APPENDIX G
REGULATORY ANALYSIS METHODS AND DATA FOR NUCLEAR FACILITIES OTHER THAN POWER REACTORS

Enclosure 2

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS AND ACRONYMS

ACNW	Advisory Committee on Nuclear Waste
BLS	Bureau of Labor Statistics
BSC	Bechtel SAIC Company
CFR	*Code of Federal Regulations*
Ci	curie(s)
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
DPC	dual-purpose canister
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
FNMC	Fundamental Nuclear Material Control
FR	*Federal Register*
GTCC	greater-than-Class C
HEPA	high-efficiency particulate air
HF	hydrogen fluoride
HLW	high-level waste
ISA	integrated safety analysis
ISFSI	independent spent fuel storage installation
ISL/ISR	in situ leach/in situ recovery
LLW	low-level waste
mrem	millirem
MTHM	metric ton(s) of heavy metal
NA	not available or not applicable
NEPA	National Environmental Policy Act
NMED	Nuclear Materials Events Database
NRC	U.S. Nuclear Regulatory Commission
OMB	U.S. Office of Management and Budget
OpE	operating experience
PRA	probabilistic risk assessment
PuO_2	plutonium dioxide
REIRS	Radiation Exposure Information Reporting System
RF	receipt facility
SFP	spent fuel pool
SNF	spent nuclear fuel
TAD	transportation, aging, and disposal
U	uranium
UF_6	uranium hexafluoride
U_3O_8	uranium (V, VI) oxide, yellowcake
UO_2	uranium dioxide
UO_2F_2	uranyl fluoride
WHF	wet handling facility
yr	year

REGULATORY ANALYSIS METHODS AND DATA FOR NUCLEAR FACILITIES OTHER THAN POWER REACTORS

G.1 PURPOSE

This appendix documents established approaches and data considerations for use in performing non-power reactor regulatory analyses. The information presented in this appendix supplements the basic concepts for conducting a regulatory analysis described in Sections 2 through 5 of this NUREG. This appendix is for use by the analyst preparing a regulatory decision making document for non-power reactor facilities and activities, including fuel fabrication facilities, independent spent fuel storage installations (ISFSIs), irradiators, uses of byproduct material, and high-level waste (HLW) repositories.

The analyst should strive to use quantitative measures when performing a regulatory analysis for regulatory changes affecting non-power reactor facilities and activities. However, many benefits of improved regulation for these facilities and activities do not lend themselves to quantification. As discussed in Appendix A, “Qualitative Factors Assessment Tools” to this NUREG, nonquantifiable costs and benefits can be significant elements of a regulatory analysis, and the analyst and decisionmaker should consider them as appropriate. The analyst should refer to “Risk-Informed Decisionmaking for Nuclear Material and Waste Applications,” as a supplement to this NUREG for guidance on making risk-informed regulatory decisions for non-power reactors.

The analyst will need to consider NRC’s backfitting regulations when evaluating changes to requirements for non-power reactor activities. These requirements are found in 10 CFR 70.76, “Backfitting,” for specific licenses involving special nuclear material; 10 CFR 72.62, “Backfitting,” for ISFSIs and monitored retrievable storage facilities; and 10 CFR 76.76, “Backfitting,” for gaseous diffusion plants. Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests”” and NUREG-1409, Revision 1, “Backfitting Guidelines,” provide guidance to NRC staff on the application of the substantial increase standard as it relates to the NRC’s backfit regulations.

This document is organized into three sections: Section G.2, “Fuel Cycle Activities,” describes the activities, hazards, and data sources for mining, milling, conversion, enrichment, and fuel fabrication for nuclear reactors, as well as activities associated with the storage and disposal of the spent nuclear fuel (SNF) and low-level radioactive waste. Section G.3, “Non-Fuel-Cycle Activities,” addresses activities involving the use of radioactive (byproduct) material, such as for medical and industrial use (e.g., irradiators, radiography, well logging, manufacturing, gauges) and for academic or research use. Section G.4, “Common Activities,” addresses those activities that are common to both groups of non-power reactor activities, such as transportation, security, material control and accounting, and emergency planning and preparedness.

G.2 FUEL CYCLE ACTIVITIES

Fuel cycle activities are those activities associated with the extraction of uranium (and other materials), production of fuel for use in nuclear reactors, and storage and disposal of SNF and associated radioactive wastes from nuclear reactor operations. The NRC regulations for these activities are 10 CFR Part 40, "Domestic Licensing of Source Material"; 10 CFR Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories"; 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste"; 10 CFR Part 63, "Disposal of High-level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada"; and parts 70 through 76 of 10 CFR[1]. The NRC's environmental protection regulations implement the National Environmental Policy Act (NEPA) and are in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions."

NUREG/BR-0280, Revision 1, "Regulating Nuclear Fuel," is an overview of the nuclear fuel cycle and includes a high-level discussion of possible hazards in various processes. Figure G-1 illustrates general fuel cycle activities.

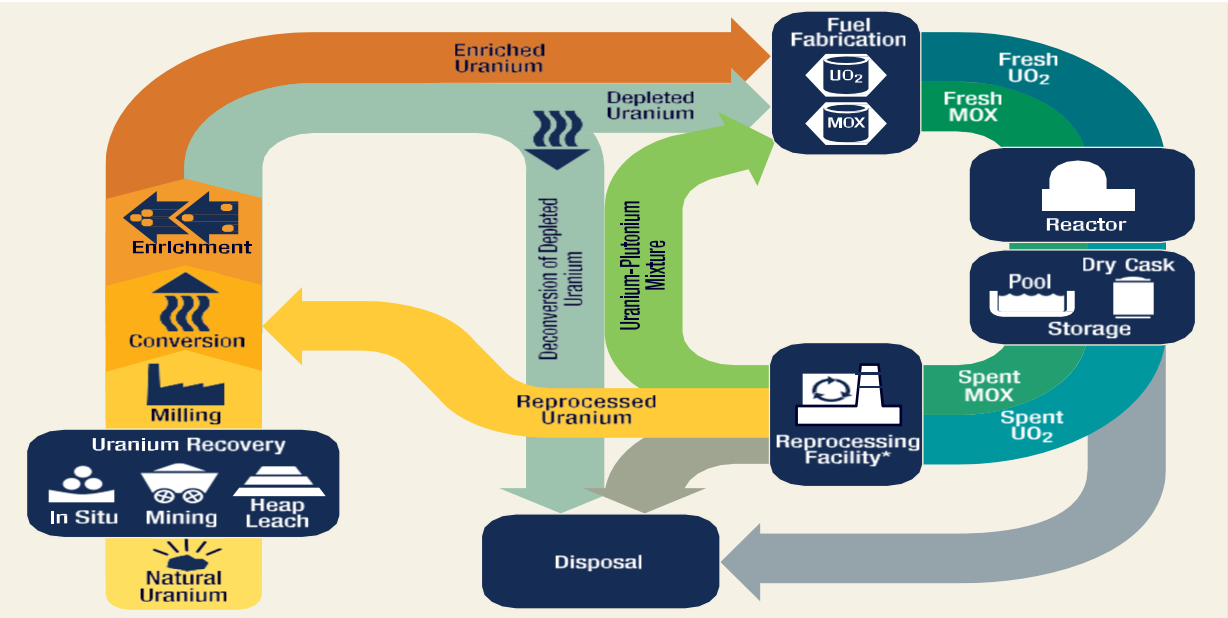

Figure G-1 The Nuclear Fuel Cycle

This section discusses the fuel activities shown in Figure G-1, the most significant hazards for each activity, and useful sources of data[2]:

- source material—uranium recovery (i.e., mining, in situ leaching, heap leaching)

---

1 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"; 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"; 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 Part 73, "Physical Protection of Plants and Materials"; 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material"; 10 CFR Part 75, "Safeguards on Nuclear Material—Implementation of Safeguards Agreements Between the United States and the International Atomic Energy Agency"; and 10 CFR Part 76, "Certification of Gaseous Diffusion Plants."

2 Operational/facility decommissioning, while not identified in Figure G-1, is discussed in Section G.2.8 of this appendix.

- milling (including reclamation of mill tailings and decommissioning)
- conversion
- enrichment
- fuel fabrication (including fuel refabrication, such as mixed oxide fuel)
- SNF storage
  - spent fuel pool (SFP)
  - ISFSI or monitored retrievable storage
- fuel reprocessing
- disposal
  - low-level waste (LLW) disposal
  - HLW geologic repository[3]

Uranium recovery by in situ leach and heap leach, discussed in Section G.2.1, fall under NRC regulatory purview; however; traditional mining is not regulated by the NRC so it is not discussed in this document. The United States currently has no licensed commercial reprocessing facilities. Should a regulatory analysis need to consider a reprocessing facility, staff should rely on NUREG/CR-7232, "Review of Spent Fuel Reprocessing and Associated Accident Phenomena." That document includes an overview of typical facility designs and describes historical accidents and the phenomena relevant to accidents at this type of facility. There also are no gaseous diffusion plants operating in the United States. Because the NRC has no knowledge of any plan to use this technology in the future, this appendix does not discuss these facilities or the associated regulations in 10 CFR Part 76.

A regulatory analysis and an environmental analysis to comply with NEPA are required for all fuel cycle facilities. Additionally, the NRC has backfitting provisions for non-power reactor licenses issued under 10 CFR parts 70, 72, and 76, for uranium enrichment, fuel fabrication, ISFSIs, HLW, and gaseous diffusion plants.

There are many sources of data available to the analyst who is developing a regulatory analysis for a regulatory action affecting specific fuel cycle activities. The list of general references below provides a starting point for performing the regulatory analysis; specific data sources are included in the discussion for each fuel cycle activity.

- NUREG/CR-2873, Volume 1, "Nuclear-Fuel-Cycle Risk Assessment: Descriptions of Representative Non-Reactor Facilities," provides a general overview of each of the fuel cycle activities and risks associated with each activity.

- NUREG/CR-2933, "Nuclear Fuel Cycle Risk Assessment: Survey and Computer Compilation of Risk-Related Literature," provides a survey and compilation of the risk-related literature for fuel cycle activities, including characterization of the specific risk/safety information.

- NUREG/CR-3682, "Nuclear Fuel Cycle Risk Assessment: Review and Evaluation of Existing Methods," provides a preliminary relative ranking of fuel cycle facilities on the basis of risk and an assessment of the adequacy of the existing (1982 timeframe) risk assessment methods.

[3] As defined in 10 CFR 63.2, "Definitions," high-level waste is composed of waste from reprocessing spent nuclear fuel and the spent nuclear fuel itself or irradiated reactor fuel.

More recent references that are generally applicable to fuel cycle activities include the annual NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Plant Reactors and Other Facilities," the Nuclear Materials Events Database (NMED), and information available to the NRC staff on the Materials Operating Experience (OpE) Gateway.

NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," provides guidance on how to calculate the characteristics and consequences of releases of radioactive materials and hazardous chemicals from fuel cycle facilities. The appendices to NUREG/CR-6410 summarize chemical and nuclear information and describe various fuel cycle facilities, details on calculating the characteristics of source terms for releases of hazardous chemicals, and other topics (e.g., high-efficiency particulate air [HEPA] filter performance and uncertainties). NUREG/CR-6410 also provides several sample problems addressing some of the more significant hazards (e.g., uranium hexafluoride [UF_6] cylinder rupture and criticality).

Additionally, a significant amount of risk assessment work has been performed to support the review and licensing of an HLW geologic repository. Section G.2.7 presents an extensive discussion of the aspects associated with the HLW risk analyses, many of which are also applicable to other fuel cycle activities. The analyst should consider this information, as appropriate, for similar activities such as SNF loading, packaging, movement, and transportation.

G.2.1 MINING AND MILLING

The nuclear fuel cycle begins with uranium recovery (i.e., mining), which is the extraction of natural uranium ore from the earth, and concentrating (i.e., milling) that ore. These recovery operations produce U_3O_8 (known as yellowcake). Uranium recovery operations also generate byproduct material waste.

NRC has determined that a license is not required for possession, use, or transfer of unrefined and unprocessed ore containing source material (10 CFR 40.13(b)). However, a license is required for a uranium mill, which is the next step in processing ore from a conventional mine. A specific license is also required for in situ recovery (ISR) (also referred to as in situ leach (ISL) mining) because uranium is extracted underground before being pumped to the surface. The NRC does not directly regulate the active uranium recovery operations in Agreement States.[4]

The majority of uranium extraction operations in the United States use ISR. Once the uranium is brought to the surface, the final stages of the ISR process produce yellowcake using low temperature vacuum dryers. A source and byproduct materials license is required to recover uranium by ISR extraction techniques under the provisions of 10 CFR Part 20, "Standards for Protection against Radiation," and 10 CFR Part 40, "Domestic Licensing of Source Material." Uranium milling and disposal of the resulting waste byproduct material by NRC licensees are also regulated under 10 CFR Part 40. Most of the regulations that the NRC has established for this type of byproduct material are found in 10 CFR Part 40, Appendix A, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the

[4] The NRC assists States expressing interest in establishing programs to assume NRC regulatory authority under the Atomic Energy Act of 1954, as amended (the Act). Section 274 of the Act provides a statutory basis under which the NRC relinquishes to the States portions of its regulatory authority to license and regulate byproduct materials (radioisotopes), source materials (uranium and thorium), and certain quantities of special nuclear material. The mechanism for the transfer of the NRC's authority to a State is an agreement signed by the Governor of the State and the Chairman of the Commission, in accordance with Section 274b of the Act.

Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content." In general, the criteria in 10 CFR Part 40, Appendix A, require uranium recovery facilities to control hazards and address waste and decommissioning concerns.

Potential accidents during processing do not yield releases much higher than those incurred during normal ISR operation. These potential accidents include various types of spills resulting from barrier breaches (i.e., failure of thickener tank, pipes, valves, or yellowcake dryer) leading to the release of pregnant lixiviant[5] or yellowcake.

The primary industrial hazards associated with uranium milling are the occupational hazards found in any metal milling operation that uses chemical extraction in addition to the chemical toxicity of the uranium itself. Because the uranium produced at these facilities is not enriched, there is no criticality hazard. Potential hazards at conventional milling facilities include a breach of the yellowcake dryer resulting in a loss of yellowcake powder (release of uranium) and the potential release of chemicals that are used in the milling process. The primary radiological hazard is attributable to the presence of radium (due to the radioactive decay of uranium) in mill tailings, which is a direct radiation hazard and causes radon emissions from the impoundment area. A collapse of the impoundment also could result in the release of hazardous material to the environment.

An applicant in a licensing action is required to provide detailed information on the facilities, equipment, and procedures used, as well as environmental information (which could be an environmental report) that discusses the effects of proposed operations on public health and safety and the environment. These submittals provide a foundation for any related analysis.

The following are some of the key data references that address uranium recovery:

- NUREG-1569, "Standard Review Plan for In Situ Leach Uranium Extraction License Applications," contains guidance for staff review of applications and provides a framework for license applications. The analyst preparing a regulatory analysis should consult the specific sections on the license application and the staff safety evaluation for basic information on hazards and effects.

- NUREG-2126, "Standard Review Plan for Conventional Uranium Mill and Heap Leach Facilities – Draft Report for Comment," provides guidance for staff review of applications for conventional uranium mills and heap leach facilities. In preparing a regulatory analysis, basic information on hazards and effects should be available in specific sections of the license application and the staff safety evaluation.

- NUREG/CR-6733, "A Baseline Risk-Informed, Performance-Based Approach for In Situ Leach Uranium Extraction Licensees," is a foundational risk characterization of general ISL facility operations. NUREG/CR-6733 describes the operations for extracting and processing uranium into yellowcake, explains the process for restoring groundwater quality subsequent to ore extraction, and identifies health and environmental hazards and risks.

---

[5] In ISL operations, a suitable injection system introduces a lixiviant into a subterranean uranium ore deposit. The lixiviant may be an acidic or alkaline medium that solubilizes uranium as it traverses the ore. The pregnant lixiviant (i.e., lixiviant containing soluble uranium) is then withdrawn and treated to recover the uranium, using suitable techniques.

- NUREG-0706, “Final Generic Environmental Impact Statement on Uranium Milling,” addresses common environmental issues associated with the construction, operation, and decommissioning of milling facilities and ISR-type facilities, as well as ground water restoration at these facilities. NRC staff use this Generic Environmental Impact Statement as a starting point for a site-specific environmental review of a license application, renewal, or amendment when addressing environmental issues common to milling and the ISR process.

- NUREG-1910, “Generic Environmental Impact Statement for In-Situ Leach Uranium Milling Facilities,” analyzes potential environmental impacts associated with the construction, operation, and decommissioning of ISR activities. The analyst can use NUREG-1910 as a starting point for the site-specific environmental review of a license application, renewal, or amendment specific to ISR activities. In particular, Appendix E, “Hazardous Chemicals,” provides an accident analysis for the more hazardous chemicals associated with ISR operations.

- NUREG-1620, Revision 1, “Standard Review Plan for the Review of a Reclamation Plan for Mill Tailings Sites Under Title II of the Uranium Mill Tailings Radiation Control Act of 1978,” discusses hazards assessment, exposure assessment, corrective action assessment, and compliance monitoring for alternative concentration limits, and addresses reclamation and stabilization cost estimates.

- “Regulatory Impact Analysis of Final Environmental Standards for Uranium Mill Tailings at Active Sites,” prepared by the Environmental Protection Agency (EPA) following the passage of the Uranium Mill Tailings Radiation Control Act of 1978, set standards that cover the processing and disposal of byproduct materials at mills that were licensed by the appropriate regulatory authorities at that time. This regulatory impact analysis examines the benefits and costs associated with the disposal of uranium mill tailings and gives some indication of what level of control of the hazards is most cost-effective. This EPA regulatory impact analysis could be useful in preparing future analyses in this area.

G.2.2 CONVERSION

After the uranium ore concentrate is produced at the milling facility, the yellowcake is packaged and shipped to a uranium conversion facility. At the conversion facility, the yellowcake is reacted with fluorine to create UF_6, which is suitable for use in enrichment operations. The UF_6 exits the conversion process as a gas that is then cooled to a liquid and drained into storage and transport cylinders. As the UF_6 cools over the course of several days, it transitions from a liquid to a solid. The cylinder containing UF_6 in the solid form can then be shipped to an enrichment facility.

As with mining and milling, the primary risks associated with conversion are chemical, rather than radiological. The process to convert uranium ore concentrate (uranium oxide or yellowcake) powder to UF_6 uses a number of flammable, volatile, and soluble chemicals, including fluorine, hydrofluoric acid, and hydrogen. These chemicals, along with UF_6, contribute to significant risks associated with inhalation if a release occurs. UF_6 is produced on a large scale for uranium enrichment. It is a white, somewhat volatile, solid at room temperature and pressure. UF_6 can form metal fluorides and react, often explosively, with organic material to form fluorinated compounds and hydrogen fluoride (HF). In the presence of water, including moisture in the air, UF_6 forms highly corrosive and toxic HF gas and particulate uranyl fluoride

(UO_2F_2), both of which are extremely toxic. The UO_2F_2 and HF, both of which form quickly during a release to the atmosphere, are readily visible as a white cloud. One of the most significant hazards for a conversion facility occurs during those stages in which the UF_6 is in liquid form and is processed. Conversion processes also use hydrogen gas, which is flammable and, if released, could create an explosive hazard. Nuclear criticality is not a hazard at these facilities because the nuclear material is unenriched (i.e., consists of natural uranium) throughout the process (NRC, 2010a).

Currently, the United States has only one licensed commercial uranium conversion facility. The NRC issued the license for this facility under 10 CFR Part 40. An applicant for a license, or for renewal or amendment of an existing license, is required to provide information consistent with 10 CFR 40.31, "Application for Specific Licenses." The applicant provides detailed information on the facilities, equipment, and procedures used, as well as an environmental report that discusses the effects of proposed operations on public health and safety and the environment. These licensee submittals provide a starting point for any related analysis. Regulatory Guide 3.55, "Standard Format and Content for the Health and Safety Sections of License Renewal Applications for Uranium Hexafluoride Production," provides high-level guidance for the preparation of the health and safety section of a renewal application. In particular, Part II of the regulatory guide addresses the safety demonstration, including Chapter 14 on accident analysis.

In addition to the above sources of information, NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities," provides broad guidance on chemical process safety issues relevant to fuel cycle facilities, including some examples for addressing chemical process safety, and sets forth the basic information needed to properly evaluate chemical process safety.

G.2.3 ENRICHMENT

As of 2018, only gaseous centrifuge technology is used in the United States for commercial nuclear fuel enrichment. In this process, the UF_6 gas is placed in a gas centrifuge cylinder and rotated at a high speed. This rotation creates a strong centrifugal force so that the heavier gas molecules (UF_6 containing uranium [U]-238 atoms) move toward the outside of the cylinder. The lighter gas molecules (containing U-235 atoms) collect closer to the center. The stream that is slightly enriched in U-235 is withdrawn and fed into the centrifuge in the next higher stage. The slightly depleted stream (with a lower concentration of U-235) is recycled back into the next lower stage. A gas centrifuge facility contains long lines of many rotating cylinders. These cylinders are connected in both series and parallel formations. Centrifuge machines are interconnected to form trains and cascades. At the final withdrawal point, the UF_6 is enriched to the desired amount. Figure G-2 illustrates the operation of a single cylinder within a gas centrifuge process.

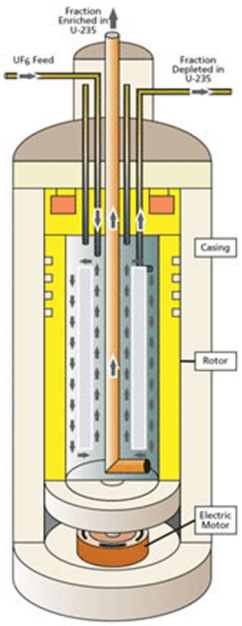

Figure G-2 Schematic of a Single Gas Centrifuge Cylinder
Source: NRC Backgrounder – Uranium Enrichment

The NRC regulates enrichment facilities under 10 CFR Part 70, including Subpart H, "Additional Requirements for Certain Licensees Authorized to Possess a Critical Mass of Special Nuclear Material." The licensee or applicant is required to provide detailed information on the facilities, equipment, and procedures used, as well as, in some cases, an environmental report that discusses the effects of proposed operations on the health and safety of the workers and the public. These licensing submittals provide a foundation for any related analysis, which includes the requirement in Subpart H for the licensee to perform an integrated safety analysis (ISA), identify items relied on for safety, and establish management measures.

Because the enrichment facilities handle and process large amounts of UF_6, the same chemical hazards exist at these facilities as at the conversion facility. In addition, the production of enriched uranium could potentially result in an inadvertent nuclear criticality under certain conditions, which would lead to high levels of radiation that can be a significant risk to workers or people in the vicinity of the facility. The primary chemical hazards for enrichment facilities are releases of UF_6, and its subsequent products (HF and UO_2F_2), from storage containers (both enriched and unenriched). The primary radiological concern is inadvertent criticality during the enrichment process. In the case of an accident, the workers have a greater chance of being impacted than the public. These facilities generally pose a low risk to the public.

The following are some of the key data references that address enrichment facilities:

- NUREG-1520, Revision 2, "Standard Review Plan for Fuel Cycle Facilities License Applications," contains guidance for staff review of license applications. The analyst preparing a regulatory analysis should consult the specific sections on the license application and the staff safety evaluation for basic information on hazards and effects. In particular, Chapter 3 of the NUREG addresses the performance of an ISA for a facility.

- NUREG-1513, "Integrated Safety Analysis Guidance Document," provides general guidance to licensees and applicants in performing an ISA. The ISA is expected to form the basis of a safety program at these facilities licensed under 10 CFR Part 70, Subpart H. The ISA involves the following:

  – performance of process hazards analysis

  – identification of accident initiating events

  – development of accident sequences and their consequences

  – the associated identification of items relied on for safety to prevent or mitigate these accident sequences

  As such, the licensee's ISA provides the bases upon which a regulatory analysis can be performed. However, the ISA itself is not part of the license and is not submitted to the NRC. The licensee submits an ISA summary annually to the NRC to reflect changes implemented at the facility in the prior year that resulted in changes to the ISA. In order to appropriately use the ISA information in a regulatory analysis, especially a facility-specific regulatory analysis or backfit analysis, the analyst may need to visit the site to review the specific process ISAs in order to fully understand the potential implications of the staff's proposed actions.

- NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," provides guidance on how to calculate the characteristics of releases of radioactive materials and/or hazardous chemicals from nuclear fuel cycle facilities and how to evaluate consequences of those releases. It contains a number of sample problems, including a liquid spill, an HF release, a UF_6 liquid cylinder rupture, and a criticality incident. NUREG/CR-6410 provides significant insight into performing new, or extending existing analyses in analyzing a regulatory action.

G.2.4 FUEL FABRICATION

Fuel fabrication facilities convert enriched uranium into fuel for nuclear reactors. Fuel fabrication for current commercial nuclear power reactors begins with the receipt of low-enriched uranium as UF_6 from an enrichment facility. The solid UF_6 is heated to gaseous form, and then the UF_6 gas is chemically processed to form uranium dioxide (UO_2) powder. This powder is then pressed into pellets, sintered (heated) into ceramic form, loaded into Zircaloy tubes, and constructed into fuel assemblies. Fuel is also fabricated for the U.S. Naval Reactors program and for non-power reactors, which typically are small reactors that do not generate electrical power but are used for research, testing, and training. Non-power reactors can include research reactors and reactors used to produce irradiated target materials (e.g., isotopes for medical use). The fuel design varies with the non-power reactor manufacturer, and there is a wide range of fuel assembly designs. Another potential fuel for commercial nuclear power reactors is mixed oxide fuel, which would comprise both UO_2 and plutonium dioxide (PuO_2). Figure G-3 depicts a typical light-water reactor fuel fabrication facility process.

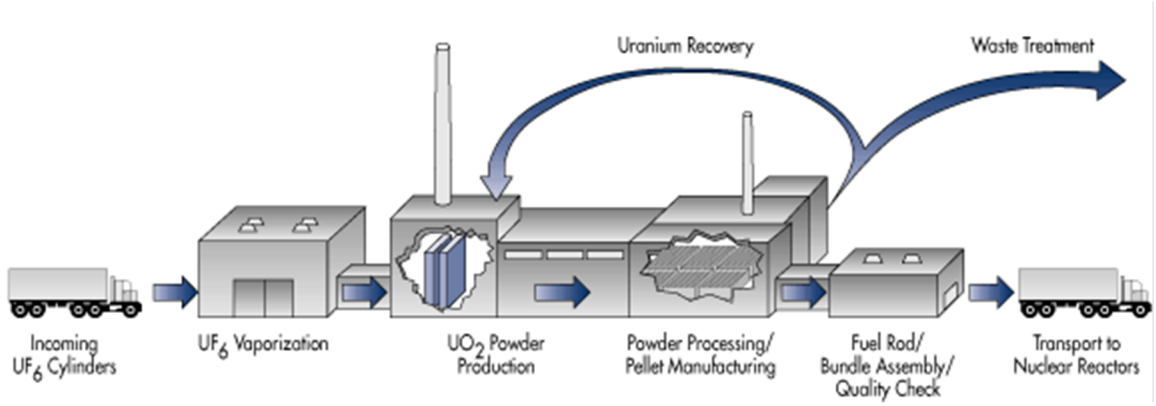

Figure G-3 Typical Light Water Reactor Fuel Fabrication Facility

Similar to its approach for enrichment facilities, the NRC regulates fuel fabrication facilities under 10 CFR Part 70, including Subpart H. The licensee or applicant is required to provide detailed information on the facilities, equipment, and procedures used, as well as, in some cases, an environmental report that discusses the effects of proposed operations on the health and safety of the workers and the public. These submittals provide a foundation for any related analysis, which includes the requirement, in Subpart H, for the licensee to perform an ISA, identify items relied on for safety, and establish management measures.

Fuel fabrication facilities have essentially the same types of hazards as enrichment facilities (i.e., chemical, radiological, and criticality hazards). The primary hazard concerns involve the release of UF_6, and its subsequent products (HF and UO_2F_2) in storage or during processing (i.e., vaporization) and inadvertent criticality during the UO_2 fuel fabrication processes. In the case of an accident, the workers have a greater chance of being impacted than the public. These facilities generally pose a low risk to the public.

The key references for enrichment facilities in Section G.2.3 of this appendix data also address fuel fabrication facilities.

G.2.5 SPENT NUCLEAR FUEL STORAGE

SNF refers to uranium-bearing fuel elements that have been used in nuclear reactors and have been removed from the reactor vessel. After the SNF is removed from the reactor, the spent fuel assemblies still generate significant amounts of radiation and heat. There are two acceptable storage methods for SNF: spent fuel pools (SFPs) and, after a suitable cooling time, dry cask storage at an ISFSI. Most SNF is stored in SFPs at individual reactor sites. The SFP maintains at least 20 feet of water above the stored fuel to provide adequate cooling of the SNF and adequate radiation shielding for plant personnel. The assemblies are moved into the SFPs from the reactor along the bottom of water canals, so that the SNF is always shielded to protect workers. The SNF accident scenarios include inadvertent criticality from misloading SNF in high-density ranks or a loss of SFP water inventory that results in SNF heatup.

The SNF that has sufficiently cooled can be stored in a variety of dry cask storage systems. There are several dry cask storage system designs, some of which can be used for both storage and transportation. The ISFSI is used to store SNF during operation when the nuclear power plant approaches the capacity limits of its SFP, or following the decommissioning of a reactor site.

An NRC-certified dry cask is one for which the NRC staff has performed a technical review of its safety aspects and has found the cask to be adequate to safely store SNF at a site that the licensee has evaluated and shown meets all of the NRC's requirements in 10 CFR Part 72. The dry cask storage license may be site specific or general. Under a site-specific license, an applicant submits a license application to the NRC and the NRC performs a technical review of all the safety and environmental protection aspects of the proposed ISFSI. The NRC has approved one license for consolidated interim storage facilities under 10 CFR Part 72 where the ISFSIis not co-located with a nuclear power reactor.

A general license authorizes a nuclear power plant licensee, without further NRC review, to store SNF in NRC-certified dry casks at a site that is licensed to operate a power reactor under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Licensees are required to perform evaluations of their site to demonstrate that the site is adequate for storing SNF in dry casks. These evaluations must show that the cask Certificate of Compliance conditions and technical specifications can be met at the licensee's site, including an analysis of earthquake events and tornado missiles. The licensee also must review its security program, emergency plan, and other programs, and make any necessary changes to incorporate the ISFSI at its reactor site. The dry cask storage Certificate of Compliance contains the technical requirements and operating conditions (e.g., fuel specifications, cask leak testing, surveillance, and other requirements) for the ISFSI and specifies what the licensee is authorized to store at the site.

The regulation related to backfitting for ISFSIs, 10 CFR 72.62, only applies to specific and general ISFSI licensees and not to certificate holders. It also uniquely requires the NRC to evaluate changes to both occupational and public radiological exposure, whereas all other backfit regulations require the NRC to evaluate changes to public exposure only. Thus, it is possible that a cost-beneficial backfit could be justified that demonstrates substantial worker benefits from averted exposure but results in small or no public benefit for these facilities.

The primary hazard associated with dry cask storage is the breach of the cask and subsequent radiological release to the environment during handling and movement from the SFP to the ISFSI area. The breach and subsequent release would be primarily an occupational hazard. Once the spent fuel is placed in the dry cask, the radiological hazard is significantly reduced as the low pressure inside the cask would not provide a significant driving force for a radiological release.

A number of studies, including the following, have considered the hazards and risks associated with SFPs:

- NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond-Design-Basis Accidents in Spent Fuel Pools."

- NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants."

- NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor."

To apply for a site-specific license to store SNF, an applicant submits an application to the NRC for review and approval. The application contains information as described in Regulatory Guide

3.50, Revision 2, "Standard Format and Content for a Specific License Application for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility," and the NRC reviews the application in accordance with NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities." The application must address the safety and operational characteristics of the facility, including the site seismic and environmental conditions, the planned storage system, accident analyses, and the radiological impact of normal operations. The following data references provide information for performing the analysis for a regulatory action related to dry cask storage, applications, and staff reviews:

- NUREG-2215 provides guidance to staff for reviewing safety analysis reports for (1) Certificates of Compliance for a dry storage for use at a general license facility and (2) a specific license for a dry storage facility that is either an ISFSI or a monitored retrievable storage installation.

- NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," provides guidance for the safety review of renewal applications for specific licenses of ISFSIs and Certificates of Compliance of dry cask storage systems, including the review of time limited aging analyses and aging management programs.

G.2.6 Low-Level Waste Disposal

Low-Level Waste (LLW) includes radioactively contaminated protective clothing, tools, filters, rags, medical tubes, and many other items. LLW disposal occurs at commercially operated LLW disposal facilities that must be licensed by either the NRC or an Agreement State. The facilities must be designed, constructed, and operated to meet safety standards. The licensee must extensively characterize the site on which the facility is located and analyze how the facility will perform for thousands of years into the future. The United States currently has four LLW disposal facilities that accept various types of LLW; all are located in and regulated by Agreement States. The Low-Level Radioactive Waste Policy Amendments Act of 1985 encouraged the States to enter into compacts that would allow them to dispose of LLW at a common disposal facility. The NRC provides additional information on the LLW disposal facilities in NUREG-1350, "NRC Information Digest."

The following principal regulations govern LLW:

- The general provision in 10 CFR 20.2002, "Method for Obtaining Approval of Proposed Disposal Procedures," allows for other disposal methods, different from those already defined in the regulations (see 10 CFR 20.2001), if doses are maintained as low as is reasonably achievable and within the dose limits of 10 CFR Part 20.

- In 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," the NRC establishes the licensing procedures, criteria, and terms and conditions for land disposal of radioactive waste.

The NRC provides an additional resource for further understanding the history of LLW operations and the associated regulations in NUREG/BR-0121, "Regulating the Disposal of Low-Level Radioactive Waste: A Guide to the Nuclear Regulatory Commission's 10 CFR Part 61." NUREG-1853, "History and Framework of Commercial Low-Level Radioactive Waste Management in the United States: ACNW [Advisory Committee on Nuclear Waste] White Paper," also provides background information on the disposal of commercial LLW.

The primary hazard from LLW disposal is the radiological release to the environment from the disposal site (e.g., resulting from the failure of the disposal liner or barrier).

In addition to the information in the license application and associated staff safety evaluation, the following data references are useful in performing a regulatory analysis associated with LLW disposal:

- NUREG-1573, "A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities: Recommendations of NRC's Performance Assessment Working Group," provides information and recommendations on performance assessment methodology as it relates to the objective of radiological protection of the general public in accordance with 10 CFR 61.41, "Protection of the General Population from Releases of Radioactivity."

- NUREG-1200, Revision 3, "Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," contains the staff guidance on reviewing applications to construct and operate LLW disposal facilities.

- NUREG-1300, "Environmental Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility: Environmental Report," provides NRC staff with guidance on performing the environmental review of an application to construct and operate a LLW disposal facility.

- NUREG-1199, Revision 2, "Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility," is the companion guidance for prospective applicants for a license to dispose of LLW pursuant to 10 CFR Part 61.

- NUREG-1241, "Licensing of Alternative Methods of Disposal of Low-Level Radioactive Waste," provides NRC staff with guidance on specific methods of disposal under consideration as alternatives to shallow land burial.

Other sources of information to consider are two U.S. Department of Energy (DOE) environmental evaluations:

- EIS-0375, "Disposal of Greater-Than-Class C (GTCC) Low-Level Radioactive Waste and GTCC-Like Waste," issued in January 2016 (DOE, 2016), which addresses the potential environmental impacts associated with constructing and operating a new facility or facilities, or using an existing facility, for the disposal of greater-than-Class C (GTCC)[6] and GTCC-like waste anticipated to be generated through 2083.

- EA-2082, "Final Environmental Assessment, " issued on October 23, 2018 (DOE, 2018), which assesses the disposal of GTCC Low-Level Radioactive Waste and GTCC-like waste at the Waste Control Specialists land disposal facility in Andrews County, Texas.

---

[6] The NRC divides LLW into three classes: Class A, Class B, and Class C. The gradation of these classes is based on the radiological hazard as determined by the quantity and type of radionuclides permitted in each class, as further delineated by concentrations of certain radionuclides. Therefore, Class A waste is the least hazardous and Class C waste is the most hazardous. In addition, some waste streams have radionuclide concentrations exceeding the limits for Class C waste and as such are referred to as "greater-than-Class C" (GTCC) waste.

G.2.7 GEOLOGICAL REPOSITORY

High-level waste is generated from DOE activities and spent nuclear fuel from commercial nuclear power plants. Disposal is currently analyzed for a mined geologic repository. This section provides the analyst with (1) the risks associated with the development of a repository, (2) a quantitative perspective on the overall risks for repository development, and (3) the risks associated with specific activities during repository development (e.g., handling, transportation, and disposal).

The development of a repository involves the handling, transportation, storage, and disposal of SNF primarily in sealed canisters. Thus, the risks for the development of a repository tend to be significantly less than for activities associated with other nuclear facilities (e.g., nuclear power reactors) that rely on active safety systems as compared to the passive safety nature of a sealed canister. However, the handling of SNF relies on active safety systems, and the transportation of SNF results in the potential for exposure to larger segments of the population than would occur near a specific site. For the purposes of this appendix: (1) only spent nuclear fuel is analyzed in detail, and (2) only mined geologic repositories are considered .

Estimating the risks related to the development of a repository includes consideration of three major activities:

(1) transporting the SNF from individual sites to a repository

(2) constructing a repository, operating the surface facilities, and emplacing the wastes (repository pre-closure period)

(3) performance of a repository after it has been permanently closed (repository post-closure period)

The analyses for repository risks in this section are based on an evaluation of various designs and site conditions (e.g., population densities, transportation routes) that are considered to be applicable to a range of potential sites in the United States. This approach provides a “generic” analysis that would be useful for application at a variety of potential sites and a comparison with other published studies on the risks from the transportation and disposal of radioactive wastes. Assumptions about the basis for the risk estimates are provided to add the appropriate context for understanding the risks in this section. Further evaluations may be necessary for site conditions and designs that vary significantly from those used for estimating the risks presented in this section.

G.2.7.1 TRANSPORTATION OF SPENT NUCLEAR FUEL TO A REPOSITORY

SNF is stored at numerous facilities around the country. Impacts for transporting spent nuclear fuel from the individual facilities to a repository is dependent upon their distance to the repository. A transportation campaign would be conducted to ship SNF from these sites to a repository; this campaign is expected to include both rail and truck transport, and result in radiation exposures from the loading of transportation casks and exposures along the transportation routes from both routine (i.e., incident-free) transportation and potential accidents.

Radiological Impacts from Loading Spent Nuclear Fuel

Impacts to workers during loading activities occur during the loading of SNF into canisters, during the loading of canisters into rail casks, and, at some sites, during the loading of SNF into truck casks. DOE/EIS-0250F-S1, "Final Supplemental Environmental Impact Statement for a Geological Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada" (DOE, 2008), states that a transportation, aging, and disposal (TAD) canister is similar to a dry storage canister in appearance, capacity, and the operational procedures that would be in use for loading. Therefore, based on historical information for the loading of SNF into TAD canisters at commercial generator sites, the DOE estimates that the average radiation occupational dose for loading commercial SNF fuel into canisters is 0.400 person-rem per canister (DOE, 2008, page G-2). The DOE also estimates that the average radiation doses for workers during (1) loading of uncanistered SNF into truck casks and loading the casks onto trailers is 0.432 person-rem per cask, and (2) transferring canisters from storage, loading canisters into rail casks, and loading the casks onto railcars is 0.663 person-rem per cask (DOE, 2008, Table G-2). The DOE used a specific number of cask shipments (i.e., 6,499 TAD canisters in rail casks, 307 dual-purpose canisters [DPCs] in rail casks, and uncanistered SNF in 2,650 truck casks) to accommodate 63,000 metric tons of heavy metal (MTHM) of SNF from commercial reactors. **Table G-** summarizes the data used to estimate radiation doses to workers for loading.

Table G-1 Data Used to Estimate Radiation Doses for Loading

Type	Operation	Radiation Dose	Number of Canisters or Casks for Operation
Rail Cask	Load commercial SNF into canister	0.400 person-rem per canister	6,499 canisters[a]
Transfer Rail Cask	Transfer canister from storage, load into rail cask, load rail cask onto rail car	0.663 person-rem per cask	6,806 casks[b,c]
Truck Cask	Load uncanistered SNF into truck cask, load truck cask onto truck trailer	0.432 person-rem per cask	2,650 casks[b]

[a] Includes only TAD canisters.
[b] Includes commercial SNF only.
[c] The estimate is based on 6,499 casks of commercial SNF containing TAD canisters and 307 casks of commercial SNF containing DPC.
Source: DOE, 2008, Table G-2

Table G- provides the collective worker dose for the loading campaign, which is obtained by multiplying the number of canisters and casks by the appropriate cask or canister collective dose rate for the workers. The DOE also estimates that a maximally exposed worker conducting loading activities at generator sites would receive an exposure rate of 500 millirem (mrem) per year based on the DOE's administrative control limit of 500 mrem per year (DOE, 2017). A similar analysis could be performed to analyze alternative disposal methods if the analyst accounts for the difference in the number of canisters and the radiation dose per canister associated with the specific canister design and loading procedures.

Radiation doses to members of the public near generator sites could occur because of the venting of radioactive gases during the handling of SNF in SFPs and the potential release of surface contamination from dry transfer casks. The DOE estimates that the population dose to members of the public within 16 kilometers (10 miles) of the generator sites would be

2.9 person-rem over the duration of loading operations. The probability of a latent cancer fatality based on the estimated dose would be 0.0017, or about one chance in 600 that one member of the exposed population would develop a latent cancer fatality. The DOE also estimated that a maximally exposed individual located 800 meters (0.5 mile) from the generator site would receive an exposure of 7.7×10^{-6} rem (DOE, 2008, page 6-12). Table G- presents the estimated exposures for workers and the public for the loading activities over a 50-year period and a shipment of 63,000 MTHM of SNF.

Table G-2 Estimated Radiological Impacts from Loading Operations

Exposure Group	Collective Dose (Person-Rem)	Latent Cancer Fatalities	Maximally Exposed Individual (Mrem/Yr)
Involved Workers[a]	8,300	5.0	500
Public	2.9	0.0017	0.0077

[a] Radiation exposures from loading operations would not occur among noninvolved workers because these workers would not be exposed to radiation from the operations.

Impacts from Transportation of Spent Nuclear Fuel to the Repository

Routine (i.e., accident-free) transportation is estimated to result in a low dose to the public. In particular, in Figure 6-3 of NUREG-2125, "Spent Fuel Transportation Risk Assessment Final Report," the NRC estimates the dose to the maximally exposed individual during transportation to be less than 0.001 mrem per shipment by either rail or truck. In NUREG-2125, the NRC estimated the risks of SNF transportation for a variety of shipping routes (both rail and truck), primarily from locations in the eastern United States to locations in the western United States, that could result in potential doses to workers and the public. The NRC estimated the average collective public dose from routine truck transport to be 0.14 rem per shipment (NRC, 2014a, Figure 6-1). The collective public dose includes doses to the population along the route (6 percent of collective dose), doses to occupants of vehicles sharing the route (38 percent of collective dose), and doses to the public at stops (56 percent of the collective dose) (NRC, 2014a, page 133). The NRC estimated the average collective dose for workers for routine truck shipments to be 0.10 rem per shipment, which includes doses to the vehicle crew and other workers (e.g., escorts, inspectors, truck stop workers) (NRC, 2014a, Figure 6-1).

NUREG-2125 also provides radiation dose estimates for two types of rail packages—a cask using lead gamma shielding and a cask using steel gamma shielding. The results reported in Table G-3 of this appendix are for the lead package because of the larger doses (i.e., approximately 30 percent larger). The NRC estimated the maximally exposed individual during routine rail transport to receive less than 0.001 mrem per shipment, similar to the estimate for a truck shipment (NRC, 2014a, Figure 6-3); however, the average collective doses were less for rail shipments than for truck shipments. The estimated average collective public dose during rail shipment using a lead cask is estimated to be 0.026 rem per shipment, and the average collective worker dose is estimated to be 0.041 rem per shipment, for a total collective dose of 0.67 rem per shipment (NRC, 2014a, Figure 6-2).

Table G-3 presents the collective dose for a transportation campaign, with the number of shipments consistent with the loading campaign discussed below (i.e., 6,806 rail shipments and 2,650 truck shipments).

Table G-1 Estimated Worker and Public Impacts from Shipping Commercial SNF to the Repository over a 50-Year Period when Waste is Expected to be Received and Emplaced in the Repository Drifts

Exposure Group	Total Average Collective Dose (Person-Rem)	Annual Average Collective Dose (Person-Rem/Yr)
Workers (Rail)	279	5.6
Workers (Truck)	265	5.3
Total Workers	544	10.9
Public (Rail)	177	3.5
Public (Truck)	371	7.4
Total Public	548	10.9

Table S-4 in 10 CFR Part 51 provides a range of doses for exposed individuals per reactor year. The values in Table S-4 would result in larger doses if they were used to generate the collective doses presented in Table G-3 of this appendix. The values in Table G-3 are based on recent analyses and represent realistic doses compared to the more conservative values used in Table S-4.

Loading and Handling Canisters and Transportation Accidents

The SNF presents a significant hazard from direct radiation for individuals that are in close proximity to SNF and from internal exposure if radioactive material is released into the environment. Thus, the loading and transportation of SNF is performed under requirements that are intended to minimize the occurrence of accidents and the consequences resulting from an accident (e.g., single-failure-proof cranes for limiting the probability of drop events, HEPA filters for buildings where handling of SNF assemblies occur, storage and transportation casks designed to withstand a variety of potential accidents). Compliance with the NRC's requirements in 10 CFR Part 71 and 10 CFR Part 72 ensures that the risks associated with loading and transportation accidents are low.

The NRC has considered a comprehensive set of initiating events for estimating the risks of handling, transfer, and storage of dry cask storage systems in NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant." NUREG-1864 estimates the probability and consequences of a loading accident based on considerations such as stresses on the fuel cladding and canister as a function of drop height, breach of the canister sealed lid, and reliability of secondary containment to isolate accidental releases during handling (e.g., containment within a handling facility). The largest identified risk occurs for a 19-foot drop during loading and handling of a canister that results in failure of the cladding and a 0.28 probability that the canister is breached (NRC, 2007b, Table 19, item 20). In the event of a breach, the average individual dose for those living within 16 kilometers (10 miles) of the facility where SNF is being loaded is 600 mrem[7] with a peak individual dose of 185 rem for individuals within 1.2 and 1.6 kilometers (0.75 and 1 mile) of the facility (NRC,

[7] NUREG-1864 reports consequences in terms of latent cancer fatality. A factor of 5×10^{-4} latent cancer fatality per rem of exposure is used to derive dose exposures from the latent cancer fatalities.

2007b, Table 18 and Table E.3). However, the probability of such a release occurring is low. NUREG-1864 provides estimated low likelihoods for a 19-foot drop that breaches the canister (i.e., 5.6×10^{-5} per lift x 0.28). Combined with a low probability for the subsequent failure of the secondary containment building to isolate the released waste after it escapes the canister (i.e., 1.5×10^{-4} per demand), the overall probability of release is 2.4×10^{-9} per lift.[8] The total risk from loading all of the SNF is the risk for a single lift multiplied by the total number of canisters that would be loaded over the transportation campaign (i.e., approximately 10,000 canisters). Thus, the total risk from a 19-foot drop during the transportation campaign for those individuals residing within 16 kilometers (10 miles) of the facility where SNF is loaded is calculated as follows:

	2.4×10^{-9}	(probability of release per lift)
x	600 mrem	(average individual dose given release occurs)
x	10,000	(number of lifts in transportation campaign)
=	0.014 mrem	(total risk from transportation campaign)

These probability weighted doses are small, partly because the canister drop is assumed to occur inside a building that would provide secondary containment to minimize potential releases in accordance with 10 CFR Part 50. If a canister drop occurs outside of a 10 CFR Part 50 building (i.e., at the ISFSI pad) the probability of a release occurring would increase to 1.6×10^{-5} or higher because a secondary containment building is not available and a single-failure-proof crane may not be used (NRC, 2007b, Table 19, note 13). The Electric Power Research Institute's Report 1002877, "Probabilistic Risk Assessment (PRA) of Bolted Storage Casks: Updated Quantification and Analysis Report," also contains information about loading, transportation, and storage of dry casks.

NUREG-2125 also provides the evaluated risks of transportation accidents. Transportation packages for radioactive materials such as SNF are designed to maintain their integrity in severe accidents. Almost all spent fuel casks are shipped without incident. However, even this routine, incident-free transportation causes radiation exposures because all loaded spent fuel casks emit some radiation (Table G-3 presents exposures for routine [accident free] transportation) whether or not radioactive material is released from the cask. This is because transportation cask shielding attenuates but does not eliminate the radiation that spent fuel emits. In NUREG-2125, the NRC concluded that the collective dose risk from accidents involving a release of radioactive material and a loss of lead shielding is negligible compared to the risk from a no-release, no-loss-of-shielding accident because of the low likelihood of accidents that affect the performance of the transportation cask (NRC, 2014a, page xxxvii). Given the robust design of the waste package, NUREG-2125 concludes that (1) only about 1 in 1,000 trips would result in an accident, (2) if an accident occurs, only about 1 in 2,000 accidents is more severe than the regulatory accident conditions, and (3) if an accident is more severe than the regulatory accident conditions, only about 3 in 1,000,000 will result in either a loss of gamma shielding or a release of radioactive material (NRC, 2014a, page xxxiv). For example, "[I]f there were an accident during a spent fuel shipment, there is only about a one-in-a-billion chance that the accident would result in a release of radioactive material" (NRC, 2014a, Figure PS-8 and page 139).

---

[8] If a release of radioactive material were to occur inside the secondary containment following a transfer cask drop, three distinct functions are designed to occur to (1) detect radioactive material, (2) isolate the secondary containment, and (3) operate the standby gas treatment system.

NUREG-2125, the NRC evaluated a variety of potential rail and truck accident scenarios over 16 truck routes and 16 rail routes and estimated that the average collective dose risk (i.e., probability weighted dose) is less than 0.003 person-rem (NRC, 2014a, Tables 6-4 and 6-5). The average collective dose for accidents with no release and no shielding loss for the rail-lead cask is 0.0022 person-rem and for the truck cask is 0.0024 person-rem. The most significant parameters contributing to the risk are the accident frequency and the length of time the intact transportation package sits at the accident location. The significant parameter for the radiological effect of the accident is not the amount or rate of radiation released but the time of exposure to the immobilized cask at the accident site. The total average collective dose risk for a transportation campaign (i.e., 6,806 rail shipments and 2,650 truck shipments) is 21.3 person-rem, which is approximately 25 times less than the collective public dose estimated for routine (incident-free) transportation shown in Table G-3 of this appendix.

G.2.7.2 REPOSITORY CONSTRUCTION, OPERATION, AND CLOSURE

The development of a repository is generally characterized by three periods that are associated with separate regulatory decisions: (1) construction (decision to grant or deny a construction authorization), (2) operations (decision to allow the receipt of radioactive waste at the repository and the start of emplacement operations), and (3) closure (decision to permanently close the facility). These three periods are typically referred to as the pre-closure period. After permanent closure, the post-closure period begins, and the repository relies on passive safety barriers. Although the DOE is required to provide continual oversight of the repository after permanent closure, this DOE oversight is not relied on for compliance with the post-closure dose limits.

Although the decisions occur at a distinct time, the periods of construction, operations, and closure will overlap in time to varying degrees. For example, the licensee can apply for a license to begin operations (i.e., receive waste) once the repository is ready to begin initial operations. However, all emplacement drifts,[9] hereafter referred to simply as “drifts,” and some surface facilities would not necessarily be completed, and thus additional surface facilities and repository drifts would continue to be developed during the initial operations.

The development of a repository has the potential to expose individuals to radiation from both human-made radionuclides in the waste and natural radionuclides associated with the underground facility (common in hard rock mines with natural radioactive materials). Exposures from the release of radionuclides into the air can affect all workers at the site and members of the public residing offsite, whereas an external dose is limited to those individuals engaged in activities near radioactive materials (e.g., handling operations for waste packages). In considering the risks from the development of a geological repository, it is useful to estimate the risks for each of the phases of development because of the potentially long time (e.g., 100 years) over which these activities will occur and the different types of activities associated with a specific phase. It is also useful to represent the risks for different exposure groups because there is the potential for significant variations in risks among exposure groups.

The DOE developed an Environmental Impact Statement (EIS) (DOE, 2008) for a geological repository at Yucca Mountain. Regardless of the site, the development of a repository will require a number of similar activities (e.g., handling and emplacement of waste packages, potential

[9] In mining terminology, a drift is a horizontal underground passage. As used in this appendix, an emplacement drift represents the horizontal excavations for the emplacement of radioactive waste packages.

repackaging of waste, excavation of drifts). Therefore, the EIS is used to assist in quantifying the potential risks from the development of a repository. The risks are generally applicable to any repository site that has similar activities. In order to estimate the potential risks, the DOE made numerous assumptions. These assumptions include the length of time for development to occur and the amount of waste that is expected to be emplaced at the repository. The EIS specifies an operating period of 105 years, beginning with construction and ending with the permanent closure of the repository, and a statutory limit of 70,000 MTHM, of which 63,000 MTHM is commercial SNF.

The EIS provides dose estimates for involved workers (i.e., workers performing physical work such as constructing, operating, monitoring, and closing the repository), noninvolved workers (i.e., managerial, technical, supervisory, and administrative personnel onsite), a maximally exposed offsite individual residing near the facility, and a potential population of 117,000 individuals residing within 84 kilometers (52 miles) of the facility. The EIS also estimates the radiation exposure from both SNF and the radioactivity from natural sources.

Exposure to Natural Sources of Radiation during the Preclosure Period

The site of an underground facility may contain certain naturally occurring radionuclides in the walls of the host rock that result in an external dose to individuals that spend a significant amount of time in the underground facility. The underground host rock may also release radon gas that will cause internal exposure to individuals in the underground facility. In addition, the ventilation of the radon gas to the surface environment results in exposure to individuals on the surface (both to onsite workers and those offsite). Radiation exposures from these natural sources of radiation will occur as the repository drifts begin to be excavated and before the presence of any radioactive waste at the site. These radiation exposures will continue as long as the underground facility is ventilated or workers are present in the underground facility. Naturally occurring radon would account for more than 99.8 percent of the radiological impacts to the offsite public (DOE, 2008, page A-4).

The release of radon from the underground facility will begin at the start of construction and continue to increase until all the drifts are completed, after which the release should remain constant until the drifts are permanently closed and the ventilation has ended. The DOE has estimated that annual radon releases would steadily increase linearly over the 27-year period needed to complete all repository drifts. They would then remain at a constant annual release through ventilation of the underground facility of 4,700 curies (Ci) until the beginning of the closure activities. At the start of a repository project only construction activities would take place, and thus the only exposure would be from natural radionuclides such as radon. Table G-4 estimates potential radon releases over the different time periods of repository development.

Table G-2 Atmospheric Release of Radon through Ventilation of the Underground Facility during Repository Development

Activities/Phase	Time Period (Years)	Annual Radon Release (Ci/Yr)		
		Initial	Ending	Annual Average
Construction Only	0 – 5	0	870	425
Construction and Emplacement	6 – 27	870	4,700[a]	2,775
	28 – 55	4,700	4,700	4,700
Monitoring	56 – 95	4,700	4,700	4,700
Closure	96 – 105	4,700	0	2,350

[a] The radon release peaks at 4,700 Ci per year in year 27 and remains at that rate until the initiating of closure activities in year 96.

Source: DOE, 2008, Figure D-1

Radon that is released through ventilation has the potential to result in exposures to underground and surface workers as well as members of the public residing offsite. The DOE identified workers as either involved workers or noninvolved workers, and estimated the potential radiation exposures for the surface and subsurface locations within the geological repository operations area. Radon is assumed to disperse sufficiently over the geological repository operations area such that the maximum exposure for surface workers (both involved and noninvolved) would be similar. Public exposures are dependent on the location of the members of the public. A repository will have an area where public access is restricted, and this would vary from site to site. The DOE determined that the location for the maximally exposed offsite member of the public, based on air dispersion of atmospheric releases from the Yucca Mountain repository, is just outside the restricted area (approximately 18 kilometers [11 miles] from the subsurface facilities) in the south-southeast direction from the repository. In determining exposure for the maximally exposed individual, the DOE assumed a hypothetical individual resided continuously for 70 years at this location and determined a population dose based on an estimated population distribution for the year 2067 that resulted in a population of approximately 117,000 individuals residing within 84 kilometers (52 miles) of the facility (DOE, 2008, page D-12). Although the maximally exposed individual is assumed to reside continuously outside the controlled area in the prevailing downwind direction from the repository at a location where the estimated dose is a maximum, the estimated population dose provides a collective dose for all individuals assumed to reside within 84 miles of the facility where the dose for a given individual within the population of 117,000 individuals will vary based on with the individuals distance from the repository as well as the location relative to the direction of the wind. Table G-5 provides unit dose conversion factors for evaluating radon exposures for workers and the public present on the land surface.

Table G-5 Internal Dose Conversion Factors for the Maximally Exposed Individuals and the Offsite Population Present on the Land Surface Based on a Release of 1 Ci of Radon

Maximally Exposed Individual Dose Conversion Factors				Offsite Population Collective Dose Conversion Factor (Person-Rem Per Ci)
Involved Surface Worker (Mrem Per Ci)	Noninvolved Subsurface Worker (Mrem Per Ci)	Noninvolved Surface Worker (Mrem Per Ci)	Offsite Individual (Mrem Per Ci)	
0.0010	0.0011	0.00097	0.0016	0.033

Source: DOE, 2008, Table D-5

Subsurface workers have the potential to receive larger external exposures because the proximity to radionuclides in drift walls. In addition, exposure to a higher concentration of radon in the underground facility results in higher internal exposure from radon in the air. The DOE assumed that the subsurface involved worker is exposed to an average radon concentration of 5.8 picocuries per liter of air, resulting in an internal dose of 70 mrem per year (DOE, 2008, page D-19). The subsurface involved worker would also receive an external dose of 50 mrem per year from radionuclides in the drift walls as a result of spending 2,000 hours underground during the work year (DOE, 2008, page 4-63). The subsurface noninvolved worker would not receive a significant external dose because of the limited amount of time spent in close proximity to the drift walls. The radon dose is estimated differently to account for the noninvolved worker who is subject to radon levels in the south portal rather than other areas of the subsurface containing higher concentrations of radon. As presented in Table G-6, the DOE estimated a larger unit exposure rate for the noninvolved subsurface worker during the construction-only period (i.e., initial 5 years of development) versus all the other times when the radon dose is comparable to the radon dose estimated for the surface workers (DOE, 2008, Table D-5). The higher rate during the construction-only period is because all ventilation exhaust exits through the south portal of the underground facility, whereas ventilation after the construction-only period is exhausted through six ventilation shafts that replace the use of the south portal for the air exhaust (DOE, 2008, pages D-12 and 13).

Table G-6 Internal and External Dose Conversion Factors for the Maximally Exposed Worker in the Subsurface from Natural Sources of Radiation throughout Repository Development

Subsurface Worker	Construction-Only Period (Years 0–5)	All Other Periods (Years 6–105)
Involved Worker (External Dose)[a]	50 mrem/year	50 mrem/year
Involved Worker (Internal Dose)[b]	70 mrem/year	70 mrem/year
Noninvolved Worker (Internal Dose)[c]	0.066 mrem/Ci radon	0.0011 mrem/Ci radon

[a] Source: DOE, 2008, page D-19
[b] Source: DOE, 2008, page 4-63
[c] Source: DOE, 2008, Table D-5

Table G-7 provides the maximum individual (i.e., workers and public) dose from natural sources of radiation over the three major periods of repository development: (1) construction and emplacement (the initial 55 years), (2) monitoring (40 years following the end of emplacement), and (3) closure (the last 10 years of the 105-year development period). The maximum annual dose to workers and the public present on the surface is estimated by multiplying the maximum radon release in Table G-4 for a given time period by the appropriate dose conversion factor in Table G-5. For example, the offsite individual has a maximum dose of 7.5 mrem per year during the monitoring period based on an annual release of 4,700 Ci of radon and a conversion factor of 0.0016 mrem per curie of radon released. The maximum annual exposure for the subsurface involved worker is based on the summation of the internal exposure (70 mrem per year) and the external exposure (50 mrem per year) in Table G-5. Maximum annual individual exposure for the noninvolved subsurface worker is estimated using the same approach for the surface workers, except Table G-6 provides the dose conversion factors. The offsite maximally exposed individual has a larger dose than the noninvolved surface worker. The DOE considered the maximally exposed offsite individual to be a hypothetical member of the public residing continuously at the site boundary. Alternatively, the onsite worker is present for 2,000 hours per year (i.e., the offsite resident at the site boundary is exposed to radon release for a longer time during the year than the site worker).

Table G-7 Maximum Annual Individual Dose for Workers and the Public from Naturally Occurring Radionuclides during the Development of a Repository

Exposure Group	Annual Individual Dose for Workers and the Public (mrem/yr)			
	Construction Only (Years 0–5)	Construction and Emplacement (Years 6–55)	Monitoring (Years 56–95)	Closure (Years 96–105)
Involved Worker (Subsurface)	120	120	120	120
Involved Worker (Surface)	0.43	3.9	4.7	2.8
Noninvolved Worker (Subsurface)	29	8.5	10	5.2
Noninvolved Worker (Surface)	0.41	3.7	4.6	2.3
Offsite Individual	0.68	6.2	7.5	3.8

Note: Rounded to two significant figures.

Collective population dose over the entire repository development period is also estimated for the worker groups (e.g., involved subsurface workers and noninvolved surface workers) and the public residing within 84 kilometers (52 miles) of the repository. The DOE estimated that approximately 86,000 worker-years would be required to develop the repository, of which approximately 66,000 worker-years are for construction and emplacement, 10,000 worker-years are for the monitoring period, and 10,000 worker-years are for the closure period. Table G-8 breaks down the workforce during the 105 years of development (DOE, 2008, Figure D-2). Table G-9 presents the collective population dose for the worker groups and the public from natural sources of radiation. The collective population dose for the worker groups is estimated by multiplying the annual individual dose for workers in Table G-7 by the appropriate number of worker-years in Table G-8 for the worker groups. The collective population dose for the offsite public is estimated by multiplying the annual average radon release for each time period (Table G-4) by the internal dose conversion for the offsite population dose (Table G-5) and the appropriate number of years for each time period to determine the person-rem for each of the time periods.

Table G-8 Population Size for Workers (Worker-Years) during Development of a Mined Repository

Worker Group	Construction and Emplacement		Monitoring	Closure
	Years 0–5	Years 6–55	Years 56–95	Years 96–105
Surface Workers				
Construction	5,000	5,000	NA	NA
Involved	NA	30,000	5,000	5,000
Noninvolved	2,000	15,000	1,000	2,000
Subsurface Workers				
Construction	300	4,000	NA	NA
Involved	NA	4,000	3,000	3,000
Noninvolved	NA	200	400	800
Overall Totals	7,300	58,200	9,400	10,800
Overall Annual Average[a]	1,500	1,200	230	1,100

[a] Overall annual average values are rounded to two significant figures.
Source: DOE, 2008, Figure D-2

Table G-3 Collective Population Dose for Workers and the Public in Person-Rem from Natural Sources of Radiation during the Development of a Mined Repository

Exposure Group	Construction Only (Years 0–5)	Construction and Emplacement (Years 6–55)	Monitoring (Years 56–95)	Closure (Years 96–105)
Surface Construction Only[a]	2	19 (ends year 32)	NA	NA
Surface (Involved Worker)	NA	120	24	14
Surface (Noninvolved Worker)	1	56	5	5
Subsurface Construction Only[a]	36	480 (ends year 32)	NA	NA
Subsurface (Involved Worker)	NA	480	360	360
Subsurface (Noninvolved Worker)	NA	2	4	4
Worker Totals	39	1,200	390	380
Offsite Population (117,000 within 84 kilometers)	2	6,400	6,200	780

[a] Considered an involved worker for the purpose of estimating dose.
[b] Fractional values rounded to nearest whole number and no more than two significant figures used.

Exposure to Radioactive Waste during Routine Operations[10]

After radioactive waste is received at the repository, potential exposures can occur from the release of radionuclides into the atmosphere resulting in an internal dose and an external dose for individuals conducting activities in close proximity to waste packages. As discussed above for radon, radionuclides released into the atmosphere can impact all workers at the site as well as offsite individuals. External dose, however, would be limited to the involved workers. The DOE considered two major sources for atmospheric releases from SNF: (1) releases that may occur when waste is being handled at a wet handling facility (WHF) and (2) releases from surface contamination on the waste canisters sitting on the surface aging pad or the waste packages that are emplaced underground but before permanent closure of the facility. The DOE estimated potential exposures for a variety of onsite locations associated with the repository facilities to account for dispersion affecting the variation in concentrations of atmospheric release of radioactive materials that a potential onsite worker might experience over the repository site (BSC, 2008a).

Individual Doses from Spent Nuclear Fuel

The DOE estimated potential exposures to atmospheric releases of radioactive material for 30 distinct surface locations of the repository. Exposure to atmospheric releases of SNF are estimated to be very low for workers, with the exception of involved workers at the WHF where uncanistered SNF would be handled. In particular, the DOE estimated that the largest internal annual dose to workers from the combined atmospheric releases (i.e., releases from the WHF, the aging pad, and the sub-surface) would be 15.3 mrem per year, whereas potential exposures at the other 29 locations would vary from 0.0248 to 0.288 mrem per year (BSC, 2008d, Table 5). The releases from the WHF are the largest contributor to worker exposures. The potential dose for the WHF workers from the WHF releases is 13.7 mrem per year (BSC, 2008a, Table 12).

As for the exposures from natural radioactive sources released from the repository to the atmosphere, the DOE considered the maximally exposed offsite individual to be a hypothetical member of the public residing continuously at the site boundary approximately 19 kilometers (12 miles) from the surface facilities of the repository (DOE, 2008). The DOE estimated that the maximally exposed individual would receive a maximum dose of 0.018 mrem per year as a result of SNF releases to the atmosphere from individual doses of (1) 0.003 mrem per year from the WHF release (assuming an annual throughput of 300 MTHM that represents 10 percent of the overall throughput of 3,000 MTHM that is assumed to be handled at the WHF),
(2) 0.012 mrem per year from aging pad releases during each year of operations, and
(3) 0.003 mrem per year from subsurface facility releases that would occur each year until final closure of the facility (DOE, 2008, Table D-5). Table G-10 summarizes the information on internal dose from atmospheric releases.

[10] These estimates deal with commercial spent fuel and do not include risk from defense waste (typically referred to as HLW) that DOE would have also disposed at the Yucca Mountain geologic repository.

Table G-4 Annual Individual Internal Dose for Atmospheric Releases of SNF during Operations

Exposure Group	Annual Dose (mrem/yr)[a]		
	Emplacement (Years 6–55)	Monitoring[b] (Years 56–95)	Closure[b] (Years 96–105)
WHF Workers	15	NA	NA
All Workers (Except WHF Workers)	0.025 to 0.29	0.011 to 0.040[c]	0.011 to 0.040[c]
Offsite Individual (Maximally Exposed)	0.018	0.003	0.003

[a] Values rounded to two significant figures.
[b] Subsurface releases only during these periods.
[c] Values based on BSC, 2008a, Table 14.

Certain involved workers performing activities sufficiently close to SNF (e.g., handling activities) can receive a measurable external dose. The DOE assumed an overall annual throughput of 500 casks (3,000 MTHM) per year with design-basis source terms to estimate the external dose to workers. However, the WHF only processes 10 percent of the overall throughput or 300 MTHM. These annual individual external doses for surface workers ranged from 200 to 1,300 mrem per year for operators and 200 to 800 mrem per year for health physics technicians (BSC, 2008b, Table 1.0). The average external dose to these involved surface workers was 480 mrem per year for operators and 358 mrem per year for health physics technicians (BSC, 2008b, Table 1). Individual external exposures for the subsurface involved workers ranged from 100 to 209 mrem per year during emplacement operations, 120 to 204 mrem per year during the monitoring period, and 8.74 to 39.4 mrem per year during closure operations (BSC, 2008c, Tables 16-18). Waste packages are emplaced in the subsurface using a transport and emplacement vehicle that is remotely controlled and monitored; thus, the subsurface workers receive lower external doses than the surface workers. Table G-11 provides the maximum and collective doses for specific worker activities.

Table G-11 External Dose for Involved Workers for Specific Activities

Activity[a,b]	Maximum Individual Dose (Rem Per Year)	Collective Dose (Person-Rem)
Receipt Facility	1.3	840
Initial Handling Facility	0.8	110
Wet Handling Facility	0.4	300
Canister Receipt and Closure Facilities	0.29	630
Aging Facility	0.30	200
Low-Level Waste Facility	0.7	310
Cask Receipt Security Station	0.4	230
Total Surface Worker External Dose	1.3	2,620
Subsurface (Operations)	0.21	510
Subsurface (Monitoring)	0.2	510
Subsurface (Closure)	0.039	80
Total Subsurface Worker External Dose	0.21	1,100

[a] Annual doses based on processing 500 casks per year or about 3,000 MTHM of commercial SNF throughput per year.
[b] Collective doses based on processing a total waste throughput of 70,000 MTHM.
Source: DOE, 2008, Tables D-9 and D-10

Table G-12 provides the estimated individual doses for site workers and an offsite maximally exposed individual over the operational period of repository development (i.e., initial construction until permanent closure of the underground facility) based on the values contained in Tables G-10 and G-11. The highest exposure is estimated for the involved worker and specifically for the workers at the receipt facility (RF) based on an external exposure from the incoming waste containers. By comparison, the annual individual dose for all worker categories from the atmospheric release of SNF is less than 1 mrem per year, except for the involved workers at the WHF who are estimated to have an individual internal dose of 15 mrem per year.

Table G-12 Largest Annual Individual Dose for Workers and the Maximally Exposed Offsite Individual from SNF during the Operational Period

Exposure Group	Exposure Source	Largest Annual Individual Exposures (mrem/yr)			
		Construction Only (Years 0–5)	Construction and Emplacement (Years 6–55)	Monitoring (Years 56–95)	Closure (Years 96–105)
Involved Worker (Subsurface)	Internal Dose	NA[a]	<1	<1	<1
	External Dose	NA[a]	210	200	390
Involved Worker (Surface)	Internal Dose	NA[a]	15 (WHF)	<1	<1
	External Dose	NA[a]	1,300 (RF)	<1	<1
Noninvolved Worker (Subsurface)	Internal Dose	NA[a]	<1	<1	<1
Noninvolved Worker (Surface)	Internal Dose	NA[a]	<1	<1	<1
Offsite Individual	Internal Dose	NA[a]	<1	<1	<1

[a] No applicable exposure because no SNF is at the site during this period.

Collective Dose from Spent Nuclear Fuel

The estimation of collective dose for worker groups should account for the variation in exposures from atmospheric releases that occur as a result of the different locations of the workers on the surface of the geological repository operations area, different periods of time for these releases to occur (e.g., only subsurface releases occur during the monitoring and the closure periods), and differing external exposures for workers because of their different work activities. Internal dose from atmospheric releases is the largest for the WHF workers. Therefore, the estimation of worker collective dose is simplified by estimating the WHF worker dose as one component of the overall collective internal dose. The second component of the overall collective internal dose represents all of the other workers and conservatively uses the largest annual internal dose estimated for the other geological repository operations area locations (e.g., the largest value for internal doses for all other workers in Table G-10). The following equation shows how to calculate the collective worker dose for these two components for internal exposure of workers from atmospheric releases of SNF:

$$CWD = AWD \times WY$$

where CWD: collective worker dose
AWD: annual worker dose (from Table G-10)
WY: number of worker-years

The number of worker-years is based on Table G-8 for all workers other than WHF workers. The number of WHF worker-years is estimated based on the assumption that 10 percent of the SNF received at the repository is processed by the WHF, and the activity requires 36 workers to process 300 MTHM in a year (BSC, 2008b). Therefore, based on the statutory limit of 63,000 MTHM of commercial fuel, the number of WHF worker-years is 756.

The DOE estimated the collective doses from atmospheric releases from SNF operations at the repository for 117,000 individuals residing within 84 kilometers (52 miles) of the facility to be (1) 0.88 person-rem from WHF releases as a result of processing 6,300 MTHM, (2) 5.5 person-rem from the aging pad releases based on an annual collective dose of 0.11 person-rem per year for each of the 50 years during the emplacement period, and (3) 3.3 person-rem from subsurface facility releases based on an annual collective dose of 0.033 person-rem per year for each of the 100 years the subsurface facility will be ventilated (DOE, 2008, Table D-5).

The DOE estimated the collective external doses for workers for the activities presented in Table G-11. Table G-13 provides the collective dose for workers and the offsite public for both atmospheric release and direct radiation associated with SNF during the operational period of a repository.

Table G-13 Collective Internal Dose for Workers and the Public from SNF during Repository Development

Exposure Group	Exposure Source	Collective Dose for the Workers and the Public (person-rem)				
		Construction Only (Years 0–5)	Construction and Emplacement (Years 6–55)	Monitoring (Years 56–95)	Closure (Years 96–105)	Totals
Surface Construction	Internal Dose	NA	1	NA	NA	1
Surface (Involved Worker)	Internal Dose	NA	20[a]	<1	<1	20
	External Dose	NA	2,620	NA	NA	2,620
Surface (Noninvolved Worker)	Internal Dose	NA	4	<1	<1	4
Subsurface Construction	Internal Dose	NA	1	NA	NA	1
Subsurface (Involved Worker)	Internal Dose	NA	1	<1	<1	1
	External Dose	NA	510	510	80	1,100
Subsurface (Noninvolved Worker)	Internal Dose	NA	<1	<1	<1	<1
Total Worker Doses	Internal Dose	NA	27	<1	<1	27
	External Dose	NA	3,183	510	80	3,773
Offsite Population (117,000 within 84 kilometers)	Internal Dose	0	8	1	<1	9

[a] The WHF workers contribute 60 percent of the total collective dose and represent less than 3 percent of the total worker-years.

Exposure to Radioactive Waste from Operational Accidents

The DOE analyzed ten postulated accident scenarios associated with SNF that could occur during the operational period of the repository[11]. The evaluations considered accidents caused by seismic events as well as internal events such as the breach of a sealed TAD canister in the WHF. The evaluations conservatively assumed the wind would blow in the direction of the largest population sector (i.e., the south-southeast sector) with a population of 104,000 within 80 kilometers (50 miles) of the facility. The DOE described this assumption as the unfavorable

[11] Only exposure to radioactive waste from operational accidents at mined repositories is discussed in this section. However, the calculation methods may be applicable for alternative disposal methods.

95th percentile meteorological conditions for the offsite public exposure assessment. Table G14 estimates the conditional consequences for these accident scenarios.

Table G-14 Conditional Doses for Accident Scenarios Involving SNF Releases during Repository Handling and Emplacement Activities

Accident Scenario	Annual Frequency	Individual Maximum Offsite Dose (Mrem)	Offsite Collective Dose (Person-Rem)	Maximum Noninvolved Worker Dose (Mrem)
Seismic event causing LLW facility collapse and failure of HEPA filters and ductwork in other facilities	2×10^{-4}	35	310	3,500
Breach of uncanistered SNF in sealed truck transport cask	2×10^{-3}	1	2.7×10^{-5}	83
Breach of uncanistered SNF in unsealed truck cask in WHF pool	1×10^{-5}	0.94	26	52
Breach of sealed DPC in air	2×10^{-6}	9.1	250	55
Breach of SNF in unsealed DPC in WHF pool	4×10^{-6}	8.4	230	740
Breach of SNF in sealed TAD in WHF pool	4×10^{-5}	5.3	140	430
Breach of SNF in unsealed TAD in WHF pool	1×10^{-5}	4.9	130	290
Drop of uncanistered SNF in WHF pool	6×10^{-3}	0.47	13	27
Breach of uncanistered SNF in WHF pool	$< 2\times10^{-6}$	0.23	6.4	14
Breach of sealed truck transport cask due to fire	4×10^{-4}	4.4	42	1,300

Source: DOE, 2008, Table 4-25

Table G-14 accounts for the risk that the accident scenarios could occur during repository development. The accident risk for the operational period is estimated by multiplying 50 years (the period during which handling and emplacement of SNF occurs) by the annual frequency for the specific scenario and the doses presented in Table G-14. Table G-15 presents the annual frequency for each of the accident scenarios and the estimated risk over the 50 year period when handling and waste emplacement activities are expected to occur.

Table G-15 Risk Associated with Accident Scenarios Involving SNF Releases during Repository Handling and Emplacement Activities

Accident Scenario	Annual Frequency	Individual Maximum Offsite Risk (Mrem/Year)	Offsite Collective Risk (Person-Rem/Year)	Maximum Noninvolved Worker Risk (Mrem/Year)
Seismic event causing LLW facility collapse and failure of HEPA filters and ductwork in other facilities	2×10^{-4}	0.35	3.1	35
Breach of uncanistered SNF in sealed truck transport cask	2×10^{-3}	0.1	2.7×10^{-6}	8.3
Breach of uncanistered SNF in unsealed truck cask in WHF pool	1×10^{-5}	4.7×10^{-4}	0.01	0.03
Breach of sealed DPC in air	2×10^{-6}	9×10^{-4}	0.02	0.006
Breach of SNF in unsealed DPC in WHF pool	4×10^{-6}	0.002	0.05	0.1
Breach of SNF in sealed TAD in WHF pool	4×10^{-5}	0.001	0.03	0.09
Breach of SNF in unsealed TAD in WHF pool	1×10^{-5}	0.002	0.06	0.1
Drop of uncanistered SNF in WHF pool	6×10^{-3}	0.1	3.9	8.1
Breach of uncanistered SNF in WHF pool	$<2\times10^{-6}$	$<2\times10^{-5}$	$<6\times10^{-4}$	$<1\times10^{-3}$
Breach of sealed truck transport cask from fire	4×10^{-4}	0.09	0.8	26

Source: DOE, 2008, Table 4-25

The risks for handling and emplacement activities presented in Table G-15 are low because of the limited conditional doses for the accident scenarios (Table G-14) and the low annual accident frequencies (e.g., all accident frequencies are 6×10^{-3} per year or less). Thus, the maximum offsite individual risks are all less than 1 mrem per year, and the offsite collective risks are under 1 person-rem per year except for two accident scenarios in which the collective risks are 3.1 and 3.9 person-rem per year. The maximum noninvolved worker risk is larger than the offsite individual dose estimate because the worker is present at the site and thus closer to any accidental release. However, the largest risk for the noninvolved worker is 35 mrem per year, and most of the accident scenarios result in a maximum noninvolved worker risk less than 1 mrem per year.

G.2.7.3 REPOSITORY RISKS AFTER PERMANENT CLOSURE

When a repository is permanently closed, all openings into the underground facility are sealed to eliminate any type of easy access to the waste. This begins the post-closure period of a repository, and the NRC expects that the multiple barriers of the repository will continue to perform their passive functions for maintaining safety (i.e., the repository site and design are intended to preserve safety without maintenance). Although a release in the distant future could

occur, it is expected that the repository will continue to function safely and any releases will not exceed the safety limits (i.e., within the post-closure dose limits). Repository programs around the world have regulatory dose limits for annual individual exposures on the order of 15 mrem for long time periods (e.g., 10,000 years). Collective doses are rarely estimated for post-closure safety because such an estimate is of limited value given the large uncertainties in projecting populations far into the future.

G.2.7.4 SUMMARY OF RISKS OF A GEOLOGICAL REPOSITORY

Table G-16 presents the collective dose for all the activities that would occur at each stage of the pre-closure period. In particular, exposures during loading come from Table G-2, exposures during transport come from Table G-3, exposures from natural occurring radionuclides during repository development come from Table G-9, and exposures from the presence of SNF during repository development come from Table G-13.

The transportation campaign results in the largest overall collective dose for workers and the public from exposure to SNF. For workers, the collective dose is primarily from the potential external dose during the loading of uncanistered fuel; whereas, public exposure is principally from the external dose along the transportation routes. The potential exposures are well within regulatory limits. Worker exposures to SNF during repository development are approximately double the worker exposure to natural radioactivity (e.g., radon release). Worker exposure to SNF during repository development is dominated by the external dose for those workers performing activities in close proximity to the SNF (see Table G-13). Public exposure to SNF during repository development is very limited (i.e., 0.09 person-rem/year); however, public exposure to radon released to the atmosphere during repository development is significantly larger (i.e., an annual collective dose of 127 person-rem, which is approximately 1,000 times greater than the dose from the SNF).

Table G-16 Collective Dose for Workers and the Public from the Transportation Campaign and Repository Development

Time Period and Activities	Exposure Source	Workers (Person-Rem)	Public (Person-Rem)
TRANSPORTATION CAMPAIGN			
Years 6–55 (Loading)	SNF	8,300	3
Years 6–55 (Transportation)	SNF	544	548
Transportation Campaign Totals	NA	8,844	551
Transportation Annual Average (based on 50 years)	NA	177	11
REPOSITORY DEVELOPMENT			
Years 0–5 (Construction only)	Natural	39	2
Years 6–55 (Construction and emplacement)	Natural	1,200	6,400
	SNF	3,183	8
Years 56–95 (Monitoring)	Natural	390	6,200
	SNF	510	1
Years 96–105 (Closure)	Natural	380	780
	SNF	80	<1
Repository Totals	Natural	2,009	13,382
	SNF	3,773	9
Repository Annual Average (based on 105 years)	Natural	19	127
	SNF	38	0.09

G.2.8 DECOMMISSIONING

The NRC's nuclear regulatory activities include decommissioning nuclear facilities, which means reducing residual radioactivity to a level that permits either of the following actions:

- Release the property for unrestricted use and terminate the license.
- Release the property under restricted conditions and terminate the license.

The NRC's decommissioning regulations are found in 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination," which applies to facilities licensed under 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," 10 CFR Part 40, 10 CFR Part 50, 10 CFR Part 52, 10 CFR Part 60, 10 CFR Part 61, 10 CFR Part 63, 10 CFR Part 70, and 10 CFR Part 72, and also provides the main requirements for license termination.

The NRC and Agreement States regulate the decommissioning of nuclear facilities, with the ultimate goal of license termination. The following facilities would be expected to undergo decommissioning:

- complex materials sites (byproduct sites)
- power reactors
- research and test reactors
- uranium recovery sites
- fuel cycle facilities

Approximately 100 materials licenses are terminated each year. Most of these license terminations are routine, and while some sites may require more extensive activities, most of the affected sites require little, if any, remediation to meet the NRC's criteria for unrestricted use. A site meeting the criteria for license termination under restricted conditions has residual radiological contamination above the levels for unrestricted release.

As with LLW disposal sites, the primary hazard at decommissioning sites would likely be a radiological release, which could occur from the failure of institutional or physical barriers.

The primary decommissioning guidance document used by licensees and the NRC is NUREG-1757, "Consolidated Decommissioning Guidance." NUREG-1757 is a three-volume series that consolidates the current policies and guidance of the NRC's decommissioning program. Volume 1 of NUREG-1757 provides guidance on the decommissioning process for materials licensees, with applicability in some areas to reactor licensees. Volume 2 contains guidance on characterization, surveys, and the determination of radiological criteria for all licensees subject to the license termination regulations of 10 CFR Part 20, Subpart E. Volume 3 addresses financial assurance, recordkeeping, and timeliness. The staff revises NUREG-1757 periodically to reflect updates to the NRC's decommissioning policy.

The NRC also provides decommissioning guidance related to non-power reactors in NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content." In addition, Section 5 of NUREG-1620 provides decommissioning guidance for uranium recovery (i.e., in situ leaching) facilities subject to 10 CFR Part 40, Appendix A.

Other studies and guidance that contain additional information on performing regulatory analyses for specific decommissioning activities include the following documents.

- NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," issued in 1988; and Supplement 1, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities: Regarding the Decommissioning of Nuclear Power Reactors," issued in 2002, address decommissioning generically. Supplement 1 incorporated technological advances in decommissioning operations, experience gained by licensees, and changes made to NRC regulations since the initial publication of NUREG-0586 in 1988.

- NUREG-1738 addresses the principal safety concern for the decommissioning of the current fleet of operating reactors, which is the storage of SNF in the SFP or at an ISFSI.

G.3 NON-FUEL CYCLE ACTIVITIES

Non-fuel cycle activities involve the use of byproduct material. The primary NRC regulations for these activities are those promulgated under 10 CFR Part 30 through 10 CFR Part 39.[12] Non-fuel cycle activities do not have backfitting regulations. Non-fuel cycle activities are diverse in scope and use, with more than 18,500 specific materials licenses and approximately 100,000 general licenses in the following general areas:

- medical use (e.g., radiation therapy and nuclear medicine)
- irradiators
- radiography
- well logging
- manufacturing
- fixed and portable gauges
- measuring systems
- academics (e.g., for education and research)

NUREG-1350 provides background information on non-power reactor activities, including the number of active licensees and general locations. The technical report series, NUREG-1556, "Consolidated Guidance about Materials Licenses," contains comprehensive reference information about the various aspects of materials licensing and materials program implementation for the non-fuel cycle activities. The Materials OpE Gateway consolidates various information sources for ease in accessing and analyzing operating experience in the regulated materials program.

The NRC regulates a wide variety of activities with diverse characteristics under 10 CFR Part 30 through 10 CFR Part 39. The materials may be solids, liquids, or gases and may be sealed or unsealed sources. Quantities in use may range from microcuries to megacuries, and access to the byproduct materials may be unlimited (e.g., for consumer products) to tightly controlled (e.g., for large irradiators). All these factors affect risk and associated impact and implementation costs.

This section summarizes the data reference materials related to byproduct materials that may be useful in preparing a regulatory analysis.

NUREG-1556 is an extensive, 21-volume document that provides program-specific guidance to assist in preparing and reviewing applications for licenses for the use of byproduct material. The program-specific guidance is intended for use by applicants, licensees, NRC staff, and Agreement States.

---

[12] These regulations are 10 CFR Part 31, "General Domestic Licenses for Byproduct Material", 10 CFR Part 32, "Specific Domestic Licenses to Manufacture or Transfer Certain Items Containing Byproduct Material", 10 CFR Part 33, "Specific Domestic Licenses of Broad Scope for Byproduct Material", 10 CFR Part 34 , "Licenses for Industrial Radiography and Radiation Safety Requirements for Industrial Radiographic Operations", 10 CFR Part 35, "Medical Use of Byproduct Material", 10 CFR Part 36, "Licenses and Radiation Safety Requirements for Irradiators", 10 CFR Part 37, "Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material," and 10 CFR Part 39, "Licenses and Radiation Safety Requirements for Well Logging." There is no 10 CFR Part 38.

NUREG/CR-6642, "Risk Analysis and Evaluation of Regulatory Options for Nuclear Byproduct Material Systems," includes activities regulated under 10 CFR Part 30 through 10 CFR Part 39.[13] The three-volume report uses insights obtained from risk analyses to identify regulatory options for the oversight of the various nuclear byproduct materials licenses. The methodology includes the following:

- organizing all nuclear byproduct materials licenses into 40 systems (i.e., groups of activities)
- describing each system in terms of tasks, hazards, barriers (i.e., physical and administrative barriers that limit doses to workers and the public), and receptors
- performing a radiation risk assessment for each system, including normal and accident doses to workers and a maximally exposed member of the public
- considering the social and economic benefits, associated costs, and the risks, for each system

NUREG/CR-6642 includes the initial information that is useful to perform a regulatory analysis for byproduct material. The primary risk associated with the studied activities is overexposure to workers and the public from the failure of one or more protective barrier(s) or lost or misplaced sources. The analyst should consider the assumptions used in the analysis to assess whether they impact the regulatory analysis. In particular, the analysis for medical systems does not consider patient doses because the individual receives a benefit from the use of the system; therefore, the NRC considers such doses a special category separate from those to workers and the public. In addition, NUREG/CR-6642 only addresses the maximally exposed member of the public and does not calculate population doses.

The analyst can find occupational exposure data in NUREG-0713, which summarizes the annual occupational exposure data that are maintained in the NRC Radiation Exposure Information and Reporting System (REIRS). The annual reports compile information from five of the seven NRC licensee categories subject to 10 CFR 20.2206, "Reports of Individual Monitoring":

(1) commercial nuclear power reactors and test reactor facilities

(2) industrial radiography

(3) fuel processing, including uranium enrichment facilities; fabricating; or reprocessing of special nuclear material above specified amounts

(4) processing or manufacturing for distributing byproduct material above specified amounts

(5) ISFSIs

Two licensee categories, facilities for land disposal of LLW and geologic repositories for HLW, do not report as there are no NRC-licensed LLW disposal facilities and there are no geologic repositories.

---

[13] In 10 CFR Part 37, the NRC addresses the level of security for nuclear byproduct material system activities, which is not covered in the report. There is no 10 CFR Part 38.

NMED is another source of information on non-fuel cycle activity events. The NMED Annual Report presents information on trending and analysis of events reported to the NRC that involve radioactive materials. NMED contains information from materials, fuel cycle, and non-power reactor licensees on events such as personnel radiation overexposures, medical misadministration, losses of radioactive material, and potential criticality events.

NUREG-1717, "Systematic Radiological Assessment of Exemptions for Source and Byproduct Materials," systematically assesses potential individual and collective (population) radiation doses associated with the current exemptions from licensing for the majority of 10 CFR Part 30, 10 CFR Part 40, and 10 CFR Part 70 licenses. The report estimates doses for the normal life cycle of a particular product or material, covering distribution and transport, intended or expected routine use, and disposal. In addition, it estimates assessments potential doses from accidents and misuse. Finally, it assesses potential radiological impacts associated with selected products containing byproduct material that currently may be used under a general or specific license and may be candidates for exemption from licensing requirements.

The regulatory analysis for the 2018 10 CFR Part 35 final rule, "Medical Use of Byproduct Material—Medical Event Definitions, Training, and Experience, and Clarifying Amendments" is a recent example of an analysis for this group of non-power reactor and non-fuel cycle activities.

G.4 COMMON ACTIVITIES

Activities common to both groups of non-power reactor activities include the following:

- transportation (10 CFR Part 71)
- security (10 CFR Part 73)
- material control and accounting (10 CFR Part 74)
- emergency planning and preparedness (10 CFR 40.31, “Application for Specific Licenses,” 10 CFR 70.22, “Contents of Applications,” and 10 CFR 76.91, “Emergency Planning”)

The following sections provide additional data references that should be used in concert with the specific non-power reactor activities being evaluated.

G.4.1 TRANSPORTATION

About 3 million packages of radioactive materials are shipped each year in the United States, either by highway, rail, air, or water. Regulating the safety of these shipments is the joint responsibility of the NRC and the U.S. Department of Transportation, as established by a Memorandum of Understanding (44 FR 38690) between the two agencies.

To apply for a Certificate of Compliance for a package design for the transportation of radioactive material, a vendor submits an application to the NRC for review and approval in accordance with 10 CFR Part 71. The application addresses the safety and operational characteristics of the package, including design analysis for structural, thermal, radiation shielding, nuclear criticality, and material containment. In addition, the application contains operational guidance, such as any testing and maintenance requirements, operating procedures, and conditions for package use. For the NRC to certify a transportation package design, actual tests or computer analyses must demonstrate that, after the tests for normal conditions of transport and hypothetical accident conditions, the package will meet the appropriate containment, dose rates, and criticality safety criteria in 10 CFR Part 71. The tests for hypothetical accident conditions are performed in sequence to determine their cumulative effects on the package. If the package design meets NRC requirements, the NRC issues a Radioactive Material Package Certificate of Compliance to the vendor.

NRC licensees are authorized to ship radioactive material in an approved package under the general license provisions of 10 CFR Part 71; Agreement State licensees ship radioactive materials under Department of Transportation regulations. Before any shipment can occur, the shipper must review the package Certificate of Compliance to determine whether any testing or maintenance is required. The shipper may be required to check or change package seals and other components or perform leak testing. In addition, the shipper must take radiation measurements at specific locations on and around the package to make sure that the levels are below the required limits. The shipper must also meet the U.S. Department of Transportation requirements for shipment of the nuclear material, including route selection, vehicle condition and placarding, driver training, package marking, labeling, and other shipping documentation.

Certain specific requirements apply to shippers of SNF, including the following:

- A licensee must use NRC-approved routes for the transport of SNF.
- The licensee must make sure that SNF is protected against radiological sabotage. Shippers that transport or deliver SNF to a carrier for transport must meet specific requirements that include the following:
    - notifying the NRC of the shipment
    - having procedures for addressing emergencies
    - having a communications center
    - having a written log of shipment events
    - making arrangements with local law enforcement agencies for shipments while en route
    - using armed escorts

Regulations governing nuclear materials transportation can be found at:

- 10 CFR Part 37, Subpart D, “Physical Protection in Transit”
- 10 CFR Part 71
- 10 CFR Part 73

The primary radiological hazards associated with transportation are the loss of containment of the hazardous material being transported, failure of the shielding to perform its function, or, for certain materials, inadvertent criticality.

The following reference materials may be useful in preparing regulatory analyses for regulatory actions affecting non-power reactor activities:

- NUREG-1609, “Standard Review Plan for Transportation Packages for Radioactive Material”
- NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel”
- NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety”
- NUREG-2125, “Spent Fuel Transportation Risk Assessment”
- NUREG/BR-0292, Revision 2, “Safety of Spent Fuel Transportation”
- NUREG-0561, Revision 2, “Physical Protection of Shipments of Irradiated Reactor Fuel”

- NUREG-2155, Revision 1, “Implementation Guidance for 10 CFR Part 37, Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material” (Subpart D)

- NUREG-0170, “Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes”

- NUREG/CR-4829, Volumes 1 and 2, “Shipping Container Response to Severe Highway and Railroad Accident Conditions”

- NUREG/CR-6672, Volume 1, “Reexamination of Spent Fuel Shipment Risk Estimates”

- Regulatory Guide 7.9, Revision 2, “Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Materials”

G.4.2 SECURITY

The NRC and Agreement States regulate the use of radioactive material in order to protect people and the environment. Materials licensees have the primary responsibility to maintain the security and accountability of the radioactive material in their possession. The NRC works with its Federal and State partners and the international community, to provide appropriate safety and security requirements for radioactive materials without discouraging their beneficial use.

In 10 CFR Part 73, the NRC prescribes requirements for the establishment and maintenance of a physical protection system that has capabilities for the protection of special nuclear material at fixed sites and in transit. Design-basis threats referenced in these regulations are used to design safeguards systems to protect against acts of radiological sabotage and to prevent the theft or diversion of special nuclear material. The provisions of 10 CFR Part 73 apply to the following:

- power reactor licensees
- research and test reactor licensees
- decommissioning reactor licensees
- independent spent fuel storage installation licensees
- fuel cycle licensees

Radioactive byproduct material provides critical capabilities in the oil and gas, electrical power, construction, and food industries. It is used to treat millions of patients each year in diagnostic and therapeutic medical procedures and is used in technology research and development. 10 CFR Part 37 contains security requirements that apply to certain quantities of radioactive byproduct material at any type of facility, including the following types of facilities:

- industrial licensees
- academic and research licensees
- medical licensees

Security for Category 1 and 2 Quantities of Radioactive Material

On March 19, 2013, the NRC published a final rule, “Physical Protection of Byproduct Material.” This rule established security requirements in 10 CFR Part 37 for the use and transport of aggregated quantities of Category 1 and Category 2 quantities of radioactive materials, as well

as for shipments of small amounts of irradiated reactor fuel. Category 1 and Category 2 quantities of radioactive materials are thresholds established by the International Atomic Energy Agency in its Code of Conduct on the Safety and Security of Radioactive Sources. The objective of this rule is to provide reasonable assurance of preventing the theft or diversion of Category 1 and Category 2 quantities of radioactive materials. The final rule incorporates lessons learned by the NRC and the Agreement States in implementing the post-September 11, 2001 security measures, as well as stakeholder input on the proposed rule. Additionally, the rule updated, clarified, and strengthened the existing regulatory requirements and thereby promotes public health and safety.

Cyber Security

High-profile cyber attacks underscore the importance of continuing to evaluate the need for a cyber security regulatory framework for all classes of NRC licensees. The NRC has gained in-depth experience with cyber security as a result of the development, implementation, and inspections performed under 10 CFR 73.54, "Protection of Digital Computer and Communication Systems and Networks." The NRC's oversight of cyber security implementation at operating reactors has positioned the agency to develop, as needed, cyber security regulations or other measures for various types of NRC licensees. An overview of the NRC's approach to cyber security for the various categories of NRC-regulated facilities is provided in SECY-17-0034, "Update to the U.S. Nuclear Regulatory Commission Cyber Security Roadmap" (NRC, 2017c).

G.4.3 MATERIAL CONTROL AND ACCOUNTING

The NRC provides the principal requirements for special nuclear material licensing in 10 CFR Part 70 and 10 CFR Part 74. In 10 CFR 70.22(b), the NRC specifies that a license application must contain a full description of the applicant's program for the control and accounting of such special nuclear material to show how compliance with the graded material control and accounting requirements of 10 CFR Part 74, Subparts B–E, will be accomplished. In 1987, the NRC revised the material control and accounting requirements for NRC licensees authorized to possess and use a formula quantity (i.e., 5 formula kilograms or more) of strategic special nuclear material (NRC, 1987a). Those revisions, issued as 10 CFR Part 74, Subpart E, "Formula Quantities of Strategic Special Nuclear Material," require timely monitoring of in-process inventory and discrete items to detect anomalies potentially indicative of material losses. Timely detection and enhanced loss localization capabilities are beneficial to resolve alarms and recover material in the event of an actual loss.

The following are useful data references for performing the regulatory analyses in this area:

- NUREG-1280, Revision 2, "Acceptable Standard Format and Content for Material Control and Accounting Plan Required for Strategic Special Nuclear Material," was first published in 1987 to present criteria that could be used by applicants, licensees, and NRC license reviewers in the initial preparation and subsequent review of fundamental nuclear material control plans submitted in response to the Reform Amendment. The report addressed general performance objectives, system capabilities, process monitoring, item monitoring, alarm resolution, quality assurance, and accounting.

- NUREG-2159, "Acceptable Standard Format and Content for the Material Control and Accounting Plan Required for Special Nuclear Material of Moderate Strategic Significance," describes the standard format and content suggested by the NRC for use

in preparing material control and accounting plans for facilities authorized to hold special nuclear material of moderate strategic significance.

- NUREG-1065, Revision 3, "Acceptable Standard Format and Content for the Material Control and Accounting Plan Required for Special Nuclear Material of Low Strategic Significance," contains information that the licensee or applicant should provide in its fundamental nuclear material control plan to implement the requirements of 10 CFR 74.31, "Nuclear Material Control and Accounting for Special Nuclear Material of Low Strategic Significance."

- NUREG/CR-5734, "Recommendations to the NRC on Acceptable Standard Format and Content for the Fundamental Nuclear Material Control (FNMC) Plan Required for Low-Enriched Uranium Enrichment Facilities," recommends information that the licensee or applicant should provide in the FNMC Plan to implement the requirements of 10 CFR 74.33, "Nuclear Material Control and Accounting for Uranium Enrichment Facilities Authorized to Produce Special Nuclear Material of Low Strategic Significance." This document also describes methods that should be acceptable for compliance with the general performance objectives.

- Regulatory Guide 5.29, Revision 2, "Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants," describes acceptable methods and procedures for the implementation and maintenance of a special nuclear material control and accounting system for non-fuel cycle facilities, including nuclear power reactors, research and test reactors, and ISFSIs.

- In 10 CFR Part 75, the NRC implements the requirements established by the safeguards agreements between the United States and the International Atomic Energy Agency. This regulation contains requirements to ensure that the United States meets its nuclear non-proliferation obligations under the safeguards agreements. These obligations include providing information to the International Atomic Energy Agency on the physical location of applicant, licensee, or certificate holder activities; information on sources and special nuclear materials; and access to the physical location of applicant, licensee, or certificate holder activities.

G.4.4 EMERGENCY PLANNING AND PREPAREDNESS

The objective of the emergency planning program is to ensure that fuel facility licensees, non-power reactor licensees, and some materials licensees are capable of implementing adequate measures to protect public health and safety in the event of a radiological emergency. As a condition of their licenses, these licensees must develop and maintain emergency plans that meet comprehensive NRC emergency planning requirements. These licensees are responsible for preventing accidents. Should an accident occur, local public safety authorities, such as fire and police departments, will act to protect the public.

After a large, toxic release of UF_6 at the Sequoyah Fuels Corporation conversion facility in 1986, the NRC decided emergency plans for fuel facilities should also account for hazardous chemical releases. At uranium conversion, enrichment and fuel fabrication facilities, the most significant accidents would be a UF_6 release, fire, or criticality (e.g., an unintended, self-sustaining nuclear chain reaction). There is likely to be little or no warning time before these accidents start. However, most can be controlled within roughly half an hour.

Regulatory Guide 3.67, Revision 1, “Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities,” contains detailed guidance on emergency planning. In general, the scope and depth of fuel cycle facility plans are more variable than and not as extensive as those of power reactors, reflecting the diverse nature of these facilities and the hazards and risks associated with their operation. For example, fuel cycle facility emergency plans have the following:

- no designated emergency planning zones
- no extraordinary provisions to alert and notify the general public
- only two levels of emergency classifications
  - Alert—requiring no offsite response
  - Site Area Emergency—could require offsite response

The Federal Emergency Management Agency has no oversight over State and local governments with regard to fuel cycle facilities. This reduced scope and depth are justified because the EPA protective action guidelines will not be exceeded beyond the site boundary.

Regulatory Guide 3.67 may be useful in preparing regulatory analyses for regulatory actions affecting emergency preparedness for certain fuel cycle, non-power reactor, and other radioactive materials licensees.

G.5 REFERENCES

42 U.S.C. 2011 et seq., Atomic Energy Act of 1954, as amended. Available at https://www.govinfo.gov/content/pkg/USCODE-2010-title42/html/USCODE-2010-title42-chap23-divsnA.htm.

42 U.S.C. 2021b, Low-Level Radioactive Waste Policy Amendments Act of 1985. Available at https://www.govinfo.gov/content/pkg/STATUTE-99/pdf/STATUTE-99-Pg1842.pdf.

42 U.S.C. 7901 et seq., Uranium Mill Tailings Radiation Control Act of 1978. Available at https://www.govinfo.gov/app/details/USCODE-2019-title42/USCODE-2019-title42-chap88-sec7901/context.

Bechtel SAIC Company (BSC), "GROA Airborne Release Dose Calculation 000-PSA-MGR0-01200-000-00C, March 26, 2008a. Agencywide Documents Access and Management System (ADAMS) Accession No. ML090770783.

BSC, "Yucca Mountain Project: Repository ALARA Goal Compliance, Informal Study," 000-30R-MGR0-04000-000, February 5, 2008b. ADAMS Accession No. ML092790253.

BSC, "Department of Energy Safety Analysis Report: Subsurface Worker Dose Assessment," 800-00C-SS00-00600-000-00B, March 26, 2008c. ADAMS Accession No. ML092790265.

BSC, "Design Calculation: GROA Worker Dose Calculation," 000-PSA-MGR0-01400-000-00C, March 27, 2008d. ADAMS Accession No. ML090770784.

Code of Federal Regulations (CFR), Title 10, *Energy*, Part 20, "Standards for Protection against Radiation." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part020/.

CFR, Title 10, *Energy*, Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part030/.

CFR, Title 10, *Energy*, Part 31, "General Domestic Licenses for Byproduct Material." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part031/.

CFR, Title 10, *Energy*, Part 32, "Specific Domestic Licenses to Manufacture or Transfer Certain Items Containing Byproduct Material." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part032/.

CFR, Title 10, *Energy*, Part 33, "Specific Domestic Licenses of Broad Scope for Byproduct Material." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part033/.

CFR, Title 10, *Energy*, Part 34, "Licenses for Industrial Radiography and Radiation Safety Requirements for Industrial Radiographic Operations." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part034/.

CFR, Title 10, *Energy*, Part 35, "Medical Use of Byproduct Material." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part035/.

CFR, Title 10, *Energy*, Part 36, "Licenses and Radiation Safety Requirements for Irradiators." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part036/.

CFR, Title 10, *Energy*, Part 37, "Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part037/.

CFR, Title 10, *Energy*, Part 39, "Licenses and Radiation Safety Requirements for Well Logging." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part039/.

CFR, Title 10, *Energy*, Part 40, "Domestic Licensing of Source Material." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part040/.

CFR, Title 10, *Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities." Available at http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/.

CFR, Title 10, *Energy*, Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part051/.

CFR, Title 10, *Energy*, Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part052/.

CFR, Title 10, *Energy*, Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part060/.

CFR, Title 10, *Energy*, Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part061/.

CFR, Title 10, *Energy*, Part 63, "Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part063/.

CFR, Title 10, *Energy*, Part 70, "Domestic Licensing of Special Nuclear Material." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part070/.

CFR, Title 10, *Energy*, Part 71, "Packaging and Transportation of Radioactive Material." https://www.nrc.gov/reading-rm/doc-collections/cfr/part071/.

CFR, Title 10, *Energy*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part072/.

CFR, Title 10, *Energy*, Part 73, "Physical Protection of Plants and Materials." https://www.nrc.gov/reading-rm/doc-collections/cfr/part073/.

CFR, Title 10, *Energy*, Part 74, "Material Control and Accounting of Special Nuclear Material." https://www.nrc.gov/reading-rm/doc-collections/cfr/part074/.

CFR, Title 10, *Energy*, Part 75, "Safeguards on Nuclear Material—Implementation of Safeguards Agreements Between the United States and the International Atomic Energy Agency." https://www.nrc.gov/reading-rm/doc-collections/cfr/part075/.

CFR, Title 10, *Energy*, Part 76, "Certification of Gaseous Diffusion Plants." Available at https://www.nrc.gov/reading-rm/doc-collections/cfr/part076/.

Electrical Power Research Institute, "Probabilistic Risk Assessment (PRA) of Bolted Storage Casks: Quantification and Analysis Report," EPRI Report 1002877, December 2003. Available at https://www.epri.com/research/products/1002877.

International Atomic Energy Agency, "Code of Conduct on the Safety and Security of Radioactive Sources," January 2004. Available at https://www-pub.iaea.org/MTCD/publications/PDF/Code-2004_web.pdf.

U.S. Department of Energy (DOE), "Final Supplemental Environmental Impact Statement: Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada," DOE/EIS-0250-S1, June 2, 2008. Available at https://www.energy.gov/nepa/downloads/eis-0250-s1-final-supplemental-environmental-impact-statement.

DOE, "Final Environmental Impact Statement for the Disposal of Greater-Than-Class C (GTCC) Low-Level Radioactive Waste and GTCC-Like Waste," DOE/EIS-0375, January 2016. Available at https://www.energy.gov/nepa/downloads/eis-0375-final-environmental-impact-statement.

DOE, "DOE Standard: Radiological Control," DOE-STD-1098-2017, January 31, 2017. Available at https://www.standards.doe.gov/standards-documents/1000/1098-AStd-2017/@@images/file.

DOE, "Environmental Assessment: Disposal of Greater-Than-Class C (GTCC) Low-Level Radioactive Waste and GTCC-Like Waste at Waste Control Specialists, Andrews County, Texas," DOE/EA-2082, October 23, 2018. Available at https://www.energy.gov/sites/prod/files/2018/11/f57/final-ea-2082-disposal-of-gtcc-llw-2018-10.pdf.

U.S. Department of Transportation and NRC, "Transportation of Radioactive Material; Memorandum of Understanding," *Federal Register*, July 1979 (44 FR 38690).

U.S. Environmental Protection Agency, "Regulatory Impact Analysis of Final Environmental Standards for Uranium Mill Tailings at Active Sites," EPA 520/1-83-010, September 1983. Available at https://nepis.epa.gov/Exe/ZyPDF.cgi?Dockey=2000XSYI.PDF.

U.S. Nuclear Regulatory Commission (NRC), "Uranium Conversion," available at https://www.nrc.gov/materials/fuel-cycle-fac/ur-conversion.html.

NRC, "Backgrounder – Uranium Enrichment," December 2020. ADAMS Accession No. ML031330159.

NRC, “Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes,” NUREG-0170, December 1977. ADAMS Accession No. ML022590355 (package).

NRC, “Final Generic Environmental Impact Statement on Uranium Milling,” NUREG-0706, September 1980. ADAMS Accession No. ML032751661 (package).

NRC, “Nuclear-Fuel-Cycle Risk Assessment: Descriptions of Representative Non-Reactor Facilities,” NUREG/CR-2873, Volume 1, September 1982.

NRC, “Nuclear Fuel Cycle Risk Assessment: Survey and Computer Compilation of Risk-Related Literature,” NUREG/CR-2933, October 1982.

NRC, “Nuclear Fuel Cycle Risk Assessment: Review and Evaluation of Existing Methods,” NUREG/CR-3682, May 1984.

NRC, “Standard Format and Content for the Health and Safety Sections of License Renewal Applications for Uranium Hexafluoride Production,” Regulatory Guide 3.55, April 1985. ADAMS Accession No. ML003739469.

NRC, “Licensing of Alternative Methods of Disposal of Low-Level Radioactive Waste,” NUREG-1241, December 1986. ADAMS Accession No. ML053010322.

NRC, “Material Control and Accounting Requirements for Facilities Licensed To Possess and Use Formula Quantities of Strategic Special Nuclear Material,” *Federal Register*, March 1987a (52 FR 10033).

NRC, “Environmental Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility: Environmental Report,” NUREG-1300, April 1987b. ADAMS Accession No. ML053010347.

NRC, “Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities,” NUREG-0586, 1988.

NRC, “Shipping Container Response to Severe Highway and Railroad Accident Conditions,” NUREG/CR-4829, October 1988. Available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr4829/.

NRC, “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask,” Regulatory Guide 3.61, February 1989. ADAMS Accession No. ML003739511.

NRC, “Regulatory Analysis for the Resolution of Generic Issue 82, ‘Beyond Design Basis Accidents in Spent Fuel Pools,’” NUREG-1353, April 1989. ADAMS Accession No. ML082330232.

NRC, “Regulating the Disposal of Low-Level Radioactive Waste: A Guide to the Nuclear Regulatory Commission’s 10 CFR Part 61,” NUREG/BR-0121, August 1989. ADAMS Accession No. ML120720225.

NRC, “Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage).” Regulatory Guide 3.48, Revision 1, August 1989. ADAMS Accession No. ML003739163.

NRC, “Backfitting Guidelines,” NUREG-1409, July 1990. ADAMS Accession No. ML032230247.

NRC, “Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility,” NUREG-1199, Revision 2, January 1991. ADAMS Accession No. ML022550605.

NRC, “Recommendations to the NRC on Acceptable Standard Format and Content for the Fundamental Nuclear Material Control (FNMC) Plan Required for Low-Enriched Uranium Enrichment Facilities,” NUREG/CR-5734, November 1991. ADAMS Accession No. ML15120A354.

NRC, “Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility,” NUREG-1200, Revision 3, April 1994. ADAMS Accession No. ML061370484.

NRC, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 1, “Format and Content,” NUREG-1537, Part 1, February 1996a. Available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1537/part1/.

NRC, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” NUREG/CR-6407, February 1996b. ADAMS Accession No. ML15127A114.

NRC, “Chemical Process Safety at Fuel Cycle Facilities,” NUREG-1601, August 1997 (non-public). ADAMS Accession No. ML093070119.

NRC, “Nuclear Fuel Cycle Facility Accident Analysis Handbook,” NUREG/CR-6410, March 1998. ADAMS Accession No. ML072000468.

NRC, “Standard Review Plan for Transportation Packages for Radioactive Material,” NUREG-1609, March 1999. Available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1609/final/.

NRC, “Risk Analysis and Evaluation of Regulatory Options for Nuclear Byproduct Material Systems,” NUREG/CR-6642, February 2000. ADAMS Accession Nos. ML003693052 and ML003693028.

NRC, “Reexamination of Spent Fuel Shipment Risk Estimates,” NUREG/CR-6672, Volume 1, March 2000b. ADAMS Accession No. ML003698324.

NRC, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel,” NUREG-1617, March 2000c. ADAMS Accession No. ML003696262.

NRC, “A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities: Recommendations of NRC’s Performance Assessment Working Group,” NUREG-1573, October 2000. ADAMS Accession No. ML053250352.

NRC, “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants,” NUREG-1738, February 2001. ADAMS Accession No. ML010430066.

NRC, “Integrated Safety Analysis Guidance Document,” NUREG-1513, May 2001. ADAMS Accession No. ML011440260.

NRC, “Systematic Radiological Assessment of Exemptions for Source and Byproduct Materials,” NUREG-1717, June 2001. ADAMS Accession No. ML011980433.

NRC, “A Baseline Risk-Informed, Performance-Based Approach for In Situ Leach Uranium Extraction Licensees,” NUREG/CR-6733, September 2001a. ADAMS Accession No. ML012840152.

NRC, “Regulating Nuclear Fuel,” NUREG/BR-0280, Revision 1, September 2001b. ADAMS Accession No. ML15104A215.

NRC, “Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities,” NUREG-0586, Supplement 1, “Regarding the Decommissioning of Nuclear Power Reactors,” November 2002. Available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0586/.

NRC, “Standard Review Plan for In Situ Leach Uranium Extraction License Applications,” NUREG-1569, June 2003a. ADAMS Accession No. ML031550302 (package).

NRC, “Standard Review Plan for the Review of a Reclamation Plan for Mill Tailings Sites under Title II of the Uranium Mill Tailings Radiation Control Act of 1978,” NUREG-1620, Revision 1, June 2003b. ADAMS Accession No. ML032250190.

NRC, “Consolidated Decommissioning Guidance,” NUREG-1757. Available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1757/.

NRC, “Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material,” Regulatory Guide 7.9, Revision 2, March 2005. ADAMS Accession No. ML050540321.

NRC, “History and Framework of Commercial Low-Level Radioactive Waste Management in the United States: ACNW White Paper,” NUREG-1853, January 2007a. ADAMS Accession No. ML070600684.

NRC, “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant,” NUREG-1864, March 2007b. ADAMS Accession No. ML071340012.

NRC, “Risk-Informed Decisionmaking for Nuclear Material and Waste Applications,” Revision 1, February 2008. ADAMS Accession No. ML080720238.

NRC, “Generic Environmental Impact Statement for In-Situ Leach Uranium Milling Facilities,” NUREG-1910, May 2009. Available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1910/.

NRC, “Issuance of Orders Imposing Trustworthiness and Reliability Requirements for Unescorted Access to Certain Radioactive Material,” EA-09-293, December 16, 2009. ADAMS Accession No. ML101670484.

NRC, “Module 7.0: Health and Safety Fundamentals and Hazards within the Nuclear Fuel Cycle,” F201S – Fuel Cycle Processes, March 2010a. ADAMS Accession No. ML12045A011.

NRC, “Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities,” Regulatory Guide 3.67, Revision 1, April 2011. ADAMS Accession No. ML103360487.

NRC, “Physical Protection of Byproduct Material,” *Federal Register*, March 2013 (78 FR 16922). Available at https://www.govinfo.gov/content/pkg/FR-2013-03-19/pdf/2013-05895.pdf.

NRC, “Physical Protection of Shipments of Irradiated Reactor Fuel: Final Report,” NUREG-0561, Revision 2, April 2013. ADAMS Accession No. ML13120A230.

NRC, “Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants,” Regulatory Guide 5.29, Revision 2, June 2013. ADAMS Accession No. ML13051A421.

NRC, “Acceptable Standard Format and Content for the Material Control and Accounting Plan Required for Strategic Special Nuclear Material: Draft Report for Comment,” NUREG-1280, Revision 2, September 2013a. ADAMS Accession No. ML13253A308.

NRC, “Acceptable Standard Format and Content for the Material Control and Accounting Plan Required for Strategic Special Nuclear Material of Moderate Strategic Significance: Draft Report for Comment,” NUREG-2159, September 2013b. ADAMS Accession No. ML13253A310.

NRC, “Acceptable Standard Format and Content for the Material Control and Accounting Plan Required for Special Nuclear Material of Low Strategic Significance: Draft Report for Comment,” NUREG-1065, Revision 3, September 2013c. ADAMS Accession No. ML13253A305.

NRC, “Spent Fuel Transportation Risk Assessment: Final Report,” NUREG-2125, January 2014a. ADAMS Accession No. ML14031A323.

NRC, “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor,” NUREG-2161, September 2014b. ADAMS Accession No. ML14255A365.

NRC, “Standard Format and Content for a Specific License Application for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility,” Regulatory Guide 3.50, Revision 2, September 2014. ADAMS Accession No. ML14043A080.

NRC, “Standard Review Plan for Conventional Uranium Mill and Heap Leach Facilities: Draft Report for Comment,” NUREG-2126, November 2014. ADAMS Accession No. ML14325A634.

NRC, “Implementation Guidance for 10 CFR Part 37, Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material,” NUREG-2155, Revision 1, January 2015. ADAMS Accession No. ML15016A172.

NRC, “Staff Requirements—SECY-14-0147—Cyber Security for Fuel Cycle Facilities,” SRM-SECY-14-0147, March 24, 2015. ADAMS Accession No. ML15083A175.

NRC, “Standard Review Plan for Fuel Cycle Facilities License Applications,” NUREG-1520, Revision 2, June 2015. ADAMS Accession No. ML15176A258.

NRC, “Cyber Security: Effective Practices for the Establishment and Maintenance of Adequate Cyber Security at Non-Power (Research and Test) Reactor Facilities,” January 2016. ADAMS Accession No. ML15253A060.

NRC, “Rulemaking for Cyber Security at Fuel Cycle Facilities,” Regulatory Basis Document, March 2016. ADAMS Accession No. ML15355A466.

NRC, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel: Final Report,” NUREG-1927, Revision 1, June 2016a. ADAMS Accession No. ML16179A148.

NRC, “Consolidated Guidance about Materials Licenses,” NUREG-1556. Available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1556/.

NRC, “Review of Spent Fuel Reprocessing and Associated Accident Phenomena,” NUREG/CR-7232, February 2017a. ADAMS Accession No. ML17045A577.

NRC, “Safety of Spent Fuel Transportation,” NUREG/BR-0292, Revision 2, February 2017b. ADAMS Accession No. ML17038A460.

NRC, “Update to the U.S. Nuclear Regulatory Commission Cyber Security Roadmap,” SECY-17-0034, February 2017c. ADAMS Accession No. ML16354A258.

NRC, “Proposed Rule—Cyber Security at Fuel Cycle Facilities,” SECY-17-0099, October 4, 2017d. ADAMS Accession No. ML17018A219.

NRC, “Cyber Security for Byproduct Materials Licensees,” *Federal Register*, May 15, 2018a (83 FR 22413).

NRC, “Medical Use of Byproduct Material—Medical Event Definitions, Training and Experience, and Clarifying Amendments,” *Federal Register*, July 16, 2018b (83 FR 33046). Available at https://www.govinfo.gov/content/pkg/FR-2018-07-16/pdf/2018-14852.pdf.

NRC, “Management of Facility-Specific Backfitting and Information Collection,” Management Directive 8.4, September 20, 2019. ADAMS Accession No.ML18093B087.

NRC, “Occupational Radiation Exposures at Commercial Nuclear Power Plant Reactors and Other Facilities 2018: Fifty-First Annual Report,” NUREG-0713, Volume 40, March 2020.

NRC, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities,” NUREG-2215, April 2020. ADAMS Accession No. ML20121A190.

NRC, “2020-2021 NRC Information Digest,” NUREG-1350, Volume 32, October 2020. Available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1350/.

NRC, “Nuclear Material Events Database” Web Page, NMED Annual Reports. Available at https://nmed.inl.gov/.