and/or D/Q) may not be the only indicator that should be reviewed. A change in impact location should also be addressed (if not already the case). Such a change could be caused by (1) an actual change in the meteorological conditions, (2) a physical change in meteorological instrumentation (i.e., mechanical versus sonic anemometry), (3) a change in data averaging approach (e.g., scalar versus vector), or (4) any combination of the above.

Invalid data should be removed from the meteorological data file prior to calculating long-term, annual-average χ/Q and D/Q values. Records of data invalidation (and if applicable, data substitution) should also be documented and retained.

The long-term, annual-average χ/Q and D/Q values should be reevaluated periodically (e.g., every 3–5 years). If the periodic reevaluation indicates the controlling/limiting long-term, annual-average χ/Q and D/Q values are substantially nonconservative (e.g., higher by 20–30 percent or more with respect to historical data), the licensee should ensure that the χ/Q and D/Q values used in the dose assessment are revised or that the ARERR addresses why such changes are not deemed necessary. Acceptable reasoning includes evaluating data anomalies, identification of failures in meteorological sensors, and documentation that the locality experienced abnormal weather patterns.

3.3 Release Height

The release height affects the dispersion (transport and diffusion) of radioactive materials, especially for "downwash" and building wake effects. For facilities with ground-level, mixed-mode, and elevated releases, an evaluation should be made to determine the proper location of the maximum exposed individual member of the public. From a dispersion perspective, when determining the maximum exposure location (submersion and/or deposition), the evaluation should consider the magnitude of the release(s) originating as an elevated release and as a ground-level release. For example, a close-in, downwind location in one sector may have a higher χ/Q (i.e., less dispersion) for a ground-level release, whereas the majority of the source term may be originating as an elevated release, causing a higher concentration (χ) at a more distant location, possibly in a different sector. RG 1.111 contains a more complete discussion of release height.

3.4 Aquatic Dispersion (Surface Waters)

Liquid radioactive effluents may be disposed in accordance with 10 CFR 20.2001 into a variety of receiving surface water bodies, including nontidal rivers, lakes, reservoirs, settling ponds, cooling ponds, estuaries, and open coastal waters. This effluent is dispersed by various mechanisms (i.e., turbulent mixing; stream flow in the water bodies; and internal circulation or flow-through in lakes, reservoirs, and cooling ponds). Parameters influencing the dispersion patterns and concentrations near a site include the direction and speed of flow of currents, both natural and plant induced, in the receiving water; the intensity of turbulent mixing; the size, geometry, and bottom topography of the receiving water; the location of effluent discharge in relation to the receiving water surface and shoreline; the amount of recirculation of previously discharged effluent; the characteristics of suspended and bottom sediments; and sediment sorption properties. RG 1.113 describes calculational models for estimating aquatic dispersion to surface water bodies. However, the dispersion characteristics may be highly site dependent, and local characteristics should be considered when performing dispersion modeling and dose assessments.

3.5 Spills and Leaks to the Ground Surface

Liquid releases onto the land surface are transported and diluted as a function of site-specific hydrologic features, events, and processes and properties of the effluent. The releases may temporarily accumulate, pool, or run off to natural or engineered drainage systems. During this process, water may also be absorbed into the soil (see Section 3.6). RG 1.113 discusses the use of simple models to estimate transport through surface water bodies and considers water usage effects. Spills or leaks of radioactive material to the ground surface should initiate characterization of the runoff. At a minimum, the characterization activities should satisfy (1) the requirements of 10 CFR 50.75(g) and (2) the effluent reporting requirements of 10 CFR 50.36a, and the guidance described in NUREG-1301 and NUREG-1302 for planned effluents (e.g., sampling before discharge to unrestricted areas). Sections 9.5.1, 9.5.2, and 9.5.9 of this RG contain recommendations on the general format for reporting abnormal releases to onsite areas and abnormal discharges to unrestricted areas.

3.6 Spills and Leaks to Groundwater

Liquid radioactive leaks and spills are sometimes released to onsite groundwater or discharged to offsite groundwater. Leaks and spills onto the ground surface can be absorbed into the soil. Depending on the local soil properties and associated liquid flux of the release, some of the material in the leak or spill may eventually reach the local water table. The dispersion of this material depends on the local subsurface geology and hydrogeologic characteristics. Liquid releases into the subsurface will be transported as a function of groundwater flow processes and conditions (e.g., hydraulic gradients, permeability, porosity, and geochemical processes) and will eventually be released to the unrestricted area.

A groundwater conceptual site model should be developed to predict the subsurface water flow parameters to include direction and rate and to be used as the basis for estimating the dispersion of abnormal releases of liquid effluents into groundwater (see RG 4.1 and RG 4.25). Section A of this RG lists references for use in developing an adequate groundwater conceptual site model.

Simple analytical models or more rigorous numerical codes (i.e., simulations) may be used to evaluate subsurface transport following a release. Appropriate use of these models and codes will depend on the release rate, depth of the release, depth to the local water table, groundwater flow directions, groundwater flow rates, geochemical conditions, and other geochemical processes (e.g., geochemical retardation). Additionally, water usage, such as groundwater pumping from wells, may create local groundwater depression(s) that can alter the natural groundwater flow.

Consistent with 10 CFR 20.1501, a basic site hydrogeological characterization, in advance of leaks or spills, is helpful for evaluating potential leaks and spills. Sites with significant residual radioactivity (see definition in the glossary) that are likely to exceed the radiological criteria for unrestricted use at the time of decommissioning (e.g., as described in 10 CFR 20.1402) should perform more extensive evaluation. Initial assessments should be conducted with relatively simple conceptual site models using scoping surveys, bounding assumptions, or a combination of both (see RG 4.25 and American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.17-2009, "Evaluation of Subsurface Radionuclide Transport at Commercial Nuclear Power Production Facilities" (Ref. 72). The complexity of the models should increase as (1) more knowledge is obtained about the system under evaluation (e.g., source of leak, plume size, concentrations, radionuclides, site characteristics, presence of preferential flow pathways) and (2) the dose estimates rise above significant residual radioactivity levels. Industry documents ANSI N2.17, as well as EPRI "Groundwater Monitoring Guidance for Nuclear Power Plants," Report No. 1011730, (Ref. 73) and EPRI "Groundwater Protection Guidelines for Nuclear Power Plants, Rev. 1," Report No. 3002000546 (Ref. 74) contain details of various industry practices that may be used as part of a groundwater monitoring program. Sites with low-level spills or leaks generally do not require extensive site characterization and monitoring.

The following are basic steps in monitoring groundwater contamination:

1. Use the conceptual site model (as necessary) to assist in monitoring, evaluating, and reporting radioactive releases and radioactive discharges.

2. Collect empirical data by one or more of the following (as necessary):

a. Sample and analyze groundwater from existing monitoring wells.

b. Conduct additional hydrogeologic testing using existing wells (or new wells) if required.

3. Test the conceptual site model and radionuclide transport predictions using groundwater sample results and data collected during hydrogeologic testing.

4. Modify conceptual site model and radionuclide transport parameters as necessary to predict discharges and assess doses to members of the public.

5. Use an iterative process and revaluate as needed.

The groundwater monitoring results should be used in the development and testing of a conceptual site model to predict radionuclide transport in groundwater. The conceptual site model is generally considered adequate when it predicts the results of monitoring (sometimes called a calibrated model). Groundwater monitoring results evaluate the validity of the conceptual site model. Following a leak or spill of licensed (radioactive) material, the conceptual site model may be used in conjunction with radionuclide transport modeling and groundwater monitoring to comprise a basis for predicting future effluents from the site. Dispersion and dilution occur over time and in three dimensions.

When used with a strategic and carefully planned monitoring program, the conceptual site model can ensure that necessary and reasonable surveys are performed (i.e., limited scoping surveys or more extensive surveys). Limited scoping surveys can determine if significant residual radioactivity exists and if there is adequate protection of public health and safety. If the limited scoping surveys identify significant residual radioactivity, then the extent of the contamination should be further evaluated by more extensive surveys (e.g., monitoring wells or other evaluations as appropriate). These survey activities may be direct (i.e., occurring at, or very near, the source of the leak) or indirect (i.e., occurring at some distance from the source of the leak) depending on the accessibility of the source of the spill or leak and the mobility of the radionuclides.

For spills or leaks occurring below the soil surface in inaccessible locations, direct scoping and characterization may not be feasible. In these cases, indirect monitoring techniques (e.g., groundwater monitoring wells in a down-gradient direction) will satisfy existing regulatory requirements. These survey activities should, at a minimum, satisfy (1) the requirements of 10 CFR 50.75(g) and (2) the effluent reporting requirements of 10 CFR 50.36a for groundwater discharges to the unrestricted area. In general, licensees should describe (report) leaks and spills of radioactive material in the ARERR for the calendar year the spill or leak occurred. Additionally, licensees should report groundwater monitoring data in the ARERR for the calendar year in which the data were collected. Sections 9.5.1, 9.5.2, and 9.5.9 of this RG contain guidance on the general format for reporting abnormal releases to onsite areas and abnormal discharges to unrestricted areas.

Although licensees may conduct a groundwater monitoring effort for different reasons, for purposes of this RG, the surveys, characterization activities, conceptual site models, and other components of any groundwater monitoring effort should be sufficient to do the following:

1. Appropriately report, for purposes of accountability, effluents discharged to unrestricted areas.

2. Document information in a format consistent with Table A-6 and Section 9.5 of this RG.

3. Provide advance indication of potential future discharges to unrestricted areas (to ensure releases are planned and monitored before discharge).

4. Demonstrate that significant residual radioactivity has not migrated off site to an unrestricted area in the annual reporting interval.

5. Communicate relevant information as described in Section 9.5 of this guide.

4. Quality Assurance

4.1 Quality Assurance Programs

The analytical process should use a range of QA checks and tests. RG 4.15 describes the QA program activities for ensuring that radioactive effluent monitoring systems and operational programs meet their intended purpose. Each licensee's licensing basis determines the applicability of Revision 1 or Revision 2. However, RG 4.15, Revision 2 contains guidance on determining appropriate sensitivity levels for analytical instrumentation based on DQOs. The use of DQOs may provide a better technical basis for determining sensitivity levels (e.g., LLDs) than the use of the default values in NUREG-1301 and NUREG-1302. A combination approach using both Revision 1 and Revision 2 of RG 4.15 may be used to determine appropriate sensitivity levels (e.g., LLDs) different (i.e., higher or numerically larger) than those listed in NUREG-1301 and NUREG-1302.

4.2 Quality Control Checks

QC checks of laboratory instrumentation should be conducted daily or before use, and background variations should be monitored at regular intervals to demonstrate that a given instrument is in working condition and functioning properly. QC records should include results of routine tests and checks, background data, calibrations, and all routine maintenance and service.

4.3 Surveillance Frequencies

Routine qualitative tests and checks (e.g., channel operational tests, channel checks, or source checks to demonstrate that a given instrument is in working condition and functioning properly) may be performed using radioactive sources that are not traceable by the National Institute of Standards and Technology (NIST). The schedule for source checks, channel checks, channel calibrations, and channel operational tests should be in accordance with NUREG-1301 and NUREG-1302, unless otherwise modified after a technical evaluation demonstrates a justifiable change in frequency. A technical evaluation that revises a surveillance frequency should include consideration of the instrument's function and the consequences of failure and not simply rely on the history of successful surveillances.

4.4 Procedures

Individual written procedures should be used to establish specific methods of calibrating installed radiological monitoring systems and grab sampling equipment. Written procedures should document calibration practices used for ancillary equipment and systems (e.g., meteorological equipment, airflow measuring equipment, in-stack monitoring pitot tubes). Calibration procedures may be compilations of published standard practices or manufacturers' instructions that accompany purchased equipment, or they may be written in house to include special methods or items of equipment not covered elsewhere. Calibration procedures should identify the specific equipment or group of instruments to which the procedures apply.

Written procedures should be used for maintaining counting room instrument accuracy, including maintenance, storage, and use of radioactive reference standards; instrumentation calibration methods;

and QC activities such as collection, reduction, evaluation, and reporting of QC data as required by the technical specifications.

4.5 Calibration of Laboratory Equipment and Routine Effluent Radiation Monitors

Calibrations (e.g., of laboratory equipment and continuous radiation monitoring systems used to quantify radioactive effluents) should be performed using the general principles for calibration of effluent monitoring instrumentation provided in ANSI N42.18-2004 and ANSI N323C-2009, "American National Standard for Radiation Protection Instrumentation Test and Calibration—Air Monitoring Instruments, American National Standards Institute" (Ref. 75), using radioactive calibration sources traceable to the NIST. Calibration sources should have the necessary accuracy, stability, and radioactivity levels required for their intended use. The relationship between concentrations and monitor readings should be determined. Performance of the monitoring system should be judged on the basis of reproducibility, time stability, and sensitivity.

Periodic inservice correlations that relate monitor readings to the concentrations, release rates of radioactive material in the monitored release path, or a combination of both, should be performed when possible to validate the adequacy of the system. These correlations should be based on the results of analyses for specific radionuclides in grab samples from the release path.

The use of NIST-traceable sources combined with mathematical efficiency calibrations may be applied to instrumentation used for radiochemical analysis (e.g., gamma spectroscopy systems) if employing a method provided by the instrument manufacturer.

4.6 Calibration of Measuring and Test Equipment

Measuring and test equipment should be calibrated using NIST-traceable radioactive sources. The source geometries should be representative of the sample types analyzed and have the necessary accuracy, stability, and activity concentrations for their intended use.

4.7 Calibration Frequency

Calibrations should generally be performed at regular intervals in accordance with the frequencies established in NUREG-1301 and NUREG-1302. A change in calibration frequency (an increase or decrease) should be based on the reproducibility and time stability characteristics of the system. For example, an instrument system that gives a relatively wide range of readings when calibrated against a given standard should be recalibrated at more frequent intervals than one that gives measurements within a more-narrow range. Any monitoring system or individual measuring equipment should be recalibrated or replaced whenever it is suspected of being out of adjustment, excessively worn, or otherwise damaged and not operating properly.

4.8 Measurement Uncertainty

The measurement uncertainty (formerly called measurement error) associated with the measurement of radioactive materials in effluents should be estimated. Counting statistics can provide an estimate of the statistical counting uncertainty involved in radioactivity analyses. Because it may be difficult to assign error terms for each parameter affecting the final measurement, detailed statistical evaluations of error are not required. Normally, the statistical counting uncertainty decreases as the amount (concentration) of radioactivity increases. Thus, for the radioactive effluent release report, the statistical counting uncertainty is typically a small component of the total uncertainty. The sampling uncertainty is likely the largest component and includes uncertainties such as the uncertainty in volumetric and flow-rate measurements and laboratory processing uncertainties.

The total or expanded measurement uncertainty associated with the effluent measurement should ideally include the cumulative uncertainties resulting from the total operation of sampling and measurement. Expanded uncertainty should be reported with measurement results. The objective should be to evaluate only the important contributors and obtain a reasonable measure of the uncertainty associated with reported results. Detailed statistical and experimental evaluations are not required. The overall objective should be to obtain an overall estimate of measurement uncertainty. The formula for calculating the total or expanded uncertainty classically includes the square root of the sum of squares of each important contributor to the measurement uncertainty. Licensees may obtain additional information from NUREG-1576 and ANSI/HPS N13.1-2011.

4.9 Calibration of Accident-Range Radiation Monitors and Accident-Range Effluent Monitors

GDC 64 requires means for monitoring radioactivity in the reactor containment atmosphere; spaces containing components for recirculation of loss-of-coolant accident (LOCA) fluids; effluent discharge paths; and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. The regulation at 10 CFR 20.1501(c) requires periodic calibration of instruments and equipment used to perform quantitative radiation measurements (e.g., dose rate and effluent monitoring).

NUREG-0737, Item II.F.1, provides guidance for monitoring radiation levels and gaseous effluent during postulated radiological emergencies. RG 1.97, Revisions 2 and 3, provide guidance on the design and performance criteria of instrumentation used to assess plant and environ conditions during and following an accident. This RG 1.21 provides further guidance on the calibration of such instrumentation based on the NRC's "Proposed Guidance for Calibration and Surveillance Requirements to Meet Item II.F.1 of NUREG-0737," issued August 1982 (Ref. 76). NUREG/CR-5569, "Health Physics Positions Data Base," Health Physics Position (HPPOS)-001, "Proposed Guidance for Calibration and Surveillance Requirements to Meet Item II.F.1 of NUREG-0737," issued February 1994 (Ref. 77), summarizes this additional guidance.

Noble Gas Monitoring - NUREG-0737, Item II.F.1-1, describes accident-range noble gas effluent monitors as monitors that are normally noble gas gross activity monitors sensitive to gamma emissions, beta emissions, or a mix of gamma and beta emissions. These monitors normally indicate (read out) in units of activity concentration, a count rate, or a dose rate (i.e., an indirect measurement of the noble gas gross activity concentration). Therefore, in order to determine the release rate of noble gas gross activity, a conversion factor (i.e., hereafter referred to as an instrument response factor) should be developed to convert the instrument output into an activity concentration for use in determining a release rate (e.g., curies per second of a mix of noble gases).

The initial vendor calibration of emergency effluent monitoring instruments may be a one-time prototype calibration based on the initial calibration of a single instrument of a certain model using NIST-traceable radiation sources. This initial prototype calibration of a single instrument model and subsequent calibration of production detectors determine the fundamental detector characteristics, such as the following:

1. a dose-rate linearity check using a radioactive gas or solid source (e.g., cesium (Cs)-137) to obtain three on-scale values separated by two decades of scale;

2. a measurement of the instrument's response factor to a calibration gas (e.g., xenon (Xe)-133 or krypton (Kr)-85);

3. a characterization of the instrument's energy-dependency characteristics, using solid sources ranging in gamma energy from low energy (e.g., 81 kiloelectron volts) to high energy (e.g., 2 megaelectron volts); and

4. a determination, using a solid source, of a transfer factor that provides a dual purpose:

    a. for use by vendors to validate that subsequent instruments produced for sale of the same model have similar performance characteristics to the initial "type" instrument model's characteristics; and

    b. for use by end users (e.g., nuclear power plants) in performing post installation and subsequent periodic calibration to verify that the instruments installed in the facility are functioning consistently with respect to initial vendor calibration of that instrument model.

Time-dependent (i.e., time since reactor shutdown) instrument response factors may be developed for each major accident type (i.e., a small-break LOCA with normal reactor coolant system activity levels, a large-break LOCA with gas gap activity levels, or a core-melt accident with noble gas activity levels arising from the fuel pellets release of noble gas). Each accident type has a characteristic, time-dependent noble gas isotopic mix. In general terms, a small-break LOCA has a substantially decayed noble gas mix from the reactor coolant system with predominantly low-energy gamma photons; a large-break LOCA has a somewhat decayed noble gas mix from the gas gap of the fuel assemblies with predominately medium-energy gamma photons; and a core-melt accident has a substantially undecayed mix of noble gas isotopes in the fuel pellets with predominately high-energy gamma photons.

The time-dependent instrument response factor accounts for the detector's energy efficiency at various gamma energies of the noble gas isotopic mix for that accident type. The instrument response factor normally has units of microcuries per cubic centimeter (μCi/cc) per count per minute or μCi/cc per milliroentgen per hour where the μCi/cc is the gross (total) summation of all the noble gas activities in the isotopic mix for each major type of accident listed above. It is also acceptable to use instrument response factors based on a single calibration gas with a low-energy gamma source (e.g., Xe-133) or beta emissions (e.g., Kr-85) for beta sensitive monitors.

The initial calibration process performed by the vendor does not need to be repeated at a nuclear power plant. Instead, a periodic single point source response check of the instrument's performance as compared to a transfer factor provided by the vendor using a solid source - see ANSI N320-1978, "Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation (Ref. 78).

Iodine and Particulate Monitoring - NUREG-0737, Item II.F.1-2, provides guidance on iodine and particulate effluent monitoring by sampling and analysis. Real-time monitoring is not required or considered practical; however, the licensees should have established procedures for collection of iodine and particulate samples and subsequent analysis to determine the release rate. For emergency dose assessment purposes, RASCAL (NUREG-1940 Section 1.2.8) can also be used to assess a real-time iodine and particulate release rate based on partitioning (scaling) factors to noble gases.

Containment High Range Monitoring - NUREG-0737, Item II.F.1-3, provides guidance on calibration of containment high-range monitors. An in-place calibration should be performed using a radioactive source at one point on the decade below 10 roentgens per hour (R/hr). Instrument scales in the range of 10 R/hr to 1E7 R/hr should be checked using electronic signal substitution with a calibrated current source to demonstrate that the system is functioning to higher radiation fields.

Containment high-range monitors should be used to assess the amount of core damage and to assess the source terms for the containment leakage release pathway. NUREG-1940, Section 1.2.4, Figures 1-1 through 1-5, provide information for PWRs and BWRs at 1 hour and 24 hours after reactor shutdown that correlates the containment radiation monitor readings to the amount of reactor damage for normal coolant, spiked coolant, cladding failure, and core melt accident scenarios.

5. Dose Assessments for Individual Members of the Public

The regulation in 10 CFR 20.1301 establishes dose limits for individual members of the public[8]. The regulations referenced in Sections 5.4–5.6 of this RG contain both dose limits and design objectives that the licensee demonstrates compliance with through calculations. Table 1 summarizes the fundamental parameters associated with the dose calculations. RG Sections 5.7 and 5.8 present important concepts for these calculations. Because of differences between NRC and EPA regulations, demonstrating compliance only with radiological effluent technical specifications (based on 10 CFR Part 50, Appendix I) does not necessarily ensure compliance with the EPA's 40 CFR Part 190, particularly if there is a direct radiation component (e.g., from BWR shine, ISFSI, or radioactive materials storage) or accumulated radioactivity from prior-year effluents.

Table 1 - Parameters Associated with Dose Calculations

	10 CFR PART 50, APPENDIX I per reactor	**10 CFR 20.1301(e) (EPA 40 CFR PART 190) Uranium fuel cycle (e.g., all reactors)**
Dose	whole body, max of any organ, gamma air, and beta air	whole body, thyroid, and max of any organ
Basis	International Commission on Radiation Protection (ICRP)-2, "Report of Committee II on Permissible Dose for Internal Radiation," issued 1959 (Ref. 79)	EPA 40 CFR Part 190
Where	unrestricted area	unrestricted area

8 For ISFSIs, 10 CFR Part 72 specifies dose limits for any real individual beyond the Part 72. controlled area boundary (excluding occupational exposures). Thus, dose assessments performed to demonstrate compliance with the 10 CFR 72.104 must include the necessary components described in10 CFR 72.104.

	10 CFR PART 50, APPENDIX I per reactor	10 CFR 20.1301(e) (EPA 40 CFR PART 190) Uranium fuel cycle (e.g., all reactors)
Individual Receptor	real person/exposure pathway (nearest real residence, real garden, real dairy/meat animal)	real person/exposure pathway (nearest real (residence, real garden, real dairy/meat animal)
Origin	liquid and gas radioactive waste	liquid and gas radioactive waste, direct radiation (e.g., nitrogen-16 shine, ISFSI, radioactive materials storage, outside tanks), accumulated radioactive material from prior-year effluents (e.g., tritium in lake water) not already included in dose estimates
Radioactive Material	licensed only (per Appendix I, Section II radioactive materials – see Section 5.4 below)	licensed and unlicensed (see Section 5.6 below)
When	current year	current and prior years' operation

5.1 Bounding Assessments

Bounding assessments may be useful if compliance can be readily demonstrated using conservative assumptions. In this RG, the term "bounding assessment" means that the reported value is unlikely to be substantially underestimated (see 10 CFR Part 50, Appendix I, Section III). Bounding assessments for the current year do not imply the absolute bounds for future conditions.

For example, licensees may use conservative bounding dose assessments in lieu of site-specific dose assessments of the maximum dose to individual members of the public. Instead of assessing dose from ground-level effluent releases to a real individual member of the public located 3.2 km (2 miles) from the site boundary, a conservative bounding dose assessment can be performed for a hypothetical individual member of the public located at the site boundary.

If bounding assumptions are made, the radioactive effluent release report should state such and should annotate the assumptions. Hypothetical exposure pathways (see definition in the glossary) and locations are sometimes used for bounding dose assessments (or hazard evaluations done in accordance with 10 CFR 20.1501).

5.2 Individual Members of the Public

Individual members of the public reside in the unrestricted area but at times may enter the controlled area or restricted area. Each licensee is responsible for classifying individuals as either members of the public or as occupational workers (see the definition of "member of the public" in 10 CFR Part 20.) The NRC annual dose limits for members of the public (regardless of their location in the restricted area, controlled area or unrestricted area) are 100 mrem total effective dose equivalent in accordance with 10 CFR 20.1301(a) and (b).

The dose criteria in Technical Specifications conforming to 10 CFR 50, Appendix I are for members of the public in the unrestricted area. In addition, in accordance with 10 CFR 20.1301(e) for uranium fuel cycle licensees (including nuclear power plants), the annual dose limits to members of the public in the unrestricted area are the EPA 40 CFR Part 190 limits of 25 mrem whole body, 75 mrem to the thyroid, and 25 mrem to any other organ while in the unrestricted area.

For demonstration of compliance with Technical Specifications conforming to 10 CFR 50, Appendix I, if bounding assessments are not used, licensees should perform evaluations to determine the dose to a real, maximum exposed member of the public in the unrestricted area. A member of the public in the unrestricted area is typically a real individual in a designated location where there is a real exposure pathway (e.g., a real garden, real cow, real goat, or actual drinking water supply) and not a fictitious fencepost resident or an exposure pathway that includes a virtual goat or cow. Licensees are encouraged (but not required) to use real individual members of the public when performing dose assessments for radioactive discharges. Table 1 in RG 1.109 allows a dose evaluation to be performed at a location where an exposure pathway and dose receptor actually existed at the time of licensing.

5.3 Occupancy Factors

For members of the public in the unrestricted area, occupancy factors should be assumed to be 100 percent at locations identified in the land use census, unless site-specific information indicates otherwise. Occupancy factors may be applied inside the controlled area based on estimated hours spent in the controlled area.

5.4 10 CFR Part 50, Appendix I, Design Objectives and Limiting Conditions for Operation

Appendix I to 10 CFR Part 50 contains numerical guidance for design objectives and limiting conditions of operation for radioactive waste systems on a per reactor basis to ensure discharges of radioactive liquid and gaseous effluents to unrestricted areas are ALARA. This numerical guidance is listed in terms of annual air doses (gamma and beta), annual total body doses, and annual organ doses (see below). Licensee technical specifications require that exposure to liquid and gaseous effluents conform to the numerical guidance in 10 CFR Part 50, Appendix I. In accordance with 10 CFR 50.34a, these numerical guides for design objectives and limiting conditions of operation are not to be construed as radiation protection standards. For these dose calculations, the following terms are generally used:

1. air doses (gamma and beta), total body doses, and organ doses (based on ICRP-2),
2. effluent discharges only (excludes direct radiation from the facility and ISFSIs),
3. current annual period (excludes accumulated radioactivity from prior-year effluents), and
4. unrestricted area (excludes individuals in the restricted areas and controlled areas).

When calculating air doses, licensees should assure that, for any location outside the site boundary, doses do not exceed the design objectives in 10 CFR Part 50, Appendix I. Calculation of air dose at the site boundary would assure the most conservative calculation of air doses for ground-level releases. This may not be true for elevated releases. Licensees should select a location that assures the most conservative calculation of air dose.

5.5 10 CFR 20.1301(a) NRC dose limits for individual members of the public

This regulation specifies dose limits for members of the public from licensed operation of the facility. These limits apply to doses resulting from licensed and unlicensed radioactive material and from radiation sources other than background radiation (see 10 CFR 20.1001, "Purpose"). The dose limits include contributions to doses from (1) current-year effluents, (2) current-year direct radiation from the facility, and (3) accumulated radioactivity from prior-year effluents[9]. The Technical Specifications establish the Radioactive Effluent Controls Program and the Environmental Monitoring Program, which establish effluent control methods sufficient to demonstrate of compliance with the NRC public dose limits in 10 CFR 20.1301(a).

5.6 10 CFR 20.1301(e) EPA Environmental Radiation Standards for the Uranium Fuel Cycle

For those facilities subject to the EPA's generally applicable environmental radiation standards in 40 CFR Part 190, licensees must assess the highest cumulative (whole body and organ) doses from the uranium fuel cycle to a real individual in the general environment (i.e., outside the site boundary). The dose limits include contributions to doses from (1) current-year effluents, (2) current-year direct radiation from the facility, and (3) accumulated radioactivity from prior-year effluents[9]. The Technical Specifications establish the Radioactive Effluent Controls Program and the Environmental Monitoring Program, which establish effluent control requirements sufficient to demonstrate compliance with the EPA public dose limits in 40 CFR Part 190 (see NUREG-0543).

These requirements include the following considerations:

1. "Whole body and organ doses" come from ICRP-2 concepts.

2. "Any member of the public" means any individual except when that individual is receiving an occupational dose.

3. The "unrestricted area" means an area, access to which is neither limited nor controlled by the licensee. The boundaries of the unrestricted area are defined by the licensee. (See also the definition of "generally applicable environmental radiation standards" in 10 CFR 20.1003.)

4. "Current-year effluents" includes both normal and abnormal discharges to the unrestricted area.

5. "Current-year direct radiation" includes all direct radiation from the facility (e.g., radioactive waste storage and ISFSIs) but excludes doses from radioactive waste shipments.

6. "Cumulative" dose means the sum of (1) current-year effluent dose, (2) current-year direct radiation dose, and (3) dose from accumulated radioactivity if not already included in the first two items.

7. "Accumulated radioactivity" includes radioactive material in the unrestricted area from prior-year discharges that remains in the environment (e.g., tritium in lake water or radionuclides).

8. The "uranium fuel cycle" excludes uranium mining, radioactive waste shipping (in the unrestricted area), operations at waste disposal sites, and reuse of nonuranium special nuclear materials. (See the definition of "uranium fuel cycle" in 40 CFR Part 190 and in the glossary of this document.)

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9 Doses from accumulated radioactivity from prior-year effluents do not need to be included in demonstration of compliance with the NRC and EPA dose limits unless the reporting levels in the environmental monitoring program associated with 10 CFR 50, Appendix I are exceeded.

5.7 Dose Assessments for 10 CFR Part 50, Appendix I

Dose assessments to show compliance with technical specification requirements for meeting the numerical values of 10 CFR Part 50, Appendix I, design objectives on a per reactor basis should include quarterly and annual doses using the considerations in Section 5.4 of this RG. The dose assessments should be reported in a format similar to that shown in Table A-4 in Appendix A to this RG and include the items listed below:

1. doses from liquid effluents

    a. total body dose, quarterly and annually;

    b. organ dose, quarterly and annually (maximum, any organ); and

    c. percent of limits for each of the above.

2. doses from gaseous effluents

    a. beta and gamma air doses, quarterly and annually;

    b. organ dose commitment from iodine, tritium, and particulate releases with half-lives greater than 8 days, quarterly and annually; and

    c. percent of limit for each of the above.

An evaluation of the local exposure pathways to determine the maximum exposed member of the public should be performed. However, maximum doses from various exposure pathways are not additive from different locations. For example, dose from a downstream drinking water exposure pathway should not be added to the dose to an upstream resident whose exposure is from gaseous effluents and direct radiation unless that individual's drinking water is obtained from the downstream location.

"Maximum" doses to real individuals should be assessed as described in RG 1.109. The locations and exposure pathways are those where real individuals are present and exposed. Maximum exposed individuals are characterized as "maximum" with regard to food consumption, occupancy, and other usage in the vicinity of the plant site. For example, licensees should make "maximum" assumptions for food consumption and occupancy factors at actual locations when assessing dose to the maximum exposed individual, unless they have determined and applied site -specific (actual) data. In lieu of assessing dose to real individuals, licensee may also use bounding dose assessments for compliance with 10 CFR Part 50, Appendix I (see Section 5.1 titled "Bounding Assessments").

The objective of 10 CFR Part 50, Appendix I, is to provide numerical guides for design objectives and limiting conditions for operation to ensure that radioactive effluent control equipment is effective in reducing emissions to ALARA levels. The numerical guidance pertains to quarterly and annual dose criteria in the unrestricted area from current-year effluent discharges. The calculations related to Appendix I do not include dose from radioactivity in prior-year, accumulated, effluent discharges (e.g., last year's radioactivity remaining in lake water is excluded). However, the dose calculations for demonstrating compliance with the EPA limits do include accumulated radioactivity (see Section 5.8 of this RG).

For purposes of demonstrating compliance with dose criteria for limiting dose to a member of the public in unrestricted areas in accordance with Technical Specifications conforming to 10 CFR 50,

Appendix I, the exposure pathways and routes of exposure identified in RG 1.109 should be considered. An evaluation of other exposure pathways (not included in dose assessments) should be performed and maintained for purposes of demonstrating compliance with the staff position on significant exposure pathways. Calculational procedures should be based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. A new significant exposure pathway should be included in the demonstration of compliance if the calculated dose from that new exposure pathway exceeds 10 percent of the 10 CFR 50 Appendix I, Section II numerical guides on design objectives. Bounding dose assessments as described in Section 5.1 of this RG may be used in evaluating the dose from any new significant exposure pathways.

Real exposure pathways are identified for routine discharges and direct radiation based on the results of the land use census. Dose calculations should typically be performed based on real exposure pathways. Conversely, dose assessments (i.e., surveillances and dose calculations) are not needed for exposure pathways that do not exist at a site. For example, if the land use census does not identify the existence of an ingestion exposure pathway involving a milk animal, the licensee is not required to assess that route of exposure for the ingestion exposure pathway. Similarly, if a licensee discharges liquid radioactive waste to a body of water (either surface water or groundwater) and that body of water is not used as a source of drinking water (either private or public), a drinking water assessment is not required. For purposes of reporting information in the ARERR, there is a distinction between dose assessments for 10 CFR Part 50, Appendix I, and hazard assessments that may be conducted for onsite spills and leaks, as outlined in 10 CFR 20.1501 (where bounding estimates may be necessary). (See the discussion of bounding dose estimates in Section 5.1 of this RG.)

5.8 Dose Assessments for 10 CFR 20.1301(e)

To show compliance with 10 CFR 20.1301(e), dose assessments should be reported according to the generally applicable environmental radiation standards in 40 CFR Part 190, with consideration of Section 5.6 of this RG, and in a format similar to Table A-5 of Appendix A to this RG.

1. The following should be reported:

    a. whole body dose to the maximum individual member of the public,

    b. thyroid dose to the maximum individual member of the public,

    c. dose to any other organ of the maximum individual member of the public, and

    d. percent of the applicable limit.

2. One means of demonstrating compliance with 40 CFR Part 190 is listed in Volume 42 of the *Federal Register*, page 2859, which states the following:

    > In the case of light water reactors, … demonstrating conformance with Appendix I of 10 CFR 50 are generally adequate for demonstrating compliance with [EPA 40 CFR Part 190].

As a result, a licensee that (1) can demonstrate that external sources of direct radiation are indistinguishable from background and (2) demonstrates compliance with the numerical dose guidance of 10 CFR Part 50, Appendix I, may cite the above reference as the basis for demonstrating compliance with 40 CFR Part 190. The NRC provides additional guidance in NUREG-0543, "Methods for Demonstrating Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190).

However, licensees that (1) have external sources of direct radiation that are above background and (2) demonstrate compliance with the numerical dose guidance of 10 CFR Part 50, Appendix I, must also include sources of direct radiation from uranium fuel cycle operations (e.g., including direct radiation from the licensed facility and co-located or nearby nuclear power facilities, as appropriate).

3. The dose contributions from direct radiation may be estimated based on either (1) direct radiation measurements (e.g., thermoluminescent dosimeters, optically stimulated dosimeters, radiation detection instruments), (2) calculations, or (3) a combination of measurements and calculations. When direct radiation dose is determined by measurement, RG 4.13 provides guidance on determining the dose to members of the public. Several sources contain additional information on background subtraction for environmental dosimeters (Refs. 29, 80, 81 and 82). Methods of determining dose from direct radiation to the maximum exposed individual member of the public may also include extrapolation methods.

Licensees must demonstrate compliance with 10 CFR 20.1301(e) for the generally applicable environmental radiation standards in 40 CFR Part 190. These include the concept of a total dose (to the whole body and to any organ) from all sources related to the uranium fuel cycle (such as adjacent or nearby nuclear power plants).

Contributions to the total dose from radioactive effluents (liquid and gaseous) and direct radiation should be included, if applicable. Other sources (e.g., accumulated radioactive materials in offsite ponds or lakes from previous years' discharges) should also be included, if applicable, when estimating the total dose. However, if the contributions from direct radiation or accumulated radioactivity are generally minor (as evaluated and documented in a licensee technical evaluation as not contributing to the total dose), these contributions need not be included in the total dose evaluation, but the basis for exclusion should be documented.

5.9 Dose Calculations

Acceptable dose assessment models, such as those provided in RGs 1.109, 1.111, 1.112, and 1.113, should be used to make dose calculations. When calculating organ doses from airborne effluents for purposes of demonstrating compliance with Technical Specifications conforming to 10 CFR 50, Appendix I, contributions from I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days should be included in the assessment. For demonstrating compliance with NRC dose limits in 10 CFR 20.1301(a) and EPA 40 CFR 190, doses from C-14 should be included in organ dose assessments.

6. Solid Radioactive Waste Released from the Unit

Section 5.6, "Reporting Requirements," in the Standard Technical Specifications normally requires reporting of "solid waste released from the unit" (see NUREG-1430, 1431, 1432, 1433, and 1434 (Refs. 83 - 87)). The data reported should be for the LLW volumes shipped from the unit (plant site).

Solid radioactive waste shipments should be reported in a format similar to that of Table A-3 in Appendix A to this RG. The total curie quantity and major radionuclides in the solid waste shipped off site should be determined and reported.

The data should be divided by the waste stream categories listed in Table A-3. The waste streams are:

(1) wet radioactive waste (e.g., spent resin, filters, sludges, etc.),

(2) dry radioactive waste (e.g., trash, paper, discarded protective clothing etc.),

(3) activated or contaminated metal or equipment, etc., and

(4) other radioactive waste (bulk waste, soil, rubble, etc., not excepted from reporting as described below).

Shipments that do not need to be reported include shipments of contaminated equipment for transfer between licensees or equipment for refurbishment, contaminated laundry (either launderable or dissolvable), or radioactive samples for analysis. Potentially contaminated dry active waste sent for resurvey and segregation (sometimes referred to as "green is clean") does not need to be reported. Equipment shipped for decontamination and free release does not need to be reported. However, records of these types of shipments should be maintained on site.

Note 1: Data on LLW disposed in licensed LLW disposal facilities is available using the Manifest Information Management System operated by the U.S. Department of Energy.

Note 2: There are no requirements for reporting storage of LLW at nuclear power plants. However, LLW storage records should be established and maintained at nuclear plants and made available for NRC inspection during routine effluent inspections consistent with applicable NRC requirements.

7. Reporting Errata in Effluent Release Reports

Errors in radioactive effluent release reports should be classified and reported as described below.

7.1 Examples of Small Errors

Small errors may be any of the following:

1. inaccurate reporting of dose that equates to ≤10 percent of the applicable 10 CFR Part 50, Appendix I, design objective or ≤10 percent of the EPA public dose criterion;

2. inaccurate reporting of curies (or release rates, volumes, etc.) that equate to ≤10 percent of the affected curie total (or release rate, volume, etc.) after correction;

3. omissions that do not impede the NRC's ability to adequately assess the information supplied by the licensee; or

4. typographical errors or other errors that do not alter the intent of the report.

7.2 Reporting Small Errors

Licensees should correct small errors within 1 year of discovery and may submit the correction with the next (normally scheduled) submittal of the ARERR, as follows. A brief narrative explanation of the errors should be included in Section 8, "Errata/Corrections to Previous ARERRs," of Table A-6. The narrative should state that the affected pages, in their entirety, are included as attachments to the ARERR. Additionally, the corrected pages, in their entirety, should be submitted as an attachment (or addendum) to the ARERR. The corrected pages should reference the affected calendar year and should contain revision bars in the margins of the page to indicate the locations of the changes. If submitting corrections to multiple ARERRs, a separate attachment (or addendum) should be made for each of the affected years. Other methods of correcting previous ARERRs may be used, provided the corrections are clearly and completely described.

7.3 Examples of Large Errors

Large errors may be any of the following:

1. inaccurate reporting of dose that equates to >10 percent of the 10 CFR Part 50, Appendix I, or EPA public dose criterion, after correction;

2. inaccurate reporting of curies (or release rate, volume, etc.) that equates to >10 percent of the affected curie total (or release rate, volume, etc.) after correction;

3. omissions that may impede the NRC's ability to adequately assess the information supplied by the licensee; or

4. typographical errors or other errors that significantly alter the intent of the report.

7.4 Reporting Large Errors

Licensee should correct large errors within 90 days of discovery. The correction may be made by special submittal or may be submitted with the next (normally scheduled) ARERR (if the next ARERR is to be submitted within 90 days of discovery of the error). If corrections are made by special submittal, the licensee should include a brief narrative explaining the errors. The narrative should state that the affected pages, in their entirety, are included as an attachment. The corrected pages should be attached in their entirety. The corrected pages should reference the affected calendar year and should contain revision bars in the margins of the page to indicate the locations of the changes. If submitting corrections to multiple ARERRs, separate attachment (or addendum) should be made for each of the affected years. If corrections are made coincident with the next (normally scheduled) submittal of the ARERR, the correction process should be used as specified in Section 7.2 (for small errors). Other methods of correcting previous ARERRs may be used provided the corrections are clearly and completely described consistent with NRC requirements on the completeness and accuracy of information.

8. Changes to Effluent and Environmental Programs

Standard Technical Specifications (e.g., Section 5.5, "Programs and Manuals") establishes requirements for the radioactive effluent controls and radiological environmental monitoring activities. The Technical Specifications establish a specific review and approval process for making changes to the ODCM. Potential changes require licensee analyses or evaluations justifying the change and a determination that the changes maintain the levels of radioactive effluent control required by10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I. The evaluation of potential changes should also consider the need for monitoring in support of decommissioning planning during operations (see RG 4.22, "Decommissioning Planning During Operations," issued December 2012 (Ref. 88).

Effluent and environmental monitoring programs may need to be modified once power operations have permanently ceased and a written certification has been submitted to the NRC in accordance with 10 CFR 50.82, "Termination of License." The evaluation of potential changes should consider the need for effluent and environmental monitoring during active decommissioning which is likely to affect principal release points and principal radionuclides. For example, the removal of effluent ventilation systems will likely change principal release points and there may be new principal radionuclides identified (e.g., Kr-85), while radioactive decay may have eliminated former principal radionuclides (e.g., I-131) (see Section C.1.8). Potential changes must be reviewed and approved by the plant manager, station manager, or as described in plant-specific Technical Specifications, with submittal to the NRC as part of the next Annual Radioactive Effluent Release Report.

If the plant has a 10 CFR Part 72 ISFSI, the licensee must maintain compliance with the requirements in 10 CFR Part 72 regarding controls of effluent(s) and an environmental monitoring program. These requirements include 10 CFR 72.44(d) for 10 CFR Part 72 specific license ISFSIs and, for 10 CFR Part 72 general license ISFSIs, any requirements specified in technical specifications of the certificate(s) of compliance for the storage systems in use at the ISFSI (to comply with 10 CFR 72.212(b)(3) and (b)(5)).

The radiological criteria for license termination are addressed in 10 CFR 20 Subpart E. The radiological criteria for unrestricted use (10 CFR 20.1402) encompass contributions from residual radioactivity in soils and remnant site components and in groundwater. While some reductions in monitoring programs may be possible when operations cease, other aspects of monitoring such as groundwater monitoring may need to be increased to adequately characterize residual radioactivity and characterize dispersion pathways to support dose assessments and to estimate the decommissioning costs. Lessons learned documented in RG 1.185 and NUREG-1757 indicate that the monitoring data from the period of operation tend to be insufficient to allow the staff to fully understand the types and the movement of radioactive material contamination in groundwater at the facility, as well as the extent of the residual radioactivity. Decommissioning reporting and recordkeeping requirements are addressed in 10 CFR 50.75(g).

Further general guidance to facilitate planning for decommissioning of power plants and facilities during operations can be found in RG 4.22, in RG 1.185 for post-shutdown decommissioning activities, in NUREG-1757 for consolidated decommissioning guidance, and in NUREG-1575, Rev. 1, "Multi-Agency Radiation Survey and Site Investigation Manual."

9. Format and Content of the Annual Radioactive Effluent Release Report

In accordance with 10 CFR 50.4, "Written communications," licensees should submit their annual report electronically or in a written communication. The report should consist of a summary of the numerical data in a tabular format similar to Tables A-1 – A-6 in Appendix A to this RG. Effluent data reported in Tables A-1, A-1A – A-1F, A-2, A-2A, A-2B, and A-4 should be summarized on a quarterly and annual basis. Tables A-3 and A-5 should be summarized on an annual basis. In addition to numerical data, the report should include additional supplemental information containing all the information in (but not necessarily in the format of Table A-6). Additional detail for the information contained in each of these tables is listed below. To comply with 10 CFR 50.36a, licensees must submit their ARERR by May 1 (unless a licensing basis exists for a different submittal date) to report on effluents and solid waste from the previous calendar year.

Radionuclides that are not detected do not need to be listed in the tables (Tables A-1A – A-1F, A-2A, and A-2B). Activity that is detected should be reported in the appropriate tables (i.e., Tables A-1, A-2, A-1A – A-1F, A-2A, and A-2B) in the ARERR, provided that the amount discharged is numerically significant with respect to the three-digit exponential format recommended for the ARERR. This should not be confused with three significant figures. Licensees may round numbers according to accepted practices (e.g., refer to ASTM E29, "Standard Practice for Using Significant Digits in Test Data to Determine Conformance with Specifications" (Ref. 89)); however, after rounding has been completed, values should be reported in the ARERR in a three-digit exponential format. Measurements should be reported for positive values. Some radionuclides that are detected in a year may not be detected in all quarters. If results are determined to be below detectable levels for an entire quarter, the table entry should include a suitable designation (e.g., N/D (not detected) and an accompanying footnote) to denote that measurements were performed but activity was not detected.

The format specified in this RG revision differs slightly from the format specified in Revision 1 and Revision 2. The format and content specified in this Revision 3 of RG 1.21 is one acceptable method of reporting the data. Other formats may be used (e.g., some tables may be combined) as long as the specified content is provided (e.g., quarterly totals and annual totals by each release category). However, licensees are encouraged to use the format listed below to maximize consistency in data reporting. This format is designed to be consistent with some commonly used electronic-data-reporting software packages. Consistency in reporting format aids review by members of the public and allows easier industrywide comparisons of the data.

10 CFR 72 licensees may also, if they choose to do so, use the format specified in this RG for independent spent fuel storage installation (ISFSI) effluent reports required by 10 CFR 72.44(d) (for specific licenses) or the storage system(s) certificate(s) of compliance (for general licenses). However, the ISFSI effluent reporting requirement is not normally satisfied by inclusion as part of the ARERR since the reporting dates may conflict. If the dates are coincident, or can be met with a single report, licensees may use the ARERR to fulfill the ISFSI reporting requirements, provided the licensee submits a copy as specified in those requirements (e.g., 10 CFR 72.44(d)(3) for specific licenses).

9.1 Gaseous Effluents

The quarterly and annual sums of all radionuclides discharged in gaseous effluents (i.e., routine and abnormal discharges, continuous, and batch) should be reported in a format similar to that of Tables A-1A – A-1F in Appendix A to this RG. The data should then be further summarized and reported in the format of Table A-1.

Table A-1, “Gaseous Effluents—Summation of All Discharges,” contains a summation of all gaseous effluent discharges from all release points and all modes of release. The data are subdivided by quarter and year for each radionuclide category: fission and activation gases, iodines/halogens, particulates, tritium, gross alpha and carbon-14.

Table A-1A, “Gaseous Effluents—Ground-Level Release—Batch Mode,” contains a summation of gaseous effluent releases from ground-level release points in the batch mode of release for six radionuclide categories: fission and activation gases, iodines/halogens, particulates, tritium, gross alpha and carbon-14. Licensees should report the following:

1. curies of each radionuclide discharged by quarter and year, and

2. total curies discharged in each radionuclide category by quarter and year.

Some licensees may have surveillance requirements allowing the non-noble gas radionuclides (e.g., iodines and tritium) for some types of batch releases (e.g., containment purge) to be reported with continuous release results. In these instances, the table entries for the affected radionuclides for batch releases should include an appropriate designation (e.g., “*”) and an accompanying footnote describing this situation.

Table A-1B, “Gaseous Effluents—Ground-Level Release—Continuous Mode,” contains a summation of gaseous effluent releases from ground-level release points in the continuous mode of release for six radionuclide categories: fission and activation gases, iodines/halogens, particulates, tritium, gross alpha and carbon-14. Licensees should report the following:

1. curies of each radionuclide discharged by quarter and year, and

2. total curies discharged in each radionuclide category by quarter and year.

Table A-1C, “Gaseous Effluents—Elevated Release—Batch Mode,” contains a summation of gaseous effluent releases from elevated release points in the batch mode of release for six radionuclide categories: fission and activation gases, iodines/halogens, particulates, tritium, gross alpha, and carbon-14. Licensees should report the following:

1. curies of each radionuclide released by quarter and year, and

2. total curies released in each radionuclide category by quarter and year.

Some licensees may have surveillance requirements allowing the non-noble gas radionuclides (e.g., iodines and tritium) for some types of batch releases (e.g., containment purge) to be reported with continuous release results. In these instances, the table entries for the affected radionuclides for batch releases should include an appropriate designation (e.g., “*”) and an accompanying footnote describing this situation.

Table A-1D, “Gaseous Effluents—Elevated Release—Continuous Mode,” contains a summation of gaseous effluent releases from elevated release points in the continuous mode of release for six radionuclide categories: fission and activation gases, iodines/halogens, particulates, tritium, gross alpha and carbon-14. Licensees should report the following:

1. curies of each radionuclide released by quarter and year, and

2. total curies released in each radionuclide category by quarter and year.

Table A-1E, “Gaseous Effluents—Mixed Mode Release—Batch Mode,” contains a summation of gaseous effluent releases from mixed-mode release points in the continuous mode of release for six radionuclide categories: fission and activation gases, iodines/halogens, particulates, tritium, gross alpha, and carbon-14. Licensees should report the following:

1. curies of each radionuclide released by quarter and year, and

2. total curies released in each radionuclide category by quarter and year.

Some licensees may have surveillance requirements allowing the non-noble gas radionuclides (e.g., iodines and tritium) for some types of batch releases (e.g., containment purge) to be reported with continuous release results. In these instances, the table entries for the affected radionuclides for batch releases should include an appropriate designation (e.g., “*”) and an accompanying footnote describing this situation.

Table A-1F, “Gaseous Effluents—Mixed Mode Release—Continuous Mode,” contains a summation of gaseous effluent releases from mixed-modes release points in the continuous mode of release for six radionuclide categories: fission and activation gases, iodines/halogens, particulates, tritium, gross alpha, and carbon-14. Licensees should report the following:

1. curies of each radionuclide released by quarter and year, and

2. total curies released in each radionuclide category by quarter and year.

9.2 Liquid Effluents

The quarterly and annual sums of all radionuclides discharged in liquid effluents (i.e., routine and abnormal discharges, continuous, and batch) should be reported in a format similar to that of Tables A-2A and A-2B. The data should then be further summarized and reported in the format of Appendix A, Table A-2.

Table A-2, “Liquid Effluents—Summation of All Releases,” contains a summation of all liquid radioactive discharges from all release points and all modes of release. The data are subdivided by quarter and year for each of the radionuclide categories: fission and activation products, tritium, dissolved and entrained noble gases, and gross alpha.

The table also includes the total volume of “primary coolant waste” (typically batch mode releases) before dilution. In this context, “primary coolant waste” means the higher activity waste that generally is not discharged directly but is instead typically processed through the liquid radioactive waste treatment system before discharge. Various methods exist for calculating the dilution water flow rate. HPPOS-099, “Attention to Liquid Dilution Volumes in Semiannual Radioactive Effluent Release Reports,” issued November 1984 (Ref. 90), indicates that licensees should use the total volume of dilution flow, not just that flow during periods of liquid effluent releases. Licensees should include information describing how this value is calculated in either the ODCM or the ARERR. Because the primary coolant waste typically accounts for the vast majority of the radioactivity in liquid waste discharges, the NRC recommends that the volume and dilution data be summarized separately from the low-activity waste described in the following paragraph.

The total measured volume or average flow rate of waste from secondary or balance-of-plant systems (e.g., steam generator blowdown, low-activity waste sumps, and auxiliary boilers) should be reported. In this context, secondary or balance-of-plant waste means the typically very low-activity waste that is generally not processed with the liquid radioactive waste treatment system and that collectively represents a very large volume of waste. Various methods exist for calculating the dilution water flow rate. HPPOS-099 states that licensees should use the total volume of dilution flow, not just that volume discharged during periods of liquid effluent releases. Licensees should include information describing how this value is calculated in either the ODCM or the ARERR. Because of the potentially high volume and extremely low activity of this type of waste, the NRC recommends the volume and dilution data be summarized separately from the higher activity waste described in the previous paragraph.

Licensees should report dilution flow rates during periods of release (before effluent is discharged to the receiving water body), as described above. If calculated differently than described above, the licensee should describe the method of calculation. Licensees may choose to report near-field dilution if they account for dilution by the receiving water body. Licensees may report the average, minimum, peak river, and stream flow rates, as applicable.

Table A-2A, "Liquid Effluents—Batch Mode," contains a summation of liquid effluent discharges in the batch mode of release. The table is divided into four radionuclide categories: fission and activation products, tritium, dissolved and entrained gases, and gross alpha. Licensees should report the following:

1. curies of each radionuclide and gross alpha discharged by quarter and year, and

2. total curies in each radionuclide category by quarter and year.

Table A-2B, "Liquid Effluents—Continuous Mode," contains a summation of liquid effluent discharges in the continuous mode of release. The table is divided into four radionuclide categories: fission and activation products, tritium, dissolved and entrained gases, and gross alpha. Licensees should report the following:

1. curies of each radionuclide and gross alpha discharged by quarter and year, and

2. total curies in each radionuclide category by quarter and year.

9.3 Solid Waste Shipments Released from the Unit (per Standard Technical Specifications)

Appendix A, Table A-3, provides an acceptable format for reporting the solid radioactive waste released (shipped) from the unit (plant site) during the reporting period. The NRC intends that licensees report the waste shipped from the site, regardless of whether the shipment is sent for waste processing or direct disposal (i.e., with or without waste processing).

Licensees should report the volume and curies of solid waste shipped (see exceptions noted in Section 6) for each of the following waste streams:

1. wet radioactive waste (e.g., spent resins, filters, sludges, etc.),

2. dry radioactive waste (e.g., trash, paper, discarded protective clothing, etc.),

3. activated or contaminated metal or equipment, etc., and

4. other radioactive waste (e.g., bulk waste, soil, rubble, etc., not excepted from reporting requirements in Section 6).

9.4 Dose Assessments

Licensees should calculate the annual evaluations of dose to members of the public using RG 1.21, Section 5 and report the data in the format of Tables A-4 and A-5. Dose assessments should demonstrate compliance with the following[10]:

1. Licensees should demonstrate compliance with 10 CFR Part 50, Appendix I (see Table A-4), by doing the following[11]:

   a. Reporting the calculated dose from liquid effluents on a quarterly and annual basis to the total body and maximum organ and the percentage of the 10 CFR Part 50, Appendix I, design objectives for the maximum exposed individual. If a particular exposure pathway is not applicable (i.e., it does not exist at a site), do not calculate the dose for that exposure pathway.

   b. Reporting the highest air dose from gaseous effluents on a quarterly and annual basis at any location that could be occupied by individuals in the unrestricted area and the percentage of the 10 CFR Part 50, Appendix I, design objectives.

   c. Reporting the organ dose from iodine, tritium, and particulates with a half-life greater than 8 days to the maximum exposed individual in an unrestricted area from all pathways of exposure (e.g., submersion and ingestion).

2. Licensees must demonstrate compliance with 10 CFR 20.1301(e) and 40 CFR Part 190 (see Table A-5) as follows:

   a. Reporting the whole body, thyroid, and highest dose to any other organ from licensed and unlicensed radioactive material in the uranium fuel cycle, excluding background, to the individual member of the public likely to receive the highest dose.

9.5 Supplemental Information

Licensees should provide supplemental information in a descriptive, narrative form (see Table A-6 or in a similar format). Relevant information and a description of circumstances should be provided as appropriate for each the following categories, adding categories as appropriate. The annotation N/A should be used if a category is not applicable.

9.5.1 Abnormal Releases or Abnormal Discharges

The reporting of abnormal releases to onsite areas and abnormal discharges to unrestricted areas should include the following:

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10 As noted in Section C.5, dose assessments for 10 CFR 72.104 should include the components necessary to appropriately demonstrate compliance with those limits.

11 The type of individual or dose receptor should be identified as a real individual or as a hypothetical individual if using bounding dose assessments; the individual/ receptor is in the unrestricted area.

1. Specific information should be reported concerning abnormal (airborne, liquid) releases on site and abnormal discharges to the unrestricted area. The report should describe each event in a way that would enable the NRC to adequately understand how the material was released and if there was a discharge to the unrestricted area. The report should describe the potential impact on the ingestion exposure pathway involving surface water and groundwater, as applicable. The report should also describe the impact (if any) on other affected exposure pathways (e.g., inhalation).

2. The following are the thresholds for reporting abnormal releases and abnormal discharges in the supplemental information section:

   a. abnormal releases or abnormal discharges that are voluntarily reported to local authorities under Nuclear Energy Institute 07-07, Revision 1, "Industry Groundwater Protection Initiative—Final Guidance Document," dated February 26, 2019 (Ref. 91);

   b. abnormal releases or abnormal discharges estimated to exceed 300 liters (100 gallons) of radioactive liquid where the presence of licensed radioactive material is positively identified (in either the onsite environs or in the source of the leak or spill) as greater than the minimum detectable activity[12] for the laboratory instrumentation;

   c. abnormal releases to onsite areas that result in detectable residual radioactivity after remediation;

   d. abnormal releases that result in a high effluent radiation alarm without an anticipated system trip occurring; and

   e. abnormal discharges to an unrestricted area.

3. Information on abnormal releases or abnormal discharges should include the following, as applicable:

   a. date and duration,
   b. location,
   c. volume,
   d. estimated activity of each radionuclide,
   e. effluent monitoring results (if any),
   f. onsite monitoring results (if any),
   g. depth to the local water table,
   h. classification(s) of subsurface aquifer(s) (e.g., drinking water, unfit for drinking water, not used for drinking water),
   i. size and extent of any groundwater plume,
   j. expected movement/mobility of any groundwater plume,
   k. land use characteristics (e.g., water used for irrigation),
   l. remedial actions considered or taken and results obtained,
   m. calculated member of the public dose attributable to the release,
   n. calculated member of the public dose attributable to the discharge,
   o. actions taken to prevent recurrence, as applicable, and
   p. whether the NRC was notified, the date(s), and the contact organization.

12 The minimum detectable activity is a post-analysis calculation of sensitivity level based on the actual sample measurement.

9.5.2 Nonroutine Planned Discharges

Discharges resulting from remediation efforts that are not identified in the ODCM should be reported. For example, the remediation effort may include pumping of contaminated groundwater in response to leaks and spills.

9.5.3 Radioactive Waste Treatment System Changes

Licensees should report any changes or modifications affecting any portion of the gaseous radioactive waste treatment system, the ventilation exhaust treatment system, or the liquid radioactive waste treatment.

9.5.4 Annual Land Use Census Changes

Licensees should report any changes or modifications affecting significant aspects of the environmental monitoring program such as receptors, receptor locations, sample media availability, or new (or changed) routes of exposure.

9.5.5 Effluent Monitoring System Inoperability

Licensees should report information on inoperable effluent monitors as follow:

1. If an effluent radiation monitor is not operable for the consecutive time period listed in the licensee's ODCM or technical specifications (typically 30 days), then the ARERR should include the radiation monitor's equipment designation, the common name of the effluent radiation monitor, the time period of the inoperability, the reason why this inoperability was not corrected in a timely manner, and any other information required by the licensee's ODCM or technical specifications.

2. In accordance with NUREG-1301 and NUREG-1302, Sections 3.3.3.10.b and 3.3.3.11.b, Generic Letter 89-01, and licensee ODCMs, the information above is required only when the minimum channels operability requirement is not achieved for the consecutive time period listed in the ODCM (typically 30 days).

9.5.6 Offsite Dose Calculation Manual Changes

Licensees should report any changes or modifications affecting significant aspects of the ODCM.

9.5.7 Process Control Program Changes

Licensees should report any changes or modifications affecting significant aspects of the ODCM.

9.5.8 Corrections to Previous Reports

When submitting corrections to previous reports, licensees should do the following:

1. Include a brief explanation of the error(s).

2. State that the affected pages, in their entirety, are included as attachments to this ARERR.

3. Ensure that a copy of the affected page(s), in their entirety, is included as an attachment to the ARERR. The attached pages should reference the affected calendar year and contain revision bars.

9.5.9 Other (Narrative Descriptions of Other Information Related to Radioactive Effluents)

Licensees should report other supplemental information (as appropriate).

D. IMPLEMENTATION

The NRC staff may use this regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this regulatory guide to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, "Backfitting," or in 10 CFR 72.62, "Backfitting," and as described in NRC Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," (Ref. 92), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

GLOSSARY

a priori—Before-the-fact limit, representing the capability of a measurement system and not as an after-the-fact (a posteriori) limit for a particular measurement.

abnormal discharge—The unplanned or uncontrolled emission of an effluent (i.e., containing plant-related, licensed radioactive material) into the unrestricted area.

abnormal release—The unplanned or uncontrolled emission of an effluent (i.e., containing plant-related, licensed radioactive material) into the onsite environs.

accumulated radioactivity—Radioactivity from prior-year effluent releases that may still be present in the unrestricted area.

background (radiation)—Means radiation from cosmic sources; naturally occurring radioactive material, including radon (except as a decay product of source or special nuclear material); and global fallout as it exists in the environment from the testing of nuclear explosive devices and from past nuclear accidents, such as Chernobyl or Fukushima, that contribute to background radiation and are not under the control of the licensee. Background radiation does not include radiation from source, byproduct, or special nuclear materials regulated by the Commission.

batch release—The release of liquid (radioactive) wastes of a discrete volume or the release of a tank or purge of radioactive gases into the site environs.

channel check—The qualitative assessment of channel behavior during operation by observation. This determination should include, where possible, comparison of the channel indication, status with other indications, and status derived from independent instrument channels measuring the same parameter.

channel operational test—The injection of a simulated signal into the channel as close to the sensor as practicable to verify operability of alarm, interlock, and trip functions, as applicable. The channel operational test should include adjustments, as necessary, of the alarm, interlock, and trip setpoints, as applicable, such that the setpoints are within the required range and accuracy.

continuous release—An essentially uninterrupted release of gaseous or liquid effluent for extended periods during normal operation of the facility where the volume of radioactive waste is non-discrete and there is input flow during the release.

controlled area (10 CFR Part 20)—An area outside of a restricted area but inside the site boundary, access to which is limited by the licensee for any reason.

controlled area (10 CFR Part 72)—The area immediately surrounding an ISFSI or a monitored retrievable storage installation (MRS) for which the licensee exercises authority over its use and within which ISFSI or MRS operations are performed.

controlled discharge—A radioactive discharge is considered to be "controlled" if (1) the discharge was conducted in accordance with methods, and without exceeding any of the limits, outlined in the ODCM or (2) if one or more of the following three items are true:

1. The radioactive discharge had an associated, preplanned method of radioactivity monitoring that assured the discharge was properly accounted and was within the limits set by 10 CFR Part 20 and 10 CFR Part 50.

2. The radioactive discharge had an associated, preplanned method of termination (and associated termination criteria) that assured the discharge was properly accounted and was within the limits set by 10 CFR Part 20 and 10 CFR Part 50.

3. The radioactive discharge had an associated, preplanned method of adjusting, modulating, or altering the flow rate (or the rate of release of radioactive material) that assured the discharge was properly accounted and was within the limits set by 10 CFR Part 20 and 10 CFR Part 50.

controlled release—A radioactive release is considered to be "controlled" if (1) the release was conducted in accordance with methods, and without exceeding any of the limits, outlined in the ODCM or (2) if one or more of the following three items are true:

1. The radioactive release had an associated, preplanned method of radioactivity monitoring that assured the release was properly accounted and was within the limits set by 10 CFR Part 20 and 10 CFR 50.

2. The radioactive release has an associated, preplanned method of termination (and associated termination criteria) that assured the release was properly accounted and was within the limits set by 10 CFR Part 20 and 10 CFR 50.

3. The radioactive release had an associated, preplanned method of adjusting, modulating, or altering the flow rate (or the rate of release of radioactive material) that assured the release was properly accounted and was within the limits set by10 CFR Part 20 and 10 CFR Part 50.

conversion factor—A factor (e.g., microcuries per cubic centimeter per counts per minute (μCi/cc/cpm) used to estimate a radioactivity concentration in an effluent based on a gross radioactivity measurement (e.g., cpm).

D/Q— A deposition parameter used for estimating the dose to an individual at a specified (e.g., controlling) location. D/Q may be described as the downwind surface or ground deposition rate (D) (e.g., in units of microcuries per square meter [$\mu Ci/m^2$]/sec of radioactive material at a location, divided by the release rate (Q) (e.g., in μCi/sec). D/Q is thus a normalized downwind surface deposition rate per unit release rate and can be used to determine the surface or ground radioactivity concentration during a measured effluent release over a specific period of time. The units of D/Q are reciprocal square meters.

determination—A quantitative evaluation of the release or presence of radioactive material under a specific set of conditions. A determination may be made by direct measurement or indirect measurements (e.g., with the use of scaling factors).

dilution water (for liquid radioactive waste)—For purposes of this RG, any water other than the undiluted radioactive waste that is mixed with undiluted liquid radioactive waste before its ultimate discharge to the unrestricted area.

discharge point—A location at which radioactive material enters the unrestricted area. This would be the point beyond the vertical plane of the unrestricted area (surface or subsurface).

drinking water—Water that does not contain an objectionable pollutant, contamination, minerals, or infective agent and is considered satisfactory for domestic consumption. This is sometimes called potable water. Potable water is water that is safe and satisfactory for drinking and cooking. Although EPA regulations only apply to public drinking water sources supplying 25 or more people (refer to the EPA for more information), for purposes of the effluent and environmental monitoring programs, the term drinking water includes water from single-use residential drinking water wells.

effluent—Liquid or gaseous waste containing plant-related, licensed radioactive material, emitted at the boundary of the facility (e.g., buildings, end-of-pipe, stack, or container) as described in the final safety analysis report.

effluent discharge—The portion of an effluent release that reaches an unrestricted area. (See also the definition for radioactive discharge.)

effluent release—The emission of an effluent into the onsite environs. (See also the definition for radioactive release.)

elevated release—A gaseous effluent release made from a height that is more than twice the height of adjacent solid structures, or releases made from heights sufficiently above adjacent solid structures such that building wake effects are minimal or absent.

exposure pathway—A mechanism by which radioactive material is transferred from the (local) environment to humans. There are three commonly recognized exposure pathways: inhalation, ingestion, and direct radiation. For example, ingestion may include dose contributions from one or more routes of exposure. One route of exposure that may contribute to the ingestion exposure pathway is often referred to as grass-cow-milk-infant-thyroid route of exposure.

general environment—An EPA 40 CFR 190.02 definition meaning the total terrestrial, atmospheric and aquatic environment outside sites upon which any (licensed) operation of a nuclear fuel cycle is conducted.

ground-level release—A gaseous effluent release made from a height that is at—or less than—the height of adjacent solid structures, or where the degree of plume rise is unknown or is otherwise insufficient to avoid building wake effects.

groundwater—All water in the surface soil, the subsurface soil, or any other subsurface water. Groundwater is simply water in the ground regardless of its quality, including saline, brackish, or fresh water. Groundwater can be moisture in the ground that is above the regional water table in the unsaturated (or vadose) zone, or groundwater can be at and below the water table in the saturated zone.

hypothetical exposure pathway—An exposure pathway in which one or more of the components involved in the transfer of a radionuclide from the environment to the human does not actually exist at the specified location, or if a real human does not consume, inhale, or otherwise become exposed to the radioactive material. For example, the grass-cow-milk-infant-thyroid route of exposure (associated with the ingestion exposure pathway) would be considered a hypothetical exposure pathway if the grass, the cow, or the milk did not actually exist at a specified location or if an infant did not actually consume the milk.

impacted areas—The areas with some reasonable potential for residual radioactivity in excess of natural background or fallout levels. The NRC discusses impacted areas in 10 CFR 50.2 and NUREG-1757. For example, impacted areas include locations where radiological leaks or spills have occurred within the onsite environs (i.e., outside of the facility's systems, structures, and components). (See also the definition for significant contamination.)

leachate—Water containing contaminants that is percolating downward from a pond or lake into the subsurface.

less-significant release point—Any location from which radioactive material is released as a liquid or gaseous effluent contributing less than or equal to 1 percent of the activity discharged from all the release points for a particular type of effluent considered. RG 1.109 lists the three types of effluent as (1) liquid effluents, (2) noble gases discharged to the atmosphere in gaseous radioactive waste, and (3) all other nuclides discharged to the atmosphere in gaseous radioactive waste.

Example: If 1,000 curies (Ci) of tritium are released in all liquid effluents in a given period of time (e.g., a typical calendar year or fuel cycle) and 0.01 Ci of tritium is released in steam generator blowdown, then the steam generator blowdown would be a less-significant release point. Similarly, for gaseous releases of radionuclides other than noble gases (i.e., iodine, particulates, and tritium), if the total effluents are 10 Ci (iodine, particulates, and tritium), and the refueling water storage tank released 0.009 Ci of iodine, particulates, and tritium, then the refueling water storage tank would be a less-significant release point. In both examples, the sample frequency can be adjusted to an appropriate frequency for the less-significant release point. Samples collected from these systems for other programs (e.g., detection of primary-to-secondary leakage) must still be collected and analyzed at the frequencies specified by the other programs.

licensed material—Source material, special nuclear material, or byproduct material received, possessed, used, transferred, or disposed under a general or specific license issued by the Commission.

lower limit of detection (LLD)—The "a priori" smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95-percent probability with only a 5-percent probability of falsely concluding that a blank observation represents a real signal (see NUREG-1301, NUREG-1302, and NUREG/CR-4007, "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," issued September 1984 (Ref. 93)).

maximum exposed individual—Individuals characterized as maximum exposed with regard to food consumption, occupancy, and other usage in the vicinity of the plant site. As such, the maximum exposed individual represents individuals with habits that are considered to be maximum reasonable deviations from the average for the population in general. Additionally, in physiological or metabolic respects, the maximum exposed individual is assumed to have those characteristics that represent the averages for the corresponding age group in the general population. (This term typically refers to members of the public.) RG 1.109 contains additional information.

member of the public (10 CFR Part 20)—Any individual except when that individual is receiving an occupational dose.

member of the public (40 CFR Part 190)—Any individual that can receive a radiation dose in the general environment, whether the individual may or may not also be exposed to radiation in an occupation associated with a nuclear fuel cycle. However, an individual is not considered a member of the public during any period in which the individual is engaged in carrying out any operation that is part of a nuclear fuel cycle.

minimum detectable concentration—The smallest activity concentration measurement that is practically achievable with a given instrument and type of measurement procedure. The minimum detectable concentration depends on factors involved in the survey measurement process (surface type, geometry, backscatter, and self-absorption) and is typically calculated following an actual sample analysis (*a posteriori*). (See NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," issued June 1998 (Ref. 94)).

mixed mode release—A gaseous effluent release made from a height higher than a ground-level release but less than an elevated release where, sometimes, because of a lack of plume rise (e.g., buoyancy, momentum, wind speed), a proper estimate of radionuclide transport and diffusion requires mathematically splitting the plume into (1) an elevated component and (2) a ground-level component to properly account for building wake effects, release, or ambient conditions (or a combination of all three). (RG 1.111 contains further guidance.)

monitoring—With respect to radiation or radiation protection, the measurement of radiation levels, concentrations, surface area concentrations, or quantities of radioactive material and the use of results of these measurements to evaluate potential exposures and doses.

nonroutine, planned discharge—An effluent release from a release point not defined in the ODCM but that has been planned, monitored, and discharged in accordance with 10 CFR 20.2001 (e.g., the discharge of water recovered during a spill or leak from a temporary storage tank).

nuclear fuel cycle—The operations defined to be associated with the production of electrical power for public use by any fuel cycle through the use of nuclear energy (see 40 CFR 190.02).

offsite environs—Locations outside the site boundary in the unrestricted area.

onsite environs—Locations within the site boundary but outside of the systems, structures, or components described in the final safety analysis report or the ODCM.

operability (operable)—The ability of a system, subsystem, train, component, or device to perform its specified safety function(s) and the ability of all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment (required for the system, subsystem, train, component, or device to perform its specified safety function(s)) to perform their related support function(s).

principal radionuclide—One of the principal gamma emitters listed in NUREG-1301 and NUREG-1302, Tables 4.11-1 and 4.11-2, or, from a risk-informed perspective, a radionuclide that contributes either (1) greater than 1 percent of the 10 CFR Part 50, Appendix I, design objective dose when all radionuclides in the type of effluent are considered, or (2) greater than 1 percent of the activity of all nuclides in the type of effluent being considered. RG 1.109 lists the three types of effluents as (1) liquid effluents, (2) noble gases discharged to the atmosphere, and (3) all other nuclides discharged to the atmosphere. In this RG, the terms "principal radionuclide" and "principal nuclide" are synonymous since this document is only concerned with measuring, evaluating, and reporting radioactive materials in effluents.

radioactive discharge—The emission of an effluent (i.e., containing plant-related, licensed radioactive material) into the unrestricted area. (See also the definition for effluent discharge.)

radioactive release—The emission of an effluent (i.e., containing plant-related, licensed radioactive material) into the onsite environs. (See also the definition for effluent release.)

real exposure pathway—An exposure pathway in which plant-related radionuclides in the environment at (or from) a specified location cause exposure to an actual individual. For example, the grass-cow-milk-infant-thyroid exposure pathway would be considered a real exposure pathway if the grass, the cow, and the milk actually existed at a specified location and an infant actually consumed the milk. For purposes of compliance with 10 CFR Part 50, Appendix I, the individual must be a member of the public in the unrestricted area.

real individual (10 CFR 72) —Any individual who lives, works, or engages in recreation or other activities close to the ISFSI/MRS for a significant portion of the year.

release source—A system, structure, or component (containing radioactive material under the licensee's control) where radioactive materials are contained before release.

release point—A location from which radioactive materials are released from a system, structure, or component (including evaporative releases and leaching from ponds and lakes in the controlled or restricted area before release under 10 CFR 20.2001). For release points monitored by plant process radiation monitoring systems, the release point is associated with the piping immediately downstream of the radiation monitor. (See also the definition for significant release point.) Several release sources may contribute to a common release point.

residual radioactivity—Radioactivity in structures, materials, soils, groundwater, and other media at a site resulting from activities under the licensee's control. This includes radioactivity from all licensed and unlicensed sources used by the licensee, but it excludes background radiation. It also includes radioactive materials remaining at the site as a result of routine or accidental releases of radioactive material at the site and previous burials at the site, even if those burials were made in accordance with 10 CFR Part 20.

restricted area—An area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Restricted area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.

route of exposure—A specific path (or delivery mechanism) by which radioactive material can eventually cause a radiation dose to an individual. The path typically includes a type of environmental medium (e.g., air, grass, meat, or water) as the starting point and a recipient's organ or body as the end point. For example, the grass-cow-milk-infant-thyroid route of exposure may contribute to the ingestion exposure pathway. Additionally, several routes of exposure may contribute to a single exposure pathway.

scaling factor—A factor used to estimate the unknown activity of a radionuclide based on its ratio to the activity of a readily measured radionuclide or other parameter (e.g., carbon-14 scaled to power generation).

significant contamination—As used for 10 CFR 50.75(g) recordkeeping, a quantity, concentration, or both, of residual radioactivity that would require remediation during decommissioning in order to terminate the license by meeting the unrestricted use criteria stated in 10 CFR 20.1402 (see NUREG-1757).

significant release point—Any location from which radioactive material is released that contributes greater than 1 percent of the activity discharged from all the release points for a particular type of effluent considered. RG 1.109 lists the three types of effluent as (1) liquid effluents, (2) noble gases discharged to the atmosphere in gaseous radioactive waste, and (3) all other radionuclides discharged to the atmosphere in gaseous radioactive waste.

significant residual radioactivity—See the definition for significant contamination.

site boundary—The line beyond which the licensee owns, leases, or otherwise controls land or property.

solid radioactive waste (solid waste)—solid material for which the licensee foresees no further use.

source check—A qualitative assessment of the channel response when the channel sensor is exposed to a source of increased radioactivity.

standard (instrument or source) (see ANSI N323C-2009 and ANSI N42.22-2006, "Traceability of Radioactive Sources to the National Institute of Standards and Technology (NIST) and Associated Instrument Quality Control" (Ref. 95):

- National standard—a standard determined by a nationally recognized, competent authority to serve as the basis for assigning values to other standards of the quantity concerned. In the United States, this is an instrument, source, or other system or device maintained and promulgated by the NIST.

- Primary standard—a standard that is designated or widely acknowledged as having the highest metrological qualities and whose value is accepted without reference to other standards of the same quantity.

- Secondary standard—a standard whose value is assigned by comparison with a primary standard of the same quantity.

- Reference standard—a standard, generally having the highest metrological quality available at a given location or in a given organization, from which measurements made there are derived.

- Transfer standard—A standard used as an intermediary to compare standards. (If the intermediary is not a standard, the term *transfer device* should be used.)

- Working standard—a standard that is used routinely to calibrate or check material measures, measuring instruments, or reference materials. A working standard is usually traceable to the NIST.

survey—An evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. When appropriate, such an evaluation includes a physical survey of the location of radioactive material and measurements or calculations of levels of radiation, or concentrations or quantities of radioactive material present.

type of effluent—A grouping of radioactive releases into one of the three categories listed in 10 CFR Part 50, Appendix I, paragraphs A–C. RG 1.109 classifies the three categories as (1) liquid effluents, (2) noble gases discharged to the atmosphere in gaseous radioactive waste, and (3) all other nuclides discharged to the atmosphere in gaseous radioactive waste.

unlicensed material—Radioactive material discharged as licensed material in effluents and background radioactivity (including global fallout). Licensed radioactive material becomes unlicensed radioactive material upon discharge in effluents, in accordance with 10 CFR 20.2001.

uncontrolled discharge—An effluent discharge that does not meet the definition of a controlled discharge. (See also the definition of controlled discharge.)

uncontrolled release—An effluent release that does not meet the definition of a controlled release. (See also the definition of controlled release).

unplanned discharge—The unintended or unexpected discharge of liquid or airborne radioactive material to the unrestricted area. Examples of an unplanned discharge include the following:

- the unintentional discharge of a wrong waste gas decay tank (or bulk liquid radioactive waste tank), or

- the failure of a radiation monitor to divert liquid to the radioactive waste system in the case where radioactivity is present and the automatic alarm/trip function fails to divert material to liquid radioactive waste and that material (or a portion of that material) instead discharges to the environment.

unplanned release—The unintended or unexpected release of liquid or airborne radioactive material to the onsite environment. An example of an unplanned release would include a plant occurrence that results in a leak or spill of radioactive material to onsite areas, requiring a report under 10 CFR 50.72, “Immediate notification requirements for operating nuclear power reactors,” or 10 CFR 50.73, “License event report system.” (See HPPOS-254, “Definition of Unplanned Release,” issued February 1994 (Ref. 96)).

For example, if a licensee has prepared documents describing an intended release (e.g., a preliminary radioactive waste release permit) in advance of the evolution, and the intended release occurs as planned, then the release is a planned release. If such documents (e.g., a preliminary release permit) are not prepared (or considered/evaluated) before the release, it is potentially an unplanned release (and additional information may be required to determine if it is an unplanned release).

unrestricted area—An area for which the licensee neither limits nor controls access.

uranium fuel cycle—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 using 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 nonuranium special nuclear and byproduct materials from the cycle.

χ/Q— Referred to as "Chi over Q," the average atmospheric effluent concentration, χ, normalized by release rate, Q, at a distance (or location) in a given downwind direction. Expressed in another way, χ/Q is the concentration (χ) of airborne radioactive material (e.g., in units of $\mu Ci/m^3$) divided by the release rate (Q) (e.g., in units of µCi/s) at a specified distance and direction downwind of the release point.

REFERENCES[13]

1. *U.S. Code of Federal Regulations* (CFR), "Standards for Protection Against Radiation," Part 20, Chapter 1, Title 10, "Energy."

2. CFR, "Environmental Radiation Protection Standards for Nuclear Power Operations," Part 190, Chapter 1, Title 40, "Protection of Environment."

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APPENDIX A—TABLES

Table A-1 - Gaseous Effluents—Summation of All Releases

SUMMATION OF ALL RELEASES	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL	UNCERTAINTY
Fission and Activation Gases	Ci						
Iodines (Halogens)	Ci						
Particulates	Ci						
Tritium	Ci						
Gross Alpha	Ci						
C-14	Ci						

Table A-1A - Gaseous Effluents—Ground-Level Release—Batch Mode

Fission and Activation Gases	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Ar-41	Ci					
Kr-85	Ci					
Kr-85m	Ci					
Kr-87	Ci					
Kr-88	Ci					
Xe-131m	Ci					
Xe-133	Ci					
Xe-133m	Ci					
Xe-135	Ci					
Xe-135m	Ci					
Xe-138	Ci					
(List Others)	Ci					
Total	Ci					

Iodines/ Halogens	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
I-131	Ci					
I-132	Ci					
I-133	Ci					
I-134	Ci					
I-135	Ci					
Total	Ci					

Particulates	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Co-58	Ci					
Co-60	Ci					
Sr-89	Ci					
Sr-90	Ci					
Cs-134	Ci					
(List Others)	Ci					
Total	Ci					

Tritium	Ci					

Gross Alpha	Ci					

C-14	Ci					

Table A-1B - Gaseous Effluents—Ground-Level Release—Continuous Mode

Fission and Activation Gases	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Ar-41	Ci					
Kr-85	Ci					
Kr-85m	Ci					
Kr-87	Ci					
Kr-88	Ci					
Xe-131m	Ci					
Xe-133	Ci					
Xe-133m	Ci					
Xe-135	Ci					
Xe-135m	Ci					
Xe-138	Ci					
(List Others)	Ci					
Total	Ci					

Iodines/ Halogens	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
I-131	Ci					
I-132	Ci					
I-133	Ci					
I-134	Ci					
I-135	Ci					
Total	Ci					

Particulates	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Co-58	Ci					
Co-60	Ci					
Sr-89	Ci					
Sr-90	Ci					
Cs-134	Ci					
(List Others)	Ci					
Total	Ci					

Tritium	Ci					

Gross Alpha	Ci					

C-14	Ci					

Table A-1C - Gaseous Effluents—Elevated Release—Batch Mode

Fission and Activation Gases	**UNITS**	**QUARTER 1**	**QUARTER 2**	**QUARTER 3**	**QUARTER 4**	**TOTAL**
Ar-41	Ci					
Kr-85	Ci					
Kr-85m	Ci					
Kr-87	Ci					
Kr-88	Ci					
Xe-131m	Ci					
Xe-133	Ci					
Xe-133m	Ci					
Xe-135	Ci					
Xe-135m	Ci					
Xe-138	Ci					
(List Others)	Ci					
Total	Ci					

Iodines/ Halogens	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
I-131	Ci					
I-132	Ci					
I-133	Ci					
I-134	Ci					
I-135	Ci					
Total	Ci					

Particulates	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Co-58	Ci					
Co-60	Ci					
Sr-89	Ci					
Sr-90	Ci					
Cs-134	Ci					
(List Others)	Ci					
Total	Ci					

Tritium	Ci					

Gross Alpha	Ci					

C-14	Ci					

Table A-1D - Gaseous Effluents—Elevated Release—Continuous Mode

Fission and Activation Gases	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Ar-41	Ci					
Kr-85	Ci					
Kr-85m	Ci					
Kr-87	Ci					
Kr-88	Ci					
Xe-131m	Ci					
Xe-133	Ci					
Xe-133m	Ci					
Xe-135	Ci					
Xe-135m	Ci					
Xe-138	Ci					
(List Others)	Ci					
Total	Ci					

Iodines/ Halogens	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
I-131	Ci					
I-132	Ci					
I-133	Ci					
I-134	Ci					
I-135	Ci					
Total	Ci					

Particulates	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Co-58	Ci					
Co-60	Ci					
Sr-89	Ci					
Sr-90	Ci					
Cs-134	Ci					
(List Others)	Ci					
Total	Ci					

Tritium	Ci					

Gross Alpha	Ci					

C-14	Ci					

Table A-1E - Gaseous Effluents—Mixed Mode Release—Batch Mode

Fission and Activation Gases	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Ar-41	Ci					
Kr-85	Ci					
Kr-85m	Ci					
Kr-87	Ci					
Kr-88	Ci					
Xe-131m	Ci					
Xe-133	Ci					
Xe-133m	Ci					
Xe-135	Ci					
Xe-135m	Ci					
Xe-138	Ci					
(List Others)	Ci					
Total	Ci					

Iodines/ Halogens	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
I-131	Ci					
I-132	Ci					
I-133	Ci					
I-134	Ci					
I-135	Ci					
Total	Ci					

Particulates	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Co-58	Ci					
Co-60	Ci					
Sr-89	Ci					
Sr-90	Ci					
Cs-134	Ci					
(List Others)	Ci					
Total	Ci					

Tritium	Ci					

Gross Alpha	Ci					

C-14	Ci					

Table A-1F - Gaseous Effluents—Mixed Mode Release—Continuous Mode

Fission and Activation Gases	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Ar-41	Ci					
Kr-85	Ci					
Kr-85m	Ci					
Kr-87	Ci					
Kr-88	Ci					
Xe-131m	Ci					
Xe-133	Ci					
Xe-133m	Ci					
Xe-135	Ci					
Xe-135m	Ci					
Xe-138	Ci					
(List Others)	Ci					
Total	Ci					

Iodines/ Halogens	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
I-131	Ci					
I-132	Ci					
I-133	Ci					
I-134	Ci					
I-135	Ci					
Total	Ci					

Particulates	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Co-58	Ci					
Co-60	Ci					
Sr-89	Ci					
Sr-90	Ci					
Cs-134	Ci					
(List Others)	Ci					
Total	Ci					

Tritium	Ci					
Gross Alpha	Ci					
C-14	Ci					

Table A-2 - Liquid Effluents—Summation of All Releases

SUMMATION OF ALL LIQUID RELEASES	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL	UNCERTAINTY (%)
Fission and Activation Products (excluding tritium, noble gases and gross alpha)	Ci						
Tritium	Ci						
Dissolved and Entrained Gases	Ci						
Gross Alpha	Ci						
Volume of Primary System Liquid Effluent (before dilution)	Liters						
Dilution Water Used for Above	Liters						
Volume of Secondary or Balance-of-Plant Liquid Effluent (e.g., low-activity or unprocessed) (before dilution)	Liters						
Quarterly Dilution Water Used for Above	Liters						
Average Stream Flow	m^3/s						

Table A-2A - Liquid Effluents—Batch Mode

Fission and Activation Products	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Cr-51	Ci					
Mn-54	Ci					
Fe-55	Ci					
Fe-59	Ci					
Co-57	Ci					
Co-58	Ci					
Co-60	Ci					
Sr-89	Ci					
Sr-90	Ci					
Nb-95	Ci					
Ag-110m	Ci					
Sn-113	Ci					
Sb-124	Ci					
Sb-125	Ci					
I-131	Ci					
I-133	Ci					
I-135	Ci					
Cs-134	Ci					
Cs-137	Ci					
(List Others)	Ci					
Total	Ci					

Table A-2A - Liquid Effluents—Batch Mode (continued)

Dissolved and Entrained Gases	**UNITS**	**QUARTER 1**	**QUARTER 2**	**QUARTER 3**	**QUARTER 4**	**TOTAL**
Kr-85	Ci					
Kr-85m	Ci					
Kr-88	Ci					
Xe-131m	Ci					
Xe-133	Ci					
Xe-133m	Ci					
Xe-135	Ci					
Xe-135m	Ci					
(List Others)	Ci					
Total	Ci					

Tritium	Ci					

Gross Alpha	Ci					

Table A-2B - Liquid Effluents—Continuous Mode

Fission and Activation Products	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Cr-51	Ci					
Mn-54	Ci					
Fe-55	Ci					
Fe-59	Ci					
Co-57	Ci					
Co-58	Ci					
Co-60	Ci					
Sr-89	Ci					
Sr-90	Ci					
Nb-95	Ci					
Ag-110m	Ci					
Sn-113	Ci					
Sb-124	Ci					
Sb-125	Ci					
I-131	Ci					
I-133	Ci					
I-135	Ci					
Cs-134	Ci					
Cs-137	Ci					
(List Others)	Ci					
Total	Ci					

Table A-2B - Liquid Effluents—Continuous Mode (continued)

Dissolved and Entrained Gases	UNITS	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	TOTAL
Kr-85	Ci					
Kr-85m	Ci					
Kr-88	Ci					
Xe-131m	Ci					
Xe-133	Ci					
Xe-133m	Ci					
Xe-135	Ci					
Xe-135m	Ci					
(List Others)	Ci					
Total	Ci					
Tritium	Ci					
Gross Alpha	Ci					

Table A-3 - Solid Waste and Irradiated Fuel Shipments

A. SOLID RADIOACTIVE WASTE SHIPPED FROM THE UNIT (not irradiated fuel)

TYPE OF WASTE	NUMBER OF SHIPMENTS	VOLUME (m^3)	ACTIVITY OF MAJOR NUCLIDES (Ci)
Wet radioactive waste (e.g., spent resins, filters, sludges, etc.)			
Dry radioactive waste (e.g., trash, paper, discarded protective clothing, etc.)			
Activated or contaminated metal or equipment, etc.)			
Other radioactive waste (e.g., bulk waste, soil, rubble, etc., not excepted per Section 6 of this RG.)			

B. IRRADIATED FUEL SHIPMENTS (Disposition)

Number of Shipments	**Mode of Transportation**	**Destination**

Table A-4 - Dose Limits[17], per Technical Specifications

	QUARTER 1	QUARTER 2	QUARTER 3	QUARTER 4	YEARLY
Liquid Effluent Dose Limit, Total Body	1.5 mrem	1.5 mrem	1.5 mrem	1.5 mrem	3 mrem
Total Body Dose					
% of Dose Limit					
Liquid Effluent Dose Limit, Any Organ	5 mrem	5 mrem	5 mrem	5 mrem	10 mrem
Organ Dose					
% of Dose Limit					
Gaseous Effluent Dose Limit, Gamma Air	5 mrad	5 mrad	5 mrad	5 mrad	10 mrad
Gamma Air Dose					
% of Dose Limit					
Gaseous Effluent Dose Limit, Beta Air	10 mrad	10 mrad	10 mrad	10 mrad	20 mrad
Beta Air Dose					
% of Dose Limit					
Gaseous Effluent Organ Dose Limit (iodine, tritium, particulates with >8-day half-life)	7.5 mrem	7.5 mrem	7.5 mrem	7.5 mrem	15 mrem
Gaseous Effluent Organ Dose (iodine, tritium, particulates with > 8-day half-life)					
% of Dose Limit					

(based on fractions of 10 CFR Part 50, Appendix I)

17 Doses based on quarterly and annual limits, on a per reactor basis.

Table A-5 - EPA 40 CFR Part 190 Dose Limits[18] to an Individual in the Unrestricted Area

	WHOLE BODY	THYROID	ANY OTHER ORGAN
Dose Limit	25 mrem	75 mrem	25 mrem
Dose[19]			
% of Dose Limit			

18 On a uranium fuel cycle basis (e.g., all reactors).

19 Dose from current year effluent discharges, current year direct radiation, and prior year effluents (if environmental reporting levels are exceeded).

Table A-6. Supplemental Information

1. Abnormal Releases and Abnormal Discharges (e.g., leaks and spills)

2. Nonroutine, Planned Discharges (e.g., pumping of leaks and spills for remediation, results of groundwater monitoring to quantify effluent releases to the offsite environment)

3. Radioactive Waste Treatment System Changes

4. Annual Land Use Census Changes

5. Effluent Monitor Instrument Inoperability

6. ODCM Changes

7. Process Control Program Changes

8. Errata/Corrections to Previous ARERRs

9. Other (narrative description of other information that is provided to the NRC, such as in the ARERR or ISFSI reports)