United States Nuclear Regulatory Commission
Protecting People and the Environment

Draft Regulatory Basis for Licensing and Regulating Reprocessing Facilities

Manuscript Completed: November 2011

Office of Nuclear Material Safety and Safeguards

Office of Nuclear Regulatory Research

Office of Federal and State Materials and Environmental Management Programs

Office of Nuclear Security and Incident Response

(Availability Page)

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ABSTRACT

The Commission directed the United States Nuclear Regulatory Commission (NRC) staff (SRM–SECY–07–0081) "Staff Requirements SECY–07–0081: Regulatory Options for Licensing Facilities Associated with The Global Nuclear Energy Partnership," dated June 27, 2007, to proceed with a regulatory gap analysis and identify changes in the regulatory requirements necessary to license and regulate a potential reprocessing facility that included an advanced burner reactor. This draft regulatory basis document details the NRC staff (staff) preliminary recommendations to address 19 high- and medium-priority regulatory gaps for licensing and regulating a reprocessing facility. Staff identified these 19 gaps in SECY–09–0082, "Update on Reprocessing Regulatory Framework—Summary of Gap Analysis," dated May 28, 2009 (ADAMS Accession No. ML09152024). This document also provides justification for the need to develop new or modify existing regulations and provides an explanation as to why alternatives to rulemaking should not be used. Staff identified different potential approaches to address each gap by considering policy, technical, and legal information. In identifying potential regulatory paths for each gap, the staff engaged stakeholders in three public meetings beginning in September 2010. Stakeholder views were considered in the development of this document. The regulatory basis also identifies plans to develop or revise guidance documents to support a new reprocessing rule (i.e., 10 CFR Part 7x) and lists documents that have been referenced or considered in the development of the regulatory basis.

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INTRODUCTION

In the staff requirements memorandum (SRM) for SECY–07–0081, “Regulatory Options for Licensing Facilities Associated with the Global Nuclear Energy Partnership (GNEP),” dated June 28, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML071800084), the Commission directed the U.S. Nuclear Regulatory Commission (NRC) staff to proceed with a regulatory gap analysis and identify changes in the regulatory requirements necessary to license a potential reprocessing facility that included an advanced burner reactor. The Commission also directed the staff to provide the gap analysis and technical bases document with recommended options on a path forward and an associated rulemaking plan, if appropriate.

In mid-2008, three nuclear industry companies informed NRC of their intent to seek a license for a reprocessing facility in the United States. Also in mid-2008, the Advisory Committee on Nuclear Waste and Materials (ACNW&M) completed a white paper on the background, status, and issues related to the regulation of advanced spent nuclear fuel recycle facilities, published as NUREG–1909, “Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Recycle Facilities” (ADAMS Accession No. ML081550505). At the time, the staff also noted that progress on GNEP initiatives had waned, and it appeared appropriate to shift the focus of the staff’s efforts from specific GNEP-facility regulations to a more broadly applicable framework for commercial reprocessing facilities.

In SECY–08–0134, “Regulatory Structure for Spent Fuel Reprocessing,” dated September 12, 2008 (ADAMS Accession No. ML082110363), the staff discussed its approach to develop the regulatory framework for commercial reprocessing facilities. The staff stated that it would defer additional work on regulatory framework development for advanced recycling reactors and focus on the framework revisions necessary to license a potential commercial reprocessing facility. The staff determined that this shift in focus warranted an additional review of the initial gap analysis. On December 19, 2008, the Nuclear Energy Institute (NEI) submitted a white paper, “Regulatory Framework for NRC Licensed Recycling Facility” (ADAMS Accession No. ML083590114), on a proposed regulatory framework for an NRC-licensed recycling facility. The NEI white paper included principal changes to existing NRC regulations required to implement NEI’s proposed regulatory framework.

In SECY–09–0082, “Update on Reprocessing Regulatory Framework—Summary of Gap Analysis,” dated May 28, 2009 (ADAMS Accession No. ML091520243), the staff summarized 23 regulatory gaps for developing the necessary framework to license and regulate reprocessing and associated facilities. For each of the 23 gaps, the staff determined a priority (i.e., identified the need for addressing each gap as high, moderate, or low) and characterized each of the gaps as resulting from one of the following: (i) lack of regulations; (ii) existing regulations that pose a significant hindrance or regulatory burden to effective and efficient licensing; (iii) licensing a production facility under 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material” (versus 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”); and (4) existing requirements that may need to be modified for clarity. The staff defined high-priority gaps (Gaps 1–14) as those that must be addressed to establish an effective and efficient regulatory framework. The staff defined moderate-priority gaps (Gaps 15–19) as those that should be addressed, but that are not essential at this stage. Based on the gap analysis, the staff proposed to develop the technical basis for a proposed rule that would resolve the high-priority gaps. The staff also addressed potential unintended consequences that ongoing rulemakings could have on the proposed reprocessing framework and noted that the U.S. Environmental Protection Agency's (EPA) dose and effluent limits in 40 CFR Part 190 could pose a challenge for reprocessing facilities.

In a memorandum to the Commission, "Annual Update on Reprocessing Activities—Timeline for Completion of Regulatory Framework," dated May 14, 2010 (ADAMS Accession No. ML101110446), the staff provided its status for developing the regulatory basis (formerly referred to as "technical basis") and stated that it anticipated completing a draft regulatory basis by September 2011. The staff also stated that the nuclear industry supports continued progress for revising the regulatory framework for reprocessing and has indicated that developing a revised framework by 2015 would keep pace with industry activities in this area. The staff described three main areas of activity for development of the regulatory framework: (i) public outreach, (ii) sharing of international knowledge, and (iii) development of the regulatory basis and subsequent rule.

This draft regulatory basis (i) explains why the current regulations should be changed; (ii) explains why options other than rulemaking should not be used to address the gaps in the regulatory framework for reprocessing; (iii) explains how changes in regulations can address the gaps in the regulatory framework and identifies different approaches that could address the gaps in the regulatory framework; (iv) provides the policy, technical, or legal information that supports NRC staff preliminary recommendations to resolve 19 high- and medium-priority regulatory gaps; (v) discusses stakeholder interactions in developing the technical portion of the regulatory basis; (vi) explains how the recommended rulemaking will support the NRC's Strategic Plan (NUREG–1614, Volume 4, which is a document that describes the NRC's mission, values, and strategic goals of safety and security) goals; (vii) explains any limitations on the scope of the regulatory basis, such as known uncertainties in the data or methods of analysis; and (viii) identifies plans to develop or revise guidance to support the recommended rulemaking and lists documents referenced or considered in the development of the regulatory basis.

The draft regulatory basis discusses Gaps 1–19 (high and moderate priority) identified in SECY–09–0082. Because the gaps relate to diverse topics, some of which are related, and the need to provide a comprehensive regulatory basis, this regulatory basis is organized in chapters that group related topics. The subsequent chapters document the results of the staff's analysis of Gaps 1–19, as well other regulatory issues that the staff identified and addressed (e.g., seismic safety) during development of the regulatory basis. The NRC staff's overall approach is risk informed, performance based, and technology neutral, to the extent possible. The staff's proposal to develop a new 10 CFR Part 7x with 10 CFR Part 50 and 10 CFR Part 70 as its foundation, reflects experience from previous licensing (more than 40 years ago) of a reprocessing facility under 10 CFR Part 50 and subsequent development of Commission regulations and policies.

Chapter 1 addresses regulatory framework options (Gap 1) and definitions for reprocessing-related terms (Gap 6). Chapter 1 also discusses issues identified during the past year that were not identified in SECY–09–0082, including licensing considerations, criticality, decommissioning, emergency planning, seismic safety, fire protection, reporting, transfer of special nuclear material, and reviews of license applications by the Advisory Committee on Reactor Safeguards. Because Chapter 1 focuses on the overall regulatory framework, the staff has included in it a discussion, for the entire regulatory basis, of regulatory alternatives, backfit analyses, overall stakeholder interactions, and the relationship of the proposed regulatory framework to the Strategic Plan. Chapter 1 also summarizes the staff's plans to develop or revise guidance documents to support the rule, while each chapter discusses the guidance documents to be revised or developed to support the topics addressed in that chapter. Each chapter also includes stakeholder views on the specific topic being addressed (e.g., baseline design criteria).

Chapter 2 addresses safety and licensing considerations. This chapter includes a discussion of risk considerations for a production facility licensed under 10 CFR Part 70 (Gap 5); licensing operators and criteria for testing and licensing operators (Gap 7); baseline design criteria (Gap 9); one-step licensing and inspection, testing, and acceptance criteria (Gap 10); and technical specifications (Gap 11).

Chapter 3 addresses waste management topics. This chapter includes discussion of independent storage of high-level waste (Gap 2), waste incidental to reprocessing (Gap 3), waste confidence for reprocessing facilities (Gap 15), waste classification (Gap 16), and effluent controls and monitoring (Gap 19).

Chapter 4 addresses operational considerations, including material control and accounting, physical protection, fees, and financial protection. This chapter discusses the exclusion of irradiated fuel reprocessing facilities in 10 CFR 74.51 (Gap 4), risk-informing 10 CFR Part 73 and 10 CFR Part 74 (Gap 8), financial protection requirements and indemnity agreements (10 CFR Part 140; Gap 12), schedule of fees (10 CFR Part 170; Gap 13), annual fees (10 CFR Part 171; Gap 14), diversion path analysis (Gap 17), and approaches to material accounting management (Gap 18).

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1 REGULATORY FRAMEWORK (GAPS 1 and 6)

1.1 Background

In SECY–09–0082, “Update on Reprocessing Regulatory Framework—Summary of Gap Analysis,” dated May 28, 2009, the U.S. Nuclear Regulatory Commission (NRC) staff provided the Commission with the staff’s summary of the regulatory gap analysis for developing the necessary framework to license a reprocessing facility.

Based on this draft regulatory basis document, the staff concludes that development of a new, reprocessing-specific regulation (i.e., 10 CFR Part 7x) would provide the most effective and efficient approach to licensing and regulating a reprocessing facility. A regulatory scheme for reprocessing facilities should include requirements similar to the existing 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.” In addition, a risk-informed, performance-based approach, coupled with defense-in-depth requirements (e.g., general design criteria), should be developed to ensure the safe handling of spent nuclear fuel (SNF), separated fission products and actinides, and associated waste streams from reprocessing operations. Moreover, the staff recommends that new reprocessing requirements should be technology neutral, to the extent possible, to reflect the different reprocessing technologies that have been proposed by industry (i.e., aqueous and electrochemical separations).

The staff has identified several topics that need a detailed analysis in light of the unique attributes of a reprocessing facility. These topics are discussed in Sections 1.3.1–1.3.9 and supplement the gaps identified in SECY–09–0082 (NRC, 2009a). The topics include licensing considerations, criticality requirements, decommissioning, emergency planning requirements, seismic safety, fire protection, reporting requirements, and transfer of special nuclear material (SNM). In addition, this chapter describes the basis for a proposed framework for licensing commercial nuclear fuel reprocessing plants (FRPs) (Gap 1) and provides definitions the staff is considering for inclusion in a potential rule on reprocessing facilities (Gap 6). Chapters 2 through 4 address gaps identified in the areas of waste, safety and licensing, and operational considerations. A regulatory framework for licensing reprocessing facilities should incorporate regulations pertaining to the identified gaps. The staff is addressing gaps in safeguards and material control and accounting in parallel rulemakings affecting 10 CFR Part 73, “Physical Protection of Plants and Materials,” and 10 CFR Part 74, “Material Control and Accounting of Special Nuclear Material.”

In summary, the staff has identified the following actions that are needed to develop an effective and efficient regulatory framework for licensing commercial reprocessing facilities:

(1) Create a new CFR part, incorporating aspects of existing regulations that should be modified (to the extent possible) to be risk-informed, performance-based, technology neutral, and applicable to the unique attributes of reprocessing facilities.

(2) Develop new regulations or modify existing regulations to resolve the regulatory gaps that have previously been identified (NRC, 2009a).

(3) Remove most references to “reprocessing” and “FRPs” in 10 CFR Part 50, including Appendix F, “Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities.” The staff recommends maintaining 10 CFR Part 50,

Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

1.2 Basis for Regulatory Framework

1.2.1 Regulatory Problem

A fuel reprocessing facility meets the definition of a production facility, as defined in the Atomic Energy Act of 1954 (as amended) (AEA), Section 11 and 10 CFR 50.2, because reprocessing will be used to separate plutonium isotopes and will produce SNM in quantities that could affect radiological health and safety and be of significance to common defense and security. Under existing regulations, the NRC could license a fuel reprocessing facility under 10 CFR Part 50. The policy statement on reprocessing in 10 CFR Part 50, Appendix F, “Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities,” has, however, not been updated recently. Sections 1.2.2 and 1.4 discuss Appendix F in more detail. Further, as discussed in SECY–09–0082, 10 CFR Part 50 has evolved into a light-water reactor-specific regulation and was developed before the Commission issued its policy on risk-informed, performance-based regulation (NRC, 1999a). Consequently, 10 CFR Part 50, for the most part, is prescriptive and deterministic. Using 10 CFR Part 50 to license a reprocessing facility would necessitate exemptions from the current rules and could result in a more protracted and less efficient licensing process. Therefore, modifying 10 CFR Part 50 into an effective and efficient regulation for a production facility that reprocesses SNF would involve considerable NRC staff resources.

In 2006, the staff made the following comment on the feasibility of licensing a commercial reprocessing facility, or a plant operated by the U.S. Department of Energy (DOE) (if mandated by Congress), using the existing 10 CFR Part 50 (NRC, 2006b):

> Part 50 is focused on LWR [light-water reactor] design and technology and would have limited applicability to commercial reprocessing facility design and technology. That is, the design and operational safety issues associated with a commercial reprocessing facility would be very different from design and operational safety issues associated with an LWR. The current Part 50 regulations would not necessarily address all commercial reprocessing facility safety issues and, conversely, are likely to contain requirements that are not applicable to a reprocessing facility. The application of the whole of Part 50 to the licensing of a commercial reprocessing facility would present significant challenges to the applicant and to the NRC. If Part 50 is used to license a commercial reprocessing facility, the regulations would have to be reviewed to determine which apply, which do not apply, and which may partially apply. Additional requirements would also need to be established to address reprocessing facility-specific design and safety issues.

In addition, 10 CFR Part 50 may not be suited to effectively regulate the chemical hazards that would be present at a reprocessing facility, such as the concentrated nitric acid used to dissolve the fuel elements.

1.2.2 Existing Regulatory Framework

The NRC has the authority, under the AEA, to license commercial spent fuel reprocessing facilities. The Atomic Energy Commission (AEC) developed Appendix F to codify its policy statement on the siting of reprocessing plants and the question of the ultimate disposal of high-level waste (HLW) (AEC, 1970). Appendix F states that liquid HLW will only be stored at a reprocessing site for 5 years before solidification and that this waste should be transferred in an AEC-approved solid form to a federal waste repository no later than 10 years following separation of fission products from irradiated fuel. The appendix also requires that reprocessing plants be designed to facilitate decontamination and removal of all significant radioactive wastes in the event a plant is retired from operational status. In addition, license applicants are required to furnish information on financial qualifications to enable the Commission to determine whether the applicant is qualified to provide for the removal and disposal of radioactive wastes in accordance with Commission regulations.

10 CFR 50.34(a)(7) requires an FRP applicant to describe in its preliminary safety analysis report (SAR) the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components (SSCs) of the facility. In 10 CFR 50.34(b)(6)(ii), the NRC requires that every applicant for an operating license for a reprocessing facility include, in the final safety analysis report (FSAR), information pertaining to the managerial and administrative controls to be used to assure safe operation.

When the AEC proposed that the quality assurance requirements in 10 CFR Part 50, Appendix B, be made applicable to reprocessing facilities, it recognized that, like a nuclear power reactor, “fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public” (AEC, 1971a). 10 CFR Part 50, Appendix B contains requirements for applicants to determine actions necessary to provide adequate confidence that SSCs will perform satisfactorily in service. It also describes how applicants are required to include information pertaining to managerial and administrative controls used to assure safe operations.

The regulation in 10 CFR 50.36 requires applicants to propose technical specifications for production and utilization facilities. Technical specifications will be derived from the analyses and evaluation included in the SAR. Technical specifications for reprocessing plants must include safety limits, limiting control settings, and limiting conditions for operation. Chapter 2 discusses the requirements for fuel reprocessing facilities in greater detail.

Facilities licensed under 10 CFR Part 50 must also comply with requirements codified in other parts of Chapter 1 including the following, for example:

- 10 CFR Part 20, “Standards for Protection Against Radiation”: Contains the NRC’s radiation protection standards based upon the International Commission on Radiological Protection system of dose limitation and the principle that all radiation exposures be kept “as low as reasonably achievable” (ALARA).

- 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions”: Codifies the NRC’s responsibilities with regard to the National Environmental Policy Act of 1969, as amended (NEPA).

- 10 CFR Part 55, "Operators' Licenses": Establishes requirements for licensing operators and senior operators at nuclear reactors.

- 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Waste, and Reactor-Related Greater Than Class C Waste": Establishes requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel, power reactor-related greater than Class C (GTCC) waste, and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI), and the terms and conditions under which the Commission will issue these licenses.

- 10 CFR Part 73, "Physical Protection of Plants and Materials": Prescribes requirements to establish and maintain a physical protection system that will allow for the protection of SNM at fixed sites and in transit, and of plants in which SNM is used.

- 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material": Contains the requirements for the control and accounting of SNM at fixed sites and for documenting the transfer of SNM.

The NRC uses 10 CFR Part 70 to regulate existing plutonium (Pu) processing and fuel fabrication facilities. This Part contains the requirements for possession and transportation of SNM. In addition, 10 CFR Part 70 provides that an applicant must include specifications (including the chemical and physical form and, if applicable, isotopic content) of the SNM the applicant proposes to use or produce and describe the physical security measures for handling the SNM, licensee training, and the hazardous chemicals that will be used on the site.

In 2000, the NRC amended 10 CFR Part 70. Included in these amendments were performance requirements and the requirement that affected licensees perform an integrated safety analysis (ISA) to identify potential accidents at the facility and the items relied on for safety (IROFS) necessary to prevent these potential accidents or mitigate their consequences (NRC, 2000). The NRC also developed the rule to reflect its policy on the use of risk-informed, performance-based regulation (NRC, 1999a). This regulatory approach balances considerations such as cost and environmental impacts against the required reduction in risk at the facility.

Reprocessing facilities involve more types and greater quantities of chemicals than other fuel cycle facilities, and their design may exacerbate chemical safety concerns. Some oxidizing chemicals, such as nitrogen tetroxide (used to convert Pu(III) to Pu(IV) in the plutonium and uranium recovery by extraction (PUREX) process), are extremely hazardous and can exhibit toxic effects for 1,000 meters (m) (equivalent) or more from a release. Unlike 10 CFR Part 50, 10 CFR Part 70 addresses chemical and facility safety in 10 CFR 70.61, 70.62, 70.64, and 70.65. These regulations require an evaluation in the ISA of an acute chemical exposure to a worker and, in the case of a new licensee, a description of how the facility design will provide adequate protection against chemical risks. Chapter 2 discusses staff's proposal on how a licensee should address and mitigate chemical hazards at reprocessing facilities.

The NRC has a memorandum of understanding with the Occupational Health and Safety Administration (OSHA), whereby the NRC is responsible for regulating chemical safety and risks as they interrelate with radiological safety (ADAMS Accession No. ML0328011621). The memorandum of understanding identifies the following as the NRC's regulatory responsibilities:

- Radiation risk produced by radioactive materials

- Chemical risks produced by radioactive materials (e.g., uranium (U) and U hexafluoride (UF_6) toxicity)

- Plant conditions that affect the safety of radioactive materials and thus present an increased radiation risk to workers (e.g., an NO_X release external to the reprocessing facility that is sucked into the plant by the ventilation system)

1.2.3 Basis for Requested Change

The development of a new part in the regulations could address the licensing and operation of a reprocessing facility. The NRC has not received an application for a reprocessing facility since the 1970s, and existing regulations in 10 CFR Part 50 have evolved to primarily focus on nuclear power plants (NPP). Consequently, if an applicant for a reprocessing facility applied for a 10 CFR Part 50 license, exemptions from reactor-specific regulations may be required.

The regulations in 10 CFR Part 70, which pertain to the licensing of fuel cycle facilities, do not include provisions sufficient to address the larger radionuclide inventory and unique chemical hazards of a reprocessing facility. Therefore, to address licensing and operation of a reprocessing facility, Part 70 would need significant revisions. A new 10 CFR Part 7x regulation could provide a regulatory framework that appropriately addresses the larger radionuclide inventory and unique chemical hazards that are associated with fuel reprocessing facilities, without affecting existing 10 CFR Parts 70 and 50 licensees.

Aspects of the existing regulations would be relocated to a new 10 CFR Part 7x, with appropriate modifications. NRC staff is currently assessing existing regulations in 10 CFR Parts 50, 70, and 55 for inclusion in a new Part. These regulations are summarized in Table 1-1.

Table 1-1. Summary of Regulations

Subject Matter of Regulation	Location in Current 10 CFR
Purpose	50.1, 70.1
Scope	70.2
Definitions	50.2, 70.4
Deliberate misconduct	50.5, 70.10
Employee protection	50.7, 70.7
Information collection requirements; OMB approval	50.8, 70.8
Interpretations	50.3, 70.6
Completeness and accuracy of information	50.9, 70.9
Communications	50.4, 70.5
License required; limited work authorization (LWA)	50.10
Exceptions and exemptions from licensing requirements	50.11, Part 70, Subpart B
Class 103 licenses; for commercial and industrial facilities	50.22
Construction permits	50.23
Filing of applications for licenses	50.30, 70.21
Combining applications	50.31
Contents of applications	50.33, 50.34, 70.22

Table 1-1. Summary of Regulations	
Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors	50.34a
Issuance of construction permits	50.35
Technical specifications	50.36, 50.36a
Environmental conditions	50.36b, 70.64(a)(4)
Agreement limiting access to classified information	50.37
Ineligibility of certain applicants	50.38, 70.40
Public inspection of applications	50.39, 70.21(d)
Common standards	50.40
Additional standard for Class 103 licenses	50.42
Standards for construction permits, operating licenses, and combined licenses (COLs)	50.45,
Emergency plans	50.47, 70.22(i)
Fire protection	50.48, 70.64(a)(3) (Baseline Design Criteria)
Issuance of licenses and construction permits	50.50, 70.31
Continuation (renewal) of licenses	50.51, 70.33
Jurisdictional limitations	50.53
Conditions of license	50.54, 70.32
Hearings	50.58(b), 70.23a
Changes, tests, and experiments	50.59, 70.72
Criticality accident requirements	50.68, 70.24
Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors	50.69
Inspections	50.70, 70.55
Maintenance of records, making reports	50.71, 70.51, 70.62(a)(2), 70.62(a)(3)
Immediate notification requirements	50.72, 70.50
Licensee event report system	50.73
Notification of change in operator or senior operator status	50.74
Reporting and recordkeeping for decommissioning planning	50.75, 70.25
Transfer of licenses	50.80, 70.36
US/IAEA Safeguards Agreement	50.78, 70.21(g)
Creditor regulations	50.81, 70.44
Termination of license	50.82, 70.38
Release of part of a power reactor facility or site for unrestricted use	50.83
Application for amendment of license, construction permit, or early site permit	50.90, 70.34
Notice for public comment; State consultation	50.91
Issuance of amendment	50.92
Revocation, suspension, or modification of licenses, permits, and approvals for cause	50.100, 70.81
Retaking possession of special nuclear material	50.101, 70.81(c)
Commission order for operation after revocation	50.102
Suspension and operation in war or national emergency	50.103, 70.82
Backfitting	50.109, 70.76

Table 1-1. Summary of Regulations	
Violations	50.110, 70.91
Criminal penalties	50.111, 70.92
Training and qualification of nuclear power plant personnel	50.120
Aircraft impact assessment	50.150
Quality assurance criteria for nuclear power plants and fuel reprocessing plants	Part 50 App. B, 70.22(f)
Earthquake engineering criteria for nuclear power plants	Part 50 App. S
Persons using special nuclear material under certain Department of Energy and Nuclear Regulatory Commission contracts	70.11
General license to possess special nuclear material for transport	70.20a(a)
Disclaimer of warranties	70.37
Reports of accidental criticality	70.52
Performance requirements	70.61
Exemptions	55.11, 55.13
Medical requirements	55.21, 55.23, 55.25, 55.27
Applications for operators' licenses	55.31, 55.33, 55.35
Implementation of written examinations and operating tests	55.40
Written examinations and operating tests	55.41, 55.43, 55.45
Simulation facilities	55.46
Waiver of examination and test requirements	55.47
Integrity of examinations and tests	55.49
Issuance of licenses	55.51
Conditions of licenses	55.53
Expiration	55.55
Renewal of licenses	55.57
Requalification	55.59
Modification and revocation of licenses	55.61
Enforcement	55.71, 55.73

1.2.4 Regulatory Alternatives

The staff considered alternatives to developing a new rule for reprocessing facilities. The staff also considered whether alternatives could be used in conjunction with the current regulations governing fuel reprocessing facilities to facilitate the licensing and regulatory process. None of the non-rulemaking alternatives outlined next were found to address the regulatory gaps in existing regulations for reprocessing facilities.

Regulatory issue summaries (a new product) are used to (i) document the NRC's endorsement of the resolution of issues addressed by industry-sponsored initiatives, (ii) solicit voluntary licensee participation in staff-sponsored pilot programs, (iii) inform licensees of opportunities for regulatory relief, (iv) announce staff technical or policy positions not previously communicated to the industry or not broadly understood, and (v) address matters previously reserved for administrative letters.

Generic letters request that addressees (i) perform analyses or submit descriptions of proposed corrective actions regarding matters of safety, safeguards, or the environment and submit, in writing, that they have completed the requests, with or without prior NRC approval of the action; (ii) submit technical information that the NRC needs to perform its functions; or (iii) submit proposed changes to technical specifications. By a generic letter, the NRC may also (i) provide to the addressees staff technical or policy positions not previously communicated or broadly understood or (ii) solicit participation in voluntary pilot programs.

Regulatory issue summaries and generic letters are, however, generally used to communicate with licensees regarding existing requirements, and are not a means to establish new regulatory requirements or revise existing regulations. Accordingly, these summaries and letters are not feasible alternatives to developing an effective and efficient regulatory framework for reprocessing facilities.

Revision of regulatory guidance documents. The NRC issued regulatory guides (RGs) for fuel reprocessing facilities in the 1970s. These RGs would need to be revised to reflect modifications and changes in the NRC's regulatory framework since the 1970s. In addition, the staff would need to consider whether guidance documents should be withdrawn because the technology described has become obsolete. For example, these guidance documents were written for aqueous separation processes; therefore, in most cases they would not be suitable for electrochemical processing. A reprocessing license applicant would not be required to comply with regulatory guidance. Therefore, modifying existing guidance would not address the issue of developing an effective and efficient regulatory framework for reprocessing facilities.

It is staff's opinion that gaps in the regulations can only be resolved through rulemaking. The staff identified the following alternative rulemakings to developing a new rule for reprocessing facilities.

- *Modifying the existing 10 CFR Part 50:* Using 10 CFR Part 50 to license a reprocessing facility would require exemptions for reprocessing licensees, as the regulation has become more applicable to NPPs over the years. Modifying the existing regulation could result in confusion for current 10 CFR Part 50 licensees (i.e., nuclear power, research, and test reactors). Therefore, staff concludes that 10 CFR Part 50 should not be modified.

- *Modifying 10 CFR Part 70:* In 2007, the staff presented a number of options to the Commission with regard to licensing facilities with the then DOE initiative, the Global Nuclear Energy Partnership (GNEP) (NRC, 2007a). In the staff requirements memorandum (SRM) for this paper (NRC, 2007b), the Commission instructed the staff to proceed with modifying 10 CFR Part 70 to accommodate reprocessing facilities. The staff's original intention was to modify the existing 10 CFR Part 70 regulations, with appropriate changes to 10 CFR Part 50, to license reprocessing facilities (i.e., removing the licensing of FRPs from 10 CFR Part 50). The staff considered developing a new subpart dedicated to licensing.

When considering possible revisions to 10 CFR Part 70, the staff determined, as stated in the first-order gap analysis (ADAMS No. ML082260223) (NRC, 2008a), that

> Part 70 currently does not address specific hazards with the reprocessing of spent nuclear fuel or any new reprocessing technology that may be proposed. Some of these hazards are…an increase in radiological risk and different

process streams than the uranium fuel processing facilities for which Part 70 was most recently revised in 2000.

Licensing a reprocessing facility under 10 CFR Part 70 would require revisions to 10 CFR Part 70, which would probably involve similar resources as developing a new regulation. Accordingly, the NRC staff determined that the development of a new, reprocessing-specific regulation (i.e., 10 CFR Part 7x) provides the most effective and efficient approach to licensing and regulating a reprocessing facility.

1.2.5 Backfit Analysis

This draft regulatory basis document provides the basis for rulemaking for reprocessing facilities. The NRC staff will consider backfit implications in its development of a final regulatory basis document.

1.2.6 Stakeholder Interaction

Nuclear Energy Institute (NEI) White Paper: NEI, the organization responsible for establishing unified industry policy on matters affecting the commercial nuclear energy industry, including the regulatory aspects of generic operational and technical issues, established a task force directed at closing the nuclear fuel cycle. This task force submitted a paper to the NRC, known as the NEI White Paper (NEI, 2008). The paper details NEI's view that a reprocessing facility is more like a complex fuel cycle facility than a reactor, and consequently, it supports creating a new regulatory part, 10 CFR Part 7x, specific to reprocessing facilities. NEI's opinion is that 10 CFR Part 7x should provide flexibility for licensing facilities associated with reprocessing operations, such as a vitrification plant or fuel fabrication and storage. The NRC should also make provisions in the regulations for a 10 CFR Part 52-type licensing, which would allow the licensee to submit a COL application for a reprocessing facility. The framework that NEI proposed is technology neutral and would therefore be applicable to aqueous solvent extraction methods of reprocessing and electrochemical separations.

NUREG–1909, "Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Recycle Facilities: ACNW&M White Paper": With the initiation of the DOE GNEP, the Commission directed the Advisory Committee on Nuclear Waste and Materials (ACNW&M) in 2006 to undertake a study of the background and issues related to the NRC role in the potential licensing of fuel reprocessing facilities (NUREG, 2008a). The resulting ACNW&M white paper provides an historical overview of previous licensing, operating, and decommissioning experience associated with commercial and government reprocessing efforts, including both domestic and international programs. In the context of this experience, the report describes the different fuel reprocessing technologies used at these facilities, including PUREX, the most common technology. The report also discusses the Uranium Extraction (UREX) advanced fuel reprocessing technology proposed for the GNEP initiative.

Based on the historical perspective, the ACNW&M identified important factors to be considered and made several recommendations for developing 10 CFR Part 7x regulations for fuel reprocessing. These include the following:

- Use a risk-informed, performance-based regulatory approach in developing a new 10 CFR Part 7x. Related to this, the ACNW&M noted that, given the evolving nature of the scope, functional requirements, size, and timing of facilities, it would be prudent to avoid the initial development of program-specific regulations that might be out of date by the time a license application is actually received.

- Use a probabilistic risk assessment (PRA) to evaluate in-plant hazards, rather than the ISA currently used in 10 CFR Part 70 regulations. The ACNW&M also identified a "companion issue" that the staff, in comparing the advantages and disadvantages of a PRA with a deterministic ISA, should consider—whether a PRA should use best estimate (more realistic) data and models or conservative assumptions that bound likely uncertainties.

- Use a two-step licensing process rather than a one-step COL process. The ACNW&M recommended this as the more appropriate approach for a new type of facility, until the NRC staff becomes familiar with the processes, equipment, and materials used in commercial fuel processing.

- The NRC staff should consider the current civilian waste classification and disposal systems [i.e., HLW; Classes A, B, and C low-level waste; and Greater-than-Class C (GTCC)] and evaluate the extent to which they apply to the waste streams generated in commercial fuel reprocessing. The ACNW&M specifically focused on the Cs and Sr waste streams that would be produced during fuel reprocessing.

The industry has stated its position regarding the development of a regulatory framework for reprocessing on several occasions. At a public meeting on May 13, 2010, NEI representatives stated that the industry wants the framework to have a structure that could evolve to accommodate advanced technologies ahead of the need to do so. They reiterated the position documented in the white paper, which is to support a new 10 CFR Part 7x and regulations that are technology neutral to the extent possible.

Other stakeholders have expressed their support of a new CFR part for FRPs and their associated facilities. At the public workshop held on September 7–8, 2010, one stakeholder stated that reprocessing plants have unique, specific attributes that are unlike those in facilities licensed under 10 CFR Part 50 or 10 CFR Part 70. The stakeholder's view was that, although the NRC could draw on aspects of 10 CFR Part 50 and 10 CFR Part 70 to develop reprocessing requirements, it should develop a new regulation.

As for being technology neutral, the same stakeholder was somewhat skeptical about the possibility of developing technology-neutral regulations. He acknowledged that it may be possible for some regulatory aspects, but made the point that, regarding waste, for example, aqueous and electrochemical processing will produce different waste streams with different associated risks . A second stakeholder supported this view and thought that, in practice, there would be so many exceptions, variations, and reserved sections in a technology rule that the NRC would essentially end up with a technology-specific regulation.

NRC staff presented their preliminary positions on how to address the regulatory gaps described in SECY–09–0082 at a public workshop held in Augusta, GA, June 21–22, 2011. The stakeholder comments that were received during the meeting and subsequent comment period will be addressed during development of the final regulatory basis and potential rulemaking.

The comments received included the following:

Regulatory Framework and Licensing

- Concern was expressed that a reprocessing facility built in the United States could be owned by a foreign entity and that foreign interests would benefit economically, while placing greater liability on the U.S. taxpayer and electric power customers.

- Development of a single set of regulations to cover all potential aspects of reprocessing operations is beyond capability of the NRC at this time.

- A mixed oxide (MOX) fuel fabrication facility should be outside the scope of regulations for a reprocessing facility, even if co-located. A comingled reprocessing plant waste and MOX plant waste would raise a unique regulatory issue that underscores that a single set of regulations for a reprocessing plant may be problematic.

- Reprocessing rulemaking should be put on hold until a thorough analysis of the Fukushima nuclear incident in Japan has been completed.

Regulatory Issues

- Emergency Planning: Requirements should be hazards/risk based and based upon a safety analysis performed for a particular facility. Another view was that emergency planning requirements should be akin to those at reactors.

- Fire Protection: An applicant should be permitted to propose use of standards appropriate for its facilities (e.g., National Fire Protection Association (NFPA) 801, 805) subject to NRC review and approval. Experiences at such facilities as Rocky Flats, West Valley, and Fukushima should be used when developing fire protection requirements.

- Seismic Safety: Use a graded seismic hazard design approach like that described in the DOE standard DOE STD–1020 (DOE, 2002).

Definitions

- “Reprocessing” should not be defined as “recycling.”

1.2.7 Strategic Plan

Development of 10 CFR Part 7x will support the NRC’s 2008–2013 Strategic Plan (NUREG, 2008b) in the areas of safety and openness. In terms of safety, the proposed rule will provide a framework for licensing reprocessing facilities that will ensure adequate protection of public health and safety and the environment. In developing the regulation, the staff will consider current requirements and whether they could be adapted for a risk-informed, performance-based regulation (Strategy 5: Use sound science and state-of-the-art methods to establish, where appropriate, risk-informed and performance-based regulations).

The development of this regulation will add CFR requirements specific to a reprocessing facility licensee that will allow more efficient and effective licensing (Strategy 8: Achieve efficiencies in the licensing process that enable the safe and secure use of nuclear material). A rulemaking for reprocessing facilities will establish a predictable regulatory program for all stakeholders. Regulations for reprocessing, which include those pertaining to risk assessment, effluent monitoring, and criticality requirements, will help support many of the NRC's strategic outcomes associated with safety, such as preventing the occurrence of any of the following: inadvertent criticality events, acute radiation exposures resulting in fatalities, and releases of radioactive materials that result in significant radiation exposures or significant adverse environmental impacts.

Involving stakeholders in the development of the regulatory basis supports the NRC's goal of openness. The staff has held a number of public meetings throughout the development of this draft regulatory basis to ensure that the public has had a reasonable opportunity to participate in the process and to submit comments. This draft document will be made available to the public, and the staff plans to conduct public meetings to discuss the contents. NRC will consider public comments in finalizing the document.

1.3 Other Regulatory Issues

In addition to the gaps identified in SECY–09–0082, "Update on Reprocessing Regulatory Framework—Summary of Gap Analysis" dated May 28, 2009 (NRC, 2009a), and discussed in greater detail in subsequent chapters, the staff identified certain regulatory issues in Table 1.1 that require further analysis to develop a suitable basis in support of rulemaking. These are described in Sections 1.3.1–1.3.9. The proposed rulemaking should also consider any outcomes or recommendations from the task force on the Fukushima incident.

1.3.1 Licensing Considerations: One- or Two-Step Process and Scope of Reprocessing Operations

One or Two-Step Process: In the SRM to SECY–06–0066, "Regulatory and Resource Implications of a Department of Energy Spent Nuclear Fuel Recycling Program," dated May 16, 2006 (NRC, 2006c), the Commission instructed staff to consider the most effective and efficient elements of the NRC licensing process to develop a process for SNF reprocessing facilities. This included the combined licensing process for nuclear power reactors under 10 CFR Part 52. The gap analysis that accompanied SECY–09–0082 (NRC, 2009a) recognized the lack of one-step licensing to facilitate effective and efficient licensing of a reprocessing facility. Chapter 2 (Gap 10) discusses this topic. Staff is considering developing a regulatory framework that provides for the licensing of a facility by either a one-step or a two-step process.

Scope of Reprocessing Operations: Staff is considering whether certain co-located facilities associated with reprocessing (e.g., fuel fabrication facility) should be regulated under the existing regulations or under a new 10 CFR Part 7x. Industry supports using a new 10 CFR Part 7x to license all reprocessing facility operations, including any co-located SNF storage and fuel fabrication operations.

1.3.1.1 Existing Regulatory Framework

One- or Two-Step Process: The current 10 CFR Part 50 contains licensing requirements for both production and utilization facilities. Licenses under 10 CFR Part 50 include Class 103 licenses, described in 10 CFR 50.22, "Class 103 Licenses; for Commercial and Industrial

Facilities," and Class 104 licenses, described in 10 CFR 50.21, "Class 104 Licenses; for Medical Therapy and Research and Development Facilities." If licensed, a commercial reprocessing facility built today would be a Class 103 facility. 10 CFR Part 50 describes a two-step licensing process: application for a construction permit and then an application for an operating license.

The key technical provisions described in 10 CFR 50.34, "Contents of Applications; Technical Information," must be included in an application for a construction permit. 10 CFR 50.34(a)(i) requires a construction permit application to include a description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. The applicant must analyze and evaluate the major SSCs of the facility, which bear significantly on the acceptability of the site, using the site evaluation factors identified in 10 CFR Part 100, "Reactor Site Criteria."

A construction permit and an operating license are issued under 10 CFR 50.23, "Construction Permits," and 10 CFR 50.57, "Issuance of Operating License," respectively. Under 10 CFR 50.58, "Hearings and Report of the Advisory Committee on Reactor Safeguards," the Commission will hold a hearing on receipt of a construction permit application. 10 CFR 50.58(b)(2) provides that a hearing may be held on amendment to a construction permit or operating license if contested.

By contrast, 10 CFR Part 52 provides for a one-step process for licensing nuclear power reactors. A one-step licensing process for reprocessing facilities is discussed in Chapter 2.

Scope of Reprocessing Operations: A reprocessing facility may be co-located with an SNF storage and fuel fabrication facility. Storage of SNF is regulated under 10 CFR Part 72, and fuel fabrication is regulated under 10 CFR Part 70.

NRC authorizes storage of SNF at an ISFSI either by a site-specific license under 10 CFR Part 72 or a general license. An applicant for a site specific license submits a license application to NRC, and the NRC performs a technical review. If the application is approved, the NRC issues a license that is valid for 40 years (10 CFR 72.42). 10 CFR Part 72 provides that a general license authorizes a nuclear power plant licensee to store spent fuel in NRC-approved casks at a site that is licensed to operate a power reactor under 10 CFR Parts 50 or 52.

NRC licenses fuel fabrication under 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The requirements in 10 CFR Part 70 provide two-step licensing for a plutonium fuel processing and fuel fabrication plant. 10 CFR 70.23(a)(7) and 70.23(b) address approvals for construction, and 10 CFR 70.23(a)(8) addresses approval for operation. Under 10 CFR 70.22(f), an application for a license to possess and use SNM must contain a description of the plant site; a description and safety assessment of the design bases of the principal structure, systems, and components (PSSC) of the plant, including provisions for protection against natural phenomena; and a description of the quality assurance program to be applied to the design, fabrication, construction, testing, and operation of SSCs of the plant.

1.3.1.2 Stakeholder Input

One- or Two-Step Process: The ACNW&M, in its 2007 letter to NRC Chairman Klein, expressed its belief that a two-step process should be used until the staff becomes familiar with processes, equipment, and materials in reprocessing facilities. A preference for a two-step licensing process has also been expressed by various stakeholders during the series of public workshops that the NRC has held during the development of the draft regulatory basis document. Much of this support stems from a concern that one-step licensing will diminish opportunities for stakeholder input. Other stakeholders did not think that there is sufficient domestic experience with reprocessing facilities to justify a one-step licensing process. Industry has expressed interest in pursuing one-step licensing for SNF reprocessing facilities; however, the NEI white paper includes a framework that could allow an applicant the flexibility for either a two-step or one-step licensing process. NEI emphasized the importance of this flexibility during the public workshops, where the concept of a first or lead facility (which could be licensed under a two-step process) followed by a subsequent application(s) (which, because of lessons learned, could be one step) was discussed. One stakeholder also stated that a one-step or two-step licensing process should be chosen based upon the level of maturity of the proposed technology.

In its white paper, NEI acknowledged that an FRP will comprise many different operations. These could include, in addition to the main reprocessing functions, fuel fabrication, vitrification, and SNF storage. NEI stated that flexibility in the regulations would allow a licensee to restrict its activities to those that are strictly separations related or to include in the application other activities taking place on a contiguous site, as mentioned previously. NEI prefers that an applicant be able to license parts of the facility early, using the existing regulatory framework (e.g., a spent fuel pool could be licensed under 10 CFR Part 72), and to later transfer to a single license without reopening, except under limited circumstances, the licensing process if the licensee desires to do so.

The language in the NEI-proposed rule includes a requirement identical to 10 CFR 50.22, “Class 103 Licenses; for Commercial and Industrial Facilities.” NEI noted that 10 CFR 50.22 was promulgated to “…define the circumstances under which research and development and training reactors will be considered to be used substantially for industrial or commercial purposes,” and thus be licensable by the Commission under AEA Section 103 (AEC, 1971b). NEI stated that this regulation would be necessary for a CFR part regulating a reprocessing facility, because it is also a production facility.

At the public workshop to discuss major issues associated with the development of a regulatory framework for a potential rulemaking for SNF reprocessing facilities (in Rockville, Maryland, on September 7, 2010), a Tennessee Valley Authority representative said that the NRC should consider formulating a rulemaking to allow one-step licensing for a more mature reprocessing plant design and a two-step process for a less established plant. A non-industry representative on the panel stated that regulatory uncertainty could ensure that the NRC would have to develop a process for determining the level of detail of design information needed in the applications submitted under one- and two-step licensing processes.

Scope of Reprocessing Operations: NEI stated in its white paper that all reprocessing operations should be licensed under a new 10 CFR Part 7x, including fuel fabrication. NEI reiterated this during the NRC public workshops. NEI explained the reason all reprocessing operations should be licensed under a new CFR part in the context of current licensee experiences at existing NPPs: SNF storage is regulated under 10 CFR Part 72, and operations for loading SNF storage casks are regulated under 10 CFR Part 50. NEI stated that licensees are finding difficulties and inefficiencies where regulations interface. Industry concludes that having all reprocessing facility operations regulated under 10 CFR Part 7x would avoid these inefficient interfaces.

1.3.1.3 Basis for Rulemaking

One- or Two-Step Process: NRC staff is considering the question of whether processes for one-step, or both one-step and two-step, licensing of reprocessing facilities should be developed. A one-step process would require an applicant to develop and essentially finalize design and operational information on a facility prior to construction, which the NRC would review and, if found to meet NRC regulatory requirements, approve. An inspection program [such as inspections, tests, analyses, and acceptance criteria (ITAAC)] would verify that the as-built facility meets the NRC-approved design and operational requirements (see Chapter 2). Staff recognizes that a two-step approach would allow the applicant flexibility in constructing the facility. Under a two-step approach, the construction permit would provide information on design and operational characteristics (i.e., design bases and principal SSCs) to define the facility and safety parameters. The applicant would then develop detailed design and operational information. Before issuing an operating license, the NRC would verify (via inspection and review) that the completed facility meets the requirements of the approved construction permit, including any potential changes identified during construction. Staff recognizes that this approach may introduce additional costs and increase the time to construct and operate the facility. This approach would require additional NRC staff resources to develop separate RGs and standard review plans to address both one-step and two-step licensing. The staff should consider, as some stakeholders suggested, whether a licensing process should be chosen based on maturity of the technology and design.

For a two-step licensing process, the requirements in 10 CFR 50.23 and 10 CFR 50.57 could be modified for use in a new regulation. A two-step licensing process for reprocessing plants could generally be structured using 10 CFR 50.33, “Contents of Applications; General Information,” and 10 CFR 50.34, “Contents of Applications; Technical Information.” This would include emergency response plans, an SAR (both preliminary and final, depending on the type of license requested), technical specifications, principal design criteria [developed from general design criteria (GDC)], and the technical qualifications of the applicant and its staff. Requirements in 10 CFR 70.22, “Contents of Applications,” regarding the possession and use of SNM should be included in a new regulation, including requirements regarding completeness of information, design, construction, testing, operations, emergency procedures, operator training, and decommissioning. Reprocessing facilities would likely involve more types and greater quantities of chemicals compared to other fuel cycle facilities, and the design of reprocessing facilities may exacerbate chemical safety concerns. Therefore, an application should be required to address chemical safety, including requirements in 10 CFR 70.65, “Additional Content of Applications,” that state, in part, that an ISA must contain “a description of the proposed quantitative standards used to assess the consequences to an individual from acute chemical exposure to licensed material or chemicals produced from licensed materials which are onsite, or expected to be onsite as described in 10 CFR 70.61(b)(4) and (c)(4).”

Scope of Reprocessing Operations: The staff considered what operations associated with a reprocessing facility should be regulated under 10 CFR Part 7x. The staff recommends that certain processes currently regulated under the existing regulations should continue to be regulated (e.g., SNF storage), provided the safety characteristics of these processes are sufficiently similar to those covered by the existing regulatory parts. The current regulatory framework ensures the safe and effective licensing of certain co-located facilities (again, assuming the characteristics are sufficiently similar) and would avoid any additional regulatory burden on existing licensees and ongoing licensing procedures.

Staff recognizes that the new fuel material resulting from reprocessing may contain fission products and other actinides in abundances that are not found in nuclear fuel materials at existing facilities; the presence of these isotopic materials increases potential hazards (by orders of magnitude), thus requiring additional safety. The amount of fuel potentially fabricated from a reprocessing facility may be significantly more than that fabricated at other NRC-regulated facilities. In addition, a technology-neutral regulation would need to consider that fuel fabrication might occur within the same building, or same hot cell, as reprocessing operations. The staff will evaluate whether existing 10 CFR Part 70 requirements would provide the most efficient and effective licensing tool and be appropriately protective for the fabrication of nuclear fuel from reprocessed material. The evaluations will include consideration of the co-located reprocessing and fuel fabrication operations, intrinsic process linkage and interactions, intrinsic hazard interactions and linkage, and (safety) protective measures needed to assure adequate protection while fabricating fuel from reprocessed materials. These evaluations may identify a threshold(s), based upon intrinsic hazards, interactions, isotopic composition, or other potential hazards, beyond which adequate assurances of safety would necessitate additional requirements under a new regulation (e.g., 10 CFR 7x). Consequently, NRC staff is continuing to develop technical and regulatory insights to determine whether fuel fabrication operations at a potential reprocessing facility can be regulated safely using existing 10 CFR Part 70 requirements, or whether these fuel fabrication operations should be regulated by specific requirements in a new 10 CFR Part 7x.

1.3.2 Criticality Requirements

In developing the regulatory basis, the staff identified the need for additional clarification of criticality requirements for reprocessing facilities.

1.3.2.1 Regulatory Problem for Criticality

10 CFR Part 7x should include regulations to account for and detect possible criticality accidents. The existing regulatory framework in 10 CFR 50.68, "Criticality Accident Requirements," and 10 CFR 70.24, "Criticality Accident Requirements," would need modification to allow for detection and monitoring of criticality incidents at reprocessing facilities, as well as the appropriate use of personnel alarm systems. At nuclear power reactors, significant amounts of SNM are generally only found in fresh and spent fuel rods. There are stringent controls in place to prevent criticality in spent fuel pools. In contrast, at an FRP, SNM would be found and handled routinely in various configurations, in addition to traditional fuel handling. Also, the reprocessing facilities handle other fissile materials (such as americium (Am) and curium (Cm) isotopes) that fall outside the definition of SNM, posing an additional criticality concern.
The variety of forms of SNM and other fissile materials and the frequency with which they are handled provide greater opportunities for an inadvertent criticality at a reprocessing facility than at an NPP.

Much of the SNM in a reprocessing plant is, however, separated from personnel by radiation shielding. An inadvertent criticality in shielded locations would not produce the immediate high radiation doses for which the criticality alarms of 10 CFR 70.24 are required to initiate evacuation in other types of facilities. The immediate consequences of a criticality event in a shielded area may be small in radiological terms but would be a loss of process or management control. It will be necessary to detect criticalities in shielded areas of a reprocessing plant to mitigate these consequences, but criticality alarms, as specified in 10 CFR 70.24, may not be necessary in all areas of an FRP, where shielding would prevent workers from receiving an excessive radiation dose.

1.3.2.2 Current Regulatory Framework for Criticality

Several regulations pertain to prevention and detection of criticality. In addition, the Commission considers criticality prevention to be a strategic outcome of the safety goals described in its Strategic Plan for Fiscal Years 2008–2013 (NUREG, 2008b). In 10 CFR 70.61(d), licensees must limit the risk of nuclear criticality accidents by ensuring that processes remain subcritical under both normal and credible abnormal conditions. This was one of the risk-informed, performance-based requirements added to the revised 10 CFR Part 70 rule in 2000, to ensure that criticality is prevented regardless of whether it would result in a dose to workers or the public (NRC, 1999b).

10 CFR 50.68 applies to NPP licensees (licensed under either 10 CFR Parts 50 or 52) and describes criticality accident requirements. An NPP licensee must comply with either the requirements in 10 CFR 70.24 or 50.68(b). 10 CFR 50.68(b)(5) states that a licensee must comply with the requirements of 10 CFR 50.68 in lieu of maintaining a monitoring system capable of detecting a criticality described in 10 CFR 70.24 if "The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass." An FRP would not be able to comply with this requirement because of the quantity of fissile material that is reprocessed.

10 CFR 70.24 requires that each licensee authorized to possess more than a specified amount of SNM maintain a criticality monitoring system "using gamma- or neutron-sensitive radiation detectors which will energize clearly audible alarm signals if accidental criticality occurs" in each area in which such material is handled, used, or stored. The regulation also specifies sensitivity requirements for these monitors and details the training that licensees must conduct in connection with criticality monitor alarms. A purpose of this section is to ensure that, if criticality were to occur during the handling of SNM, personnel would be alerted and would take appropriate action.

The criticality accident requirements in 10 CFR Part 76, "Certification of Gaseous Diffusion Plants," are similar to those of 10 CFR 70.24, in that a criticality monitoring and alarm system must be employed in all areas of the facility (10 CFR 76.89). The regulation provides specific criteria related to the amount of dose (20 rads) that the system is required to detect and the length of time in which it should do so.

1.3.2.3 Basis for Rulemaking for Criticality

To develop a new rule that is risk informed and has universal applicability to the variety of processes at an FRP that have the potential for a criticality incident, the staff suggests implementation of 10 CFR 70.24 requirements in a new regulatory part only where evacuation is appropriate. The staff recommends using American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.3, "Criticality Accident Alarm System," as guidance in determining evacuation areas (ANS, 1997). ANS (1997) suggests that "an excessive radiation dose" to personnel corresponds to an absorbed dose greater than or equal to 0.12 gray or 12 rad in free air. For heavily shielded areas, FRP licensees should be required to develop requirements for monitoring and detecting criticality, in keeping with the NRC requirement of limiting the risk of nuclear criticality accidents, which is codified in 10 CFR 70.61(d). In addition, to support this goal, the FRP licensee should apply, throughout the facility, the double contingency principle, which requires at least two "unlikely," independent, and concurrent process changes before a criticality might occur. ANSI/ANS 8.10, "Criteria for Nuclear Criticality

Safety Controls in Operations with Shielding and Confinement," allows the number of contingencies to be reduced to one if the principals of the standard are followed (ANS, 1983); therefore, the staff concludes that this guidance would not be applicable to an NRC-licensed reprocessing facility.

As Chapter 2 discusses, the NRC should also develop GDC regarding nuclear criticality safety and should address the reliability of process monitoring equipment. The reasoning behind this is that many historical criticality accidents in the United States and abroad have been the result of defective monitoring equipment. Therefore, any equipment required to monitor areas of the reprocessing plant where fissile materials exist in sufficient concentrations to cause criticality should have a reasonable expectation of reliability.

A new regulation should address the fissile radionuclides present, other than those defined as SNM, as discussed in the previous section. The NRC should develop a general requirement for placing criticality detectors wherever there is a critical mass of fissionable materials and where the potential exists for workers to be exposed to an excessive radiation dose. ANSI/ANS 8.15, "Nuclear Criticality Control of Special Actinide Elements" (ANS, 1981), provides subcritical mass limits for many isotopes that would be present at a reprocessing facility, including those of neptunium (Np), Am, and Cm. The standard does not, however, provide criticality limits for mixtures of these isotopes. During rulemaking, the NRC may need to devise a research program to develop such values. This would be necessary if applicants proposed using advanced separations, such as the isolation of Am and Cm.

1.3.2.4 Guidance Documents for Criticality

RG 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," currently endorses a number of ANSI/ANS–8 nuclear criticality safety standards, including ANSI/ANS 8.10. As discussed in the basis section, ANSI/ANS 8.10 would not apply to an NRC-licensed reprocessing facility. Therefore, if the NRC adopted the staff proposals in this document, it would need to revise this RG to reflect the conditions under which the ANSI/ANS standard is endorsed.

1.3.3 Decommissioning

In developing the regulatory basis, the staff identified the need for additional clarification on decommissioning for reprocessing facilities.

1.3.3.1 Regulatory Problem for Decommissioning

As part of its overall regulatory framework development, the NRC recognized the need for decommissioning requirements for the efficient and effective oversight of fuel reprocessing facilities. The current regulations do not address the decommissioning of reprocessing facilities.

To reflect the unique aspects of reprocessing facilities with respect to decommissioning, new regulations should incorporate 10 CFR Part 70 financial requirements, including additional cost estimates for environmental remediation. The NRC should adopt 10 CFR Part 50 requirements regarding planning to reflect the greater complexities of decommissioning reprocessing facilities than currently licensed fuel cycle facilities.

1.3.3.2 Current Regulatory Framework for Decommissioning

In 1988, the Commission amended its regulations in 10 CFR Parts 30, 40, 50, 51, 70, and 72 to establish specific technical and financial criteria for decommissioning licensed nuclear facilities (NRC, 1988):

- Acceptable decommissioning alternatives
- Plans for decommissioning
- Assurance of the availability of funds
- Environmental review requirements

The Commission's objective is to safely decommission the facility, resulting in license termination for unrestricted use or under restricted conditions. In this rulemaking, NRC determined that the acceptable alternatives for decommissioning were DECON, SAFSTOR, and ENTOMB.

DECON: The licensee has removed or decontaminated equipment, structures, and portions of the facility and site that contain radioactive contaminants to a level that permits termination of the license after cessation of operations.

SAFSTOR: The licensee has placed the facility in a safe stable condition and maintained it in that state until it is subsequently decontaminated and dismantled to levels that permit license termination. During SAFSTOR, a facility is left intact, but, in the case of an NPP, the fuel has been removed from the reactor vessel and radioactive liquids have been drained from systems and components and then processed. Radioactive decay occurs during the SAFSTOR period, thus reducing the levels of radioactivity in and on the material and potentially reducing the quantity of material that must be disposed of during decontamination and dismantlement.

ENTOMB: The licensee encases radioactive SSCs in a structurally long-lived substance, such as concrete. The entombed structure is maintained and continued surveillance is carried out until the radioactivity decays to a level that permits termination of the license. This option might be acceptable for facilities that can demonstrate that radionuclide levels will decay to unrestricted use levels in about 100 years. If using the ENTOMB method, the provisions in 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination," related to unrestricted or restricted use still apply.

In 1996, the NRC published a final rule in the *Federal Register* (FR) (NRC, 1996), amending the regulations on decommissioning procedures pertaining to nuclear power reactors affecting 10 CFR Parts 2, 50, and 51. The rulemaking recognized that the degree of regulatory oversight required during the decommissioning stage is considerably less than that required for the facility during its operating stage. The NRC amended 10 CFR 50.59, "Changes, Tests, and Experiments," to include decommissioning activities and to allow licensees to make changes to facilities undergoing decommissioning using the process described in 10 CFR 50.59. The amendment also allowed a licensee to include decommissioning activities in the FSAR. The same rulemaking introduced requirements that a power reactor licensee submit a post-shutdown decommissioning activities report (PSDAR) before or within 2 years following cessation of operations. The aim of this was, in part, to provide a mechanism for timely NRC oversight. Similar to a decommissioning plan, the PSDAR document must include the following:

- A description and schedule of planned decommissioning activities

- An estimate of the costs

- A discussion that provides means for concluding that the environmental impacts associated with the decommissioning activities will be bounded by appropriately issued environmental impact statements

The NRC will notice the receipt of the PSDAR in the FR and make the document available for public comment. In addition, the NRC will hold a public meeting near the licensee's facility to discuss the PSDAR.

10 CFR Part 50, Appendix F contains requirements for FRP licensees to provide financial qualification information for decommissioning and the removal and storage of radioactive waste. It states, "A design objective for fuel reprocessing plants shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned. Criteria for the extent of decontamination to be required upon decommissioning and license termination will be developed in consultation with competent groups," and the public will be afforded an opportunity to comment before such criteria are made effective.

10 CFR 50.33(f), "Contents of Applications; General Information," requires that information must be provided to the Commission that demonstrates that the applicant is financially qualified to carry out the activities regulated under 10 CFR Part 50 for which the license or permit is sought.

10 CFR Part 20, Subpart E contains similar requirements that would apply to FRP licensees. Regulations in 10 CFR 20.1406, "Minimization of Contamination," require license applications to describe how the design of the facility and its operations will minimize, to the extent possible, contamination of the facility and the environment and facilitate eventual decommissioning. Decommissioning of any commercial reprocessing plant must meet the criteria in 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination."

Requirements regarding decommissioning for materials licensees are in the following sections of 10 CFR Part 70: 10 CFR 70.25, "Financial Assurance and Recordkeeping for Decommissioning"; 10 CFR 70.38, "Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas"; and 10 CFR 70.9, "Completeness and Accuracy of Information." Regulations in 10 CFR 70.25 and 10 CFR 70.38 specify the requirements for certain licensees to provide financial assurance for decommissioning. Regulations in 10 CFR 70.9 address, in part, the completeness and accuracy of information provided to the NRC. A licensee is required to submit a plan in certain cases, such as where workers would be entering areas not normally occupied and where surface contamination and radiation levels are significantly higher than routinely encountered during operation [10 CFR 70.38(g)(1)].

In September 2003, the Office of Nuclear Material Safety and Safeguards consolidated and updated the policies and guidance of its decommissioning program in a three-volume NUREG series (NUREG, 2007a). This series provides guidance on planning and implementing license termination under the NRC's License Termination Rule (LTR), complying with the radiological criteria for license termination (in 10 CFR Part 20, Subpart E), and with the requirements for financial assurance and recordkeeping for decommissioning and timeliness in decommissioning materials facilities. The guidance extends NRC's risk-informed, performance-based regulatory philosophy to decommissioning.

Both 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning," and 10 CFR 70.25 require a license applicant to provide assurance of funds for decommissioning and submit a report to the NRC demonstrating this. In 10 CFR Part 50, this is called a decommissioning report. In 10 CFR Part 70, it is a decommissioning funding plan. Both allow for the use of insurance or sinking funds. Differences between 10 CFR Parts 50 and 70 include, for example, Section 50.75(f), "Reporting and Recordkeeping for Decommissioning Planning," which allows accumulation of decommissioning funds over time, and Section 70.25, "Financial Assurance and Recordkeeping for Decommissioning," which requires that the licensee provide the appropriate financial assurances for decommissioning after the license has been approved and issued, but prior to the receipt of licensed material. In addition, 10 CFR 50.75(c)(2) specifies a floor for a site-specific decommissioning cost estimate and requires that it be escalated annually according to a specific formula defined in NUREG–1307, "Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities," Revision 14, issued November 2010 (NUREG, 2010a), while 10 CFR 70.25 does not specify a floor for the site-specific decommissioning cost estimate. In the case of NPPs, Pacific Northwest National Laboratory derived cost estimates from studies that it performed for the NRC (NUREG, 1978a; NUREG, 1980). Similar studies were done on a number of fuel cycle facilities, including a reference nuclear FRP (NUREG, 1977). However, 10 CFR Part 50 did not include decommissioning cost estimates for these production facilities.

1.3.3.3 Basis for Requested Change for Decommissioning

The staff recommends that rulemaking for fuel reprocessing facilities take the form of a new regulation that incorporates information from the regulations in 10 CFR Part 50 and 10 CFR Part 70 on decommissioning, including financial assurances. The staff suggests that the NRC base the financial requirements on those in 10 CFR Part 70 rather than 10 CFR Part 50. Under 10 CFR 70.25(a), an applicant for certain fuel cycle facilities must demonstrate financial assurances at the beginning of the licensing process, whereas a licensee under 10 CFR Part 50 does not. The historical reason for allowing power reactors to build up decommissioning financing over time stems from the fact that the original power reactors belonged to public utilities. Therefore, the rate payers would be responsible for decommissioning costs if utility financing failed. Because fuel cycle facilities are private companies, they do not have this option and therefore have to demonstrate their funding ability up front. This will also be the case for reprocessing licensees.

Because there is relatively little domestic experience in decommissioning reprocessing facilities, it would not be practical to develop a fee structure like that established for NPPs. Uncertainties surrounding the precise method of reprocessing, coupled with the lack of operating fuel reprocessing facilities in the United States, make determining decommissioning costs challenging. Therefore, the staff recommends that a licensee be required to provide the NRC with a cost estimate for decommissioning activities, as is currently required in both 10 CFR 50.75(d)(2)(i) and 10 CFR 70.25(e). Because decommissioning a reprocessing plant would be site specific, the NRC staff would need to develop a standard review plan to provide guidance for evaluating the financial requirements for decommissioning. This could be challenging, because an FRP will have higher and more variable costs for decommissioning than other fuel cycle facilities, taking into consideration the possibility of fission product contamination of process equipment and the environment. NUREG–0278, "Technology, Safety, and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant," issued

October 1977 (NUREG, 1977), may provide useful insights into the challenges of Decontamination and Decommissioning (D&D) for a reprocessing facility.

The decommissioning of reprocessing plants would be similar to the decommissioning of reactors because of the large radiological inventory at each type of facility. Therefore, planning requirements for decommissioning FRPs should be similar to 10 CFR Parts 50 and 52 facilities, including the submittal of a PSDAR. There are some significant differences between NPPs and FRPs, however, because reactors are usually contaminated with induced activity, whereas reprocessing facilities are contaminated with fission products U and Pu. Consequently, decontamination methods are different for reprocessing plants. With regard to overall decommissioning and radiation levels, removal of the core of the reactor eliminates the most significant source of radiation and risk to worker and public safety. This was recognized in the 1996 rulemaking discussed in the previous section (NRC, 1996), which stated "The degree of regulatory oversight required for an NPP during its decommissioning stage is considerably less than that required during its operating stage." This is not necessarily the case for an FRP, because there is the possibility of a significant radiological dose at various parts of the plant even after operations have ceased. Residual radioactivity could exist in the liquid waste storage tanks, including contamination adhering to tank walls and internals, materials deposited on the tank bottom, and the residual solution (heel) that cannot be removed from tanks easily (Brooksbank, 1976). Licensees will have to conduct remote radiation surveys to identify residual radioactive species and determine their chemical speciation for safety and material control and accountability purposes. There is also a concern that some residual radioactive species may not be detected, leading to an unknown criticality risk. In fact, some methods of decommissioning could represent a significant risk to the health of workers and the public. In 1981, the NRC invited comments on the "Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" [later finalized as NUREG–0586, issued August 1988 (NUREG, 1988a)], which included a section on FRPs (NRC, 1981). The document compared the two decommissioning alternatives, DECON and SAFSTOR, for a reference plant based upon the Barnwell reprocessing facility. The largest radiation dose to the maximally exposed individual (MEI) from a postulated accident during DECON was the failure of the ventilation system during chemical decontamination of the high-level liquid waste tank. The estimated 60 millicuries (mCi) released to the atmosphere was postulated to cause a maximum annual dose of 15 millirem (mrem) to the lung and a 50-year dose of 160 mrem to the bone of the MEI. Therefore, the degree to which a facility would be completely dismantled would depend on a cost/risk-benefit analysis, where the cost/risk is the cost in dollars plus the cost of health effects to personnel who carry out the decontamination activities, and the benefit is the reduction of the risk to the general public and residual radioactivity in the facility.

Another significant concern in decommissioning a fuel reprocessing facility is the issue of radioactive waste management: many of the wastes generated will be HLW and contain radionuclides with half-lives considerably longer than the period of operation for a reprocessing facility. Any regulation regarding decommissioning will have to consider that, while the main reprocessing operations onsite are decommissioned, some areas may still be required for long-term storage of wastes. This is analogous to power reactors undergoing decommissioning; an NPP licensee may need to apply for an ISFSI license for ongoing storage of SNF. For example, the reactor at Humboldt Bay, currently undergoing decommissioning, applied for and was granted an ISFSI license in 2003 and 2005, respectively.

The Commission recognized the concern of long-term reprocessing waste management in 10 CFR Part 50, Appendix F, which addresses, in part, the ultimate disposal of these wastes. In the 1970s, preparation of the document "Design Objectives for Decommissioning of Nuclear

Reprocessing Facilities" (ANSI N300–1975) was industry's attempt at interpreting Appendix F for the design of nuclear reprocessing facilities (Graham, 1975). Appendix F states, in part, that "a design objective for reprocessing plants shall be to facilitate decontamination and removal of all significant radioactive waste at the time the facility is permanently decommissioned." Section 1.4 discusses the applicability and relevance of Appendix F to a modern reprocessing facility.

Part of the ANSI standard N300 contained a requirement that, during the design phase of the plant project, the applicant evaluate the levels of radiological contamination expected to be present at the time of decommissioning and identify the general procedures and equipment to be used to decontaminate the affected area. The standard also required that, at the time of construction, the applicant incorporate into the facility the special design features necessary to safely carry out the proposed decontamination.

Environmental contamination is also a concern, particularly in light of U.S. experience in reprocessing, both in the governmental and commercial arena. For example, in the case of West Valley, considerable contamination was found at the site. Radionuclides included the fission products Sr-90 and Cs-137, along with U, Pu-238, Pu-239, Pu-241, and Am-241. Substantial contamination levels have been found in many of the cells and rooms of the process building, and some contamination is present inside other facilities (DOE, 2009). Subsurface soil and groundwater contamination is widespread. The current reactor decommissioning funding requirement in 10 CFR 50.75(c) does not address the cost to clean up environmental contamination (e.g., soil, groundwater). The regulations in 10 CFR 20.1406, which require a facility to minimize contamination and facilitate decommissioning, also do not address such financial assurances. A new regulation for reprocessing facilities should require a licensee to estimate the cost of environmental remediation. This will be particularly important for a licensee proposing an aqueous separations process. The Decommission Planning Rule (NRC, 2009b) recognized the need for environmental cleanup. Under its provisions, a materials licensee must factor in subsurface contamination when estimating its decommissioning cost. The Decommission Planning rule was approved by the Commission (NRC, 2010) and issued in June 2011 (NRC, 2011).

The staff is also proposing the development of GDC for decommissioning. These would include criteria relating to facilitation of decontamination inventory limitations, and decommissioning planning. Chapter 2 (Gap 9) discusses this subject.

1.3.4 Emergency Planning Requirements

As part of developing the regulatory basis, the staff indentified the need for additional clarification on emergency planning for fuel reprocessing facilities.

1.3.4.1 Regulatory Problem for Emergency Planning

Because a spent FRP meets the definition of a "production facility," it would currently be subject to the regulations of 10 CFR Part 50, which include emergency planning requirements. However, the potential offsite impacts of an accident at an FRP can be smaller than those of an accident at a power reactor. The NRC should develop regulations for reprocessing facilities that include emergency preparedness requirements commensurate with the risks posed by accidents at reprocessing plants.

1.3.4.2 Current Regulatory Framework for Emergency Planning

10 CFR 50.34(b) requires an application for an operating license to include a discussion of the applicant's plans for coping with emergencies. 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," provides minimum requirements of an emergency plan. The contents of an emergency plan shall include organization (who responds to radiological accidents and how); assessment actions (determining the impact of the event in terms of radioactive release); activation of the emergency organization; notification procedures (local, State, and Federal agencies; public); emergency facilities and equipment; training; and recovery (when operations could be resumed at the facility). 10 CFR Part 50, Appendix E specifically mentions the emergency plan's relevance to research and test reactors and other fuel facilities licensed under 10 CFR Parts 50 and 70, respectively. It states that, for these facilities, the applicability of the appendix, including the size of an emergency planning zone (EPZ), should be decided on a case-by-case basis.

Facilities licensed under 10 CFR Part 70 may be required to develop an emergency plan. 10 CFR 70.22, "Contents of Applications," requires that an application to possess enriched U or Pu for which a criticality accident alarm system is required, or an application to possess Pu in excess of 2 Ci in unsealed form, to contain either (i) an emergency plan or (ii) an evaluation showing that the maximum dose to a member of the public offsite caused by a release of radioactive materials would not exceed 1 rem effective dose equivalent or an intake of 2 milligrams (mg) of soluble U. Emergency plans for 10 CFR Part 70 licensees must contain, for example, facility description, classification of accidents, mitigation of consequences, responsibilities, information to be communicated, safe shutdown, hazardous chemicals, types of potential accidents, detection of accidents, assessment of releases, notification and coordination, training, and exercises.

Emergency plan requirements in 10 CFR Part 70 are not as stringent as emergency plan requirements in 10 CFR Part 50. For example, emergency plan requirements in Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," establishes minimum requirements, including requirements for EPZs. EPZs typically have a radius of 10 miles for sheltering and evacuation and 50 miles for protection from ingesting contaminated food. Unlike Part 50, 10 CFR 70.22(i)(3) does not address EPZs (10 CFR 70.22(i)(3)). 10 CFR Part 50 requires offsite emergency response plans in 10 CFR 50.47(b). Offsite emergency response plans are not required by 10 CFR Part 70. As stated in NUREG–1140, accidents at fuel cycle facilities would involve only relatively small doses within a mile or two of the facility (NUREG, 1988b). NUREG–1140 states that offsite doses large enough to cause a fatality or early injury are not plausible from an accident at a fuel cycle facility.

The U.S. Environmental Protection Agency (EPA) also has protective action guides (PAGs), which help State and local authorities make decisions regarding offsite protective actions during emergencies. The PAGs suggest precautions that State and local authorities can take during an emergency to keep people from receiving an amount of radiation that might be dangerous to their health. The PAGs provide guidance only: they do not determine an acceptable level of risk for normal (nonemergency) conditions. PAGs also do not represent the boundary between safe and unsafe conditions; rather, they are the approximate levels at which the associated protective actions are justified. In the case of an airborne release, for example, the PAGs currently advise that evacuation (or, for some situations, sheltering) should normally be initiated at projected doses of 1–5 rem. Doses to emergency workers should be limited to 5 rem for general response activities, 10 rem for protecting valuable property, and 25 rem for life saving and protection of large populations (EPA, 1992). These values are based upon the sum of the external effective dose equivalent and the committed effective dose equivalent to nonpregnant adults from exposure and intake during an emergency situation. Note that, as the NRC was developing this document, the EPA was revising its PAGs, and it has not released them to the public at this time.

1.3.4.3 Stakeholder Input on Emergency Planning

NEI proposes license application requirements for emergency preparedness in the NEI white paper (NEI, 2008). NEI stated that this proposal is consistent with the emergency planning criteria of 10 CFR 70.22(i) for fuel fabrication facilities unless it is determined that there is a need for a General Emergency Classification in which case the requirements of 10 CFR Part 50, Appendix E are invoked.

When NEI developed its white paper and the emergency planning requirements therein, it focused on the likelihood of having an offsite situation. NEI endorsed the 10 CFR Part 70 approach, unless the potential exists for a general emergency classification event, defined as an offsite release that could be expected to exceed EPA's PAGs for more than the immediate site area. If a facility could have substantial offsite consequences, NEI stated that the more formal emergency plan in 10 CFR Part 50 would be more appropriate.

A representative of Sellafield, Ltd., in the United Kingdom stated at a public workshop in Albuquerque, NM, in October 2010 that emergency plans differ from one plant to the other in the United Kingdom. The plan reflects the results of the safety analysis, particularly regarding the design basis for the emergency plan. The licensee must be prepared to deal with emergencies that are outside the design basis. The plan has to be integrated with local authorities and national agencies and has to be demonstrated twice a year. Once every 3 years, a national emergency plan has to be carried out.

1.3.4.4 Basis for Requested Change on Emergency Planning

NUREG–1140, "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees" (NUREG, 1984), contains a review of accidents that could occur at fuel cycle facilities, including FRPs, and the implications for emergency planning requirements. It answers the question of whether the NRC should impose additional emergency preparedness requirements on certain fuel cycle facilities for responding to accidents that might have offsite releases of radioactive material. The document concluded that the releases during these accidents would not result in a maximum offsite individual dose commitment that exceeded PAGs. This was primarily attributed to a lack of strong driving forces and extensive containment systems. The overall conclusion from the regulatory analysis was that accidents at fuel cycle facilities and other radioactive materials licensees (reprocessing plants included) pose a small risk to the public. It concluded that serious accidents would be infrequent and would generally involve relatively small radiation doses to few people located in small areas. When it added emergency planning requirements to 10 CFR Part 72 for ISFSIs and monitored retrievable storage (MRS) facilities, the Commission determined that emergency planning requirements should be similar to those in 10 CFR Part 70 (NRC, 1995). In addition to NUREG–1140, the Commission considered the analysis of potential MRS accidents in developing emergency planning requirements (NUREG, 1984).

NUREG–1140 considered an analysis of three major accident scenarios at reprocessing facilities, taken from the “Generic Environmental Statement on the Use of Recycled Plutonium in Mixed Oxide Fuel in Light-Water Reactors” (NUREG, 1976): (i) criticality, (ii) HLW concentrator or calciner explosion, and (iii) Pu product concentrator explosions. The analysis considered the dispersal of 150 gallons of HLW solution from a waste concentrator explosion and assumed the same concentration of aerosol from an explosion in the Pu concentrator. However, NUREG–1140 did not include details of the analyses, except for the following: the analyses assume the filtration systems are not affected by the explosions and use a reduction in the fraction of material released to 3.6×10^{-8}. It estimated the material leaving the final filter at 30.5 mg of HLW solution (as an aerosol). It estimated the maximum offsite bone dose commitment that could result from this hypothetical accident to an individual at about 2.6 mrem for uranium oxide fuel.

Current NRC guidance in the accident analysis handbook (NUREG/CR–6410) indicates reduced performance of high-efficiency particulate air (HEPA) filters because of the effects of accidents and mentions a range of 95–99 percent (as compared to 99.95 percent removal when new and undamaged). Guidance used for the review of the MOX fuel fabrication Facility (NUREG–1821) limits damaged HEPA filter performance to the 99–99.9 percent range, depending on the number of HEPA filter banks in series.

Following this guidance and applying an efficiency of 99 percent for degraded HEPA filter performance, and assuming no plating out of the aerosol in the cell, the resultant offsite dose would be 7.2×10^{6} mrem. If the accident was sufficiently energetic to completely degrade HEPA filter performance or to create an unfiltered release pathway (e.g., from a potential hydrogen explosion) in which all the material was released (i.e., assuming a 100 percent failure of the filter systems and no plating out), the result would be a dose of 7.2×10^{10} mrem. Either result is greater than the 2.6 mrem cited in NUREG–1140 and exceeds the EPA’s PAG; thus, the potential for an event classified as a general emergency exists for potential accidents at reprocessing facilities.

NUREG–1140 did not use higher burnup fuel with correspondingly greater radiotoxic inventories per unit mass. The analysis uses a burnup of 33,000 megawatt-days/metric ton of initial heavy metal (MWD/MTIHM), which is approximately half of current day burnups and therefore underestimates the concentration of Pu-238 by about 75 percent. Pu-238 is the dominant contributor to inhalation dose. For SNF storage, the analysis only looks at the doses from krypton-85 and iodine-129, caused by a breach of the fuel rod plenum(s). There is no analysis for disruptive events, such as fires and explosions that can damage and potentially turn fuel constituents to aerosols (e.g., Cs, Pu, Am, and Cm isotopes). There is no mention of a possible significant inventory (1,000+ MTIHM) of spent fuel being stored in a wet pool at a reprocessing facility—much larger than the small number of assemblies used in the NUREG. There is also no mention of the other events that have occurred at reprocessing facilities (e.g., loss of filters, explosions, and fires blowing out cells and glove boxes) and waste accidents and analyses (e.g., Kyshtym and the AEC/DOE HLW tanks).

Since publication of the NUREG in 1988, other incidents at nuclear facilities have occurred that could affect the analysis. On April 6, 1993, an accident occurred at the Tomsk-7 reprocessing facility in the Russian Federation. Overpressure occurred in a tank containing U nitrate solution that caused gases to burst through the top of the tank, displacing the cover of the containment cell and leading to a forceful explosion. Release of radioactive materials to the local environment took place through the large holes in the side walls and roof of the room and through the side wall of the galley. A ventilation system also released radiation through a

150 m-high stack. The initial release of radioactive materials caused contamination near the building over an area of 1,500 square meters (m^2), and the localized release was said to be 150 gigabecquerels of beta and gamma emitters. The major release occurred through the 150-m stack to the atmosphere until the ventilation flow could be rearranged to curtail the release. The total beta and gamma activity of material released was said to be 1.5 terabecquerels. In addition, extensive contamination was found: the spread of radioactive material into the environment extended 8 kilometers (km) to the perimeter fence and an additional 20 km beyond the perimeter in a northeasterly direction. Radioactivity above background levels was detected in the ground, forested areas, and surface waters.

There were indications of a malfunction for about 60 minutes before the actual explosion. Data from the pressure transducer showed that the pressure within the installation was starting to rise. Twenty minutes before the explosion, red smoke was observed coming out of a vent tube (IAEA, 1988). Therefore, the NRC staff concludes that there would arguably have been enough time to evacuate nonessential personnel from the site when the abnormal conditions in the tank were first detected.

The most recent event at the Fukushima Daiichi plant in Japan regarding the spent fuel pools will also affect any future analyses of accident scenarios concerning such storage facilities. Loss of power, loss of water supply, temperature rise, reduced level of water, exposed fuel, and high radiation fields have been of concern at the troubled site. This is another example of an event during which time evacuation plans could be implemented.

Consequently, to develop a new rule that reflects adequate emergency planning requirements for fuel reprocessing facilities, it will be necessary to update NUREG–1140 to account for changes in process (e.g., larger fuel burnups, spent fuel inventories) and incidents that have occurred at reprocessing plants and their associated facilities. Potential radiochemical plumes have been demonstrated to go beyond 1.6 km (1 mi) (though the recorded dose rate was not very high, it does illustrate the potential for extensive radioactive contamination), and it has been shown that a reprocessing facility can have fast-moving events (a radiochemical release, typically 30–90 minutes), intermediate-length events (e.g., loss of cooling to process units, about 1–8 hours), and longer events (e.g., loss of fuel pool cooling or water, as in Japan). Chemical explosions should also be considered to account for the large quantities of hazardous process chemicals that may be stored onsite.

1.3.4.5 Alternatives to Rulemaking on Emergency Planning

Instead of developing a new rule reflecting the unique accident scenarios and inventory of a reprocessing facility, the NRC could include reprocessing facilities in the more formal emergency planning requirements of 10 CFR Part 50 (including Appendix E). This could be overly conservative, however, and would not be consistent with the Commission's policy on the use of risk-informed, performance-based regulation.

1.3.4.6 Regulatory Guides on Emergency Planning

The NRC would need to develop a new RG for emergency planning at reprocessing facilities to account for the complexities of the site (i.e., several facilities and unique safety attributes). It may be appropriate to incorporate some of the contents of both RG 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," and RG 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," into such a guide.

(See Appendix D for an assessment of the applicability of the existing suite of RGs to reprocessing facilities.)

1.3.4.7 Relevant Documents on Emergency Planning

NUREG–0396, “Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans In Support of Light Water Nuclear Power Plants,” issued 1978 (NUREG, 1978b).

NUREG–75/111 (reprint of WASH–1293), “Guide and Checklist for the Development and Evaluation of State and Local Government Radiological Emergency Response Plans in Support of Fixed Nuclear Facilities,” issued 1974 (NUREG, 1974).

1.3.5 Seismic Safety Requirements

As part of developing this draft regulatory basis, the staff identified the need for additional clarification of seismic safety requirements for reprocessing facilities.

1.3.5.1 Regulatory Problem on Seismic Safety Requirements

Seismic hazard has a major impact on both plant construction and the licensing of plant sites. The plant must be sited in a location where plant integrity can be retained during a projected earthquake of reasonable probability. Parts of the reprocessing facility that contain heavy shielding and the highest levels of radioactivity must be capable of withstanding earthquakes with no loss of containment integrity (NUREG, 2008a). Consequently, regulations must provide adequate assurance that there will not be a release of radioactive material as a result of seismic activity.

The regulation should clearly define the seismic category that the reprocessing facilities belong to and the seismic design criteria applicable to reprocessing facilities. It should also define the safety evaluation method. Chapter 2 discusses the staff recommendations regarding safety evaluations.

1.3.5.2 Existing Regulatory Framework on Seismic Safety Requirements

10 CFR 50.34(a)(3)(i) requires construction permit applications to include preliminary facility designs including the principal design criteria for the facility. 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” provides minimum requirements for the principal design criteria for water-cooled NPPs. Appendix A, “General Design Criteria for Nuclear Power Plants,” and Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” to 10 CFR Part 50 contain the detailed GDC and the earthquake engineering criteria, respectively. 10 CFR Part 50, Appendix A, GDC 2, “Design Bases for Protection Against Natural Phenomena,” requires that NPP SSCs important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. This is a fundamental regulation for the seismic safety of the most demanding seismically designed facilities (seismic Category I).

In 10 CFR 70.22, “Contents of Applications,” a licensee must include provisions for protection against natural phenomena. In 10 CFR 70.64, “Requirements for New Facilities or New Processes at Existing Facilities,” a licensee must address baseline design criteria (BDC), one of which is to provide adequate protection against natural phenomena, considering the most severe documented historical events for the site. In 10 CFR Part 72, there are geological and seismological characteristics for applications (in 10 CFR 72.102, “Geological and Seismological Characteristics for Applications before October 16, 2003, and Applications for Other Than Dry Cask Modes of Storage,” and 10 CFR 72.103, “Geological and Seismological Characteristics for Applications for Dry Cask Modes of Storage on or after October 16, 2003”). The regulations describe an acceptable generic (not site-specific) standard design earthquake ground motion (DE), which is described by an appropriate spectrum anchored at 0.25 g (acceleration due to gravity) for a site east of the Rocky Mountains that is not in areas of known seismic activity. Alternatively, a site-specific DE determination may be established as specified in 10 CFR 72.103(a)(2).

For a stationary reactor site application submitted on or after January 10, 1997, 10 CFR 100.23, “Geologic and Seismic Siting Criteria,” describes the seismic regulations, with references to 10 CFR Part 50, Appendices A and S.

1.3.5.3 Basis for Regulation Changes on Seismic Safety Requirements

This section provides the regulatory basis for seismic requirements for those SSCs important to safety, including foundations and supports, of a reprocessing facility designed as seismic Category I. Reprocessing facility operations involve highly radioactive and toxic materials in large inventories in a complicated system. A Category 1 reprocessing facility should withstand the effects of the safe-shutdown earthquake (SSE) ground motion and remain functional. The seismic design requirements, as stated in 10 CFR Part 100, Appendix A and in 10 CFR Part 50, Appendices A and S, require that the SSCs important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. As discussed next, similar requirements should be included in proposed 10 CFR Part 7x.

10 CFR Part 50, Appendix A, GDC 2 requires that SSCs important to safety be designed to withstand the effects of earthquakes without losing the capability to perform their safety functions.

This criterion should be incorporated into 10 CFR Part 7x, because the fundamental safety principle of preventing earthquakes from causing major releases of radionuclides is the same for reprocessing facilities as for NPPs. The SSCs of a reprocessing facility important to safety should be classified and designed as seismic Category I, according to the previously described GDC. The main process involves highly radioactive, toxic, and corrosive solutions or molten salts and metals, which have to be contained in vessels, pipes, or cells. This process equipment should be designed to prevent major releases of radionuclides under conditions assumed to be credible. Process equipment should provide confinement integrity for design-basis accidents and naturally occurring events, such as earthquakes and tornadoes.

The ACNW&M white paper (NUREG, 2008a) provides some further details about the design of reprocessing facilities, considering the characteristic nature of reprocessing facilities. The ACNW&M white paper states the following:

- Because reprocessing facilities consist mainly of vessels and pipes containing highly radioactive, toxic, and corrosive aqueous materials or molten salts and metals, the process equipment should be fabricated from materials that are resistant to corrosive failure and that operate very reliably. Process equipment designed to prevent major releases of radionuclides under conditions assumed to be credible was designated as being of “Q” design or Items Relied on For Safety

(IROFS). These systems must provide confinement integrity for design-basis accidents and naturally occurring events, such as earthquakes and tornadoes. In other less critical areas, the design membrane stress of the equipment had been established at 80 to 90 percent of the yield stress during a design-basis earthquake. Structural barriers are designed to contain process materials if primary equipment barriers are breached. The principal structural barriers are constructed of heavily reinforced concrete.

- The structural barriers for process equipment, generally termed “radioactive process cells,” are usually surrounded by maintenance or operating areas. The process cells where the spent fuel is chopped and dissolved and where high-level liquid wastes are concentrated have very high radiation levels. These cells were designed for remote maintenance (i.e., maintenance from outside the cell by the use of in-cell cranes, shielding windows, and manipulators). Similarly, a cell was also provided for remote packaging of radioactive wastes and for performing remote decontamination and maintenance of equipment removed from other process cells. The rest of the process cells were designed to permit direct personnel entry and contact maintenance, but only after appropriate remote decontamination has been completed to allow safe entry. These cells were designed to minimize maintenance requirements.

- The process and support equipment used in handling radioactive materials is contained in cells or glove boxes. Spent fuel assemblies are stored and transported under water in pools. The cells, glove boxes, and pools provide a barrier between the highly contaminated or radioactive environment within and the habitable environment. Cells with thick concrete shielding walls or pools with deep water cover are provided where protection is required against penetrating (gamma) radiation. Glove boxes are used to isolate radioactive material when radiation levels are low and contact operations are permitted.

The design and operation of fuel reprocessing facilities are particularly challenging compared with NPPs because of their complexity.

A reprocessing facility will require extensive and expensive operator training, a very complex plant, and diverse equipment types. Nuclear accident data show that a half dozen existing reprocessing facilities worldwide have had more accidents than a few hundred NPPs in a much shorter operating period. Therefore, the likelihood of nuclear accidents at reprocessing plants is higher than at NPPs (NUREG, 2008a, Tables 7 and 8).

All the high-level nuclear accidents, according to the International Nuclear Accident Scale (from Levels 4 to 6) have occurred at reprocessing facilities (see Chapter 2, Table 2-4). The only level accident that has not occurred at a reprocessing facility is the major accident (the highest, Level 7) at Chernobyl in the former Union of Soviet Socialist Republics. Thus, nuclear accidents at reprocessing facilities are not less severe than accidents at NPPs.

Reported nuclear accidents at reprocessing facilities worldwide are not seismically related; none of the facilities have experienced a major earthquake like the March 11, 2011, Japanese earthquake. Vessels and pipes or cells of reprocessing facilities are vulnerable to strong ground shaking from earthquakes, which causes the collapse of buildings and structures, cracks in walls and vessels, and ruptures of pipes. Secondary hazards, such as fires and gas and water releases, can follow the initial damage.

During normal operations of reprocessing facilities, the conditions required for the release and dispersal of significant quantities of radioactive materials are always present. There are many components carrying fluids at high temperatures or pressures, during normal operations or under design-basis accident conditions, to cause the release and dispersal of radioactive materials. Volatile radioactive materials are readily available for release to the environment. To withstand these conditions, the reprocessing facilities should be designed to the highest seismic standard, like NPPs.

The staff recommends that reprocessing facilities be designed to seismic Category I. 10 CFR Part 50, Appendices A and S, which contain the GDC and the earthquake engineering criteria for NPPs, respectively, should apply to reprocessing facilities. Because of the characteristics of reprocessing facilities, with vessels and pipes containing a large inventory of radioactive, toxic, and corrosive aqueous solutions or molten salts and metals, special designs such as the "Q" design should be considered or required. These "Q" systems can provide confinement integrity for design-basis accidents and naturally occurring events, such as earthquakes and tornadoes.

1.3.5.4 Regulatory Guides on Seismic Safety Requirements

Because the staff is recommending that reprocessing facilities be designed to seismic Category 1 standards, like NPPs, most of the RGs developed to support the NPP seismic regulations should also apply to the seismic regulations for reprocessing facilities. For example, RG 1.29, "Seismic Design Classification," describes an acceptable method of identification and classification of those SSCs that should be designed to withstand the SSE. RG 1.29 states that systems and components required for NPP safe shutdown, including their foundations and supports, are designated as seismic Category I and should be designed to withstand the effects of the SSE and remain functional. In addition, this guide recommends that systems, other than radioactive waste management systems, that contain, or may contain, radioactive material and the postulated failure of which would result in potential offsite whole body (or equivalent) doses that are more than 0.005 sieverts (Sv) (0.5 rem), should also be classified as seismic Category I. Following the guidance in RG 1.29 can help ensure that, by designing the SSCs identified in the guide to withstand the effects of an SSE, a designed-in-safety margin is provided for bringing the reactor to a safe, shutdown condition while also reducing potential offsite doses from seismic events. This guidance and the recommendations therein should be applicable to reprocessing facilities. The staff has also reviewed the following guidance for applicability (see Appendix 6):

- RG 1.29: "Seismic Design Classification"
- RG 1.61: "Damping Values for Seismic Design of Nuclear Power Plants"
- RG 1.100: "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants"
- RG 1.166: "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions"
- RG 1.167: "Restart of a Nuclear Power Plant Shut Down by a Seismic Event"

- RG 1.198: “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites”

- RG 1.208: “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion”

1.3.5.5 Stakeholder Interactions on Seismic Safety Requirements

In its white paper (NEI, 2008), NEI does not propose specific regulations pertaining to seismic requirements. Instead, it developed BDC, derived from 10 CFR 70.64(a)(2) and 10 CFR 72.122(b), to ensure the protection of the facility, and hence the public, from potentially large source term releases from reprocessing facilities that could result from the occurrence of natural phenomena. It also developed these criteria to ensure that the design of IROFS or systems or components that support IROFS consider the impact of natural phenomena that are known to exist at the reprocessing facility’s site location. The criteria list natural phenomena to be addressed, including earthquakes.

1.3.6 Fire Protection

As part of developing the regulatory basis, the staff indentified the need for additional clarification on fire protection for fuel reprocessing facilities.

1.3.6.1 Regulatory Problem for Fire Protection

A potential risk to the public health and safety and plant personnel at a nuclear fuel reprocessing facility is the release and dispersal of radioactive and related chemical materials from a fire or explosion. The NRC requires fire protection programs for these facilities to prevent, detect, extinguish, limit, or control fires and explosions and their concomitant hazards and damaging effects.

SSCs important to safety should be designed and located so they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Heat-resistant and noncombustible materials should be used wherever practical throughout the facility, particularly in locations vital (i) to the functioning of confinement barriers and systems, (ii) to methods of controlling radioactive materials within a facility, and (iii) to the maintenance of safety control functions. The adverse effects of fires and explosions on SSCs important to safety can be minimized by providing systems with sufficient capacity and capability for detecting and suppressing explosions and fires and for transmitting alarms to one or more central control areas. Adverse effects may result from normal operation, malfunction, or failure of a fire protection system. It is important to recognize these potentially adverse effects and eliminate or mitigate them through proper design and installation.

The principal purpose of a fire protection program for a nuclear fuel reprocessing facility should be to ensure adequate protection of public health and safety and the environment from the potentially adverse radiological and chemical consequences of a fire.

1.3.6.2 Existing Regulatory Framework on Fire Protection

The regulations currently applicable to a nuclear fuel reprocessing facility are in 10 CFR 50.48, “Fire Protection”; this also references 10 CFR Part 50, Appendix A, GDC 3, “Fire Protection,” which states

> Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

10 CFR 50.48(a) provides that a fire protection plan must describe the overall fire protection program for the facility; identify the various positions within the licensee's organization that are responsible for the program; state the authorities that are delegated to each of these positions to implement those responsibilities; and outline the plans for fire protection, fire detection, and suppression capability, as well as the limitation of fire damage. Additionally, the fire protection plan must describe specific features necessary to implement the referenced fire protection program, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities; automatic and manually operated fire detection and suppression systems; and the means to limit fire damage to SSCs important to safety so that the capability to shut down the plant safely is ensured.

Although the NRC developed these regulations primarily to address NPPs, it is staff's opinion that they are generically written and therefore can be viewed as establishing a minimum level of fire protection for nuclear fuel reprocessing facilities. However, as described previously, portions of these regulations are not directly applicable to nuclear fuel reprocessing, and there are several hazards specific to reprocessing that these regulations do not address.

Each holder of an operating license issued under Part 50 or 52 must have a fire protection plan that satisfies 10 CFR Part 50, Appendix A, GDC 3. The definition of "structures, systems, and components important to safety," as used in GDC 3, does not clearly state that it is applicable to reprocessing facilities. Additionally, 10 CFR 50.48 does not discuss hazards specific to reprocessing facilities that could lead to the release and dispersal of radioactive and related chemical materials caused by a fire or explosion.

Based on the previous information, staff concludes that 10 CFR 50.48 lacks the detail needed to ensure adequate protection of public health and safety and the environment from the potentially adverse radiological and chemical consequences caused by fire at a nuclear fuel reprocessing facility.

1.3.6.3 Basis for Requested Change on Fire Protection

Several regulations for related nuclear facilities can be used as models for developing new regulations for fuel reprocessing facilities. Fire protection is a key element in the safety of (i) existing nuclear power reactors [10 CFR 50.48, GDC 3; 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979" and 10 CFR 50.54(hh)(2)]; (ii) new nuclear power reactors (10 CFR 52.46, 52.79, 52.80, and 52.137); and (iii) nuclear fuel cycle facilities [10 CFR 70.4, "Definitions" (definition of ISA), 10 CFR 70.62(c)(2), and 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities"]. Considering the potential impact of a fire on public health and safety, fire protection should be a key element in the proposed rulemaking for the design, construction, and operation of reprocessing facilities.

The referenced regulations vary in content from performance-based requirements, with supporting regulatory guidance, to specific deterministic requirements. For example, 10 CFR 70.64(a)(3) states that "the design must provide for adequate protection against fires and explosions," while the acceptance criteria for fire protection are in the standard review plans for fuel cycle facilities (NUREG, 2010b) and the MOX fuel fabrication facility (NUREG, 2000). In contrast, 10 CFR Part 50, Appendix R, Section III.I requires specific instruction, practice, and recordkeeping for fire brigades.

The staff recommends that the fire protection portion of the reprocessing regulation be similar to the performance-based option in 10 CFR 50.48(c). National Fire Protection Standard (NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," is a performance-based standard that describes the methodology for applying performance-based requirements and fundamental fire protection program design and elements, as well as for determining fire protection systems and features, for all phases of plant operation, including decommissioning and permanent shutdown. It provides for the establishment of a minimum set of fire protection requirements but allows licensees to use performance-based or deterministic approaches to meet performance criteria. Requirements in 10 CFR 50.48(c) provide for licensees to use this standard to meet the fire protection requirements for nuclear power reactors, with specified exceptions, modifications, and supplementation.

NFPA 801, "Standard for Fire Protection for Facilities Handling Radioactive Materials," should be incorporated into the reprocessing regulation, just as NFPA 805 is incorporated into the reactor regulations. NFPA 801 is a performance-based standard that describes the methodology for the application of fundamental fire protection program design and elements, determination of fire protection systems and facility features, and evaluation of special nuclear hazards (including those at reprocessing facilities) for all phases of plant operation, including decommissioning and permanent shutdown. It provides for the establishment of a minimum set of fire protection requirements but allows the use of performance-based or deterministic approaches to meet performance criteria. A vital element of NFPA 801 is the fire hazards analysis (FHA), which must be initiated at the beginning of the design process, or when configuration changes are made, to ensure that the fire protection requirements of the standard have been evaluated. The FHA identifies fire hazards that are directly applicable to risk-evaluation methods, such as the ISA, described in 10 CFR 70.62(c).

Furthermore, NFPA 801 references additional NFPA codes and standards that provide industry-accepted methods for fire protection safety for the design, testing, inspection, and maintenance of related systems and equipment (examples of these codes are found in Appendix 4). Following NFPA codes and standards is a method to meet many of the acceptance criteria specified in NUREG–1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility—Final Report," Revision 1, issued May 2010 (NUREG, 2010b), and NUREG–1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," issued August 2000 (NUREG, 2000).

The staff recommends supplementing NFPA 801 with language/requirement(s) similar to 10 CFR 70.64(b), which requires defense-in-depth practices to be used by incorporating (i) selection of engineered controls over administrative controls and (ii) features that enhance safety by reducing challenges to IROFS. Combustible loading controls are commonly credited in the fire protection of fuel cycle facilities. Although this is an administrative control that can provide a significant level of safety, it lacks the robustness of an engineered control, like a sprinkler system. The defense-in-depth practices and preference of engineered controls over administrative controls should be viewed as a vital component in the proposed fire protection regulations for reprocessing facilities.

The staff also recommends supplementing NFPA 801 with the acceptance criteria from NUREG–1718, Section 7.4.3.2. Many of these acceptance criteria were taken from DOE standards and orders for facilities containing Pu. Although NFPA 801 briefly discusses reprocessing facilities, the acceptance criteria provided in NUREG–1718, Section 7.4.3.2 for a MOX fuel fabrication facility include additional hazards commensurate with those found in a nuclear reprocessing facility.

The staff recommends providing the guidelines for an FHA found in NUREG–1718, Appendix D within the regulatory guidance document produced for the reprocessing rulemaking. The staff determined that the overall requirements that NFPA 801 establishes for an FHA are sufficient; however, the staff suggests that several details discussed in NUREG–1718 elaborate on the requirements in NFPA 801 and should be incorporated into the Standard Review Plan for reprocessing facilities. Note that the NRC will need to develop a separate standard review plan for reprocessing facilities, which will include acceptance criteria pertaining to fire protection.

Additionally, some process-specific fire hazards may need regulatory guidance beyond the sources listed. Other hazards may require new guidance. The staff recommends consideration of specific fire hazards during the development of regulatory guidance.

1.3.7 Reporting Requirements

Both 10 CFR Parts 50 and 70 contain reporting requirements, and both require the licensee to notify the NRC in case of an emergency. Under 10 CFR 50.73, “Licensee Event Report System,” licensees must file an event report with the NRC within 60 days of an incident. This subpart describes incidents covered under this requirement, many of which are specific to nuclear power reactors. Some would be applicable to FRPs, such as safety threat, required emergency power system, plant shutdown, and radioactive release. Consequently, aspects of these regulations should be considered for inclusion in 10 CFR Part 7x. 10 CFR 50.72, “Immediate Notification Requirements for Operating Nuclear Power Reactors,” and 10 CFR 50.74, “Notification of Change in Operator or Senior Operator Status,” should also be considered for inclusion in Part 7x as these regulations also address reporting requirements and would be applicable to FRPs. 10 CFR 70.50, “Reporting Requirements,” requires that licensees notify NRC within 4 hours of the discovery of an incident that leads to uncontrolled releases of radioactive material. This should be followed by a written report within 30 days; these regulations should also be considered in 10 CFR Part 7x.

1.3.8 Transfer of Special Nuclear Material

An FRP will require the receipt of SNF. The SNM that is generated through reprocessing operations will be used to produce fresh reactor fuel (e.g., MOX fuel). If a fuel fabrication plant is not co-located with a reprocessing facility, regulations would be needed for the shipment of SNM to a fuel fabrication plant. Staff recommends that 10 CFR 70.42, “Transfer of Special Nuclear Material,” be used, in modified form, in a new regulation pertaining to FRPs.

1.4 Modifications to 10 CFR Part 50

1.4.1 Regulatory Problem

Several paragraphs and appendices in 10 CFR Part 50 make specific reference to "fuel reprocessing plants." These provisions, if not modified, may cause confusion and regulatory uncertainty if the NRC issues a new part specifically written for the licensing of commercial reprocessing plants.

1.4.2 Basis for Requested Change

The staff recommends that several references to FRP be removed from the 10 CFR Part 50 regulations to avoid confusion and uncertainty regarding whether to use 10 CFR Parts 50 or 7x. The staff identified the following paragraphs of 10 CFR Part 50 that contain references to FRPs and has made the following recommendations with regard to their removal:

- 10 CFR 50.30(f): The term "fuel reprocessing plant" could be removed with no effect on the overall rule regarding the requirement for an environmental report for other facilities that are licensed under this part.

- 10 CFR 50.34(a)(7); (b)(6)(ii): These paragraphs refer to Appendix B in terms of how the licensee should satisfy its requirements. Because none of the requirements pertain specifically to an FRP, references to fuel reprocessing facilities could be removed from the aforementioned sections in 10 CFR Part 50.34.

- 10 CFR 50.36(c)(1)(i)(B): This section requires safety limits for FRPs. The staff may consider adapting the contents of this paragraph for use in 10 CFR Part 7x as a condition for technical specifications. The staff also recommends removing this paragraph from 10 CFR Part 50 to avoid confusion and repetition in the regulations.

- 10 CFR 50.36(c)(1)(ii)(B): This section requires limiting control settings for FRPs. The staff may consider adapting the contents of this paragraph for use in 10 CFR Part 7x as a condition for technical specifications. The staff also recommends removing this paragraph from 10 CFR Part 50 to avoid confusion and repetition in the regulations.

- 10 CFR 50.36(c)(2)(i): References to FRPs could be removed with no effect on the rule as it pertains to nuclear power reactors. The NRC should consider replacing the term "fuel reprocessing plant" with "production facility."

- 10 CFR 50.54(ee)(3): This paragraph could be left in 10 CFR Part 50, as it does not pertain to fuel reprocessing but rather to the receipt of material from reprocessing.

With regard to 10 CFR Part 50 Appendices B and F, the latter of which the NRC developed solely for the regulation of FRPs, the staff recommends the approaches described next.

The current regulations in 10 CFR 50.34(a) require an applicant for an FRP to include in the preliminary SAR a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the SSCs of the facility. Under 10 CFR 50.34(b), every applicant for an operating license for such a facility must include in the FSAR information pertaining to the managerial and administrative controls to be used to ensure safe operation.

The AEC issued these requirements in 1971, when it recognized that, like a nuclear power reactor, "fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public" (AEC,1971a). The purpose of the amendments was to provide "…explicit quality assurance requirements for the design, construction, and operation of these structures, systems, and components by making Appendix B [to 10 CFR Part 50] applicable to fuel reprocessing plants" (AEC, 1971a).

The staff recommends that this appendix remain applicable to FRPs and be referenced in the new rule that is proposed to address these facilities. There is already precedence for this: 10 CFR 70.22(f), which states

> Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain, in addition to the other information required by this section, a description of the plant site, a description and safety assessment of the design bases of the principal structure, systems, and components of the plant, including provisions for protection against natural phenomena, and a description of the quality assurance program to be applied to the design, fabrication, construction, testing and operation of the structures, systems, and components of the plant.

The footnote at the end of this requirement states that "The description of the quality assurance program should include a discussion of how the criteria in Appendix B of Part 50 of this chapter will be met."

In 10 CFR Part 70, the reference to 10 CFR Part 50, Appendix B demonstrates that the criteria therein are generic enough to apply to any fuel cycle facility and are suitably technology neutral. The requirement was also recognized as being appropriate for a risk-informed, performance-based regulatory structure for future licensing of NPPs (NUREG, 2007b).

The current regulations in 10 CFR Part 50, Appendix F address onsite storage of both liquid and solid HLW, siting and HLW disposal, waste form criteria, financial qualifications, design objectives to facilitate D&D, and currently licensed reprocessing facilities (of which there are none in the United States). However, the waste management practices described in Appendix F have been supplemented or replaced over time with more general radioactive waste management and disposal practices described elsewhere in the Commission's regulations; specifically, in 10 CFR Parts 50, 20, and 61, "Licensing Requirements for Land Disposal of Radioactive Waste," and 10 CFR Part 63, "Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada."

The staff recommends removal of Appendix F, as it has limited applicability to modern reprocessing facilities. For example, paragraph (6) was added to the policy statement as a backfit provision to ensure that the facilities operating at the time Appendix F was issued would have their licenses appropriately conditioned to carry out the stated policy. As there are no commercial reprocessing facilities currently operating and paragraph (6) would not apply to new facilities, this requirement is not needed. The policy on the storage time limits for both liquid and solid HLW is not consistent with a risk-informed, performance-based regulatory structure. The requirements for solidification of HLW and its consequent storage before shipping to a Federal repository are discussed in Chapters 2 and 3 (under GDC and storage of HLW, respectively) of this document. Requirements addressing the performance of waste packages at a Federal repository are in 10 CFR Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories," and 10 CFR Part 63. Therefore, the staff recommends that these aspects of paragraph (2) not be incorporated into a new regulation.

Paragraphs (1), (2), and (3) contain statements regarding ownership, payment by the licensee to the Federal Government for receipt of HLW, and ultimate disposal, respectively. Because these statements are derived from statute (i.e., the AEA and the Nuclear Waste Policy Act of 1982 (NWPA) (as amended)), it is not necessary to incorporate them into a new rule.

The 5- and 10-year limits with regard to onsite storage of liquid and solid HLW, respectively, were imposed in the interest of minimizing any potential hazard to the public health and safety. As previously stated, requirements in paragraph (2) stipulate times that HLW can be stored onsite. When the NRC issued Appendix F, it did not regard storage of liquid HLW in tanks as constituting an acceptable method of long-term storage. The importance of solidifying waste was underlined in the FR notice that accompanied the final rule (10 CFR Part 50, Appendix F). It stated that "…wastes in liquid form offer a much more serious potential for dispersal in the environment in the event of an accident, no matter how unlikely such an accident may be" (AEC, 1970). The staff's position is that this statement is still true. Experience with tank storage of liquid HLW at DOE sites, in particular, has highlighted the problems that can be encountered with this type of storage; namely, deterioration and consequent leakage. Liquid wastes have a more serious potential for dispersal in the environment than solid wastes and would prove more difficult to recover. As stated in the same FR notice (AEC, 1970), "Tank storage requires extensive surveillance, and often requires mechanical cooling apparatus to be functioning continuously"; loss of cooling would present a severe safety concern. However, the FR notice did acknowledge that "Some period of in-tank storage of liquid wastes at the reprocessing plant site may be required for cooling purposes depending upon the solidification process to be used." The staff agrees with this assessment but proposes that, instead of using the 5 years stipulated in the current regulation, requirements for solidification be adapted into the technical specification requirements (see Chapter 2). Liquid wastes will not be a consequence of electrochemical processing. However, HLW will be generated in the form of molten salts (NAS, 2000). Although the concerns with molten waste are not equivalent to those of liquid wastes in terms of potential mobility through the environment, a licensee should solidify these wastes in a timely fashion to immobilize the HLW.

Paragraph (4) states the following:

> A design objective for fuel reprocessing plants shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned. Criteria for the extent of decontamination to be required upon decommissioning and license termination will be developed in consultation with competent groups. Opportunity will be afforded for public comment before such criteria are made effective.

Although 10 CFR 20.1406 addresses the first aspect of this paragraph and criteria for the extent of decontamination were never developed, this requirement should be considered for inclusion in a new regulation for FRP decommissioning requirements or GDC. Similarly, paragraph (5), which requires that an applicant demonstrate its financial qualification, should be integrated into an overall rule relating to decommissioning (see Section 1.3.3 on decommissioning).

The NRC has received stakeholder input on the applicability of Appendix F. The NEI white paper (NEI, 2008) calls for the removal of Appendix F from the CFR. However, it does suggest adapting certain provisions from the appendix for the BDC in proposed rule 10 CFR 7x:

- *Criterion 4: Site Selection*. The provision concerning private land is in Appendix F, paragraph (1).

- *Criterion 14: Inventory Limitation*. This criterion is in Appendix F, paragraph (2). It ensures that liquid waste products will not be allowed to accumulate onsite beyond a reasonable inventory, defined in the criterion as that waste which is 5 years of age or less.

1.5 Guidance Documents

A number of active RGs developed in the 1970s pertain to FRPs. They address various topics, including technical specifications, offgas systems, and SARs. During development of the regulatory framework, these will have to be updated, combined, or withdrawn. Appendix C contains the full list of the RGs.

The staff reviewed the existing suite of RGs to assess their relevancy, as they are or as a source of information for reprocessing-specific RGs. Appendix D lists these findings and provides the title of the RG, the gap to which it relates, and its applicability.

The staff also recommends development of new guidance for fuel reprocessing facilities to aid licensees and to support the staff's review of a facility. These could include the following:

- *A Standard Review Plan for an FRP*: The NRC should use existing NUREGs, such as NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"; NUREG–1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility"; and NUREG–1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," in developing the SRP.

- *Regulatory Guidance Concerning Risk Assessment of a Reprocessing Facility*: The NRC should quantify more thoroughly and realistically those systems with very high risk/consequence/material-at-risk event sequences and systems important to safety. Guidance is needed on how to perform both highly quantified and qualitative risk assessments. The staff should consider using existing regulatory guidance (e.g., RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance"; IAEA–TECDOC–1267, "Procedures for Conducting Probabilistic Safety Assessment for Non-Reactor Nuclear Facilities") as a basis for developing a guide.

- *Regulatory Guidance on Fire Protection:* Some process-specific fire hazards may need additional regulatory guidance. The staff recommends considering specific fire hazards during the production of regulatory guidance.

- *Regulatory Guidance on the Environmental Qualification Process*: The environmental qualification process should show that IROFS (and possibly other SSCs) will perform their safety functions during their operating life while exposed to normal operating environmental conditions and while exposed to the hostile environments encountered during accident scenarios.

- *Technology-Specific Guidance*: This may be necessary for issues associated with different reprocessing technologies.

This list is not exhaustive. The need for RGs pertaining to the individual gap issues is addressed in the appropriate sections of this chapter and subsequent chapters. The staff should also consider the need to develop separate RGs for the different reprocessing technologies.

In addition to the RGs identified previously, a number of NUREG guidance documents, related to both NPPs and fuel cycle facilities (see Appendix E), may provide useful information on which to base similar NUREGs for reprocessing facilities.

1.6 Definitions

1.6.1 Reprocessing

Existing regulations in 10 CFR Parts 20, 50, 51, 60, 63, 70, and 72 use the term “reprocessing.” However, this term is not defined in any 10 CFR part. The term is also not defined in applicable laws, although “reprocessing” is used in, for example, the AEA and the NWPA. In SECY–08–0134, (NRC, 2008a), the staff stated that one of “(t)he most significant issues identified thus far related to the regulation of a reprocessing facility, where the regulations may need to be amended or clarified (by guidance) to effectively and efficiently process an application for a spent fuel reprocessing facility…” is the “[d]efinition of reprocessing.” This was reiterated in the subsequent Commission paper (NRC, 2009a), which identified the lack of definitions for certain reprocessing-related terms as a significant gap in establishing an effective and efficient regulatory framework for reprocessing (Gap 6).

The need to include a definition for reprocessing stems from the classification of a fuel reprocessing facility as a “production facility,” as defined by the AEA and 10 CFR Part 50. Currently, the NRC would regulate a production facility—and therefore an FRP—under 10 CFR Part 50. Establishing definitions for “reprocessing” is important to prevent any regulatory confusion.

The NRC is considering three definitions of “reprocessing” for inclusion in a new regulation:

(1) *Staff Proposal*: The separation of SNF into its constituent components of isotopes of U, fission products, and transuranic (TRU) nuclides by aqueous and nonaqueous chemical separation processes for the purpose of recovering fissile and fertile material. (This definition encompasses the types of materials that would be produced in reprocessing and the varying methods of separation that have been proposed.)

(2) *IAEA Safety Glossary (2007 Edition):* A process or operation, the purpose of which is to extract radioactive isotopes from spent fuel for further use.

(3) The NRC and its predecessor, the AEC, considered amendments to 10 CFR Part 50 during the 1970s and early 1980s that would add GDC for FRPs (39 FR 26293; July 10, 1974). Included in the proposed Appendix P was the following definition for FRP:

> A Fuel Reprocessing Plant means the structures, systems, and components required for the separation, recovery, storage, and handling of fissile and fertile nuclear material, byproducts, and waste from irradiated nuclear fuels or materials, and includes those structures and protection systems or components required to provide reasonable assurance that the plant can be operated without undue risk to the health and safety of the public.

In 10 CFR Part 110, "Export and Import of Nuclear Equipment and Material," Appendix I, "Illustrative List of Reprocessing Plant Components Under NRC Export Licensing Authority," a note provides the following description of reprocessing, with reference in particular to the PUREX process:

> Note—Reprocessing irradiated nuclear fuel separates plutonium and uranium from intensely radioactive fission products and other transuranic elements. Different technical processes can accomplish this separation. However, over the years Purex has become the most commonly used and accepted process. Purex involves the dissolution of irradiated nuclear fuel in nitric acid, followed by separation of the uranium, plutonium, and fission products by solvent extraction using a mixture of tributyl phosphate in an organic diluent.

1.6.2 Modification of Existing Definitions

The staff recommends changes to other definitions for use in a new 10 CFR Part 7x, including the definition for "controls." A definition for this term appears in both 10 CFR Parts 50 and 55. The definition in 10 CFR Part 55 specifies that "Controls when used with respect to nuclear reactors means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor." 10 CFR Part 50 also uses this definition but further defines the term with regard to nonpower reactors: "Controls when used with respect to any other facility means apparatus and mechanisms, the manipulation of which could affect the chemical, physical, metallurgical, or nuclear process of the facility in such a manner as to affect the protection of health and safety against radiation." The NRC removed the second definition from 10 CFR Part 55 in a 1987 rulemaking that deleted references to production facilities, as it determined that there were no operators at currently licensed production facilities (NRC, 1987).

NEI, in its white paper, proposed the following definition: "Controls means the apparatus and mechanisms the manipulation of which affects the prevention or mitigation of high-consequence events, as defined in §7x.32 involving fission product releases to an individual outside the controlled area." For use in the regulations, the staff recommends the following modified version of this definition: "Controls means the apparatus and mechanisms the manipulation of which affects the prevention or mitigation of very high consequence events."

This reflects the staff recommendation that the NRC only license those personnel who manipulate controls and apparatus and mechanisms of systems that are required to prevent and mitigate very high consequence accident sequences (see Chapter 2).

1.6.3 Proposed Definitions for Inclusion in 10 CFR Part 7x

Rulemaking for a reprocessing facility also presents an opportunity to introduce reprocessing-related terms into the regulations that are not currently defined. In addition to "reprocessing," the NRC should consider including the following definitions in the regulation during rulemaking:

- *Recycling*: The systematic life cycle process of (i) reprocessing SNF into its constituent isotopic components; (ii) fabrication of fresh fuels containing plutonium, minor actinides, and possibly some fission products; (iii) management of solid, liquid, and gaseous waste; and (iv) storage of spent fuel and wastes.

- *Pyroprocessing*: A nonaqueous reprocessing process in which spent fuel is subjected to high temperatures [typically over 600 °C (equivalent)] to facilitate physical or chemical processes for the purpose of separating and recovering fissile and fertile materials.

- *Pyrochemical/Electrochemical Processing*: A high-temperature chemical operation involving selective reduction and oxidation in molten salts or metals to recover nuclear materials.

- *Conditioning*: Conditioning involves transforming radioactive waste into a form suitable for handling, transportation, storage, and disposal. This may include immobilizing radioactive waste, placing waste into containers, and providing additional packaging. Common immobilization methods include solidification of low-level waste and intermediate-level liquid radioactive waste in cement, for example, and vitrification of HLW in a glass matrix. Immobilized waste may be placed in steel drums or other engineered containers to create a waste package.

- *Very High Consequence Event*: The set of high consequence events defined in 10 CFR 7x.61 that have the potential for consequences which significantly exceed the high-consequence thresholds or have steep dose and mass curves with the potential for uncertainties that significantly exceed the high-consequence thresholds, such as the following:

    - Fission products, reactor-grade Pu, and TRU isotopes have dose conversion factors that are orders of magnitude greater than low-enriched U materials. In addition, small changes and uncertainties would have relatively large changes in consequences, and, thus, extra scrutiny is needed.

    - U can have a fission product or TRU content that exceeds or otherwise does not meet current enrichment feed standards or fresh low-enriched U fuel criteria. Again, such materials have dose conversion factors significantly higher than materials meeting the purity criteria.

    - Facility and chemical hazards can affect the safety or safeguards of licensed radioactive materials because of the unique designs of reprocessing facilities (e.g., impeded access and egress, labyrinthine designs, negative pressures, airlocks). For example, a toxic chemical release to the outside of a reprocessing facility could be sucked back in to the facility because of the negative pressures from exhaust systems used to prevent radionuclide contamination (e.g., heating, ventilation, and air conditioning systems and HEPA filters).

- Multiple receptors in an accident sequence could create high-consequence events.

- Exposure of the individual outside the controlled area boundary to radiation doses exceeding 100 rem or chemical levels could endanger life.

- *Reactor-Grade Pu:* Pu present in the SNF from nuclear reactors used to generate commercial power. Such SNF usually has a high burnup, which results in a fissile Pu percentage below 90 percent.

- *Item Supporting Safety (ISS)*: An SSC required to meet ALARA or as low as reasonably practicable requirements.

- *Hazardous Chemicals Regulated by the NRC*: Chemicals with hazardous properties that contain licensed radioactive materials, that are produced from licensed radioactive materials, and that affect the safety and safeguards of licensed radioactive materials.

1.6.4 Definition of High-Level Waste

The staff considered the need to develop a definition of high-level radioactive waste (HLW) for inclusion in a new 10 CFR Part 7x that addresses reprocessing. This definition would include many of the diverse waste streams that would result from both aqueous and nonaqueous separation processes.

The definition in the NWPA states the following:

The term “high-level radioactive waste” means

(a) The highly radioactive material resulting from the reprocessing of SNF, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and

(b) Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.

Staff concluded that, at this time, a definition in 10 CFR Part 7x is not needed. Staff may need to reassess this conclusion during development of the final regulatory basis or during rulemaking.

1.6.5 Existing Definitions in the Code of Federal Regulations

The NRC is considering including definitions from existing CFR parts (with appropriate modifications highlighted) in a new 10 CFR Part 7x (see Appendix H).

1.6.6 Stakeholder Input

In its white paper, NEI suggested including a number of definitions in a new 10 CFR Part 7x dedicated to the licensing of a fuel reprocessing/recycling plant. These include defining HLW, replacing the word reprocessing with recycling, and clarifying the phrase from the NWPA, "liquid waste produced directly in recycling," by adding aspects of 10 CFR Part 60:

> HLW is the highly radioactive material resulting from recycling of spent nuclear fuel, including liquid wastes produced directly in recycling (i.e., liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent) and any solid material derived from such liquid waste that contains fission products in sufficient concentrations. HLW does not include Waste Incidental to Reprocessing (WIR).

A definition of "waste incidental to recycling" was also included in the NEI white paper to clarify what was not HLW. The definition came from the Commission's policy statement approving the NRC's LTR as decommissioning criteria for the West Valley Demonstration Project (NRC, 2002) and The Ronald Reagan National Defense Authorization Act (NDAA) for Fiscal Year 2005, Section 3116:

> Waste material resulting from recycling of spent nuclear fuel, including liquid wastes produced directly in recycling and any solid material derived from such liquid waste that contains fission products that is not so highly radioactive or contains insufficient concentrations of fission products to be classified as HLW. Such waste is not so highly radioactive or of sufficient concentration if it (1) has been processed to remove key radionuclides to the maximum extent that is technically and economically practical, and (2) either meets Class C concentrations under 10 CFR Part 61 or will meet the performance objectives in 10 CFR Part 61, Subpart C if disposed of in a near surface disposal site based on a site specific performance assessment. This definition does not relieve the Department of Energy from its responsibility for the disposal of radioactive material which is greater than Class C under the Low-Level Radioactive Waste Policy Act of 1985.

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2 SAFETY AND RISK, AND LICENSING CONSIDERATIONS (GAPS 5, 7, 9, 10, and 11)

2.1 Introduction

The U.S. Nuclear Regulatory Commission's (NRC's) "Strategic Plan Fiscal Years 2008–2013" (NRC, 2008c) states the mission of the NRC is to license and regulate the nation's civilian use of byproduct, source, and special nuclear materials (SNM) to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment. The Strategic Plan has two principal goals: ensuring adequate protection of public health and safety and the environment and ensuring adequate protection in the secure use and management of radioactive materials. Appropriate safety, risk, and licensing approaches for operation and decommissioning of reprocessing facilities are required to meet the NRC's mission and the principal goals stated in the Strategic Plan.

This chapter provides the regulatory bases and the NRC staff's proposed approaches for addressing key safety, risk, and licensing issues. These are organized using the gap designations identified in SECY–09–0082, "Update on Reprocessing Regulatory Framework—Summary of Gap Analysis" dated May 28, 2009 (NRC, 2009a), as follows:

(1) Safety and risk considerations associated with structures, systems, and components (SSCs)—Gaps 5, 11, and 9

(2) Safety and risk considerations associated with operators—training, testing, and licensing requirements—Gap 7

(3) Safety considerations associated with one-step licensing—Gap 10

Gap 5 involves the development of safety requirements for adequately identifying and controlling radiological and NRC-regulated chemical hazards and accidents at a reprocessing facility and minimizing any resulting risks to the public, workers, and the environment. Gap 11 involves the need for requiring technical specifications that provide the overall design and operational envelope for the safe operation of reprocessing facilities. Technical specifications identify safety limits (SLs), limiting control settings (LCSs), limiting conditions of operations, and surveillance and design requirements. As mentioned in Chapter 1 (Section 1.2), a reprocessing facilities meets the definition of "production facility," as defined in Section 11 of the Atomic Energy Act of 1954 (as amended) (AEA) and 10 CFR 50.2. Thus, the AEA requires reprocessing facilities to have technical specifications [AEA Section 182(a)]. Gap 9 identifies the development of general design criteria (GDC) for reprocessing facilities. GDC represent minimum design requirements at a facility that support or enhance safety, usually by providing defense in depth. Gap 7 identifies the need for regulations for the training, testing, and licensing of the operators at reprocessing facilities, as required by Section 107 of the AEA. Licensed operators are those operators whose actions can have a significant impact on safe operation of reprocessing facilities, in a manner analogous to licensed operators at nuclear power reactors. Gap 10 identifies the need for establishing requirements that would allow a one-step licensing process for reprocessing facilities by combining the authorization of a construction permit and a license for the operation of the facilities into a licensing process called the combined license (COL). As part of one-step licensing, the NRC needs to establish the regulations for a safety-based inspection program to verify that the constructed facility conforms to the approved, licensed design and to ensure the reprocessing facilities operate as designed and constructed, prior to issuance of the COL.

In summary, the safety and risk analyses from Gap 5 identify specific safety controls. The technical specifications (Gap 11) provide parameters, bounds, envelopes, and actions to take for those safety controls. The Gap 9 GDC establish minimum requirements beyond the safety controls and technical specifications for defense in depth. Operators are trained and licensed to safely run the facility (Gap 7). The approaches from Gap 10 consider one-step licensing results in the safe operation of the reprocessing facilities.

2.2 Safety and Risk Assessment Methodologies, and Considerations for a Reprocessing Facility (Gap 5)

2.2.1 Regulatory Issue

Requirements are needed to ensure that radiological hazards (and NRC-regulated chemical hazards) and accidents are identified and any associated risks to the public, workers, and the environment are adequately minimized. Currently, the NRC regulates reprocessing facilities as production facilities under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Reprocessing facilities meet the 10 CFR 50.2 definition of production facilities because they are designed and used for the separation of the isotopes of plutonium and will produce SNM in quantities that could affect radiological health and safety and be of significance to common defense and security. Over time, 10 CFR Part 50 has evolved to focus primarily on light-water reactors (LWRs). As previously discussed in Chapter 1, current 10 CFR Part 50 requirements do not match well with the safety and risk attributes of reprocessing facilities. For example, LWR analyses focus on a design basis accident (DBA) approach where a single DBA is the worst case and bounds all other accidents. Thus, designing for the DBA necessarily addresses all other accidents. In contrast, reprocessing facilities have many potential DBAs that do not bound each other due to the multitude of complex operations involved, and these have to be analyzed individually. Many interactions with applicants and multiple exemptions to 10 CFR Part 50 requirements would be needed for reprocessing facilities licensed under 10 CFR Part 50, and regulation would be less efficient and effective.

At the same time, reprocessing facilities are also fuel cycle facilities. The NRC regulates fuel cycle facilities that process plutonium and fabricate fuel containing isotopes of plutonium and SNM under 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." When the NRC developed 10 CFR Part 70, fuel cycle facilities were recognized as having significantly fewer amounts of hazardous materials than reprocessing facilities. 10 CFR Part 70 was developed prior to the Commission's policy on risk-informed and performance-based regulation (SECY–98–144). The NRC revised 10 CFR Part 70 in 2000 to include an integrated safety analysis (ISA) approach, primarily for SNM containing enriched uranium and low-enriched uranium (LEU) (NRC, 2011d). Consequently, 10 CFR Part 70 and its approaches, if applied to the regulation of reprocessing facilities, do not adequately address the potentially larger releases of hazardous material (source terms), the higher dose impacts from the radioactive materials, the greater number of scenarios, more event sequences, and consequences that can exceed the thresholds in 10 CFR 70.61, "Performance Requirements," that are associated with reprocessing facilities as compared to existing fuel cycle facilities. For example, the dose conversion factors per unit mass of the radioactive materials present at reprocessing facilities can be several orders of magnitude higher than for the radioactive materials present at existing

10 CFR Part 70 facilities because of the different isotopic composition of the materials present. As shown by Table 2-1, mixed-oxide (MOX) fuel prepared from weapons-grade plutonium has dose conversion factors approximately 20,000 times higher than LEU materials. Spent nuclear fuel (SNF) materials and MOX fuel prepared from reactor-grade plutonium have even higher dose conversion factors: some 200,000 times greater than LEU materials. Thus, a potential scenario at a facility involving LEU materials might have low consequences, but the same type of event at a reprocessing facility could potentially have consequences orders of magnitude larger because of this greater radiotoxicity of materials (see Table 2-1 for comparisons), thus requiring additional safety controls to achieve the same level of safety. Likewise, the greater number of possible event sequences at a reprocessing facility would increase the total risk associated with the facility, even if the consequences of the individual event sequences were comparable. Note that the reactor-grade plutonium MOX stream at a reprocessing facility is almost an order of magnitude more radiotoxic than weapons-grade plutonium MOX because of the different plutonium isotopic compositions.

The NRC's regulations require licensed facilities to demonstrate adequate safety assurance and practices to limit risk to acceptable levels (NRC, 2008c). The analysis of risk involves interactions between regulated activities, their potential hazards, the potential consequences if an unanticipated event occurs, and the probabilities of occurrence. The NRC has typically defined risk as the product of consequence and probability (NRC, 2008a). In addition to ensuring acceptable levels of safety and risk, the NRC is authorized by AEA Section 53e(7) to not only protect but also minimize danger to life and property. This is codified in 10 CFR Part 70.23, "Requirements for the Approval of Applications." This minimization may require measures that increase safety and reduce risk further within acceptable levels (i.e., "safer"). As discussed further in the NRC's gap analysis (NRC, 2009a) and noted previously, the existing performance requirement regulations in 10 CFR Part 70, Subpart H, "Additional Requirements for Certain Licensees Authorized To Possess a Critical Mass of Special Nuclear Material," do not adequately address the hazards, consequences, and risks of potential accidents that incorporate fission products, activated metals, and actinides from potentially high burnup power reactor fuel at reprocessing facilities, including distinguishing potentially life-threatening events from lesser ones, and minimization of risks, property loss, and environmental damage.

Safety and risk considerations consist of two parts. First, limits are needed for risk. Such limits would need to be developed for types of sequences [e.g., for high consequence events (HCEs)] and receptors (e.g., worker, public, environment), and for "total" risk. "Total risk" would have to be defined; it could mean one or more limits based upon receptors, (e.g., member of the public, a maximally exposed individual [MEI], an average worker, a "site" worker), categories of potential accidents (e.g., HCEs, all events, events involving certain radionuclides), or combinations thereof. Second, a methodology is needed to assess the risks and show compliance with the limit or limits.

2.2.2 NRC Staff Approach and Recommendation

This section addresses the NRC staff's preliminary recommendations on developing safety and risk criteria and risk assessment methodologies for reprocessing facilities.

2.2.2.1 Safety and Risk Criteria

Risk information is one factor considered in this systematic regulatory decision-making process. Other factors include defense in depth and cost-benefit-like analyses that evaluate the risk

Table 2-1. Comparison of Unit Mass Dose Conversion Factors		
Specific and Relative Inhalation Doses		
Isotope/Mixture	**Specific Inhalation Dose, rem/gram**	**Relative Dose, Ratio to “Ideal LEU”**
Uranium-234 (U-234)	8.21E5	1.64E4
Uranium-235 (U-235)	2.58E2	5.16
Uranium-238 (U-238)	3.91E1	7.81E-1
Depleted Uranium (DU) U-235: 0.25%, U-234: 0.00194%, balance U-238	5.55E1	1.11
Natural Uranium U-235: 0.71%, U-234: 0.0055%, balance U-238	8.58E1	1.72
Low-Enriched Uranium (LEU) U-235: 5%, U-238: 95%	5.00E1	1 (reference)
LEU: U-235: 5%, U-234: 0.0055%, balance U-238 (similar to laser enrichment product)	9.72E1	1.9
LEU: U-235: 5%, U-234: 0.03873%, balance U-238 (similar to GC/gaseous diffusion plant enrichment product)	3.68E2	7.36
High-enriched uranium: U-235: 80%, U-234: 0.88%, balance U-238	7.44E3	1.49E2
Mixed oxide (MOX): plutonium (Pu)-239: 5%, U-238: 95%	9.55E5	1.91E4
MOX: weapons Pu, 5% Puf, balance DU	1.27E6	2.54E4
MOX: reactor Pu, 5% Puf, balance DU	1.00E7	2.01E5
MOX: reactor Pu, 5% Puf, 0.25% Am-241, balance DU	1.40E7	2.81E5
Spent nuclear fuel: 60,000 MWD/MTIHM Only fission products considered are Cs-135, Cs-137, and Sr-90 isotopes	1.11E7	2.2E5
Cs-135, Cs-137, and Sr-90 isotopes from 60,000 MWD/MTIHM spent nuclear fuel	2.05E6	4.1E4
Inhalation doses are based upon 50-year committed effective dose equivalent (see ADAMS Accession No. ML102720167, slide 5). Specific data for isotopes are from EPA-520/1-88-020, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988 (http://www.epa.gov/rpdweb00/federal/techdocs.html#report11).		

reduction effects (value) of different approaches as compared to their costs and impacts (this is often called value-impact analysis). This section focuses on addressing staff activities to review and develop the safety and risk criteria for a reprocessing facility.

In the staff requirements memorandum for SECY–04–0182, “Status of Risk-Informed Regulation in the Office of Nuclear Material Safety and Safeguards,” dated January 18, 2005 (NRC, 2005), the Commission approved the staff’s plan to continue applying risk-informed methods on materials and waste repository activities. The Commission directed the staff to keep the Commission informed on significant activities and results.

The NRC staff document, "Risk-Informed Decision-Making for Nuclear Material and Waste Applications" (RIDM) Revision 1, issued February 2008 (NRC, 2008a), describes program activities and provides general concepts related to the use of risk information in the regulation of nonreactor nuclear practices. While the document was being developed, the Advisory Committee on Nuclear Waste recommended (NRC, 2006) "that the staff consider the feasibility of applying the draft guidance to…fuel cycle issues, including design criteria for reprocessing spent nuclear fuel…" In addition to general concepts, the RIDM document provides specific decision criteria for the use of quantitative risk to individual workers and members of the public in decisions involving setting new regulatory requirements or relaxing existing ones. This document describes the general concept of three regions of risk to individuals: unacceptable, tolerable, and negligible (in decreasing order of magnitude). The RIDM contains generic quantitative health guidelines (QHGs), which represent the boundary between tolerable and negligible. These values are:

(1) Public individual risk of acute fatality (QHG 1) is negligible if it is less than or equal to 5×10^{-7} fatality per year.

(2) Public individual stochastic dose risk (QHG 2) is negligible if it is less than or equal to 4 millirem per year.

(3) Public individual risk of serious injury (QHG 3) is negligible if it is less than or equal to 1×10^{-6} injury per year.

(4) Worker individual risk of acute fatality (QHG 4) is negligible if it is less than or equal to 1×10^{-6} fatality per year.

(5) Worker individual stochastic dose risk (QHG 5) is negligible if it is less than or equal to 25 millirem per year.

(6) Worker individual risk of serious injury (QHG 6) is negligible if it is less than or equal to 5×10^{-6} injury per year.

Note that the risk metrics used to compare to these guidelines are the probability-weighted sum of risk to an individual from all accident scenarios to which he or she is exposed. For a member of the public or resident near a facility, this could be the risk from all scenarios producing significant effects offsite, whereas for the worker, the risk may be dominated by accidents that could occur in processes or buildings near the normal work station. These total individual risk metrics are expressed differently from the per-accident-sequence likelihoods used in the ISA approach in 10 CFR Part 70, Subpart H but equate to comparable levels of risk and, from a practical perspective, are the same for LEU facilities.

A different type of risk, collective risk, is one contributor to a metric called "net value impact" that is used as part of the decision process called Regulatory Analysis. This analysis considers multiple factors, including cost impacts, in reaching an optimal decision. This analysis is summarized in the RIDM document mentioned previously and is laid out in more detail in NUREG/BR–0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Revision 4, dated May 9, 2011 (NRC, 2011b). A related NRC document, NUREG/BR–0184, “Regulatory Analysis Technical Evaluation Handbook,” issued January 1997 (NRC, 1997), includes technical information useful in evaluating risk associated with reprocessing facilities. These documents provide additional insights and information for applying safety and risk analyses to reprocessing facilities.

Other agencies and institutions have developed potential safety/risk limits and/or goals for nuclear facilities. In general, these have similarities to the RIDM and other values discussed previously in this document.

When developing an efficient and effective quantitative risk-informed and performance-based regulation, the staff plans to develop safety and risk criteria to assess reprocessing facilities that align with the existing guidance, as discussed previously. Such criteria and methodologies should be consistent with NRC policies and accepted approaches, and the criteria might include separate values for different classes of receptors, such as workers, members of the public, and the environment. The staff will also investigate safety and risk criteria and methodologies that can be applied to reprocessing facilities and evaluate those criteria and methodologies using a hypothetical, generic reference design of a reprocessing facility.

2.2.2.2 Risk Assessment Methodologies

During the past two decades, both the NRC and the nuclear industry have recognized that probabilistic risk assessment (PRA) has evolved to the point that it can be used increasingly as a tool in regulatory decision making. The NRC Commission has established a policy statement on the use of PRA methods in nuclear regulatory activities (FRN 60 FR 42622, “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,” *Federal Register,* Volume 60, Number 158, p. 42622, Washington, DC, August 16, 1995). The policy notes the Commission believes that the agency should establish an overall approach on the use of PRA methods in nuclear regulatory activities so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. The policy endorses the use of PRA technology in all regulatory matters, to the extent supported by the state of the art, and in a manner that complements the NRC’s deterministic approach and supports the agency’s traditional defense-in-depth philosophy.

The policy statement on PRA encourages greater use of this analysis technique, which augments the defense-in-depth regulatory programs in order to improve safety decision making and regulatory efficiency. Current or planned activities to expand the agency’s use of risk information can be found in the Risk-Informed and Performance Based Plan (RPP, formerly known as the Risk-Informed Regulation Implementation Plan) as indicated in SECY–10–0143, “Annual Update of the Risk-Informed and Performance-Based Plan” dated October 28, 2010 (NRC, 2010c), and the NRC's Internet site at http://www.nrc.gov/about-nrc/regulatory/risk-informed.html. In its approval of the policy statement, the Commission articulated its expectation that implementation of the policy statement will improve the regulatory process in three areas:

(1) Foremost, through safety decision making, enhanced by the use of PRA insights
(2) Through more efficient use of agency resources
(3) Through a reduction in unnecessary burdens on licensees

In August 1995, the NRC adopted the following as part of the policy statement (FRN 60 FR 42622) regarding the expanded use of PRA:

(a) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

(b) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed.

(c) PRA evaluations in support of regulatory decisions should be as realistic as practicable, and appropriate supporting data should be publicly available for review.

(d) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

The NRC also has a policy statement on risk-informed, performance based (RIPB) regulation (NRC 1998). The RIPB policy encourages risk assessment methods as an additional method that complements performance-based approaches and the NRC's traditional requirements for defense in depth.

The NRC advisory committees favor increased use of quantitative risk assessment, specifically PRA methodologies, in the NRC's regulatory analyses (NRC, 2011a, 2008b). Specific preliminary recommendations with respect to reprocessing facilities are discussed next and in Section 2.3.3.

The trend in safety analyses for both nuclear and nonnuclear facilities incorporates more quantitative risk assessment methodologies. PRA is increasingly being used in safety analyses of nuclear and chemical processing facilities.

2.2.2.3 Staff Recommendation

The staff recommends developing approaches that incorporate more quantitative risk assessment and PRA methods to adequately address safety and risk at reprocessing facilities. Such approaches are consistent with Commission policies and guidance. The staff also considers that a PRA can be most effective during the design process of a new facility to provide insights to optimize safety, operations, and risk.

Domestic safety and risk information related to commercial reprocessing facilities is some 30 years old. Only general and overview information is available for the modern reprocessing facilities overseas, although informally the NRC staff has been informed that some PRA approaches have been applied at these facilities. The staff plans to address the lack of design data and information by developing generic reference designs of reprocessing facilities suitable for preliminary, top-level safety and risk analyses, and leveraging this with information from overseas facilities as it becomes available. Subsequently, staff will pursue and evaluate PRA scopes and methods for potential implementation as part of the rulemaking for reprocessing facilities.

To identify the pros and cons of various risk-informed options, the staff is considering two basic approaches: a hybrid ISA-PRA approach and a PRA approach based upon recommendations by the Advisory Committee on Reactor Safeguards (ACRS) and the existing Commission policies identified previously. The staff also recognizes there may be options available that incorporate more PRA methodologies into the analysis than the hybrid approach.

The hybrid ISA-PRA approach has four main themes:

(1) Quantify to the extent practical.

(2) Identify all accident sequences, and categorize them by consequence.

(3) Apply PRA methodologies to HCEs and very high consequence events (VHCEs), as defined next, and calculate risk.

(4) Apply safety controls and applicable design changes to reduce and minimize total risk from the reprocessing facility.

This hybrid process includes the following steps, in approximate order:

(1) Quantify all analyses to the extent practical and as supported by the state of the art.

(2) Use, in a manner analogous to 10 CFR 70.61, a quantified ISA to identify all credible accident sequences that, when uncontrolled, could exceed the consequence thresholds (Table 2-2). Such accidents would fall into one of the “Not Acceptable” bins of Table 2-3. The quantified ISA may use some conservative values as part of the binning process.

(3) Identify a subset of HCEs based upon attributes that significantly increase consequences above the high-consequence thresholds in 10 CFR 70.61, and designate this subset as VHCEs. At a minimum, these attributes would include a potential for offsite acute radiation or chemical effects, or significant contamination resulting in the loss of the use of large areas of the environment for an extended period of time. Other attributes could include the presence of reactor-grade plutonium, other transuranic (TRU) isotopes, and/or fission products, or other characteristics (e.g., multiple receptors, loss of property or use, or environmental degradation) that potentially increase the consequences significantly above 10 CFR 70.61 thresholds. Many accident sequences that have low consequences with LEU materials would likely be categorized as VHCEs when handling many of the radioactive materials occurring at a reprocessing facility simply because of the orders of magnitude increases in dose conversion factors (Table 2-1). Potential examples of VHCEs include large fires, red oil explosions, and SNF pool fires.

(4) Apply safety controls [e.g., items relied on for safety (IROFS)] to render the likelihood of intermediate events, HCEs, and VHCEs acceptable, including a lower likelihood value for VHCEs as compared to HCEs because of the greater consequence of VHCEs (i.e., a lower frequency limit is required for the same level of risk with a higher consequence event).

(5) Conduct probabilistic (i.e., quantitative) risk analyses on HCEs and VHCEs to the extent practicable and consistent with the state of the art, based upon more realistic consequence and frequency information from the reprocessing facility design.

(6) Use the PRA results to aggregate risk from a subset of accident sequences (e.g., the VHCEs and HCEs) for potential receptors (at a minimum, for a member of the public).

(7) Adjust (reduce) risk as needed to meet the appropriate NRC risk limits and criteria (these risk limits/criteria would need to be developed, and they would be informed by the QHGs). This would be accomplished by applying additional controls (e.g., IROFS) or by modifying the facility's design, and then analyzing the effect of these controls on PRA results. The PRA may be used to rank and prioritize IROFS as a function of their contribution to reducing the risk, as recommended by the ACRS (NRC, 2011a).

(8) Further, minimize the total risk to receptors beyond the minimum requirements consistent with NRC guidance, based on a value-impact (consequence-benefit) analysis.

(9) Identify GDC (see Section 2.4) and/or other controls (e.g., defense-in-depth measures) that reduce the risk beyond the minimum requirements as items supporting safety (ISS) for accident situations.

(10) Require routine updates to the safety analyses, and establish a facility-specific program to generate and collect data to refine and support risk quantification.

(11) Identify processes for ranking the various IROFS and events according to their risk importance.

(12) Identify processes for risk-informed safety review, inspection, and surveillance programs.

The staff also anticipates developing thresholds for environmental releases; environmental contamination; economic, schedule, and availability impacts; and loss of property or land use, for HCEs and VHCEs [e.g., analogous to the requirements in 10 CFR 70.23(a)(3) and (a)(4)]. Several draft thresholds are included in Table 2-2. Guidance will also be needed to support the application of quantitative risk analysis approaches to reprocessing facilities. The staff also anticipates criticality safety will follow an approach similar to that in 10 CFR Part 70 (e.g., double contingency) that requires a minimum of two independent controls to prevent criticality events. In a similar manner as 10 CFR Part 70, criticality events will be considered HCEs.

Table 2-2. Conceptual Criteria and Consequence Thresholds*		
	Receptor	
Event	**Worker**	**Individual Outside Controlled Area Boundary/Environment**
VHCE— Very-High-Consequence Event: – Prevent to very highly unlikely – PRA required	– >> †100 rem (total effective dose equivalent (TEDE)) – > endanger life (chemical) – HCEs due to the presence of fission products, reactor-grade plutonium, TRU above established thresholds, >> one receptor, unique chemicals – aggregate (consider total risk) – value-impact analysis	– > 100 rem (TEDE) – endanger life (chemical) – HCEs due to the presence of fission products, reactor-grade plutonium, TRU above established thresholds, >> one receptor, unique chemicals – aggregate to ensure total risk is acceptable – value-impact analysis – > 500,000 times values in 10 CFR Part 20, Appendix B, Table 2 – >> EPA's Protective Action Guidelines – > $1 billion in damages
HCE—High-Consequence Event: – Prevent to highly unlikely – Prevent or mitigate to intermediate or low – PRA may be required	– > 100 rem (TEDE) – > endanger life (chemical) – aggregate (consider total risk) – value-impact analysis	– > 25 rem – > 30 milligrams soluble uranium – irreversible or serious long-lasting health effects (chemical) – aggregate to ensure total risk is acceptable – value-impact analysis – > 50,000 times values in 10 CFR Part 20, Appendix B, Table 2 – > EPA's Protective Action Guidelines (PAG) – > $100 million in damages
ICE—Intermediate-Consequence Event: – Prevent to unlikely – Mitigate to low	– > 25 rem – irreversible or serious long-lasting health effects (chemical)	– > 5 rem – mild transient health effects (chemical) – > 5,000 10 CFR Part 20, Appendix B, Table 2
LCE—Low-Consequence Event	mild transient health effects or less	lesser effects
*The preliminary values contained in this table are conceptual and drawn from various sources. For example, the values of 5, 25, and 100 rem come from 10 CFR Part 70. During rulemaking, the agency will select and justify final values, including threshold values. †As the dose to an individual increases farther from the threshold (100 rem), the consequences increase at a much greater rate.		

The staff considered and reviewed several different assessment methodology options for safety and risk:

(1) Option 1 considered qualitative approaches using multiple consequence and likelihood categories. The staff found more quantification was needed to avoid differences in qualitative judgments, improve consistency, and provide a reasonable basis for regulatory decisions involving reprocessing facilities, and, thus, the staff did not consider this option further.

(2) Option 2 evaluated semi-quantitative methods, such as using indices. The staff found that this option also relied heavily on judgments and did not provide an adequate calculational continuum for reprocessing facilities; thus, the staff did not consider this option further.

Table 2-3. Conceptual Performance Requirements					
Consequence		Likelihood (Events Per Year)			
		Very Highly Unlikely (< 1E-6)	Highly Unlikely (< 1E-5)*	Unlikely (< 1E-4)*	Not Unlikely (> 1E-4)
	VHCE	Acceptable	Not Acceptable	Not Acceptable	Not Acceptable
	HCE	Acceptable	Acceptable	Not Acceptable	Not Acceptable
	ICE	Acceptable	Acceptable	Acceptable†	Not Acceptable
	LCE	Acceptable	Acceptable	Acceptable	Acceptable

*These proposed likelihood numbers are based on those in NUREG–1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," Revision 1, issued May 2010, and apply to individual accident sequences. As part of the rulemaking effort, the staff will evaluate these and other likelihood criteria, which may or may not be on a per accident sequence basis. The final numbers may or may not be in the rule; however, at a minimum, the staff recommends that they be included in the Statement of Considerations.
†This category of accident sequences may involve both prevention (reduce likelihood) and mitigation (reduce consequence) strategies). Additional information may be necessary to show the workability of such approaches.

(1) Option 3 investigated a quantified ISA. This approach provided greater consistency but did not provide adequate rigor and differentiation for some of the higher consequence events that could potentially occur at reprocessing facilities.

(2) Option 4 evaluated a hybrid ISA-PRA approach, which uses an ISA approach for some accident sequence categorizations and PRA approaches for other accident sequence categories (e.g., VHCEs). This approach was one of several discussed at public workshops in September 2010 (NRC, 2010a, p. 156, lines 22–25; p. 157, lines 1–8), October 2010 (NRC, 2010b, p. 162, lines 13–15; p. 167, lines 20–22), and June 2011 (NRC, 2011d, pp. 22–24).

(3) Option 5 considered a full PRA approach. This approach was one of several discussed at public workshops in September 2010 (NRC, 2010a, p. 116 et seq.), October 2010 (NRC, 2010b, p. 107, et seq.), and June 2011 (NRC 2011d, p. 11 et seq.).

The use of more PRA methods and/or a PRA approach (although not explicitly identified as Option 5, described previously) is recommended in NUREG–1909, "Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Recycle Facilities—Advisory Committee on Nuclear Waste and Materials (ACNW&M) White Paper issued June 2008 (NRC 2008b), and in the ACRS's February 17, 2011, letter to the Commission dated February 17, 2011 (NRC, 2011a). This approach would apply PRA methodologies to identified accident sequences at reprocessing facilities. Specifically, the ACRS stated (NRC, 2011a)

> "It is more likely that greater benefits will be achieved for complex facilities with high consequence events [HCEs]. Moving ISA towards PRA can begin with a more rigorous treatment of dependencies and human error. This would be followed by a more structured approach to allow ranking of scenarios and integration of overall risk calculations."

This approach enhances the hybrid method (Option 4) and more closely aligns with the existing use of PRA for reactors. It provides a better measure for ranking the IROFS and accident events according to risk, and provides for more risk-focused safety reviews and inspection programs.

Option 5 is the methodology the advisory committees recommended. In particular, in NUREG–1909 (NRC, 2008b), ACNW&M provided important insights on risk. The ACNW&M stated the following:

> Use of an integrated safety analysis (ISA): 10 CFR Part 70 calls for the use of an ISA to evaluate the in-plant hazards and their interrelationship in a facility processing nuclear materials. The Committee [ACNW&M] and the Advisory Committee on Reactor Safeguards have previously recommended that a regulation based on probabilistic risk assessment (PRA) is preferable to one based on ISA because the latter has significant limitations in its treatment of dependent failures, human reliability, treatment of uncertainties, and aggregation of event sequences.
>
> Best estimate versus conservative approach: A companion issue to that of ISA versus PRA approaches is whether analyses should be based on data and models that represent the best estimate of what might really occur with an associated uncertainty analysis to explore the effects of incorrect data or models, or should be based on demonstrably conservative data and models. The Committee has letters on record pointing out problems with using the latter approach. Some of the most important problems arise because very conservative assumptions can mask risk-significant items, and most conservative analyses are not accompanied by a robust uncertainty analysis.

The staff agrees that PRA methodologies should be used to the extent practicable and consistent with the state of the art and the availability of data, corresponding to either the hybrid or PRA approach discussed previously (Options 4 and 5). As a result of the lack of data for a domestic facility, the staff concludes that the proposed rule for reprocessing should include a requirement for the licensee to collect information and data from actual plant performance during operations to update and improve its safety and accident analysis as more experience is gained from actual plant operations, thus improving the PRA component of either Option 4 or 5.

In accordance with the NRC's PRA policy statement of 1995, the NRC staff concludes PRA should be used to the extent practicable and consistent with the state of the art, if data are available to support a PRA. The staff found that methods relying extensively on PRAs usually could address VHCEs and HCEs that could potentially affect members of the public. However, such methods may not add value in assessing low-consequence sequences and non-binary logic events, such as adverse chemical reactions that may be dependent on a large number of physical and chemical variables, at reprocessing facilities. Also, sufficient data may not be available for PRA analyses of all potential events. Reprocessing facilities will probably have significant chemical hazards that could be adequately addressed by quantitative ISA methods. The staff intends to continue its evaluation of applying PRAs to reprocessing facilities throughout the rulemaking process. As noted previously, the United States does not have recent design and operating experience in commercial reprocessing facilities. Thus, the staff plans to develop a generic reference design and build upon decades of international experience with reprocessing operations when developing a generic analytic reprocessing PRA model. Developing a generic risk model of a reprocessing facility will require leveraging international experiences and data acquisition as well as developing the source term, dispersion models, and analytic methods for quantifying risk for mixed chemical and radiological processes. The staff will pursue, and will keep the ACRS updated, on the staff's activities.

The proposed regulation (Part 7x) should be risk-informed, not risk-based, in accordance with the Commission's longstanding policy. Consequently, the staff envisions a rule that encompasses either (i) Option 4, a hybrid ISA-PRA methodology for assessing accidents, with PRA applied, at a minimum, to VHCEs and HCEs involving a member of the public or (ii) Option 5, a broader PRA approach based upon the ACRS recommendations that would be applied to all significant event sequences. The staff will assess both options (pros and cons) in developing the final regulatory basis.

As noted previously, Tables 2-2 and 2-3 summarize the hybrid ISA-PRA approach.

2.2.3 Rationale for the Safety and Risk Approach

In the late 1990s, the NRC formulated 10 CFR Part 70, Subpart H, which required a safety program for major fuel cycle facilities based on an ISA. At that time, the affected facilities only processed uranium, which has very low radioactivity. A reprocessing facility would possess and process large amounts of SNF containing highly radioactive fission products and TRU isotopes, which have dose conversion factors per unit mass that are orders of magnitude greater than LEU (i.e., Table 2-1). The 10 CFR Part 70, Subpart H rule did not include provisions to address situations that might involve accidents with consequences far above the high-consequence thresholds of 10 CFR Part 70, that have large acute radiation doses, or that result in large radiation doses (internal and external); such large doses could adversely affect members of the public or have negative consequences for the environment that are different from those anticipated by 10 CFR Part 70, Subpart H (NRC, 2011d). The highest category of consequences in 10 CFR Part 70, Subpart H for individuals outside the controlled area (the public) was doses exceeding 25 rem, or long-lasting or other serious chemical health effects. However, a reprocessing plant could present a higher level of possible consequences for the public, including life-threatening consequences. In addition, certain accidents may have the capability to produce large-scale land contamination and cause large property damage costs. Thus, a new regulation for reprocessing facilities should address situations and consequences not anticipated by 10 CFR Part 70.

Given the possible importance of such VHCEs, and the higher dose conversion factors that magnify the effect of uncertainties in the calculations, it is reasonable to require that the analysis of such scenarios be of a greater technical depth than the analysis of lower consequence events to meet the statutory requirements of the AEA; namely, to protect health and to minimize danger (i.e., risk) to life and property. In addition, the systems to prevent or mitigate such accidents, and their supporting management measures, should be of a higher reliability and quality. Thus, the staff recommends an analysis with greater technical depth to provide this additional assurance. The analysis should be as quantitative as possible to ensure that event consequences and likelihoods have been properly characterized. In addition, because one does not know with assurance whether a potential accident sequence will result in very-high-consequences unless one has a quantitative basis, the assignment of all accidents that could be very high consequences to a consequence category should be supported by quantitative consequence analysis to the extent practicable.

In addition, the evaluation of likelihoods for those scenarios categorized as "very high consequences" should be supported by the best available information, preferably quantitative data. Of course, the likelihood criterion for such events should be more stringent than for the "highly unlikely" events, as defined in 10 CFR Part 70, because a lower likelihood (probability) for a higher consequence event is necessary for the same level of risk. Because the evaluation is to be quantitative, risks to receptors from each accident can be aggregated to ensure that the total risk to an individual is not excessive.

Reprocessing facilities will likely have many potential accident sequences requiring evaluation. While the sequences will likely fall into known categories (e.g., criticality, fires), the specific sequences and their characteristics may not be known. Therefore, the specific sequences will need to be identified and characterized (e.g., by using an ISA that incorporates a method to identify and characterize sequences, or other hazards analysis methods). For lower consequence event sequences, there may be little difference between the uranium fuel facilities originally subject to the ISA requirement and a reprocessing facility. Hence, at this time, the staff recommends a hybrid approach where VHCEs (and, if there are a very large number, HCEs) are identified by an ISA approach and analyzed with more rigor with a PRA approach, while lower consequence events are identified and analyzed using ISA methodologies (Option 4).

The NRC staff also recommends that a reprocessing facility applicant establish an IROFS prioritization scheme, perhaps as either a regulatory requirement or a license condition. Prioritization of IROFS based on their importance to safety will facilitate risk-informing the NRC's licensing and inspection activities, as well as an operator's application of management measures to the IROFS. The ACRS noted (NRC, 2011a) that "Without such rankings, licensees and regulatory bodies will find it challenging to apportion their resources for inspecting, monitoring, and maintaining IROFS in complex facilities."

The staff also notes that several accidents related to reprocessing and the storage of SNF have resulted in significant consequences, as ranked by the International Nuclear Event Scale (INES) developed by the International Atomic Energy Agency (IAEA, 2009). Table 2-4 summarizes the scale and some of the more significant events. Note that several events with the potential for very high consequences and fatality have occurred at reprocessing and related facilities. This reemphasizes the need for a rigorous approach to safety analyses.

Table 2-4. The International Nuclear Event Scale

Level and Descriptor	Accidents and Nature of Events	Location and Type of Facility
7 Major accident	External release of a large fraction of the radioactive material in a large facility, in quantities radiologically equivalent to more than tens of thousands of terabecquerels of iodine-131.	1986 Chernobyl, nuclear reactor, USSR 2011 Fukushima, nuclear reactor, Japan
6 Serious accident	External release of radioactive material in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of iodine-131 and likely to result in full implementation of countermeasures to limit serious health effects.	1957 Kyshtym reprocessing plant, USSR
5 Accident with offsite risk	External release of radioactive material in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of iodine-131 and likely to result in partial implementation of countermeasures to lessen the likelihood of health effects.	1957 Windscale Pile, nuclear reactor, UK 1979 Three-Mile Island, nuclear reactor, USA

Table 2-4. The International Nuclear Event Scale		
Level and Descriptor	**Accidents and Nature of Events**	**Location and Type of Facility**
4 Accident without significant offsite risk	External release of radioactivity resulting in a dose to the critical group on the order of a few millisieverts. Significant damage to the nuclear facility. Irradiation of one or more workers that results in an overexposure where a high probability of early death occurs.	1973 Windscale Reprocessing Plant, UK 1980 Saint-Laurent, nuclear reactor, France
3 Serious incident	External release of radioactivity resulting in a dose to the critical group on the order of tenths of millisieverts. Onsite event resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination (e.g., a few thousand terabecquerels), but releases in a secondary containment where the material can be returned to a satisfactory storage area. Incidents in which a further failure of safety systems could lead to accident conditions if certain initiators were to occur.	1989 Vandellos nuclear reactor, Spain 1993 Tomsk, Reprocessing Plant, Russian Federation 2005 Sellafield, Reprocessing Plant (Thorp), UK
2 Incident	Incidents with significant failure in safety provisions but with sufficient defense in depth remaining to cope with additional failures. An event resulting in a dose to a worker exceeding a statutory annual dose limit and/or an event which leads to the presence of significant quantities of radioactivity in the installation in areas not expected by design and which require corrective action.	
1 Anomaly	Anomaly beyond the authorized operating regime but with significant defense-in-depth remaining.	

2.2.4 Stakeholder Views

Stakeholders generally have two types of viewpoints on this subject. The Nuclear Energy Institute (NEI) and industry representatives consider reprocessing facilities (sometimes called recycling facilities by NEI) to be more similar to existing fuel cycle facilities than nuclear reactors. Therefore, they believe that the NRC should regulate these facilities in a manner analogous to 10 CFR Part 70, including the use of an ISA and without specific risk aggregation or comparisons to total risk limits of any type. In a white paper (NEI, 2008), NEI proposed that HCEs involving a release of fission products affecting members of the public would require additional analyses, potentially including quantitative risk analyses. At the June 2011 public meeting (NRC 2011c), NEI reiterated its position favoring ISA methodology for reprocessing facilities, as outlined in its 2008 white paper, with very limited use of PRA methodologies The NEI viewpoint is similar to a blend of Options 2 and 3. The other stakeholder viewpoint generally considers reprocessing facilities to be more analogous to reactors and indicates that the NRC should regulate them in a manner analogous to 10 CFR Part 50 (NRC, 2010a). This would require more quantitative analyses and PRA. This view also holds that the rigor and structure of a PRA would provide additional design and safety insights and, ultimately, benefits for both the licensee and the NRC. This viewpoint approximately correlates with Options 4 and 5.

Staff Assessment: After considering these disparate views, the staff concludes a hybrid ISA-PRA approach (Option 4) is preferable because this approach uses a risk-graded approach to determine when to apply a PRA. This approach involves aggregating the risk from higher consequence accident sequences to ensure that the total risk from these accidents is not excessive and within the risk criteria. As noted previously, the staff concluded that risk criteria would be needed due to the unique hazards and characteristics of a reprocessing facility and the potential for a relatively large number of HCEs and VHCEs. Such total risk criteria would be consistent with existing NRC policy and guidance (e.g., RIDM) and likely correspond to circa 1×10^{-6}/yr.

The staff notes that the NEI white paper's approach calls for defining a subset of HCEs that required additional, quantitative analyses, based upon fission product releases affecting members of the public. Although not identified in the white paper as VHCEs, NEI's approach is directly analogous to the staff's approach using VHCEs—the staff included additional criteria for VHCEs because of their potential for consequences comparable to or exceeding those of fission product releases (see Table 2-1). The staff also concluded it is necessary to include other potential receptors and consequence attributes beyond radiation dose, such as chemical exposures, land contamination, and monetary impacts from events. This ultimately results in analyzing VHCEs in a manner more analogous to nuclear reactors and with the rigor of PRA methodologies, corresponding to the second major viewpoint from stakeholders.

The staff observes the NEI white paper is defining a de facto accident category based upon both performance (equivalent to 10 CFR 70 thresholds) and type of material (fission product; i.e., source) for additional quantitative analyses. The NRC staff builds upon NEI's definition by including additional criteria for VHCEs. Staff considered other potential performance (dose) criteria and concluded no NRC precedent exists for such higher consequence criteria as they approach consequences with potentially significant fractions of fatality. Consequently, the staff approach includes both performance and source criteria to characterize VHCEs.

2.2.5 Guidance Documents

The staff envisions that significant effort will be needed to develop and revise guidance documents to address the safety and risk of accidents at reprocessing facilities. These activities are likely to include the documents and subjects discussed next.

The staff will develop a reprocessing standard review plan (SRP) that describes how the NRC staff will review a license application for a reprocessing facility. An essential part of this review will deal with safety and risk. The reprocessing SRP should address how the NRC staff would review the applicant's risk assessment in the license application and what the staff expects from the risk assessment (e.g., level of detail, screening arguments, events considered, source terms, and consequence analysis).

The NRC may need to issue other guidance documents identifying the appropriate risk assessment methodology for applicants to use. This would likely include an example risk assessment (hybrid ISA-PRA) of a generic reference reprocessing facility using a methodology that is acceptable to the staff for use in analyzing a reprocessing facility. Accident analysis would likely be included as part of this guidance.

The staff would also likely need to develop additional guidance to clarify the safety and risk criteria the applicant's analysis will need to meet or for demonstrating compliance. These guidance documents would establish approaches and methodologies that would be suitable to demonstrate compliance with 10 CFR Part 7x. For example, the NRC would need to develop guidance on criteria associated with land contamination, threshold values (such as which plutonium isotopic compositions the NRC would consider to be reactor grade), risk minimization, and value-impact analysis.

2.2.6 Conclusions

The NRC requires the demonstration of adequate assurances of safety for licensed activities. Consequently, safety analyses must appropriately analyze and address the potential hazards and complexities of the licensed activities. Some areas and processes at reprocessing facilities have potential hazards and characteristics more similar to reactor facilities, while other areas and processes are more similar to those of existing fuel cycle facilities. The NRC staff concludes that it is appropriate for areas and processes at reprocessing facilities with hazards and characteristics more similar to reactors (10 CFR Part 50 and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants") to be analyzed with the same degree of scrutiny and rigor as is applied in addressing hazards and characteristics associated with reactors. At the same time, areas at reprocessing facilities with hazards and characteristics more similar to other fuel cycle facilities (10 CFR Part 70) should be analyzed with the same degree of scrutiny and rigor as is applied in addressing hazards and characteristics associated with those fuel cycle facilities. This allows hazards and characteristics to be analyzed appropriately, in a risk-informed, performance-based manner.

The staff review concluded reprocessing facilities are currently regulated under 10 CFR Part 50. However, use of 10 CFR Part 50 for regulation of reprocessing facilities would be difficult due to its almost exclusive focus upon LWRs, and many exemptions would likely be necessary to accommodate the different functions, needs, and types of hazards associated with reprocessing facilities. The staff considers the ISA method required by 10 CFR Part 70 to be appropriate to address the types of hazards and accident sequences associated with existing fuel cycle facilities. However, the presence and processing of large quantities of fission products and TRU isotopes at reprocessing facilities have the potential to greatly increase consequences far above the 10 CFR Part 70 high-consequence thresholds for some accident sequences (e.g., fires, explosions), and, therefore, 10 CFR Part 70 is not appropriate for reprocessing facilities. These VHCEs require more rigorous analyses and controls to reduce their probability (e.g., to very highly unlikely or incredible) and ultimately reduce their risks to acceptable levels. The staff also agrees with the ACRS recommendation (NRC, 2011a) that "…for more complex facilities (such as reprocessing facilities), especially those with the potential for large radiological exposure releases, the use of a PRA approach is advantageous because it provides a basis for prioritization of safety systems and maintenance activities."

The staff also notes that there is the potential for a large number of high-consequence-events and VHCE. The number of these events at reprocessing facilities is likely to far exceed the number of such events at existing fuel cycle facilities. Consequently, this would increase the total risk above that considered in 10 CFR Part 70. The staff concludes aggregation of the risk from these accident sequences and the requirement to meet a risk criterion are necessary to ensure that the total risk from potential accidents at a reprocessing facility is commensurate with risks from other NRC-licensed facilities.

The NRC staff concludes approaches that incorporate more quantitative risk assessment, including PRA, are needed to adequately address safety and risk at reprocessing facilities. The staff is considering two basic approaches—a hybrid ISA-PRA approach (Option 4) and a PRA approach based on ACRS recommendations (Option 5). The staff considers the hybrid approach is a reasonable starting point at this preliminary stage of the staff's efforts in support of potential future rulemaking activities. The staff also recognizes that more PRA methodologies may also be relevant to the analysis, and staff will evaluate these methodologies further.

The above considerations lead the staff to conclude a new regulation (Part 7x) is needed to adequately address the safety and risk approaches of reprocessing facilities in an efficient and effective manner.

2.2.7 References

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U.S. Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Domestic Licensing of Special Nuclear Material," Part 70, Chapter I, Title 10, "Energy."

2.3 Technical Specifications (Gap 11)

As mentioned in Chapter 1 (Section 1.2), a fuel reprocessing facility meets the definition of a "Production Facility," as defined in Section 11 of the Atomic Energy Act of 1954 (as amended) (AEA) and 10 CFR 50.2. Section 182a of the AEA of 1954 (as amended) mandates the inclusion of technical specifications for production facilities. Technical specification requirements for production facilities appear in 10 CFR Part 50; however, 10 CFR Part 70 does not require technical specifications. The NRC staff is proposing to use the technical specification requirements in 10 CFR Part 50 as a model for new technical specification requirements for reprocessing facilities to use in 10 CFR Part 7x.

2.3.1 Regulatory Issue

The technical specifications requirements for utilization and production facilities (10 CFR 50.36) would require modification to reflect the safety and risk attributes specific to fuel reprocessing facilities. See Section 2.2 of this document for a discussion of the staff's proposed approach to addressing accident risks. Any 10 CFR Part 50 or other technical specification requirements incorporated into a new 10 CFR Part 7x should enable the NRC to find that construction, operation, or decommissioning of a reprocessing facility will provide adequate protection to the health and safety of the public and will be in accord with the common defense and security.

2.3.2 Existing Requirements

AEA Section 182a

Section 182a of the AEA states the following: "In connection with applications for licenses to operate production or utilization facilities, the applicant shall state such technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization or production of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued."

10 CFR Part 50

The regulation in 10 CFR 50.36, "Technical Specifications," requires technical specifications for production and utilization facilities. Technical specifications must be derived from the analyses and evaluation included in the safety analysis report. According to 10 CFR 50.36(c), technical specifications for reprocessing plants must include safety limits (SLs), limiting control settings (LCSs), limiting conditions for operation (LCOs), surveillance requirements, design features, administrative controls, initial event notifications, and written reports of events.

Safety Limits

The NRC SLs requirements for reprocessing facilities are in 10 CFR 50.36(c)(1)(i)(B). SLs for production facilities are bounds within which process variables must be maintained for adequate control of the operation. These limits must not be exceeded in order to protect the physical integrity of the physical system that guards against the uncontrolled release of radioactivity. If an SL is exceeded, corrective action must be taken as stated in the technical specifications, or the entire process or affected part of the process must be shut down, unless this action further reduces the margin of safety. An example of an SL is a temperature limit in a tank, beyond which safety of the system cannot be assured.

Limiting Control Settings

LCSs requirements for reprocessing facilities are in 10 CFR 50.36(c)(1)(ii)(B). LCSs for production facilities are settings for automatic alarm or protective devices related to those variables having significant safety functions. Where an LCS is specified for a variable on which an SL has been placed, the setting must be chosen so that protective action, either automatic or manual, will correct the abnormal situation before an SL is exceeded. If, during operation, the automatic alarms or protective devices do not function as required, then action shall be taken to maintain the variables within the LCS values and promptly repair the automatic devices, or to shut down the affected part of the process and, if required, to shut down the entire process for repair of the automatic devices. An example of an LCS is a safety system actuation temperature setpoint for the contents in a tank that includes a sufficient margin to ensure that the temperature SL for the tank is not exceeded.

Limiting Conditions for Operation

Requirements for LCOs for production and utilization facilities are found in 10 CFR 50.36(c)(2)(i). LCOs are the lowest functional capability or performance levels of equipment required for safe operation. If an LCO for a production facility is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specifications until the condition can be met. An example of an LCO is the operability of a safety system that detects a high temperature in a tank and alarms or takes automatic action, such as cooling of the system, to restore it to a safe condition.

Other Technical Specifications

Other requirements for technical specifications include surveillance requirements, design features, and administrative controls. Surveillance requirements in 10 CFR Part 50 are requirements related to tests, calibration, or inspections to ensure that the necessary quality of systems and components is maintained, that facility operation will be within SLs, and that the LCOs will be met. Design features to be included are those features of the facility, such as materials of construction and geometric arrangements, that, if altered or modified, would have a significant effect on safety and that are not covered in categories described in 10 CFR 50.36(c)(1) to (c)(3). Administrative controls are the provisions related to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to ensure operation of the facility in a safe manner.

<u>10 CFR Part 76</u>

The NRC developed regulations in 10 CFR Part 76, “Certification of Gaseous Diffusion Plants,” in the mid-1990s. Gaseous diffusion plants (GDPs) are more similar to fuel cycle facilities licensed under 10 CFR Part 70 than to reactors. 10 CFR Part 76 requires technical safety requirements (TSRs) that are similar, in many respects, to technical specifications required by 10 CFR Part 50. Therefore, the framework and contents of the GDP TSRs provide a good reference to the NRC staff for considering technical specification requirements for reprocessing facilities.

At the time 10 CFR Part 76 was being developed, the NRC recognized that it would need to establish operational SLs and requirements for the GDPs similar to technical specifications for operating reactors. The staff also recognized that because the GDPs were more akin to fuel cycle facilities than reactors, their operational SLs and requirements would differ from reactor technical specifications. As such, the NRC introduced the term “technical safety requirement” (TSR). Operational safety requirements addressing protection of individuals onsite and offsite under U.S. Department of Energy (DOE) regulations and the results of the deterministic accident analyses contained in the safety analysis reports for the GDPs at that time became the basis for the GDP TSRs. The format used for addressing safety system functions in the TSRs was based on the format used for power reactors.

For the GDPs, in addition to any SLs, LCSs, and LCOs, specific process TSRs addressing safety system functions include an actions table that identifies a failure condition, actions required to address the condition, and time allowed for completing the actions. In addition, each process TSR identifies the operational modes for which the TSR is applicable and any surveillance requirements to confirm the operability of the safety system. Individual TSRs also include a basis section that discusses the need for and safety significance of the SSCs important to safety. The basis section is not considered to be part of the TSR.
In 10 CFR 76.87, "Technical Safety Requirements," the NRC requires the GDP operator to establish TSRs to address the following safety topics:

- Effects of natural phenomena
- Building and process ventilation and offgas
- Criticality prevention
- Fire prevention
- Radiation protection
- Radioactive waste management
- Maintenance
- Environmental protection
- Packaging and transporting nuclear materials
- Accident analysis
- Chemical safety
- Sharing of facilities and SSCs
- Utilities essential to radiological safety
- Operations

Similar to 10 CFR 50.36, 10 CFR 76.87 requires that TSRs include SLs, LCSs, and LCOs. 10 CFR 76.87 also requires that TSRs include design features, surveillance requirements, and administrative controls.

The NRC certified the Portsmouth and Paducah GDPs in late 1996, and after a transition period of several months, assumed regulatory oversight of the two GDPs from DOE in March 2007. Because criteria for establishing TSRs for the GDPs were not clearly established in regulation or guidance at that time, the TSRs for the GDPs were based primarily on the operational safety requirements that existed under DOE and the safety analysis reports that were submitted as part of the certification applications. Nevertheless, the framework and contents of the GDP TSRs provide a good reference to the NRC staff for considering technical specification requirements for reprocessing facilities.

2.3.3 Staff Recommendation

Operational Technical Specifications

For certain licensees authorized to possess a critical mass of SNM, 10 CFR Part 70 requires an ISA and implementing and maintaining IROFS identified in the ISA to ensure safety from potential radiological and certain chemical accidents. The staff considers the ISA required by 10 CFR Part 70, Subpart H to be appropriate for addressing the types of hazards and accident sequences associated with existing fuel cycle facilities. However, the presence and processing of large quantities of fission products and TRU isotopes at a reprocessing plant may introduce credible hypothetical accident sequences (very-high-consequence accident sequences) with consequences much higher than consequences from credible hypothetical high-consequence accident sequences at large 10 CFR Part 70 facilities. As noted in the regulatory basis for Gap 5, very-high-consequence accident sequences would be made very highly unlikely by applying IROFS. Operational technical specifications would then establish a formalized means for demonstrating that very-high-consequence hypothetical accident sequences have been made very highly unlikely. In addition, as discussed in the regulatory basis for Gap 5, an applicant would be required to quantify risks to receptors to the extent practicable and to use a risk-informed approach to prioritize IROFS. The resulting risk information and priority of an IROFS used to prevent or mitigate very-high-consequence accident sequences would inform the contents of operational technical specifications, including surveillance frequencies for the IROFS and technical specification action statements specifying an operator's response if a limiting condition of operation is exceeded.

General Technical Specifications

Because reprocessing facilities would involve large quantities of highly radioactive and other hazardous materials, the NRC staff considers it reasonable to establish, as is established for power reactors and GDPs, general technical specifications that may not necessarily trip the very-high-consequence accident sequence criteria but would still be important from the standpoint of providing protection for the health and safety of the public. Examples of such technical specifications may be a time limit for storing liquid high-level radioactive waste (HLW), and as low as is reasonably achievable (ALARA) environmental effluent limits, which would minimize any environmental impacts that may result from storing HLW or from routine environmental effluents.

Environmental Technical Specifications

Reprocessing spent fuel would involve de-encapsulation (e.g., decladding, dissolving) and release into process vessels of large quantities of radioactive materials, including fission product gases, such as krypton-85, and particulates, such as iodine-129 and TRU radionuclides that could become airborne. The NRC staff is considering requiring technical specifications for effluents for production facilities, as is required for power reactors (utilization facilities) in 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," to make releases of such gases and particulates to the environment ALARA to ensure minimization of allowable releases of radioactivity to the environment.

Waste-Related Technical Specifications

10 CFR Part 50, Appendix F, "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities," (see also Gaps 1, 2, and 3) discusses wastes from reprocessing facilities and provides requirements for their safe handling, storage, and ultimate disposal. As discussed for Gaps 1, 2 and 3, as part of transitioning reprocessing regulations from 10 CFR Part 50 to 10 CFR Part 7x, the NRC staff anticipates deleting Appendix F. However, the NRC staff recommends converting certain Appendix F requirements into technical specification requirements to meet the intent of the AEA, Section 182a, such as the Appendix F requirements that pertain to the management of HLW. These are summarized below:

- Liquid HLW inventory is limited to that created in the previous 5 years.

- Liquid HLW shall be converted to a dry solid form.

- The dry, solid HLW must be chemically, thermally, and radiolytically stable and placed in a sealed container before transfer to a Federal repository.

- The dry, solid HLW will not generate a pressure exceeding the rating of the sealed container for at least 90 days after its receipt at the Federal repository.

- HLW will be transferred to a Federal repository no later than 10 years after its generation.

If not managed properly, waste in general, and specifically HLW, can result in adverse short-term and long-term impacts to the environment and the health and safety of the public. Because a reprocessing facility is expected to handle large quantities of waste, including HLW, specific requirements in the form of technical specifications for waste management, akin to those listed above, are needed. However, the NRC staff notes that Appendix F requirements may need revision. For example, Appendix F requires HLW to be transferred to a repository no later than 10 years after its generation. However, the existence of a repository authorized to accept HLW from a reprocessing facility within this time-frame is currently uncertain. In addition, the staff may need to add new technical specification requirements such as requiring consideration in the design, operational, and decommissioning phases of all reasonably practicable steps that would minimize the amounts of liquid HLW at any given time.

LCO Criteria

10 CFR 50.36(c)(2)(ii) establishes four criteria for LCOs for reactors:

- *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

- *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- *Criterion 4.* A structure, system, or component that operating experience or PRA has shown to be significant to public health and safety.

No such criteria are specified in 10 CFR 50.36(c)(2) for reprocessing facilities. 10 CFR 50.36(c)(2)(i) requires that LCOs for reprocessing facilities be "the lowest functional capability or performance levels of equipment required for safe operation of the facility." The NRC staff proposes that LCOs for reprocessing facilities also have a clear link to safety, such as applying to IROFS for preventing and/or mitigating VHCE accident sequences. The NRC staff also proposes to include a requirement that the applicant develop, as required for reactors and the GDPs, administrative technical specifications that may not necessarily trip the VHCE criterion but would still be important to protect the health and safety of the public.

Application of Technical Specifications

The NRC regulates SNM under 10 CFR Part 70. As discussed in the NRC's gap analysis (SECY–09–0082), 10 CFR Part 70 does not require technical specifications for fuel facilities. Instead, it requires a less prescriptive approach to appropriately address accidents. 10 CFR Part 70, Subpart H requires fuel facilities that possess greater than a critical mass to conduct an ISA. ISAs involve the use of risk-informed methods for establishing limits and controls for preventing and/or mitigating all credible high-consequence and intermediate-consequence accident sequences (defined in 10 CFR 70.61). Such limits and controls protect against intermediate- and high-consequence accidents similar to 10 CFR Part 50 technical specification requirements. For example, operational SLs are based on the ISA and established in plant documentation, such as procedures to prevent a high-consequence or intermediate-consequence accident sequence from occurring. Similarly, surveillance requirements for an IROFS are based on the ISA. These are set to ensure adequate availability and reliability of IROFS such that an HCE that the IROFS is preventing or mitigating is highly unlikely and an intermediate-consequence event that the IROFS is preventing or mitigating is unlikely.

2.3.4 Stakeholder Views

The NEI white paper on the regulatory framework for recycling nuclear fuel (ADAMS Accession Nos. ML083590115 and ML083590129) recommended developing technical specifications for those IROFS that will be applied to protect against or mitigate the potential accident consequences that could result in an HCE involving fission product releases to an individual located outside the controlled area.

Because very-high-consequence accident sequences are a subset of the high-consequence accident sequences referred to in NEI's white paper, NEI's recommendation is, to a certain extent, consistent with the NRC staff's recommendation that technical specifications be developed for very-high-consequence accident sequences. The NRC staff notes that NEI's recommendation for technical specifications does not directly address the safety of workers and the protection of property and the environment from very-high-consequence accident sequences.

On July 7, 2011, the NRC received several public comments, including those from NEI, on the NRC staff's draft recommendations and alternatives for resolving the regulatory gaps, as discussed in the June 21–22, 2011, public meeting in Augusta, Georgia. Prior to the meeting, the NRC staff had also summarized its draft recommendations in a *Federal Register* Notice (76 FR 34,007, June 10, 2011).

In its July 7, 2011 comments, NEI reiterated its position that technical specifications should be developed only for IROFS needed to protect against HCEs involving fission product releases to a member of the public. According to NEI, "The requirement to establish and maintain IROFS to be available and reliable when needed obviates the need for a lengthy set of Tech Specs. Similarly NEI does not agree that the five "categories" of Tech Specs identified by the staff (e.g., safety limits and limiting control settings, etc.) are necessary or appropriate for recycling facilities." The NRC staff agrees with NEI that IROFS for high-consequence and intermediate-consequence accident sequences need not be addressed specifically by operational technical specifications.

NEI also opposes the NRC staff's recommendation for establishing effluent technical specifications to keep radioactive effluents ALARA based on the concepts of 10 CFR 50.36a. According to NEI, 10 CFR Part 7x, Baseline Design Criterion 13, "Control of Releases of Radioactive Materials to the Environment," proposed in its December 2008 white paper and the plant's safety program that will ensure compliance with the provisions of 10 CFR Part 20 would be sufficient to adequately address radioactive effluents. NRC notes that it is important to ensure that any radioactive effluents meet the regulations and are minimized. Therefore, the NRC staff recommends that, as is required in 10 CFR Part 50 for reactors, 10 CFR Part 7x require technical specifications that will keep average annual releases of radioactive material in effluents at small percentages of the dose limits in 10 CFR 20.1301.

2.3.5 Guidance Documents

The following existing guidance documents on technical specifications may be applicable to a reprocessing facility:

- Regulatory Guide 3.6, "Content of Technical Specifications for Fuel Reprocessing Plants" issued April 1973 (NRC, 1973)

- Regulatory Guide 3.19, "Reporting of Operating Information for Fuel Reprocessing Plants" issued February 1974 (NRC, 1974)

- Regulatory Guide 1.177, (draft was issued as DG–1065) and Draft Guide–1227, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" issued August 1998 (NRC, 1998)

Regulatory Guide 3.6, which was developed for reprocessing facilities in 1973, describes the recommended content of technical specifications required by 10 CFR 50.36. The NRC staff considers a large portion of this regulatory guide to be applicable because it expects to include and modify, as appropriate, in 10 CFR Part 7x the 10 CFR 50.36 provisions applicable for production facilities. However, the NRC staff anticipates developing, during rulemaking activities, additional technical specification requirements in 10 CFR Part 7x, such as criteria for LCOs, environmental effluent ALARA dose limits, and waste management technical specifications. Guidance would be needed to address such additional requirements. This could be done by expanding Regulatory Guide 3.6 or by developing new Regulatory Guides.

Regulatory Guide 3.19, which was developed for reprocessing facilities in 1974, describes the routine and non-routine operational reports that should be developed during and following the startup phase of a new reprocessing facility. The NRC staff would need to study the applicability of the recommendations contained in Regulatory Guide 3.19 due to the proposed change from a predominantly deterministic safety approach for reprocessing under 10 CFR Part 50 to a predominantly risk-informed safety approach under 10 CFR Part 7x. If the NRC staff pursues the withdrawal of Regulatory Guide 3.19, it will need to ensure that any applicable recommendations contained in Regulatory Guide 3.19 are otherwise addressed. The NRC staff anticipates the need to develop a new regulatory guide on risk-informing technical specifications for reprocessing facilities. Regulatory Guide 1.177 addresses risk-informing technical specifications for reactors by applying PRA results and insights to reactor technical specifications. The NRC staff anticipates incorporating applicable insights from Regulatory Guide 1.177 into the new regulatory guide for reprocessing.

The staff will need to develop a new SRP for reprocessing facilities. The SRP should include a chapter on technical specifications, with explicit acceptance criteria, that the NRC staff would use to review technical specifications in any license application for a reprocessing facility. The contents of this chapter could also benefit an applicant in developing the technical specifications for its license application.

The NRC has established standard technical specifications for each type of power reactor system historically operated in the United States (Westinghouse, General Electric, Babcock and Wilcox, and Combustion Engineering) in guidance documents (NRC, 2004a–e). The NRC staff does not anticipate that standard technical specifications will be developed for reprocessing at this time, because of the uncertainty in the type of reprocessing design that may be used and the expectation that not more than a few reprocessing plants will be built in the United States within several decades after 10 CFR Part 7x is issued. The staff anticipates that an applicant for a reprocessing facility would develop technical specifications for its facility based on the requirements in 10 CFR Part 7x and any associated NRC guidance. In addition, the staff anticipates that an applicant would propose these to the NRC as part of its license application.

2.3.6 Conclusions

The NRC staff recommends that the technical specifications for production facilities contained in 10 CFR 50.36 be transferred to 10 CFR Part 7x. The NRC staff recommends that operational technical specifications (SLs, LCSs, LCOs, design features, and surveillance requirements) be established for IROFS relied on to prevent or mitigate very-high-consequence accident sequences at a reprocessing plant. In addition, the NRC staff recommends requiring general administrative technical specifications that may not trip the very-high-consequence accident sequence threshold to provide adequate protection for the health and safety of the public. The NRC staff is also considering requiring technical specifications for effluents for reprocessing facilities, as is required for power reactors, to make releases of such gases and particulates to the environment ALARA to ensure minimization of allowable releases of radioactivity to the environment.

2.3.7 References

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NRC (1974). U.S. Nuclear Regulatory Commission, “Reporting of Operating Information for Fuel Reprocessing Plants,” Regulatory Guide 3.19, February 1974.

NRC (1998). U.S. Nuclear Regulatory Commission, “An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications,” Regulatory Guide 1.177, August 1998.

NRC (2004a). “Standard Technical Specifications Babcock and Wilcox Plants,” NUREG–1430 Vols. 1 & 2, Rev. 3, June 2004.

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NRC (2004c). “Standard Technical Specifications Combustion Engineering Plants,” NUREG–1432 Vols. 1 & 2, Rev. 3, June 2004.

NRC (2004d). “Standard Technical Specifications General Electric Plants, BWR/4,” NUREG–1433, Vols. 1 & 2, Rev. 3, June 2004.

NRC (2004e). “Standard Technical Specifications General Electric Plants, BWR/6,” NUREG–1434, Vols. 1 & 2, Rev. 3, June 2004.

2.4 General Design Criteria (Gap 9)

2.4.1 Regulatory Issue

The NRC establishes minimum requirements for proposed facilities (e.g., 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants”; 10 CFR Part 72, Subpart F, “General Design Criteria for Storage of Spent Nuclear Fuel and High-Level Radioactive Waste”) or applications for licensed radioactive materials (e.g., 10 CFR 70.64(a), “Baseline Design Criteria”). These minimum requirements, or GDC, generally provide

- Assurance that SSCs that directly support safety (e.g., important to safety, IROFS) will have the ability and reliability to perform their intended safety functions
- Assurance that uncertainties and errors, from design and analysis and unknowns, are adequately addressed
- Adequate defense in depth
- Redundancy and diversity
- Assurances that balance of plant and unanalyzed situations do not negatively impact safety
- Potential beyond-design-basis considerations

NRC regulations frequently identify these minimum requirements by terminology such as GDC or baseline design criteria (BDC) in the NRC regulations [e.g., 10 CFR 50, Appendix A; 10 CFR 70.64(a); 10 CFR Part 72, Subpart F]. The GDC in 10 CFR Part 50 are very specific to LWR design and operations, and require a licensee to specifically address each GDC. The GDC in 10 CFR Part 72, Subpart F also require specific GDC and require a licensee to specifically address each GDC. In contrast, the BDC for fuel cycle facilities regulated under 10 CFR Part 70 are very general and do not comprehensively address potential hazards posed by the design and operations of reprocessing facilities, such as shielding, containment, control room, decommissioning, and waste management. The BDCs in 10 CFR Part 70 are reviewed in a programmatic manner. Licensees are required to maintain the application of the BDC unless the ISA analyses [10 CFR 70.62(c)] demonstrate that a given SSC is not relied on for safety or does not require adherence to the BDC. Thus, the BDC are not truly minimum requirements. The more detailed GDC in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 only apply to nuclear power plants. However, because 10 CFR Part 50 does not contain a GDC specific to reprocessing facilities, this presents a regulatory gap.

2.4.2 Staff Recommendation

The NRC staff reviewed and evaluated several different sources of information on potential design criteria, including existing regulations (10 CFR Parts 20, 50, 70, and 72), previously proposed regulations [proposed 10 CFR Part 50, Appendix P, "General Design Criteria for Fuel Reprocessing Plants" (39 FR 26293; July 18, 1974)], and Appendix Q, "Design Criteria for Protection of Fuel Reprocessing Plants and Licensed Material Therein" (39 FR 26296, July 18, 1974), and stakeholder information [the NEI white paper on the regulatory framework for recycling nuclear fuel (ADAMS Accession Nos. ML083590115 and ML08359019] and ACNW&M (NUREG–1909, "Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Facilities"). The proposed 10 CFR Part 50, Appendices P and Q, were indefinitely deferred on April 19, 1984 until needed for NRC's regulation of a reprocessing facility (49 FR 16699, April 19, 1984).

The NRC staff considered using GDC from each of the following separate sources of information for the proposed GDC for reprocessing facilities, due to potential similarities in the types of operations, hazards, and safety requirements:

(1) Existing 10 CFR Part 50 GDC

(2) GDC previously proposed for 10 CFR Part 50 for reprocessing facilities (proposed Appendices P and Q)

(3) 10 CFR Part 70.64(a) BDC,

(4) GDC in 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," Subpart F, "General Design Criteria"

(5) The proposed BDC in the NEI white paper

The NRC staff also considered potential thresholds for applying GDC, such as the presence of fission products, reactor-grade plutonium, other TRU isotopes, and specific hazards and operations (e.g., high-temperature vitrification); if the threshold was not met, the GDC would not apply.

On the basis of its review and evaluation, the NRC staff concluded none of the listed sources, by itself, properly addressed the potential safety characteristics and issues for reprocessing facilities. It was necessary to separately select from the previously described sources appropriate GDC areas and develop GDC issues, each of which might be developed into a GDC as part of the rulemaking process. These GDC issues are largely derived from GDC for irradiated materials in 10 CFR Part 50 and 10 CFR Part 72. The NRC staff identified 78 potential GDC issues within these 10 categories from its review of GDC for irradiated materials (Table 2-5), each of which could become a GDC. The proposed 10 categories are Overall, Confinement and Containment, Process Safety, Criticality Safety, Radiological Protection, Physical Security, Material Control and Accounting (MC&A), Fuel and Radioactive Waste, Siting, and Decommissioning.

Table 2-5. Summary of Proposed GDC for Reprocessing Facilities

Draft GDC Categories with Associated Proposed GDC

Overall	Confinement and Containment	Process Safety	Criticality Safety	Radiological Protection	Physical Security	Material Control and Accounting	Fuel and Radioactive Waste	Siting	Decommissioning
1 Quality assurance and records 2 Defense in depth 3 Inherent protection 4 Preference for engineered controls 5 Anticipated operational occurrences 6 Minimize risks and impact 7 Independence between systems and facilities 8 Proximity or collocation with other nuclear facilities 9 Fire and explosion protection 10 Environmental and dynamic effects 11 Instrumentation and control 12 Emergency capability	13 Confinement design 14 Leakage monitoring 15 Inspection and testing 16 Negative pressure 17 Piping systems penetrating confinement and containment areas 18 Control and monitoring of flammable gas 19 Flammable gas in ullage and pipes 20 Habitability monitoring and control 21 Heat control and removal 22 Atmosphere cleanup	23 Functions 24 Reliability and testing 25 Independence 26 Failure of systems leads to safe states 27 Separation of process safety features from control systems 28 Process boundary quality standard 29 Inspection and testing boundary 30 Residual heat removal 31 Emergency heat removal 32 Inspection and testing heat removal 33 Control rooms 34 Chemical protection 35 Electrical power systems	36 Prevent criticality 37 Methods of control 38 Neutron absorbers 39 Adequate safety margins 40 Monitors and alarms 41 Safety control 42 Control accumulation	43 As low as reasonably achievable 44 Access control 45 Shielding 46 Monitoring and alarms 47 Minimize contamination 48 Effluent monitoring and control 49 Waste management	50 Physical barriers 51 Plant isolation 52 Lighting 53 Person, package, vehicle control 54 Equipment design and placement 55 Shipping and receiving 56 Surveillance 57 Emergency monitoring 58 Intrusion alarm 59 Essential communication 60 Cybersecurity 61 Design-basis threat 62 Aircraft impact	63 Material control areas 64 Data processing 65 Equipment 66 Measurement 67 Waste accountability 68 Special nuclear material storage	69 Spent nuclear fuel and radioactive waste storage 70 Waste form	71 Site selection 72 Seismic 73 Wind 74 Other natural phenomena hazards	75 Design, construct, and operate to facilitate decontamination and decommissioning 76 Inventory limitations 77 Time limits 78 Decontamination and decommissioning plan

The staff found that most GDC in the NRC regulations have an overall or general category that establishes general design principles (e.g., 10 CFR Part 50 Appendix A and 10 CFR Part 72 Subpart F). Such principles cover design attributes such as quality assurance, defense in depth, system independence, and emergency capability. These general attributes are also appropriate for reprocessing facilities. Consequently, these would comprise the first category for potential GDC.

The second category includes confinement and containment GDC. Reprocessing facilities will likely include multiple confinement and/or containment systems, and areas within their facilities for controlling contamination internally and releases to the environment, and for addressing off-normal and accident conditions. Criteria will need to address key hazardous aspects of the facilities, such as multiple TRU and fission product isotopes, hazardous chemicals, reactivity, and self-heating properties that would challenge the confinement and containment systems.

The third category of GDC represents specific process safety features and design concepts. These should include structures, systems, and components (SSCs) that (i) ensure that specified, acceptable operating design limits are not exceeded as a result of routine, off-normal, and accident conditions; and (ii) detect potentially hazardous situations and mitigate their risk to the workers, the public, and the environment. Audio and visual alarm systems may be used to alert operators, identify plant status, and initiate corrective actions. The process safety features should allow for periodic inspection and testing. This category would likely include criteria for control areas and rooms.

The fourth category addresses GDC for criticality safety. Reprocessing facilities will handle significant quantities of fissile materials, some of which will be outside of the current definition of Special Nuclear Material (SNM) in 10 CFR 50.2 and 10 CFR 70.4. Some examples of fissile material include LEU, plutonium, neptunium, americium, and curium. Processing will change properties that affect criticality. The staff currently concludes adequate assurances of safety require that criticality should be prevented by specific controls as well as GDC, such as neutron absorbers and alarms. In the United States, criticality safety is primarily a potential onsite hazard due to the larger sites involved and significant distances to members of the public.

The fifth category concerns GDC for radiological protection. A reprocessing facility will contain large quantities of radioactive material, including significant quantities of fission products and TRU isotopes. From a dose perspective, some of the more noteworthy fission products include cobalt-60, technetium-99, ruthenium-106, iodine-129, strontium-90, and cesium-137. Radioactive lanthanides may be present, depending on the time after discharge. Stable isotopes (e.g., lanthanides, noble metals) may be present in some processing operations in sufficient concentrations to affect radionuclide partitioning, accident scenarios (e.g., noble metals may function as catalysts), and releases. Similarly, some of the more significant TRU isotopes are neptunium-237, plutonium-238, plutonium-241, americium-241, curium-242, and curium-244. A large facility may even have significant quantities of californium isotopes. Steps must be taken to ensure that the workers, the general public, and the environment are protected from high doses of radiation resulting from uncontrolled releases of radioactive materials. Releases may occur because of equipment failure, human error, or sabotage. Preventive measures must be taken to ensure that removal of radiological material from the site is mitigated. Typical GDC would include ALARA, shielding, and effluent controls.

The sixth category covers physical security aspects. Physical security is necessary for the protection of the nuclear material that will be held on site. Some of the material on site will consist of fissile isotopes (such as plutonium-239 formed through neutron capture of uranium-238) which could be attractive to terrorist and criminal organizations wishing to divert and use such material in a malevolent manner or for extortion. Diversion may involve both external and insider scenarios. Measures must be taken to ensure that no unauthorized access to and/or removal of nuclear materials occurs. In addition to diversion, there is also a potential threat from such groups for sabotage of the reprocessing facilities, a consequence of which could result in a release of radioactive materials. The risks from such potential events can be mitigated by having such controls in place that only allow authorized personnel to access controlled areas. Such methods would include GDC for limited authorized access to certain areas of the facility, lighting, and monitoring.

The seventh category represents GDC for material control and accounting (MC&A). Reprocessing facilities will be required to adhere to the MC&A regulations in 10 CFR Part 74, “Material Control and Accounting of Special Nuclear Material.” These regulations are necessary to ensure that only authorized individuals are allowed access to SNM and that any loss or theft of such material can be recognized, and continuous knowledge of and control over the locations and quantities of nuclear materials, with the objective of detecting, identifying, quantifying, and resolving loss, theft, diversion, or unauthorized production of nuclear materials. While, at this time, it is not clear if some design approaches allow for inherently better MC&A and proliferation resistance as compared to others, some of the MC&A requirements translate into GDC. For example, NUREG–1909 states that no accumulation of separated plutonium should occur and indicating that no recovery of plutonium in a pure form would take place (i.e., the plutonium would always be mixed with something else, such as uranium). In designing an MC&A program, the facility should be able to assess MC&A feature vulnerabilities and weaknesses that identify and analyze adversary scenarios and related mitigating measures. During design of the MC&A program, the applicant should consider a “safeguards by design” approach. Safeguards by design is a process, incorporated in the early design phase, that makes the implementation of national and international safeguards features at a new nuclear facility more effective and efficient, and avoids the need and cost to redesign or retrofit the facility at a later date; this design approach also correlates with potential GDC. Besides the benefits of a proliferation-resistance assessment, and avoiding redesign and retrofitting aspects, consideration of safeguards features early in the design process for a new facility could better provide adequate implementation of safeguards requirements and optimize facility features that can be favorably gained (i.e., for MC&A purposes) in the facility’s layout and selection of material processing technologies.

The eighth category addresses GDC related to fuel and radioactive waste. Reprocessing facilities will conduct radiochemical processing operations and separations. The different radionuclides will partition in different ways, relative to percentages and quantities during facility operations. Partitioning will vary with the types of processes used, their design, and the way the facility is operated. Some radioactive materials may be recovered for use in thermoelectric generators, or for medical uses (cancer treatment, medical tests, etc). Some nonradioactive materials may also be recovered, such as xenon (anesthetic uses), and platinum group materials (catalysis). These GDC would address criteria for appropriate forms of these materials, and their safe storage and associated processes.

The ninth category includes GDC related to the site and site parameters. Reprocessing facilities will likely have site radionuclide inventories equivalent to several nuclear power plants. For example, the British and French commercial facilities (three reprocessing plants) have approximately 8 gigacuries of fission products in vitrified HLW and the equivalent of over 100 reactor core equivalents of spent nuclear fuel, in addition to in-process storage.

Consequently, siting issues become important and include GDC related to natural phenomena and associated hazards; site parameters, such as proximity to other facilities and population centers; and impacts of the site environment upon releases and doses (e.g., wind, rain, marine effects).

The tenth category concerns GDC related to decommissioning. In a manner similar to reactors and other fuel cycle facilities, an overall design objective for a reprocessing plant would likely be to facilitate the decommissioning of the plant by decontamination and removal of all significant radioactive materials and wastes from the facility and/or site at the time the facility is permanently decommissioned. However, this invokes design, construction, and operational considerations, such as selection of materials (e.g., metals and coatings that facilitate decontamination), radionuclide form and inventory limits, cell inspection and access ports, manipulators and windows, means to drain cell sumps and tanks, etc. GDC related to decommissioning may also invoke requirements for materials left on site for interim storage after facility operations cease but before transportation to another facility for treatment or disposal. For example, there may be GDC related to long-term storage of solidified HLW after the reprocessing facility ceases operations and is decommissioned.

2.4.3 Rationale for GDCs

The NRC staff noted that the agency's risk-informed, performance-based policy (NRC, 1998) requires defense-in-depth and related GDC attributes in addition to risk and performance insights. Many of the NRC regulations invoke defense-in-depth and the need for defense-in-depth, in addition to risk informed and performance based approaches. This approach has been reemphasized in the NRC's recent report by the Japan Reactor Task Force (NRC, 2011a).

The NRC staff has preliminarily concluded that reprocessing facilities have many design and hazard characteristics similar to 10 CFR Part 50 and to 10 CFR Part 72-regulated facilities that handle SNF and irradiated materials; thus, the GDC requirements should reflect the insights of those GDC identified in 10 CFR Part 50 and 10 CFR Part 72. Consequently, the NRC staff proposes GDC areas that largely follow the GDC in 10 CFR Part 50 and 10 CFR Part 72. The staff modified this initial list of proposed GDC to address the pertinent characteristics of reprocessing facilities and those GDC proposed for reprocessing facilities in the 1970s and 1980s (proposed 10 CFR Part 50, Appendices P and Q).

The NRC staff found that 10 CFR Part 50 and 10 CFR Part 72 do not have explicit limits or thresholds for applying GDC, and that current reprocessing facilities are integrated, with only nominal physical and process separation between areas. Thus, the NRC staff concludes a basis for a limit or threshold for applying GDC to reprocessing facilities does not exist.

2.4.4 Stakeholder Views

NEI proposed GDC for reprocessing facilities in its white paper (NEI, 2008) and at the public meeting in May 2010 (NRC, 2010). These GDC are largely a subset of the 10 CFR Part 50 GDC for nuclear power plants, with some additions from 10 CFR Part 70 and 10 CFR Part 72. NEI establishes a threshold for some proposed GDC—if the threshold is not met, the GDC would not apply. This threshold is based upon a fission product release that could result in a HCE to an individual outside the controlled area boundary. Thus, a TRU isotope release that could result in an HCE to such a receptor might not invoke these GDC. With NEIs proposed approach, the applicant would need to explain in a summary format (i.e., a programmatic approach) how the GDC are addressed to achieve the performance requirements. NEI does not envision that the applicant would necessarily be required to describe on a system-by-system basis how the GDC are met. In addition, NEI states there may be processes or aspects of a fuel recycling facility for which some of the GDC may not be necessary or appropriate, based on the results of the ISA.

The staff reviewed the stakeholder views. The staff analysis found the NEI proposed GDC approach might not adequately address the assurance of safety needed at a reprocessing facility, which, by its nature, contains many processes and associated hazards from processing irradiated materials that have unit dose conversion factors significantly higher than LEU materials (Table 2-1). In lieu of specific designs and experience from modern reprocessing facilities in the United States, the staff perceived that a programmatic approach might not ensure that all potential hazards would be adequately addressed and concluded that only a facility-specific approach to GDC would demonstrate adequate assurances of safety. Staff noted excluding GDC based upon risk analyses, such as ISA results, defeats the purpose of GDC and their important contributions to safety. The staff generally concluded GDC should be addressed in a manner analogous to those for reactors and SNF, as envisioned in the current 10 CFR Part 50 and 10 CFR Part 72, because of similarities in the types of operations, hazards, and safety requirements for handling irradiated materials. This approach identifies specific GDC and their requirements, specifically described by the applicant in its license application. The staff approach is in alignment with the second stakeholder viewpoint noted previously.

2.4.5 Conclusions

The staff concludes that GDC are needed at reprocessing facilities to do the following:

- Support safety systems (some GDC may actually result in or become safety systems).
- Support the functionality of safety systems.
- Enhance safety with defense in depth and redundancy and diversity.
- Address uncertainties, errors, and unknowns (particularly unintentional omissions, oversights, and unanalyzed effects in safety analyses).
- Establish minimum facility requirements.
- Avoid common-mode failure effects from balance of plant activities.
- Consider beyond-design-basis events.

The staff analysis concluded an approach analogous to those in 10 CFR Part 50 and 10 CFR Part 72 is appropriate for addressing potential GDC at reprocessing facilities because of similarities in the types of operations, hazards, and safety requirements for handling irradiated materials.

2.4.6 Guidance Documents

The NRC does not have guidance documents specific to GDC for reprocessing facilities. As noted previously, 10 CFR 50.34(a)(3)(i) states that GDC for chemical processing facilities are being developed. The NRC regulations in 10 CFR Parts 20, 50, 52, and 72 require GDC, and these regulations, with their respective guidance, provide some perspectives on GDC use.

Two proposed appendices to 10 CFR Part 50 also include insights similar to guidance on GDC and might be applicable to reprocessing facilities:

- Proposed Appendix P to 10 CFR Part 50—"General Design Criteria for Fuel Reprocessing Plants" (NRC, 1974a) (39 FR 26293; July 18, 1974)

- Proposed Appendix Q to 10 CFR Part 50—"Design Criteria for Protection of Fuel Reprocessing Plants and Licensed Material Therein" (NRC, 1974b) (39 FR 26296, July 18, 1974)

In addition, two recent documents from NEI (NEI, 2008) and ACNW&M (NRC, 2008) provide insights on potential GDC for reprocessing facilities.

Consequently, the staff concludes new guidance will be needed for GDC at reprocessing facilities that encompasses insights from the documents listed previously.

The new SRP for reprocessing facilities will contain a section explaining how the NRC staff will review a license application for a reprocessing facility. An essential part of this review will deal with the GDCs, either as an explicit chapter or as part of other sections of the SRP (e.g., for technical review areas such as fire, criticality, chemical). The reprocessing SRP must address how the NRC staff would review the applicant's specific approach to meeting the GDC requirements and what staff would expect from the applicant's submittal (e.g., level of detail, grading schemes, and any GDC items identified as IROFS).

The staff may also need to develop at least one additional guidance document specific to GDC. This would discuss the GDC requirements in detail, their rationale, and how they might apply to different processes and facility designs and would include examples. The NRC may need to issue other guidance documents on specific GDC or GDC areas, identifying specific GDC approaches that staff would find acceptable for analyzing a reprocessing facility application.

Potentially applicable regulatory guides are identified in Appendix F of this draft regulatory basis document. Staff concluded that many of these regulatory guides are outdated and would likely require revision to address GDC and GDC-related issues associated with modern reprocessing facilities.

2.4.7 References

NEI (2008), Nuclear Energy Institute, Letter to Michael F. Weber, U.S. Nuclear Regulatory Commission, December 19, 2008 (NEI White Paper). (ADAMS Accession No. ML0835901300)

NRC (1974a), Proposed Appendix P to 10 CFR Part 50—"General Design Criteria for Fuel Reprocessing Plants" 39 FR 26293; July 18, 1974.

NRC (1974b), Proposed Appendix Q to 10 CFR Part 50—"Design Criteria for Protection of Fuel Reprocessing Plants and Licensed Material Therein" 39 FR 26296, July 18, 1974.

NRC (1998). U.S. Nuclear Regulatory Commission, "White Paper on Risk-Informed and Performance-Based Regulation," SECY-98-144, June 22, 1998. (ADAMS Accession No. ML992880068)

NRC (2008). U.S. Nuclear Regulatory Commission, "Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Recycle Facilities—ACNW&M White Paper," NUREG–1909, June 2008. (ADAMS Accession No. ML082100043)

NRC (2010), Category 2 Meeting - U.S. Nuclear Regulatory Commission (NRC) and Representatives of the Nuclear Energy Institute (NEI), NRC Headquarters, Rockville, Maryland, On May 13, 2010 (ADAMS Accession No. ML101450520)

NRC (2011a) Japan Reactor Task Force Report, "Recommendations for Enhancing Reactor Safety in the 21st Century," July 12, 2011. (ADAMS Accession No. ML111861807). (http://pbadupws.nrc.gov/docs/ML1118/ML111861807.pdf)

2.5 Licensed Operators and Criteria for Testing and Licensing Operators (Gap 7)

2.5.1 Regulatory Issue

Atomic Energy Act (AEA) Section 107 requires both production and utilization facilities to have licensed operators (42 U.S.C. 2137). A reprocessing facility meets the definition of a production facility as defined in Section 11 of the AEA and 10 CFR 50.2 because reprocessing will be used for the separation of isotopes of plutonium and will produce SNM in quantities that could affect radiological health and safety and be of significance to common defense and security. Consequently, in the 1970's, licensing of a reprocessing facility was incorporated into 10 CFR Part 50. However, a regulatory gap exists because the NRC does not currently have regulations for licensing operators of a reprocessing facility. The current regulations for operator testing and licensing in 10 CFR Part 55, "Operators' Licenses," apply to utilization facilities (reactors) and are not applicable, in whole, to operators of reprocessing facilities. Before its revision in 1987 (52 FR 9453; March 25, 1987), 10 CFR Part 55 did include consideration of both production and utilization facilities (28 FR 3196; April 3, 1963). However, production facilities were removed from Part 55 in 1987 because "there [were] no operators at production facilities currently licensed by the Commission" (52 FR 9453). Therefore, the NRC should develop criteria for testing and licensing operators of reprocessing facilities.

2.5.2 Staff Recommendation

The NRC staff recommends creating an operator licensing subpart to the proposed 10 CFR Part 7x that is based on 10 CFR Part 55 and includes previous versions of 10 CFR Part 55 that applied to reprocessing. The NRC would license personnel whose actions are clearly related to safety, such as actions required by technical specifications. In the staff's approach, the applicant for a reprocessing facility would identify those systems with the potential for VHCEs. This includes the identification of the controls and the parameters that need to be controlled, as well as the technical specifications to prevent and mitigate VHCEs.

The staff would review the identification of event sequences and controls, IROFS, and technical specifications.

The staff's general approach to personnel training requirements is to require that all personnel be trained using a "systems approach to training," which is defined in 10 CFR 55.4, "Definitions," as "a training program that includes the following five elements:

(1) Systematic analysis of the jobs to be performed.

(2) Learning objectives derived from the analysis which describe desired performance after training.

(3) Training design and implementation based on the learning objectives.

(4) Evaluation of trainee mastery of the objectives during training.

(5) Evaluation and revision of the training based on the performance of trained personnel in the job setting."

The applicant would provide the NRC with the details of the training program for licensed operators and any other licensed personnel. The staff would review the development and implementation of the training program for licensed operator and senior operator candidates to ensure that the training program is based on a systems approach, as defined in 10 CFR 55.4, and is approved by the NRC or accredited by an independent entity. This approach is consistent with the current approach to training programs in 10 CFR 55.31(a)(4).

Personnel at reprocessing facilities licensed under 10 CFR Part 7x should also meet the requirements of 10 CFR Part 26, "Fitness for Duty Programs." However, 10 CFR Part 26 does not mention reprocessing facilities. To ensure that these requirements are applied, the scope of 10 CFR Part 26 needs to be expanded to include consideration of reprocessing facilities, and 10 CFR Part 7x would require compliance with 10 CFR Part 26.

One difference between reprocessing facilities and reactors is that operating reprocessing facilities typically involves several control rooms (e.g., separations, vitrification, plutonium line, uranium line) and extensive balance of plant facilities that would interact with the systems controlled by a licensed operator during potential accident sequences. The presence of multiple control rooms and extensive balance of plant facilities would complicate interactions between the control rooms, and between operators who may only be licensed for one or some of the control rooms, thus necessitating coordination between control rooms during some accident sequences. The NRC staff recommends licensing the individuals who have overall responsibility over a control room and those with responsibility for coordinating between control rooms. These individuals would have an important safety role in coordinating licensed operator actions and would be the equivalent of senior operators under 10 CFR Part 55. In the past (see NUREG–1909, Section 4.4.1 (NRC, 2008a)), senior operators were licensed to operate reprocessing facilities. Therefore, it is reasonable to conclude that senior operators would continue to be needed, as the operation of commercial reprocessing facilities has not become significantly simpler in the elapsed time. The staff also recommends that personnel who meet the additional training requirements (10 CFR 55.43(b)) for senior operators be on staff at the licensed facility, because of the variety of facility controlled parameters and systems that may have VHCEs. This is similar to existing requirements in 10 CFR Part 55. The NRC staff identified two main approaches:

(1) The NRC staff's recommended approach is to adopt the current 10 CFR Part 55 responsibilities and training for senior operators, with minor changes to make them more applicable to reprocessing (e.g., removing mention of power level). In this approach, the NRC would license as senior operators those personnel who are in charge of the control rooms and of coordinating interaction between control rooms.

(2) An alternative approach is to increase the training of the operators by incorporating the training requirements that apply to senior operators into the requirements that apply to all operators. The facility licensee's management could choose from among these trained operators those who would have the responsibilities of senior operators.

The NRC staff recommends that the regulation establish the requirements for training and testing candidates based on requirements contained in 10 CFR Part 55, with additional or altered requirements to address the particular safety aspects unique to reprocessing facilities. Candidates would be tested on the areas that are applicable to the position(s) and facility for which an operator license is requested.

The NRC staff recommends that 10 CFR Part 7x contain requirements such as those in 10 CFR Part 55 for testing licensed operator and senior operator candidates, with both written examinations and operating tests. Each test would need to be prepared, proctored, and graded, preferably by a person or group other than the candidate's immediate supervisor or the person or group that provided the training. Similar to the requirement in 10 CFR 55.40(c), if the NRC disapproves the facility licensee's testing, the NRC will prepare, proctor, and grade the required tests. Therefore, the NRC would need to maintain its proficiency in developing examinations and tests. Similar to the requirement in Section ES-201 of NUREG–1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1, issued October 2007 (NRC 2007a), this could be accomplished by developing a certain number of written examinations and operating tests per year, or by maintaining a suitable training and qualification program for NRC staff. The requirements for requalification would be based on the requirements in the current 10 CFR Part 55, the 1986 version of Appendix A to 10 CFR Part 55, and DOE orders (DOE 2010, 2011) and on the operator's role in safe operations, such as preventing and mitigating VHCEs.

Table 2-6 summarizes the staff's recommended approach to testing candidates.

Two alternative approaches to the NRC's role in testing candidates, as well as the NRC staff's recommended approach for written examinations and operating tests, are shown in Table 2-6.

The NRC staff's recommended approach for written examinations (see Table 2-6) is the same as that in 10 CFR Part 55. For licensee-developed operating tests, staff's approach would be based primarily on 10 CFR Part 55 but would have some variations. The NRC staff would review and approve the tests before they are used. The licensee may then choose to administer and grade the approved operating tests, and the NRC would observe the tests and co-evaluate the candidates. Significant differences in the grades assigned by the licensee and the NRC observer must be resolved before the grades could be used in operator licensing. The NRC would grade NRC-administered operating tests.

Table 2-6. NRC Staff's Recommended Approach			
Written	**Prepares**	**Proctors**	**Grades**
Examinations	Licensee; the NRC can elect to prepare but it must review and approve the test. (The examination standards would require some examinations to be developed by the NRC.)	Licensee; an NRC contact is available during the exam.	Licensee (usually for licensee-developed exams) or the NRC for NRC-developed exams. The NRC must review and approve the licensee-recommended grades.
Operating Tests	Licensee; the NRC can elect to prepare but it must review and approve the test. (The examination standards would require some tests to be developed by the NRC.)	Licensee administers, with NRC co-evaluation, or the NRC administers.	Licensee grades licensee-administered tests, with comparison to NRC evaluation grades; the NRC grades NRC-administered operating tests.

The first alternative approach staff identified used the existing framework in 10 CFR Part 55 without significant changes. In this approach, the NRC is solely responsible for administering and grading the operating tests. In the second alternative, similar to NEI's approach (NEI, 2008), the facility licensee would be responsible for certifying the candidates. The NRC would audit the facility licensee's certification program to ensure that it adequately trained and tested candidates. Although this approach would consume significantly fewer NRC resources, it would be hard to show that an examination or test was deficient after it had been given and to correct any deficiencies. The most obvious solution to the problem of a deficient test would be to retest the operators who were licensed/certified based on that test. However, finding a test deficient would cause the licensee significant problems, such as having inappropriately licensed operators operating controls, and the logistical problems associated with retesting. Staff's recommendation that the NRC review and approve of operating tests, would work better, in-part because everyone must agree that the test is acceptable before it can be used in operator licensing.

Under either approach, a significant training and support program would be needed because the NRC does not currently have the regulatory framework and staff qualified to license operators for reprocessing facilities. A disadvantage of the NRC staff's recommended approach is that it requires significantly more resources (in terms of maintaining the qualified personnel to review and approve, develop, and conduct the necessary examinations and tests) than NEI's approach. The resources for such an operator licensing program include the need to develop and conduct training classes for NRC staff. These resources may be greater for reprocessing facilities than for reactors because of the larger number of diverse systems and areas that would relate to operator licensing.

Years of industry and NRC experience with simulators at reactor units have proven the value of simulators in the training of licensed operators at nuclear power plants. Simulation facilities are used to perform operating tests and can be used to meet experience and training requirements. The use of simulators to train for accident conditions is especially important because it may be difficult to safely accommodate accident training by using the facility alone. However, a facility licensee can choose how to meet the experience and training requirements. The facility licensee will have to show that its training and qualification program and the system on which operating tests are performed is acceptable for such use, so as to not result in negative training.

The training and testing requirements for reprocessing facility operators in 10 CFR Part 7x could be met either by using a simulator or by using the facility as a simulator. The NRC staff should further consider the need to require simulator training and testing. The requirements should include describing how the simulation facility or facility itself should be used in training and tests and in meeting experience requirements. Requirements for ensuring simulator fidelity should also be included in 10 CFR Part 7x. These requirements should be based on those in 10 CFR 55.46(c) and (d), which prescribe the minimum scope and fidelity requirements for simulators. Criteria for the acceptability of various simulation facilities would be based on demonstrating fidelity during normal and accident sequences such that negative training is avoided. This may include demonstrating that the facility can be adequately used to train operators for accident conditions. While the Commission has not approved any reprocessing facility simulator for use in training, the staff expects that proper modeling of the basic chemistry and physics of the processes can overcome this limitation.

2.5.3 Rationale

The requirements for reprocessing operator licensing should be based on 10 CFR Part 55 because the requirements for both reprocessing operator and reactor operator licensing are based on AEA Section 107, 10 CFR Part 55 previously applied to reprocessing facilities. The staff's recommended approach (i.e., creating an operator licensing subpart to the proposed 10 CFR Part 7x) has the advantage of not changing 10 CFR Part 55. This eliminates difficulties in using a single regulation to license both reactor and reprocessing operators. Many sections of 10 CFR Part 55 are applicable with little or no modification (e.g., medical requirements, conditions of licenses, renewal of licenses), while other sections (e.g., written examinations and operating tests, requalification) would have to be adapted appropriately.

In 10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel," and 10 CFR 76.95, "Training," the NRC requires the licensee to train unlicensed personnel at power reactors and GDPs using a systems approach to training. 10 CFR 50.120(b)(2) provides a list of nonlicensed reactor personnel who must be trained using a systems approach. Many of the positions on this list would also be applicable to a reprocessing facility. This supports the staff's position that personnel, including licensed operators, should be trained using a systems approach.

It is the staff's position that the NRC should license personnel whose actions are clearly related to safety, such as being relied on to control the important parameters of systems that, if not controlled, could lead to accident sequences with VHCEs. NEI (NEI, 2008) also recommends a similar threshold for its recommended approach. In a reactor, the core is the primary system that can cause serious accidents of concern, so the regulations in 10 CFR Part 50 and 10 CFR Part 55 concerning licensed operators focuses on "the reactivity or power level" of the reactor as the important parameter that the licensed operators control. However, a reprocessing facility may have multiple systems located throughout the reprocessing facility that can have VHCEs. In addition, operators may need to control a broader range of parameters than at a reactor; therefore, the definition of "controls" in 10 CFR Part 50 and the old 10 CFR Part 55 (28 FR 3196; April 3, 1963) for production facilities uses a catch-all definition of controls such that the controlled parameters may be any "chemical, physical, metallurgical, or nuclear process of the facility" that affects "the protection of health and safety against radiation."

The regulations in 10 CFR Part 55 specify the content of candidate written exams and operating tests. This ensures that the candidates have received training in these areas that are considered important for safe operations of the facility. Similar training requirements are needed for operators of reprocessing facilities to ensure the safe operation of their facility. The scope of 10 CFR Part 55 once included reprocessing facilities. The current scope of 10 CFR Part 55 is limited to utilization facilities, but the training requirements in 10 CFR Part 55 may be applicable to reprocessing because the requirements for operators' licenses have not been changed significantly since 10 CFR Part 55 was applicable to reprocessing facilities. Also, 10 CFR Part 55 requires that the criteria in NUREG–1021 (NRC, 2007a) be used to evaluate and prepare the required examinations and tests. As discussed in Section 2.5.5, reprocessing facility operators would also need an examination standard. The requirements for areas on which to train and test candidates will be placed in 10 CFR Part 7x or in a separate examination standards document with which 10 CFR Part 7x would require compliance. Establishing the areas of training and testing makes it easier for the staff to perform a consistent and predictable review and provides clearer direction to applicants.

2.5.4 Stakeholder Views

Appendix E, Section III, Part E of the NEI white paper (NEI, 2008) briefly discussed licensing operators:

> The AEA requires that operators be licensed for production facilities but does not specify the process. The proposed framework requires that the applicant will certify to the NRC the operators as trained and technically, medically, and physically competent based on an NRC approved certification program. Based on this NRC approved certification program, NRC will approve and issue an operator license to the certified operator allowing an individual to perform licensed activities.
>
> The application will include, in addition to the management measures that require training and qualification of the non-licensed operators and other facility staff, the applicant's program for training, periodic proficiency training, requalification, and certification of operators who will be approved by NRC as licensed operators. It is expected that NRC, as part of the process to license operators who have been certified, will audit the facility's certification process to be satisfied that the certified operators are competent and capable of performing licensed operations. Recognizing that fuel cycle facility operators are not required to be licensed, the threshold for licensed operators under the proposed framework are operators whose actions are necessary to prevent or mitigate identified and defined accident scenarios involving fission products that could result in a high consequence event to an individual outside the controlled area.

In Appendix A, Section IV, NEI also stated that "To the extent practical, a systems approach to training would be utilized. If available, training would include use of a plant referenced simulator." NEI later stated that the "certification program is designed to ensure technical competency. The facility licensee should decide which of its certified operators should be supervisors." The staff has considered these views in developing its regulatory basis and concludes that an approach based primarily on 10 CFR Part 55 that licenses operators is appropriate.

2.5.5 Guidance Documents

The NRC examination standards for reprocessing would be based on NUREG–1021 and NUREG–1478, “Operator Licensing Examiner Standards for Research and Test Reactors” Revision 2, dated June 20, 2007 (NRC, 2007b). The examination standards for reprocessing would establish the policies, procedures, and practices for administering the required initial and requalification written examinations and operating tests. Examination standards are needed to ensure the equitable and consistent administration of examinations to all applicants at all facilities that are subject to the regulations.

The staff considers it necessary to create technology-specific knowledge and abilities catalogs for reprocessing facilities. These documents would either be generated by an applicant and submitted for NRC review, or generated by an industry working group and endorsed by the NRC. The catalogs are relied on to ensure that the content of the examinations is valid. NRC would have to endorse or approve any knowledge and abilities catalog, or substitute, and enable “uniform conditions for licensing individuals as operators” (Section 107 of the AEA; 42 U.S.C. 2137). The existing catalogs, such as NUREG–1122, “Knowledge and Abilities Catalog for Nuclear Power Plant Operators: Pressurized Water Reactors,” Revision 2, Supplement 1, issued on October 2007 NRC, 2007c), could be used as a model for the reprocessing catalogs.

The staff may need to develop a regulatory guide based on Regulatory Guide 1.149, “Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations,” Revision 4, issued April 2011 (NRC, 2011), that would endorse the relevant sections of American National Standards Institute (ANSI)/American Nuclear Society (ANS)-3.5-2009, “Nuclear Power Plant Simulators for Use in Operator Training” (ANSI/ANS, 2009), clarifying how to use the standard to demonstrate that a simulator meets the requirements of 10 CFR Part 7x. If a consensus standard specifically for reprocessing simulators becomes available, this proposed regulatory guide may endorse that standard as applicable.

Other NRC guidance documents that may apply without modification include Regulatory Guide 1.134, “Medical Evaluation of Licensed Personnel at Nuclear Power Plants” Revision 3, issued March 1998 (NRC, 1998); Regulatory Guide 1.114, “Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit” Revision 3, issued October 2008(NRC, 2008b); and Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants” Revision 3, issued May 2000 (NRC, 2000).

2.5.6 Conclusions

Licensed operators will be needed for any licensed reprocessing facility. The criteria for determining which operators are licensed should be based on the operator’s role in ensuring safe operations. All personnel, including licensed operators, should be trained using a systems approach to training. The NRC should be involved in the testing process for licensed operators to ensure that the operators have demonstrated that they have been adequately trained and are qualified for their positions. NRC staff concludes that 10 CFR Part 7x should contain provisions for senior operators and testing requirements that are similar to the provisions and requirements in 10 CFR Part 55. Staff recommends that 10 CFR Part 7x allow the applicant to meet the training and testing requirements either by using a simulator or by using the facility as a simulator.

2.5.7 References

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NRC (2007c). U.S. Nuclear Regulatory Commission, "Knowledge and Abilities Catalog for Nuclear Power Plant Operators: Pressurized Water Reactors," NUREG–1122, Revision 2, Supplement 1, October 2007. (ADAMS Accession No. ML072970334)

NRC (2008a). U.S. Nuclear Regulatory Commission, "Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Recycle Facilities—ACNW&M White Paper," NUREG–1909, June 2008. (ADAMS Accession No. ML082100043)

NRC (2008b). U.S. Nuclear Regulatory Commission, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit," Regulatory Guide 1.114, Revision 3, October 2008. (ADAMS Accession No. ML082380236)

NRC (2011). U.S. Nuclear Regulatory Commission, "Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations," Regulatory Guide 1.149, Revision 4, April 2011. (ADAMS Accession No. ML110420119)

U.S. Code of Federal Regulations, "Fitness for Duty Programs," Part 26, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Operators' Licenses," Part 55, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Certification of Gaseous Diffusion Plants," Part 76, Chapter I, Title 10, "Energy."

2.6 One-Step Licensing and Inspections, Tests, Analyses, and Acceptance Criteria Requirements (Gap 10)

2.6.1 Regulatory Issue

The current regulations for reprocessing facilities are in 10 CFR Part 50; 10 CFR Part 50, Appendix F requires applicants to submit an application for construction permit followed by an application for an operating license. This approach results in significant regulatory uncertainty because it is possible to receive a construction permit for a facility that, when constructed, is not allowed to operate. The Combined License (COL) process under 10 CFR Part 52 addresses this uncertainty by using a one-step process that combines the application for a combined license (combining the construction permit and operating license issuance).

In an effort to improve licensing and regulatory efficiency for nuclear power plants, the NRC established regulations for a one-step licensing process in 10 CFR Part 52. In the one-step licensing process, the NRC evaluates the information in the application through the use of inspections, tests, analyses, and acceptance criteria (ITAAC) to ensure that the plant operates as designed and constructed before the agency authorizes fuel loading. 10 CFR Part 52, (Subpart A) also allows for Early Site Permits (ESP) and Limited Work Authorizations (10 CFR Part 52, Sections 52.17(c), 52.27, and 52.91). However, 10 CFR Part 52 does not apply to SNF reprocessing facilities.

Section 70.23, "Requirements for the Approval of Applications,'' provides for one-step licensing for certain fuel cycle facilities. However, 10 CFR Part 70.23(b) provides for a two-step process for plutonium processing and fuel fabrication plants. 10 CFR 70.23(a)(7) and 70.23(b) address approvals for construction, and 10 CFR 70.23(a)(8) addresses approval for operation. Under 10 CFR 70.22(f), an application for a license to possess and use SNM must contain a description of the plant site; a description and safety assessment of the design bases of the principal structure, systems, and components (PSSC) of the plant, including provisions for protection against natural phenomena; and a description of the quality assurance program to be applied to the design, fabrication, construction, testing, and operation of the SSCs of the plant.

The NRC currently has no regulations for one-step licensing of a reprocessing facility. A new 10 CFR Part 7x could be developed to provide a one-step licensing process for a reprocessing facility and provide reasonable assurance that the facility will be constructed and operate in conformity with the Commission's regulations and licensing requirements.[1]

Gap 10 in SECY–09–0082, "Update of Reprocessing Regulatory Framework-Summary of Gap Analysis" (NRC, 2009a), describes a one-step licensing approach for fuel reprocessing facilities.

2.6.2 Staff Recommendation

The NRC staff proposes creating application requirements for a SNF reprocessing facility that are similar to those in 10 CFR Part 52 for the construction and operation of a nuclear power plant. A licensing process similar to 10 CFR Part 52 for reprocessing facilities would provide predictability for a COL and reduce regulatory uncertainties.

The NRC must ensure that after issuance of a COL, the reprocessing facility is constructed according to the approved application. Therefore, the one-step licensing process for reprocessing facilities will also include an inspection process to confirm that the facility meets the reprocessing design and construction requirements. This requires a significant commitment of NRC inspection resources, as well as the resources needed for the associated training and support. In addition, the COL licensing process may require ITAAC or voluntary ITAAC, to confirm that the facility meets the design and construction licensing requirements. ITAAC are conditions of the license and must be met before operation.

The proposed 10 CFR Part 7x may provide the requirements for applicants and procedures for the Commission to issue an ESP for approval of a site for a reprocessing facility. This will be separate from the filing of an application for a COL. Thus, a COL application could reference an ESP. If an ESP is not referenced, the applicant would be required to provide an equivalent level of information in the one-step license application.

A limited work authorization may be permitted at the applicant's request after the ESP has been issued. A 10 CFR Part 52 applicant may request that an LWA be issued in conjunction with the ESP under 10 CFR 52.17(c).

The regulations in 10 CFR Part 52 also include options for standard design certifications, standard design approvals, and manufacturing licenses, and 10 CFR Part 52 has appendices containing design certification rules for reactor designs. The NRC staff's proposal for reprocessing facilities does not include these options. Instead, the one-step licensing application process would incorporate information on confirming reprocessing designs, requirements for design approvals by the NRC staff, and requirements and approvals for the manufacturing of spent fuel reprocessing facilities components.

The proposed regulation should contain sections that address specific requirements for the aqueous and electrochemical separation processes. If the requirements are not known when the regulations are written, the regulation could contain a blank section labeled "Reserved." The

[1]Staff will consider and examine any changes needed to be made to 10 CFR Part 2 and will discuss its findings in the final Regulatory Basis document.

staff may add separate appendices to the regulation to describe unique design requirements that must be addressed for liquid-liquid aqueous separation processes and electrochemical separation processes.

The staff's regulatory basis will also address general and technical information to be included in the contents of a COL application. The general information in the contents of the application will be derived from 10 CFR 52.77, "Contents of Applications; General Information," and additional technical information will come from 10 CFR 52.80, "Contents of Applications; Additional Technical Information." This latter section will also provide the guidance for standards for the review of the application. The staff will also identify the technical information that must be incorporated in the applicant's final safety analysis report describing the SNF reprocessing facility. Much of this information will be derived from 10 CFR 52.79, "Contents of Applications; Technical Information in the Final Safety Analysis Report."

10 CFR Part 7x may include requirements for prototype reprocessing facilities. A prototype reprocessing facility is a reprocessing facility that is used to test new safety features before a full-scale reprocessing facility is licensed to operate. A prototype reprocessing facility is similar to a first-of-a-kind facility in all features and size, but it may include additional safety features to protect the public and the facility staff from the possible consequences of accidents during the testing period.

The application should be required to describe the reprocessing plant-specific risk assessment (see Options 4 (hybrid ISA-PRA) or Option 5 (PRA) in Section 2.2). The purpose of the plant-specific risk assessment is to assess risk, aid in the design of the facility, and provide a comparison with the general or standard risk assessment of the plant design. Information should be provided to demonstrate that the design of the facility falls within the site characteristics and design parameters specified in the ESP. In addition, the plant-specific risk assessment information must use the risk assessment information for the design approval and should be updated to account for site-specific design information and any design changes or departures.

2.6.3 Rationale

Under 10 CFR Part 52, the NRC authorizes applicants to construct and operate nuclear power through a one-step licensing process, or COL. A COL for a nuclear power plant is valid for 40 years from the date on which the Commission makes a finding that acceptance criteria are met under Section 52.103(g). A COL can be renewed for an additional 20 years.

The staff is considering a similar licensing approach for reprocessing facilities because the NRC expects to receive a limited number of applications for reprocessing facilities. The NRC staff concludes it is appropriate that a one-step licensing approach with an inspection process and ITAAC be implemented for licensing reprocessing facilities. This framework would simplify the licensing process for a reprocessing facility by permitting an applicant, in developing its facility, to combine both the licenses for construction and operation of the facility in a one-step licensing process.

Section 185b of the AEA establishes the role of a combined construction and operating license and the incorporation of ITAAC in the COLs for production facilities:

> After holding a public hearing under Section 189a.(1)(A), the Commission shall issue to the applicant a combined construction and operating license if the

> application contains sufficient information to support the issuance of a combined license and the Commission determines that there is reasonable assurance that the facility will be constructed and will operate in conformity with the license, the provisions of this Act, and the Commission's rules and regulations. The Commission shall identify within the combined license the inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that, if met, are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of this Act, and the Commission's rules and regulations. Following issuance of the combined license, the Commission shall ensure that the prescribed inspections, tests, and analyses are performed, and, prior to operation of the facility, shall find that the prescribed acceptance criteria are met. Any finding made under this subsection shall not require a hearing except as provided in Section 189a.(1)(B).

In the staff requirements memorandum to SECY–06–066, "Regulatory and Resource Implications of a Department of Energy Spent Nuclear Fuel Recycling Program" dated May 16, 2006 (NRC, 2006), the Commission directed the staff to consider the most effective and efficient elements of the NRC's licensing process to develop such a process for SNF reprocessing facilities. The Commission directed the staff to review the one-step licensing provisions for enrichment facilities as described in AEA Section 193 and the features of nuclear power plant combined licensing under 10 CFR Part 52 (i.e., construction authorization, operating license hearing process, design certification process, and ESP process).

In SECY–09–0082 (NRC 2009a), the NRC staff identified that clarity is needed to provide reasonable assurance that a reprocessing facility undergoing a one-step licensing process will be constructed and will operate in conformity with the AEA and the Commission's rules and regulations. In addition, SECY–09–0082 noted that to accommodate one-step licensing, it may be necessary to establish a requirement for one-step applications to submit a plan akin to that required under 10 CFR Part 52 for ITAAC. The staff believes that an inspection process and voluntary ITAAC are suitable replacements for the traditional ITAAC. One commercial company has written a letter to the Commission expressing interest in constructing a reprocessing facility and indicating that a one-step licensing process is needed for the same reasons that support the reactor COL process [AREVA (2008), ADAMS Accession No. ML081280528].

The staff will assess the flexibility to combine in a single license activities that would otherwise be licensed separately. This is similar to the process laid out for reactors in 10 CFR 52.8, "Combining Licenses; Elimination of Repetition" [see Gap 1 (Section 2.3)]. This flexible framework may be particularly advantageous for a reprocessing facility that may contain several different types of facilities on a contiguous site, such as spent fuel storage, waste solidification including vitrification, processing of plutonium or minor actinides or both and fuel fabrication, waste storage and processing, and storage of new fuel. The staff believes that this approach would provide efficiencies in the licensing process for reprocessing facilities.

2.6.4 Alternative Approaches

The NRC staff considered a number of alternative approaches to licensing a SNF reprocessing facility. The NRC could apply the COL process with different options. One option would be to revise 10 CFR Part 52 to include reprocessing facilities. This may be appropriate, as several sections of 10 CFR Part 52 already contain references to reprocessing facilities. However, this process could be cumbersome and require many exemptions and exceptions if only a few reprocessing facilities are constructed and no standard designs are available. Another approach would have the proposed rule for reprocessing facilities reference the appropriate section of 10 CFR Part 52. This approach would entail editing 10 CFR Part 52 so as to expand its scope to cover reprocessing facilities. In a third option, the staff would base sections of the proposed 10 CFR Part 7x on the applicable sections of 10 CFR Part 52. This seem the most likely approach as those sections of 10 CFR Part 52 that apply to reprocessing facilities could easily be incorporated in the new 10 CFR Part 7x.

(1) The NRC could adopt a traditional two-step license approach in which an applicant submits an application to construct a reprocessing facility, followed by an application to operate the reprocessing facility. This two-step licensing approach would allow the applicant the flexibility to construct and operate a reprocessing facility and would allow for additional time to address uncertainties in the construction and operation of the facility, should they arise. However, this approach may introduce additional costs, increase the uncertainty of getting a license after the reprocessing facility has been constructed, and increase the time to construct and operate the facility.

(2) The NRC would not permit an applicant for a reprocessing facility to reference an ESP, and the applicant would be required to address all siting issues in the COL application.

(3) The staff could allow preapproval of standard designs; require an applicant to obtain standard design certificates and standard design approvals; and require manufacturing licenses to be obtained, where the Commission issues a license authorizing manufacture of reprocessing facility components to be installed at sites not identified in the manufacturing license application. Because the staff expects only a few reprocessing facilities representing two different separation processes to be designed and constructed, these approaches appear costly and cumbersome, and the same information can be incorporated within the COL license application.

(4) The proposed regulation could have several appendices, similar to the current appendices in 10 CFR Part 52, that provide the requirements for design certifications for specific reprocessing separation processes and facility designs. This alternative is not needed if only a few reprocessing facilities are constructed. Requirements for information on the different reprocessing separation processes and facility designs can be incorporated into the proposed regulation. If the technical information is not known when the regulations become available, the staff will leave the section of the regulation blank and label it “Reserved.” The staff could complete such a section later as the information becomes available.

2.6.5 Stakeholder Views

NEI has issued a white paper (NEI, 2008) that includes a framework which could allow an applicant the flexibility for either a two-step or a one-step licensing process. NEI also supports the development of ITAAC to ensure that a reprocessing facility is built and operated in accordance with the application.

Comments from participants and the public at three public meetings held in September 2010 (NRC 2010a, p. 52, lines 9–19; pp. 53–54, lines 23–3; p. 91, lines 16–19), October 2010 (NRC 2010b, p. 16, lines 18–25; p. 17, lines 1–18); and June 2011 (NRC 2011a, p. 127, et seq.) on whether one-or two-step licensing processes should be used were mixed. Some said that they liked the flexibility of the two-step processes, while others favored a one-step (COL) process.

In a letter expressing its interest in constructing a reprocessing facility, one company indicated that a one-step licensing process is needed for the same reasons that support the reactor COL process [AREVA (2008) ADAMS Accession No. ML081280528].

2.6.6 Guidance Documents

Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued June 2007 (NRC, 2007), could form the basis of similar guidance for reprocessing facilities. Revised guidance would explain how to prepare and submit a COL application for a reprocessing facility.

Regulatory Guide 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52," issued October 2009 (NRC, 2009b), could form the basis for guidance for ITAAC closure for reprocessing facilities.

The staff will update Regulatory Guide 3.26, "Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants" (NRC, 1975), for use by applicants to identify more detailed information and the depth of detail in the safety analysis reports required in the COL regulations.

2.6.7 Conclusions

The NRC staff concludes that a COL process similar to the requirements in 10 CFR Part 52 that combines the construction authorization and operating license into a single application is appropriate for licensing SNF reprocessing facilities. The staff further concludes that before the commencement of operation of a reprocessing facility, the Commission should verify through an inspection process and ITAAC that the reprocessing facility has been constructed in accordance with the requirements of the license for construction and operation. The NRC should provide requirements and procedures to allow for the issuance of an ESP for approval of a reprocessing facility site, separate from the filing of a COL application. This would allow the one-step license application to reference an ESP. The COL process should not include preapproved standard designs and design certifications. The NRC should address specific requirements for aqueous and electrochemical separation processes in separate sections of the proposed regulation.

2.6.8 References

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2.6.9 Bibliography

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(2) Commission Papers Involving COLs (10 CFR Part 52) and ITAAC

SECY–92–214, "Development of Inspections; Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications," 1992.

SECY–00–0092, "Combined License Review Process," April 20, 2000.

SECY–02–0067, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC)," April 5, 2002.

SRM–04–0032, "Programmatic Information Needed for Approval of a Combined License without Inspections, Tests, Analyses and Acceptance Criteria," February 26, 2004.

SECY–05–0197, "Review of Operation Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005.

SECY–08–0117, “Staff Approach to Verify Closure of Inspections, Tests, Analyses, and Acceptance Criteria and to Implement Title 10 CFR 52.99, ‘Inspection During Construction,’ and Related Portion of 10 CFR 52.103(g) on the Commission Finding,” August 7, 2008.

SECY–09–0119, “Staff Progress in Resolving Issues Associated with Inspections, Tests, Analyses, and Acceptance Criteria,” August 26, 2009.

(3) Commission Meetings Involving 10 CFR Part 52 and ITAAC
Commission Meeting, “Periodic Briefing on New Reactor Issues, September 22, 2009

(4) ACRS Meetings on 10 CFR Part 52 and ITAAC: October 8, 2008, “New Plant Licensing Process, an Overview,” by Office of New Reactors

(5) Public Meetings

Meeting with Industry’s Design-Centered Working Groups to Discuss Generic Topics Related to Combined License Applications—US NRC; Rockville, MD, January 20, 2009.

(6) Waste Management Symposia

“Then and Now—and Where to? New Reactor Licensing from a Regulator’s Perspective,” WM2010, March 7–11, 2010.

(7) NEI View, Comments, and Reports

NEI 06-14, “Quality Assurance Program Description,” Revision 7, July 2009.

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3 WASTE AND ENVIRONMENTAL CONSIDERATIONS (GAPS 2, 3,15, 16, and 19)

3.1 Introduction

The civilian reprocessing of spent nuclear fuel (SNF) involves the separation of desirable radionuclides from other elements in the fuel. These separation processes generate liquid and solid radioactive wastes and could release gaseous radionuclides into the atmosphere. Consistent with the Atomic Energy Act of 1954, as amended (AEA), the U.S. Nuclear Regulatory Commission (NRC) is authorized to establish regulations for ensuring that civilian nuclear facilities, such as a potential reprocessing facility, can operate safely and securely. The NRC's Strategic Plan (NRC, 2008) expands upon this concept by incorporating protection of public health and safety, promotion of the common defense and security, and protection of the environment as essential mission functions for the NRC. A fundamental goal of this mission is ensuring adequate protection in the secure use and management of radioactive materials, which includes the radionuclides associated with SNF reprocessing.

In this chapter, the staff will discuss the following five technical gaps that define issues specific to waste and environmental considerations:

(1) Gap 2: Independent storage of high-level waste (HLW)
(2) Gap 3: Waste incidental to reprocessing
(3) Gap 15: Waste confidence
(4) Gap 16: Waste classification
(5) Gap 19: Effluent control and monitoring

For the independent storage of HLW, the staff noted that the existing regulatory framework does not accommodate the interim storage of HLW and SNF at a civilian reprocessing facility. Resolving Gap 2 is a high priority for rulemaking, as SNF and HLW storage are integral components of a potential reprocessing facility. In Gap 3, the NRC staff identified a longstanding issue related to the storage of liquid HLW at legacy reprocessing sites, which involves a determination of whether generated wastes are HLW or low-level waste (LLW). Although onsite disposal is not being considered for a potential reprocessing facility, the staff determined that developing an appropriate basis to distinguish between HLW and LLW streams is a high-priority issue.

The three remaining gaps represent technical issues that can be accommodated within existing regulatory frameworks but that can be enhanced by additional considerations. In Gap 15, the staff recognized that the existing regulation for long-term storage, Title 10 of the *Code of Federal Regulations* (10 CFR 51.23, "Temporary Storage of Spent Fuel after Cessation of Reactor Operation—Generic Determination of No Significant Environmental Impact" (Waste Confidence Rule), applies only to SNF generated in a reactor. The staff identified the need to address the potential environmental impacts from long-term storage of HLW resulting from reprocessing. In Gap 16, the NRC's LLW classification tables in 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," do not consider some of the radionuclides resulting from reprocessing SNF, which can lead to uncertainty on a safe disposal pathway. The staff also recognized, in Gap 19, that the presence of fission products in gaseous and liquid forms heightens the need for appropriate effluent controls and monitoring at a potential reprocessing facility.

3.2 Independent Storage of High-Level Waste (Gap 2)

3.2.1 Regulatory Issue

This section addresses independent storage of HLW. HLW should not be confused with reactor-generated spent fuel. The term high-level waste is used in this section to refer to radioactive materials, other than spent reactor fuel, that would require geologic disposal. A reprocessing facility for SNF will likely generate quantities of HLW that will require safe and secure storage before disposal. The Nuclear Waste Policy Act of 1982 (NWPA) and 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," provide for general or specific licenses for the storage of SNF at licensed commercial sites. Under NWPA Section 141, HLW can be stored at a monitored retrievable storage (MRS) installation, which the U.S. Department of Energy (DOE) designed and managed (42 U.S.C. 10161). DOE has not stated that it plans to develop an MRS installation; thus, changes may be needed for licensing storage of commercially generated HLW from reprocessing.

3.2.2 Basis for Requested Change

The staff proposes to incorporate the requirements for safe and secure HLW interim storage as part of the general license for a potential reprocessing facility, an approach that is similar to regulation of SNF interim storage at nuclear power plants. A fundamental premise underlying this approach is that a potential reprocessing facility, like a nuclear reactor facility, would already contain the physical and managerial infrastructure needed to meet the requirements for safe interim storage in 10 CFR Part 72. Authority to issue this general license, however, would require modifying 10 CFR Part 72, Subpart K, "General License for Storage of Spent Fuel at Power Reactor Sites," to include the interim storage of both SNF and HLW at a licensed reprocessing facility. Currently, the NRC can authorize a general license to store SNF to nuclear power reactors licensed 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," pursuant to 10 CFR 72.210.

Similar to the regulatory approach in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC # 61, 62, and 63), the NRC staff proposes to include general design criteria (GDC) in a new 10 CFR Part 7x to address safety considerations specific to a reprocessing facility, such as (i) providing SNF and HLW storage, handling, and radioactivity control; (ii) preventing criticality in storage and handling; and (iii) monitoring conditions under a proposed 10 CFR Part 72 general license. Proposed GDC for reprocessing facilities are detailed in Chapter 2. Establishing GDC for SNF and HLW management for reprocessing waste, in addition to the requirements established under a proposed 10 CFR Part 72 general license, would provide a level of protection that is consistent with the potential large radionuclide inventories and waste forms anticipated in a reprocessing facility.

The Commission has a long-established safety policy (codified in 10 CFR Part 50, Appendix F, "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities") that liquid HLW from reprocessing operations should be solidified before storage (NRC, 1970). Consistent with this policy, the staff believes that a new 10 CFR Part 7x should include the requirements in 10 CFR Part 50, Appendix F (Item 2) for HLW stability and storage. Chapter 1 contains additional discussion of Appendix F. The staff does not consider it necessary to propose either the restrictions in Appendix F (Item 2) on the amount of solidified

HLW that can be stored at a potential reprocessing facility or the time requirements for removing solidified HLW to a disposal facility, because the staff will require licensing of storage canisters for HLW that meet 10 CFR Part 72 requirements. The staff's view is that a new 10 CFR Part 7x should require the removal of all significant radioactive wastes at the time of facility decommissioning, which is consistent with existing requirements in 10 CFR Part 50, Appendix F (Item 4).

The staff assumes that solidified HLW from reprocessing would likely be stored in the same types of canister systems that are currently used to store commercial SNF. 10 CFR Part 72, Subpart L, "Approval of Spent Fuel Storage Casks," provides regulatory requirements for the approval of SNF storage casks but has no provisions for approving casks for HLW storage. The NRC would need rulemaking to amend Subpart L to allow for cask certification for HLW storage and to identify technical requirements for the safe interim storage of HLW from reprocessing (e.g., 10 CFR 72.236, "Specific Requirements for Spent Fuel Storage Cask Approval").

The staff is considering the need to establish reasonable limits on the amount of SNF stored at a potential reprocessing facility. Staff is concerned that without limits a licensee could establish a de facto ISFSI at the facility. Storing SNF beyond that needed for efficient reprocessing operations could potentially impact facility safety. SNF storage could be authorized to allow for efficient reprocessing operations by providing a stockpile of raw materials to accommodate variations in the rate that material can be processed through the facility. This limit would distinguish the proposed facility from an independent spent fuel storage installation that would require a separate 10 CFR Part 72 license.

As an alternative to the 10 CFR Part 72 "general" license approach, which applies to nuclear power reactors and is issued along with the reactor's 10 CFR Parts 50 or 52 license, the staff considered initiating an additional rulemaking in 10 CFR Part 72 to develop requirements for issuing a "specific" license for a separate installation to store HLW associated with the reprocessing facility. Although 10 CFR Part 72 allows interim storage of SNF at a commercial reactor or independent interim spent fuel storage installation, this regulation does not authorize commercial storage of HLW. Provisions exist in 10 CFR Part 72 for licensing a DOE-operated MRS installation to store solidified HLW from reprocessing; however, there is no national program to develop such an installation. Thus, revisions to 10 CFR Part 72 would be needed to develop an appropriate regulatory framework for specific licensing of an independent storage installation for HLW from reprocessing. These revisions would supplement revisions identified for 10 CFR Part 72, Subparts K and L referenced previously. The staff determines that, although it could revise 10 CFR Part 72 to provide for a specific license for commercial HLW and SNF storage at an independent waste storage installation associated with a reprocessing facility, it anticipates very few applications for a reprocessing facility in the near term. Thus, the staff concluded that an extensive revision to 10 CFR Part 72 is not necessary or efficient at this time to meet the HLW storage needs of these potential reprocessing facilities.

The NRC has considered the technical challenges associated with safe interim storage of solidified HLW as part of the 10 CFR Part 72 rulemaking to regulate an MRS installation, as authorized in the NWPA. In that rulemaking, the NRC stated that "From a technical stand-point storage of solidified waste is not significantly different from storage of spent fuel …" (NRC, 1986). The basis for this view was that "(1) HLW would be solidified in containers which can be handled and stored in the same manner as spent fuel containers, (2) the HLW form will be at least equivalent to spent fuel as a potential leaching barrier, (3) the heat and radioactivity associated with the HLW package will be equivalent or less than the heat and radioactivity associated with the packaged spent fuel, (4) there is no criticality problem because the special nuclear material content is so low, and (5) no radioactive gases and little radioactive iodine are associated with solidified HLW" (NRC, 1986).

This reasoning led the NRC staff to conclude that the existing 10 CFR Part 72 was "… generally applicable to the design, construction, operation, and decommissioning of a monitored retrievable storage installation" (NRC, 1986). Additionally, detailed DOE assessment (DOE, 1979) concluded that there were no significant differences in potential health effects between various fuel cycle management options, including the generation of additional HLW from SNF reprocessing.

More recent NRC and DOE assessments continue to support the conclusion that the interim storage of HLW is not significantly different from SNF interim storage (DOE, 2008). Potential hazards and risks from dry cask HLW storage have been analyzed, for example, as part of the NRC staff's review of the Private Spent Fuel installation (NRC, 2001).

Thus, the staff concludes that solidified HLW from reprocessing would likely be stored in the same types of canister systems that are currently used to store commercial SNF. Technical information for SNF storage provides a reasonable basis to develop appropriate requirements for a general license to store HLW at a commercial reprocessing facility, with general license authority granted through a revised 10 CFR Part 72. Currently, 10 CFR Part 72, Subpart K, permits the issuance of a general license for SNF storage at the site of a nuclear power reactor licensed under 10 CFR Parts 50 or 52. Subpart K also contains regulatory requirements for issuing this general license. To issue a general license for HLW storage at a commercial reprocessing facility, rulemaking would be needed to amend Subpart K to authorize this action. This rulemaking would also need to allow issuance of a general license for SNF storage at a reprocessing facility licensed under a new 10 CFR Part 7x. 10 CFR Part 72, Subpart L, "Approval of Spent Fuel Storage Casks," provides regulatory requirements for the approval of SNF storage casks, but has no provisions for approving casks for HLW storage. Based upon the previous technical information, the staff finds it reasonable that solidified HLW from reprocessing would likely be stored in the same types of canister systems that are currently used to store commercial SNF. The NRC would need rulemaking to amend 10 CFR Part 72, Subpart L to allow for cask certification for HLW storage and to identify technical requirements for the safe interim storage of HLW from reprocessing, similar to the technical requirements found in 10 CFR 72.236, "Specific Requirements for Spent Fuel Storage Cask Approval."

3.2.3 Stakeholder Views

Stakeholder views expressed in public meetings focused primarily on the need for a safe disposal pathway for wastes resulting from reprocessing. Much of this concern arises from the ongoing challenges of waste treatment, storage, and disposal from legacy reprocessing sites and the desire to ensure that such challenges do not occur at a future reprocessing facility. Some view redirection of the national repository disposal program as a fundamental argument against the production of any additional HLW, although other stakeholders view reprocessing as beneficial in reducing SNF volumes and, thus, facilitating repository development. Overall, stakeholders are concerned that a clear pathway remains elusive for the disposal of HLW resulting from reprocessing and worry that the default solution to this problem would be simply to allow the HLW to remain onsite indefinitely.

Kellar (2008) advocated, on behalf of the Nuclear Energy Institute (NEI), an approach to HLW storage that is similar to the approach used in 10 CFR Part 50 for storage of SNF at power reactors. This proposed approach uses baseline design criteria to provide technical requirements for the storage of both SNF and HLW:

> (10) Criterion 10—Fuel storage and handling and radiological control. This criterion is derived from 10 CFR Part 50, Appendix A, Criterion 61 and is to ensure that systems designed to handle radioactive materials, including both fuel product and wastes, are designed to ensure adequate safety under normal and postulated accident conditions. To ensure that the systems accomplish these objectives the design must address several attributes which are described in the Criterion. Achieving these objectives during the design of these systems will ensure that safety for the public and for the facility operators is achieved and maintained.
>
> (11) Criterion 11—Monitoring fuel and waste storage. This criterion is derived from 10 CFR Part 50, Appendix A, Criterion 63. With large source terms and the associated decay heat produced, heat removal systems must be available at a recycling facility to remove the heat and maintain the materials within the design temperature limitations of the facility. This criterion ensures that the needed systems to accomplish these objectives as described in Criterion 10 are provided with adequate and reliable monitoring systems to ensure that adequate protection is available.

On October 28, 2009, the NRC held a Category 2 public meeting with the NEI staff to discuss waste and safeguards issues. During that meeting, NEI provided its perspectives on the storage of HLW. NEI concluded that the storage of both SNF and HLW could be accomplished as a general license under a new 10 CFR Part 7x, with appropriate design basis criteria.

3.2.4 Guidance Documents

The regulatory process for the issuance of a general license authority to store SNF at a licensed reactor is outlined in NUREG–1571, “Information Handbook on Independent Spent Fuel Storage Installations” (NRC, 1996). As discussed in NUREG–1571, 10 CFR 72.212(b)(2) requires that the licensee shall establish, in written evaluations, that applicable regulatory requirements in 10 CFR Part 72 are met. These requirements include additional review to determine that site parameters are enveloped by the design basis of the storage casks, that appropriate operating and training procedures are developed, that physical protection has been established, and that the site emergency plan includes the storage facility. Although the same fundamental licensing approach is envisioned for SNF and HLW interim storage at a potential reprocessing facility, NUREG–1571 would require revision to expand the scope of the licensing review to encompass both HLW storage and a reprocessing facility.

Dry cask storage systems for storage of SNF must have a certificate of compliance. A similar certification should be required for the storage of HLW. NUREG–1536, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility” (NRC, 2010), provides guidance to the NRC staff conducting the safety review of an application for a certificate of compliance for SNF dry cask storage systems for use at a general license facility. The review is primarily conducted using the information an applicant or cask vendor submits in a safety analysis report that is required by 10 CFR 72.230, “Procedures for Spent Fuel Storage Cask Submittals.”

The NRC would need a standard review plan for an HLW storage system and would use applicable information in NUREG–1536 (NRC, 2010). Particular aspects that pertain to specific fuel-related issues, such as damaged cladding and criticality concerns, would not be retained, because these issues are not applicable to storage of HLW.

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3.3 Waste Incidental to Reprocessing (Gap 3)

3.3.1 Regulatory Issue

In NWPA Section 2 (12)(A), Congress defined HLW as the “highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations.” Because of the high levels of radioactivity that may persist for long periods of time, Congress determined in NWPA Section 111(a) that such wastes required disposal in a deep geologic repository to protect public health and safety.

The staff recognizes that wastes resulting from reprocessing can originate through a broad range of processes, including the physical separation of fuel assemblies and the chemical separation of radionuclides from SNF. The concentration of radionuclides in these wastes will necessarily vary. Radioactive wastes from a commercial reprocessing facility will need to be disposed of at an appropriate waste disposal facility because differences in radioactivity present different public health and safety concerns. Therefore, a method of distinguishing between HLW and LLW associated with reprocessing SNF is necessary to ensure that appropriate safety requirements are met for both interim storage and ultimate disposal.

3.3.2 Basis for Requested Change

In 1969, the Atomic Energy Commission (AEC) published a draft policy statement that assumed that SNF would be reprocessed and the residual uranium and plutonium would be recycled as fuel (AEC, 1969). The draft 10 CFR Part 50, Appendix D (AEC, 1969) proposed that certain reprocessing wastes did not have to be subject to geologic disposal as HLW in a federally operated facility. The intention was to dispose of these other radioactive wastes, meaning the non-HLW, into what today would be a commercial LLW near-surface disposal facility, provided that it met the requirements of 10 CFR 20.302 (i.e., the predecessor of today’s 10 CFR Part 61, “Licensing Requirements for Land Disposal of Radioactive Waste”) (AEC, 1969). Paragraphs 6 and 7 of the proposed Appendix D (AEC, 1969) stated that other types of waste, such as radioactive hulls and other hardware and solid waste resulting from reprocessing operations, could be disposed of in licensed waste burial facilities on Federal- or State-owned land. In particular, Appendix D, Paragraph 7 (AEC, 1969) stated

> [O]ther solid wastes resulting from operation of commercial fuel reprocessing plants, such as ion-exchange beds, asphalted sludges, vermiculited sludges, and contaminated laboratory items, clothing, tools, and equipment must be disposed of in accordance with Commission regulations for the disposal of such materials

> in Part 20 of this chapter (e.g., disposal at a licensed waste burial facility located on land owned by the Federal Government or by a State Government).

In 1970, the AEC finalized the proposed policy statement as Appendix F to 10 CFR Part 50 (AEC, 1970) and defined HLW for the purposes of that policy statement as

> Those aqueous wastes resulting from the operation of a first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels.

Portions of the draft policy statement (AEC, 1969) that discussed incidental wastes were omitted, as the Commission noted that it wanted to preserve its flexibility on how such wastes would be treated in the future (AEC, 1970). Given this policy, HLW was defined as any material left after fuel reprocessing. Consequently, HLW was defined as the liquid wastes resulting from a particular source (i.e., reprocessing) rather than the waste's constituents or radiological properties.

In 1982, the NWPA provided further clarification regarding the earlier definition of HLW. This clarification included specific legislative reference to SNF. NWPA Section 2(12) defined "high-level waste" as

> …(A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and
>
> (B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation….

However, because the terms "highly radioactive" and "sufficient concentrations" are not defined, various interpretations of these terms have been used to determine what wastes from reprocessing might be considered HLW. For example, the NRC generic regulations for geologic disposal (10 CFR Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories") included in the definition of HLW those "… liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel …." (10 CFR Part 60.2). The first cycle solvent extraction system criterion was not included, however, in the site-specific disposal regulation for geologic disposal at Yucca Mountain, which maintained the NWPA Criterion "A" in the HLW definition (10 CFR Part 63, "Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada").

In 1990, the States of Oregon and Washington petitioned the Commission to amend its regulations to redefine HLW to be other than source based. The petition was motivated by concerns that the bulk of contents from several underground radioactive waste storage tanks at the Hanford Nuclear Reservation, which were undergoing remediation at the time, contained HLW and, thus, were subject to NRC licensing jurisdiction. DOE was planning to remove, solidify (i.e., vitrify), and dispose of the high-activity fraction of the tank wastes as HLW but planned to treat (i.e., grout) the residual low-activity fraction in the tank and dispose of it in place. The petition requested that the Commission establish a procedural framework for determining, on a tank-by-tank basis, whether the contents of the Hanford tanks were HLW or incidental, low-activity wastes (NRC, 1990). In 1993, the Commission denied the petition, arguing that the principles for waste classification were well established and could be applied on a case-by-case basis to the tanks in question without revising the regulations, as the petitioners proposed (NRC, 1993).

Soon thereafter, and in connection with the planned remediation of the Hanford tanks, the NRC staff (Bernero, 1993) informed DOE that it would consider the residual fraction of the separated wastes removed from the tanks as "incidental" (so named for the first time) provided that the waste

(1) Has been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical

(2) Will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR Part 61

(3) Will be managed, pursuant to the Atomic Energy Act, so that safety requirements comparable to the performance objectives set out in 10 CFR Part 61 are satisfied

Note that the NRC does not have authority to regulate wastes at the DOE Hanford site but provided an independent assessment of DOE's technical approach for determining its waste incidental to reprocessing (WIR). Nevertheless, in its final policy statement for decommissioning the West Valley site (NRC, 2002), the Commission took the position that Criteria 1 and 3 should be applied to the WIR determinations at that site. Criterion 2 was not included in the Commission's position, because this criterion was not integral to safety concerns.

The staff is considering various options to classify certain types of wastes resulting from reprocessing as LLW instead of HLW. The staff determines that there is a need to develop a practicable approach to determining what materials are considered "highly radioactive" in the definition of HLW and, thus, require deep geologic disposal, in contrast to those lower activity wastes that could be safely disposed of in a near-surface facility that meets the radioactive disposal requirements of 10 CFR Part 61. The staff believes wastes that are not "highly radioactive" can be safely disposed of in a near-surface disposal facility, provided the wastes in question could meet the requirements for disposal specified in 10 CFR Part 61.

Consistent with current decommissioning practices, the staff will require removal of all significant amounts of radioactive waste from a potential reprocessing facility as a condition for decommissioning the site. Consistent with 10 CFR Part 50, Appendix F(4), a new 10 CFR Part 7x would also require the removal of all significant radioactive wastes at the time of facility decommissioning. The NRC decommissioning practice is different than DOE's approach to management of WIR. Previous DOE WIR evaluations have focused on determining what radioactive materials resulting from reprocessing could be safely disposed of at the site (in situ) where they were produced. The staff is concerned that adopting a WIR definition in a new 10 CFR Part 7x could be misinterpreted as facilitating onsite disposal of wastes. In addition, existing criteria for the removal of key radionuclides "to the extent practical" do not appear applicable to the safe management and disposal of future wastes from reprocessing. Thus, the staff does not recommend including a WIR definition in future regulations for a reprocessing facility.

The staff anticipates that much of the waste resulting from reprocessing SNF will be highly radioactive and will contain fission products in sufficient concentrations to be considered HLW, as defined in the NWPA, and require disposal in a mined geologic repository. Many of the first stages in radionuclide separations generate highly concentrated liquid extracts, which require careful management during operations to achieve appropriate chemical and thermal stability. In addition, many parts of fragmented SNF assemblies (e.g., ion exchange resins and filters) would likely be highly radioactive and require management and disposal as HLW. Nevertheless, from a risk-informed perspective, the staff is open to the possibility that some materials resulting from reprocessing would not be so highly radioactive as to unilaterally require disposal in a mined geologic repository. For those reprocessing wastes that have radionuclide concentrations that would permit shallow disposal under 10 CFR Part 61, the staff concludes that public health and safety is protected by allowing such disposal.

To develop a practical basis for distinguishing HLW and LLW from reprocessing, the staff is considering the following three alternatives:

(1) Develop a legislative proposal to Congress that would provide exceptions to the definition of HLW similar to those implemented in the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005, redefining HLW to exclude much of the waste that is not highly radioactive.

(2) Issue a regulation to clarify the meaning of “highly radioactive” and “sufficient concentrations” in the context of HLW. This rulemaking would allow for differentiation between the types of reprocessing wastes that would be considered HLW and those types that might be considered LLW, thereby allowing for different disposal strategies.

(3) Take no action so that all highly radioactive waste streams associated with reprocessing SNF would still be considered HLW, as defined in the NWPA.

3.3.3 Stakeholder Views

The staff held several interactions with stakeholders, including a number of Category 2 public meetings with the NEI staff and two public workshops (Rockville, Maryland, on September 18–19, 2010, and Albuquerque, New Mexico, on October 19–20, 2010). During these meetings, the waste issues discussed included the subject of non-HLW resulting from reprocessing operations. Among the participants in these sessions were representatives of other Federal agencies, potential licensees, members of the international nuclear community, and public interest groups. Transcripts of some of these meetings are available on the NRC’s Web site at http://www.nrc.gov/materials/reprocessing.html.

The industry opinion expressed throughout these public proceedings was that the NRC should develop a hazard-based methodology for waste classification, which would be applicable to different types of reprocessing technology and their subsequent waste management technologies. In this methodology, the NRC should include a category of WIR, consistent with the way in which it has been applied to some of the DOE wastes.

The NEI white paper (Kellar, 2008) contains a definition of “waste incidental to recycling (WIR),” which was added to clarify what NEI believed was not HLW. NEI derived the definition from the

Commission's decision in the decommissioning criteria for the West Valley Demonstration Project (NRC, 2002) and of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 Section 3116. The NEI-preferred definition of WIR is the following:

> Waste material resulting from recycling of spent nuclear fuel, including liquid wastes produced directly in recycling and any solid material derived from such liquid waste that contains fission products that is not so highly radioactive or contains insufficient concentrations of fission products to be classified as HLW. Such waste is not so highly radioactive or of sufficient concentration if it (1) has been processed to remove key radionuclides to the maximum extent that is technically and economically practical, and (2) either meets Class C concentrations under 10 CFR Part 61 or will meet the performance objectives in 10 CFR Part 61, Subpart C if disposed of in a near surface disposal site based on a site specific performance assessment. This definition does not relieve the Department of Energy from its responsibility for the disposal of radioactive material which is greater than Class C under the Low-Level Radioactive Waste Policy Act of 1985.

NEI believes that a definition of WIR is needed to clarify the terms "highly radioactive" and "sufficient concentration" that are included in the current source-based definition of HLW as set forth in the NWPA. NEI concludes that such a definition would provide regulatory certainty, predictability for the industry, and transparency for both the licensee and the regulator.

In support of the public meeting on October 28, 2009, NEI submitted a paper to the NRC on HLW insights (Lieberman and Greeves, 2009). This paper provides a history of the regulatory development of the definitions of HLW and "waste incidental to recycling" (i.e., WIR or incidental waste) and a perspective for concluding that WIR is not HLW. The development of the definitions of HLW and WIR supported the definitions of HLW and WIR in the proposed 10 CFR Part 7x submitted to the NRC in the NEI white paper (Kellar, 2008). As discussed in Lieberman and Greeves (2009), there is no intent to apply WIR to liquids, as the concept of WIR is that such material is suitable for near-surface land disposal under 10 CFR Part 61, which generally requires that disposed waste be in solid form to provide stability, in accordance with 10 CFR 61.56, "Waste Characteristics." This position is consistent with the WIR criteria NRC cited in denying a petition for rulemaking (NRC, 1993), which requires WIR to be in solid form. Lieberman and Greeves (2009) do not propose that waste from reprocessing would be disposed of onsite but rather offsite, at licensed disposal sites meeting the requirements of 10 CFR Part 61 or equivalent Agreement State regulations.

Some meeting participants opposed including a definition for WIR in NRC reprocessing regulations. One concern was related to the waste management of longer lived radioisotopes, such as iodine-129, which has a half-life of 1.57×10^7 years. The stakeholder questioned the ability to accurately model the performance of certain waste forms for several thousand years and, consequently, stated very long-lived radioisotopes should not be permitted for shallow burial, because of the inability to model the behavior of the waste form. In response, the industry recognized that categories of waste that fall within WIR, which are greater than Class C waste, would require some kind of engineered disposal other than a repository but more constrained than shallow-land disposal. Other public interest representatives expressed concern that any NRC implementation of the WIR concept would result in waste remediation problems like those experienced at the Savannah River National Laboratory and Idaho National Laboratory sites.

3.3.4 Guidance Documents

The section of this report on Gap 2 discusses guidance applicable to the storage of SNF from reprocessing. Although the same fundamental licensing approach is envisioned for SNF and HLW interim storage at a potential reprocessing facility, applicable guidance in NUREG–1571, "Information Handbook on Independent Spent Fuel Storage Installations" (NRC, 1996), would need revision to expand the scope of the licensing review to encompass both HLW storage and a reprocessing facility.

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3.4 Waste Confidence for Reprocessing Facilities (Gap 15)

3.4.1 Regulatory Issue

SECY–09–0082 (NRC, 2009) identified that applicants for reprocessing facility licenses will need to address the environmental impacts of long-term storage of HLW. Two options to address the environmental impacts of long-term storage of HLW are (1) revise the NRC's Waste Confidence Rule (10 CFR 51.23) to incorporate HLW generated at a reprocessing facility or (2) have applicants for a reprocessing facility license address the potential environmental impacts of long-term waste storage in the environmental reports submitted as part of a license application. Under the latter approach, the NRC would need to evaluate the long-term storage of reprocessing wastes in the environmental assessment or environmental impact statement for a reprocessing facility license.

3.4.2 Basis for Proposed Approach

The staff is proposing to require license applicants to include an evaluation of the potential environmental impacts from the long-term storage of waste from reprocessing in the environmental reports submitted as part of a license application. Similarly, the NRC staff would evaluate the environmental impacts of long-term storage of reprocessing waste in its environmental assessment or environmental impact statement.

In the original 1984 Waste Confidence Decision (NRC, 1984a) and subsequent updates in 1990 (NRC, 1990a) and 2010 (NRC, 2010a), the NRC examined available information and determined that the safe disposal in a mined geologic repository of either SNF or HLW, including solidified HLW resulting from reprocessing, is technically feasible. In addition, NWPA Section 302 establishes that DOE is authorized to enter into contracts with any domestic producer of HLW to take title, transport, and dispose of the waste. Section 302 also requires that before the NRC can issue a license, the applicant must have a contract with DOE or be in active negotiations for a contract with DOE for disposal of HLW. Thus, the NRC's confidence that safe disposal of HLW from reprocessing can occur is founded on both technical information and an established legal framework.

The NRC also stated in its Waste Confidence Decision Update (NRC, 2010a) that it has confidence that both SNF and HLW can be managed safely until disposal occurs (Finding 3). The basis for this finding is that the management of HLW and SNF will occur at a licensee's site and that compliance with applicable NRC regulations and specific license conditions will provide the assurance of safety.

In contrast, the generic environmental determination in the Waste Confidence Rule (NRC, 2010b) only touches on the long-term storage of SNF. Findings 2 and 4 (NRC, 2010a) provide the basis for the generic determination in the Waste Confidence Rule (NRC, 2010b). These findings are supported by technical information derived from several decades of nuclear power plant operating and licensing experience and associated SNF storage.

The staff considered recommending rulemaking to expand the existing Waste Confidence Rule in 10 CFR 51.23 to encompass the solidified HLW resulting from reprocessing at any facility licensed under the requirements of a new 10 CFR Part 7x. As discussed in the section of this report on Gap 2, the staff recognizes that substantial experience has been gained worldwide in licensing, operating, and regulating dry storage of solidified HLW, including HLW from reprocessing. This information suggests that the existing technical requirements for safe long-term storage of SNF might encompass the requirements for safe long-term storage of HLW from reprocessing.

However, several factors prevent the staff from recommending rulemaking to expand the Waste Confidence Rule to encompass long-term storage of HLW from reprocessing. In contrast to the decades of nuclear power plant licensing that preceded the original Waste Confidence Rule (NRC, 1984b), the NRC does not have comparable experience licensing commercial reprocessing facilities. The technical bases to support, or challenge, safe long-term HLW storage have not benefited from the scrutiny and review of a rigorous licensing process. Although the solidified HLW from reprocessing commonly is in a vitrified form, other waste forms are possible. Additionally, casks used to store SNF have not undergone a licensing certification to identify technical requirements for the safe storage of HLW. Although no single factor prevents the staff from concluding that the environmental impacts from long-term storage of HLW from reprocessing might be small or low, the scope and magnitude of existing knowledge gaps currently do not give the staff confidence that a sufficient technical basis exists, or could be developed in the near term, to support an expanded waste confidence rulemaking. Thus, the staff concluded that a recommendation for rulemaking to expand the Waste Confidence Rule was not an effective solution to closing Gap 15.

To meet applicable National Environmental Protection Act requirements, an applicant for a license to construct and operate a commercial SNF reprocessing facility would be required to evaluate all potential environmental impacts associated with the storage of HLW produced at the facility until DOE takes title to, and removes, the waste from the facility. The staff notes that the applicant's environmental report might consider the timeframe of the post-licensed life (60 years) evaluated for SNF in the Waste Confidence Decision and rule. The staff would evaluate the long-term storage of reprocessing wastes in the environmental assessment or environmental impact statement for a reprocessing facility license.

3.4.3 Stakeholder Views

In public meetings NRC has organized since 2008, the issue of waste confidence for reprocessing facilities has received few direct comments. Most of these comments addressed the overall framework of the proposed approach for waste management and tended to focus on uncertainties in the disposal pathway for HLW and SNF. Some comments expressed a lack of confidence in the generic waste confidence finding in 10 CFR 51.23, or in the national policy for SNF and HLW management. Taken as a whole, these comments tend to indicate support for the staff's proposal to require the evaluation of potential environmental impacts from long-term storage of HLW from reprocessing as part of the licensing process.

NEI (Kellar, 2008) expressed the view that the license application for a reprocessing facility would contain an environmental report meeting the requirements of 10 CFR Part 51. Kellar (2008) did not directly address the need to consider potential environmental impacts from the long-term storage of HLW if the requirements in 10 CFR 51.23 were not applicable. In an October 28, 2009, public meeting, the NEI staff expressed the view that the application for a potential reprocessing facility will address the environmental impacts of storage of solidified HLW.

3.4.4 Guidance Documents

The Gap 2 portion of this report discusses guidance applicable to the storage of HLW from reprocessing.

3.4.5 References

Kellar (2008). Kellar, Felix M., Nuclear Energy Institute, Letter to Michael F. Weber, U.S. Nuclear Regulatory Commission, December 19, 2008. (ADAMS Accession No. ML0835901300)

NRC (2010a). U.S. Nuclear Regulatory Commission, "Waste Confidence Decision Update," *Federal Register*, Vol. 75, No. 246, December 23, 2010, pp. 81037–81076.

NRC (2010b). U.S. Nuclear Regulatory Commission, "Consideration of Environmental Impacts of Temporary Storage of Spent Fuel After Cessation of Reactor Operation," *Federal Register*, Vol. 75, No. 246, December 23, 2010, pp. 81032–81037.

NRC (1990a). U.S. Nuclear Regulatory Commission, "Waste Confidence Decision Review Waste Confidence Decision Review," *Federal Register*, Vol. 55, No. 181, September 18, 1990, pp. 38474–38514.

NRC (1984a). U.S. Nuclear Regulatory Commission, "Waste Confidence Decision," *Federal Register*, Vol. 49, No. 171, August 31, 1984, pp. 34658–34688.

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NRC (2009). U.S. Nuclear Regulatory Commission, "Update on Reprocessing Regulatory Framework—Summary of Gap Analysis," SECY–09–0082, May 28, 2009.

NWPA, *Nuclear Waste Policy Act of 1982*, as Amended, Pub. L. No. 97-425, 96 Stat. 2201 (1983).

U.S. Code of Federal Regulations, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Part 51, Chapter I, Title 10, "Energy."

3.5 Waste Classification (Gap 16)

3.5.1 Regulatory Issue

The NRC based the development of 10 CFR Part 61 in the early 1980s on several assumptions regarding the types of wastes likely to be disposed of in a commercial LLW disposal facility. To better understand the likely inventory of such wastes, the NRC conducted a survey of existing LLW generators. The survey, documented in NUREG–0782, Chapter 3, “Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste: Main Report” (NRC, 1981), revealed that there were 37 distinct commercial waste streams consisting of 24 radionuclides of potential regulatory interest. The waste streams represented the types of commercial LLW being generated at the time. The survey did not consider waste streams associated with DOE’s nuclear defense complex, because those wastes would be disposed of at DOE-operated sites.

The suite of 25 radionuclides was subsequently assessed through a series of technical evaluations, which later demonstrated that only 12 were risk significant. These 12 radionuclides later formed the basis for the concentration information in 10 CFR 61.55, “Waste Classification,” Tables 1 and 2. Table 1 provides the limiting concentrations for certain long-lived radionuclides; Table 2 provides them for certain short-lived radionuclides.

The licensing and operation of any commercial reprocessing facility will produce several radioactive waste streams. Gaseous effluents released by a reprocessing facility would be regulated under 10 CFR Part 20, “Standards for Protection against Radiation,” using standards similar to the U.S. Environmental Protection Agency’s (EPA’s) National Emissions Standards for Hazardous Air Pollutants (NESHAP) regulations. Similarly, the NRC would regulate disposal of SNF and other HLW from reprocessing facilities under 10 CFR Parts 60 and 63. Finally, the disposal of any waste streams determined to be LLW will be regulated under 10 CFR Part 61.

The LLW classification tables in 10 CFR 61.55, Tables 1 and 2 include many radionuclides that may be associated with reprocessing commercial SNF. However, depending on the particular reprocessing technology, some SNF reprocessing waste streams may contain radionuclides that were not considered in the development of those tables. They include, for example, krypton-85 that would be separated from gaseous effluents, certain noble metals, and some isotopes from the lanthanide series. In addition to reprocessing-related LLW, other unevaluated waste streams identified for possible disposal in a near-surface disposal facility licensed under 10 CFR Part 61 include large quantities of highly concentrated depleted uranium; large-scale blended LLW; and possibly certain DOE-generated, defense-related LLW streams.

3.5.2 Basis for Requested Change

To address the potential impact of the disposal of large quantities of depleted uranium in a 10 CFR Part 61 disposal facility, the Commission directed the staff to undertake a limited rulemaking that would require 10 CFR Part 61 licensees to conduct a site-specific analysis before the disposal of large quantities of depleted uranium and other unique waste streams (i.e., NRC, 2008). The Commission also directed the staff to conduct public workshops to discuss issues being considered in the rulemaking and invite stakeholder input. During these workshops, the staff received significant comments regarding the scope of the rulemaking. Specifically, the staff was encouraged not to limit the scope of the rulemaking to depleted uranium but to allow the disposal of radionuclides on the basis of their risk.

Consistent with that approach, the rulemaking evolved into an analysis to evaluate the disposal of LLW streams at a 10 CFR Part 61 facility under a performance-based, risk-informed framework. The analysis would ensure that the LLW streams met the performance objectives in 10 CFR Part 61, Subpart C, "Performance Objectives," and would identify any additional measures that would enhance the protection of public health and safety. The NRC expects to complete this proposed rulemaking in 2012.

In a staff requirements memorandum (NRC, 2009), the Commission also directed the staff to incorporate large-scale LLW blending into the limited 10 CFR Part 61 rulemaking. The staff recommended soliciting stakeholder views on whether there should be amendments to the current 10 CFR Part 61 and, if so, what the nature of those amendments should be, before the NRC started the rulemaking process. The purpose of these meetings was to gather information from a broad spectrum of stakeholders concerning their continued support for the existing 10 CFR Part 61, recommendations for specific changes to the existing rule, or suggestions for possible new approaches to commercial LLW management.

The staff believes that it incorporated the original technical issue raised in Gap 16 into the subsequent 10 CFR Part 61 rulemaking (NRC, 2008). This rulemaking addresses the need to develop an appropriate basis for evaluating the safety of LLW disposal for radionuclides not included in 10 CFR 61.55, Tables 1 and 2 (e.g., radionuclides resulting from reprocessing SNF). Depending on the technology selected, there could also be some isotopes that are produced during reprocessing that are not currently in 10 CFR 61.55, Table 1 or 2. If such isotopes are produced, the disposal needs for these isotopes will be addressed in the ongoing 10 CFR Part 61 rulemaking.

3.5.3 Stakeholder Views

Stakeholder views are part of the ongoing rulemaking for unique waste streams (NRC, 2008). The NRC documented comments regarding this rulemaking at www.regulations.gov, under docket NRC–2009–0257. Stakeholder views expressed in public meetings on potential reprocessing regulations have supported the general approach staff outlined for the rulemaking on unique waste streams.

3.5.4 Guidance Documents

The staff anticipates developing new guidance documents to support the review of analyses for the disposal of unique waste streams. Existing guidance pertaining to LLW disposal does not appear to provide the scope and depth of information needed to support such reviews.

3.5.5 References

NRC (2010). U.S. Nuclear Regulatory Commission, "Blending of Low-Level Radioactive Waste," SECY–10–0043, April 7, 2010. (ADAMS Accession No. ML090410246)

NRC (2009). U.S. Nuclear Regulatory Commission, "Staff Requirements: SECY–08–0147—Response to Commission Order CLI–05–20 Regarding Depleted Uranium," SRM–SECY–08–0147, March 18, 2009. (ADAMS Accession No. ML090770988)

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U.S. Code of Federal Regulations, "Standards for Protection Against Radiation," Part 20, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Disposal of High-Level Radioactive Wastes in Geologic Repositories," Part 60, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Licensing Requirements for Land Disposal of Radioactive Waste," Part 61, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada," Part 63, Chapter I, Title 10, "Energy."

3.6 Effluent Controls and Monitoring (Gap 19)

3.6.1 Regulatory Issue

NRC regulations do not specifically address effluent control and monitoring at reprocessing facilities. The regulations in 10 CFR Part 50 that relate to this subject are geared primarily toward nuclear power reactors: 10 CFR 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors," and 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors"; Appendix A, "General Design Criteria for Nuclear Power Plants"; GDC 60, "Control of Releases of Radioactive Materials to the Environment"; and GDC 64, "Monitoring Radioactivity Releases". The NRC specifically developed 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," which provides quantitative values for design criteria for keeping offsite doses as low as reasonably achievable (ALARA), for light-water nuclear power reactors (LWRs). Regulations in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," may not be appropriately protective, because they were developed for facilities that do not have significant radioactive effluents and waste streams as nuclear power plants. Because reprocessing facilities have a relatively large inventory of radionuclides and a higher potential than both reactors and fuel cycle facilities to adversely affect the surrounding environment through effluent leakage, additional requirements in 10 CFR Parts 50 and 70 for effluent control and monitoring are warranted to protect public health, safety, and the environment.

3.6.2 Basis for Requested Change

SNF reprocessing facilities are expected to generate a larger and more varied source term, with more radionuclides in potentially mobile forms (e.g., liquids and gases) than at other fuel cycle facilities and nuclear power plants. In a nuclear power plant, for example, volatile fission products are contained, to a large extent, in the fuel assembly because of the effectiveness of cladding as a barrier. In a fuel reprocessing plant, however, the SNF is dismantled, releasing these radionuclides into the environment of the process vessel. The main gaseous fission products of concern are krypton-85, hydrogen-3 (i.e., tritium), carbon-14, and iodine-129. Therefore, reprocessing facilities have the potential to release considerable quantities of radionuclides into the environment through the offgas system if steps are not taken to mitigate releases and keep any potential releases within 10 CFR Part 20 dose limits. Regulations are required to ensure that adequate controls are put in place to protect the health and safety of people and the environment.

The staff determined that 10 CFR 50.34a is an appropriate model on which to develop regulations for effluent control, because it requires a licensee to describe the methods and equipment that it will use to control effluents and to stipulate the quantities of radioactive isotopes that will be released under normal operating conditions. The requirements are not specific to a type of technology and so would be consistent with a technology-neutral framework.

The staff determined that the NRC should also develop GDC, based upon those in 10 CFR Part 50, Appendix A, but with added requirements relating to holdup capabilities of both waste and effluents. Depending upon the location of a reprocessing facility, the Commission may want to consider implementing a rule that only allows a licensee to discharge effluents under certain meteorological conditions to ensure that dose limits are being met. Chapter 2 of this report discusses GDC in greater detail.

ALARA has been a central tenet of radiation protection for many years and is a requirement in many NRC regulations. Therefore, ALARA requirements should be integral to any regulation regarding effluents from reprocessing plants. ALARA requirements for reprocessing facilities could be based upon the existing requirements in 10 CFR 50.34(a), which require licensees to describe how, through stated design objectives and the means to be employed, they will keep radioactive releases to unrestricted areas ALARA. Based on published information, dose impacts from reprocessing facilities in other countries are low and well within applicable NRC regulatory dose limits. From the staff's perspective, these operations appear to appropriately consider ALARA principles. Based on this information, the staff anticipates that a potential reprocessing facility in the United States could reasonably be expected to have low levels of radionuclide releases and to meet applicable ALARA requirements.

In 1971, the AEC published for consideration 10 CFR Part 50, Appendix I, which provided numerical guidance for design objectives and technical requirements for limiting conditions for LWR operations to keep radioactivity in effluents as low as practicable (AEC, 1971). The NRC issued Appendix I in 1975 (NRC, 1975), after a lengthy comment process. Appendix I was developed prior to the Commission's adoption of the risk-informed and performance-based regulatory policy. Consequently, the staff would examine Appendix I to consider a risk-informed and performance-based approach to ALARA values for reprocessing facilities.

If the NRC staff decides that Appendix I-type regulations are necessary in 10 CFR Part 7x, it will perform studies to develop release limits and numerical guidelines that incorporate current information. One example is a study by Finney, et al. (1977), which contains a cost-benefit analysis to determine the effectiveness of radioactive waste treatment systems in decreasing the release of radioactive materials from a model nuclear fuel reprocessing facility for LWR fuels. This study also determined the radiological impact (i.e., dose commitment) of the released materials on the environment. The Finney, et al. (1977) study was designed to assist in defining the term ALARA in relation to limiting the release of radioactive materials from reprocessing facilities. An additional investigation by McMahon, et al. (2010) compared the technical bases EPA used in developing 40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations,” with current methods used to evaluate health effects. This investigation determined that EPA’s use of a collective dose approach makes it difficult to project health effects and that EPA overestimated its projections for the growth of nuclear power when making collective dose estimates. The McMahon, et al. (2010) study also highlighted numerous improvements in dose modeling made since the issuance of 40 CFR Part 190. The NRC would need to consider these and other studies if it developed Appendix I-type regulations. Such regulations also would need to comply with EPA’s regulations regarding radionuclides in 40 CFR Part 61, “National Emissions Standards for Hazardous Air Pollutants,” Subpart I, “National Emission Standards for Radionuclide Emissions from Federal Other Than Nuclear Regulatory Commission Licensees and Not Covered by Subpart H.”

Mandating the use of appropriately aged SNF in reprocessing is one potential approach to resolving issues about the capture of krypton-85. This approach could also apply to tritium because of its comparably short half-life; krypton-85 and tritium have half-lives of approximately 11 and 12 years, respectively. However, an ageing management strategy would not resolve concerns regarding carbon-14 and iodine-129, because of the relatively long half-lives for these fission products (5,700 and 1.57×10^7 years, respectively). Nevertheless, the use of older SNF also reduces the hazards associated with self-heat generation, some additional short-lived fission products (e.g., ruthenium-106), and the need for shielding.

The disadvantages of using older SNF for reprocessing are that the fuel value in a thermal spectrum is decreased and the associated americium-241 ingrowth increases; both of these are caused by the decay of plutonium-241 (14.2-year half-life). French regulators have taken a compromise approach to reprocess and recycle the plutonium into mixed-oxide fuel, with an optimized ageing timeframe of 5–7 years. NRC staff plans to develop regulations that are flexible in terms of the age of SNF to be reprocessed, because much of the SNF in the stockpile has been withdrawn from the reactor for more than 5 years. This plan is consistent with a performance-based regulatory framework. However, the age of the SNF likely will require license conditions and, potentially, technical specifications to address this linkage and ensure safety. Also, the capture and storage of some volatile radionuclides introduce hazards that require additional controls for safe operations.

In particular, tritium has garnered much public attention in recent years because of the nonroutine releases that have occurred at several U.S. power plant sites. The NRC groundwater task force report (NRC, 2010a) highlighted the widespread public concern about tritium releases. In addition, the tritium releases from reprocessing facilities might exceed releases from nuclear power plants. For example, data in the OSPAR Commission report (2009) show that total tritium releases for 2007 were 2,936 terabecquerels (TBq) (79,351 curies) for nuclear power plants and 12,628 TBq (341,298 curies) for reprocessing plants. The NRC will consider these stakeholder concerns during rulemaking, in addition to the safety concerns associated with effluent control and treatment of gaseous radionuclides from reprocessing operations.

In its white paper (NRC, 2008), the NRC Advisory Committee on Nuclear Waste and Materials (ACNW&M) stated, “Establishing release limits for volatile radionuclides could be a particularly lengthy process because of the likely need to perform engineering design, cost, and risk studies as a basis for the limits.” The ACNW&M also recommended that NRC hold interagency discussions with EPA on whether (i) existing release limits for krypton-85 and iodine-129 need to be reexamined to reflect current technology and (ii) release limits need to be established for tritium and carbon-14 (NRC, 2008). As part of the potential rulemaking for a new 10 CFR Part 7x, the staff is considering how best to respond to ACNW&M's recommendations on establishing release limits.

Regulations in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," include requirements to meet EPA's generally applicable environmental radiation standards in 40 CFR Part 190, which include both dose and normalized quantity units. Although the dose limits appear practicable for a reprocessing facility to meet and demonstrate compliance with 10 CFR Part 20 requirements, meeting the quantity limits appears to be more challenging. As discussed in McMahon, et al. (2010), the quantity limits are based on an idealized pathway and dose assessment analysis that involves very small doses to large populations, with a collective dose calculation. Portions of the calculations in the regulatory basis for 40 CFR Part 190 are not consistent with current radiological assessment approaches [i.e., McMahon, et al. (2010), and some of the assumptions in the calculation appear to be conservative [e.g., all crops grown (1 mi) from the facility]. In March 2011, the EPA's Radiation Protection Division published a notice on the Federal Business Opportunities Web site to solicit support for developing a document that would provide technical recommendations to EPA on targeted technical issues to be addressed in a potential revision of 40 CFR 190.10(b).

3.6.3 Stakeholder Views

NEI (Kellar, 2008) addresses the issue of effluent monitoring and control in its proposed basic design criteria. These criteria are, for the most part, derived from the equivalent requirements in 10 CFR 50.34a, GDC 60 and GDC 63, "Monitoring Fuel and Waste Storage," and 10 CFR Part 50, Appendix A, GDC 64. NEI (Kellar, 2008) provided additional requirements in its proposed framework to address the specific waste issues that would be involved in reprocessing nuclear fuel; namely, the possible need for holdup capabilities to control the release of gaseous and liquid effluents.

NEI (Kellar, 2008) did not adapt any of the requirements in 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," it derived its proposed reporting requirements regarding effluent releases from 10 CFR Part 70.

The NRC held two public workshops in the autumn of 2010 at Rockville, Maryland, and Albuquerque, New Mexico, to obtain stakeholder input on the development of the regulatory basis for reprocessing. Discussions on effluent control focused on the quantitative limits in 40 CFR Part 190 and the potential challenges licensees faced in meeting these limits. Industry representatives believed that the requirements in this EPA regulation should be revised in line with an updated technical basis. They also highlighted concerns with the existing requirements in 40 CFR Part 190; namely, difficulties in converting the quantitative limits into a meaningful indicator of the actual limits allowed. As currently written, the 40 CFR Part 190 limits are calculated "per gigawatt-year of electrical energy produced by the fuel cycle."

Other stakeholders felt it was reasonable to impose release limits on certain radionuclides, even if doses are low, because of concerns regarding collective impacts. Some stakeholders stated that special attention should be paid to radioactive atmospheric releases and to the siting of the facility.

3.6.4 Guidance Documents

EPA establishes radiation standards for the protection of the general environment from radioactive materials; NRC implements and enforces these standards. As stated previously, a reprocessing facility will have to meet the effluent limits EPA established in 40 CFR Part 190. These limits apply to "uranium fuel cycle" operations, which include the "reprocessing of spent uranium fuel" (40 CFR 190.02). The requirements consist of two parts: a dose limit and a release limit. The NRC implements these standards in 10 CFR Part 20. The NRC, in collaboration with EPA, may need to consider developing a regulatory guide (RG) that would explain to licensees how to interpret the quantitative limits in 40 CFR Part 190 for reprocessing facilities.

To determine the applicability of RGs relating to effluents for fuel cycle facilities and nuclear power plants to fuel reprocessing plants, the staff compared RG 4.16, "Monitoring and Reporting Radioactive Materials In Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities" (NRC, 2010b), with RG 1.21, "Monitoring and Reporting Radioactive Materials In Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities" (NRC, 2009). Both RGs address dose assessments to members of the public in accordance with 10 CFR 20.1301. Licensees are permitted to use conservative bounding dose assessments to determine the maximum dose to individual members of the public. The significant difference between the two RGs is that RG 1.21 addresses effluent dispersion, including unplanned releases and discharges, whereas RG 4.16 does not address dispersion. An RG for monitoring effluents from reprocessing likely would need to address the potential for leaks and the effects on the surrounding environment, including groundwater, as well as subsequent licensee actions.

The NRC staff determined it would develop a separate RG for effluent monitoring from reprocessing that incorporates the more risk informed, less descriptive aspects of RG 4.16 and the effluent dispersion recommendations from RG 1.21. The guidance should also consider addressing the specific radionuclides of concern in gaseous effluents; namely, iodine-129 and krypton-85. The NRC staff has not identified a need to develop separate guidance for aqueous and electrochemical separations for this particular area.

3.6.5 References

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NRC (2010b). U.S. Nuclear Regulatory Commission, "Monitoring and Reporting Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities," Regulatory Guide 4.16, Revision 2, December 2010. (ADAMS Accession No. ML101720291)

NRC (2009). U.S. Nuclear Regulatory Commission, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," Regulatory Guide 1.21, Revision 2, June 2009. (ADAMS Accession No. ML091170109)

NRC (2008). U.S. Nuclear Regulatory Commission, "Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Recycle Facilities—ACNW&M White Paper," NUREG–1909, June 2008. (ADAMS Accession No. ML082100043)

NRC (1975). U.S. Nuclear Regulatory Commission, "Part 50—Licensing of Production and Utilization Facilities—Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," *Federal Register*, Vol. 40, No. 87, May 5, 1975, pp. 19439–19443.

OSPAR Commission (2009). OSPAR Commission, "Liquid Discharges from Nuclear Installations in 2007," ISBN 978-1-906840-96-9, London, United Kingdom, 2009.

U.S. Code of Federal Regulations, "Standards for Protection Against Radiation," Part 20, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "Domestic Licensing of Special Nuclear Material," Part 70, Chapter I, Title 10, "Energy."

U.S. Code of Federal Regulations, "National Emission Standards for Hazardous Air Pollutants," Part 61, Chapter I, Title 40, "Protection of Environment"

U.S. Code of Federal Regulations, "Environmental Radiation Protection Standards for Nuclear Power Operations," Part 190, Chapter I, Title 40, "Protection of Environment.

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4 OPERATIONAL CONSIDERATIONS (GAPS 4, 8, 12, 13, 14, 17, and 18)

4.1 Introduction

This chapter addresses operational considerations, including material control and accounting (MC&A), physical protection, fees, and financial protection. Section 4.2 discusses the exemption of irradiated fuel reprocessing facilities in 10 CFR 74.51, "Nuclear Material Control and Accounting for Strategic Special Nuclear Material" (Gap 4). Section 4.3 discusses risk informing 10 CFR Part 73, "Physical Protection of Plants and Materials," and 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material" (Gap 8). Section 4.4 discusses diversion path analysis requirements (Gap 17). Section 4.5 discusses approaches toward material accounting management (Gap 18). Section 4.6 discusses the regulations in 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" (Gap 12). Section 4.7 discusses the regulations in 10 CFR Part 171, "Annual Fees for Reactor Licenses and Fuel Cycle Licenses and Materials Licenses, Including Holders of Certificates of Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licensed by the NRC" (Gap 14). Section 4.8 discusses the schedule of fees in 10 CFR Part 170, "Fees for Facilities, Materials, Import and Export Licenses, and Other Regulatory Services under the Atomic Energy Act of 1954, as Amended" (Gap 13).

4.2 Exclusion of Irradiated Fuel Reprocessing Facilities in 10 CFR 74.51 (Gap 4)

This section addresses Gap 4. The U.S. Nuclear Regulatory Commission (NRC) requires licensees to provide assurance that special nuclear material (SNM) is protected in accordance with regulations for MC&A, as described in 10 CFR Part 74. Regulations for MC&A follow a tiered structure based on the type of material and quantity a licensee possesses. These levels correspond to the requirements for physical security described in 10 CFR Part 73. Licensees possessing SNM of low strategic significance, as defined in 10 CFR Part 74, are commonly referred to as Category III facilities. Licensees authorized to possess SNM of moderate strategic significance are referred to as Category II facilities, and those authorized to possess strategic special nuclear material (SSNM) are called Category I facilities. Category I facilities are subject to the strictest MC&A requirements.

10 CFR 74.51 states that each licensee authorized to possess SNM—other than a nuclear reactor licensed pursuant to 10 CFR Part 50, an irradiated fuel reprocessing plant, an operation involved with waste disposal, or an independent spent fuel storage facility licensed pursuant to 10 CFR Part 52—shall establish, implement, and maintain a Commission-approved MC&A system that achieves specified objectives. A reprocessing facility would possess Category I quantities of SNM. Consequently, based on 10 CFR 74.51, a Category I reprocessing facility may not have the same MC&A requirements as other Category I facilities. MC&A requirements comparable to other Category I facilities may, however, be needed to protect against loss, theft, or diversion of separated SNM and other materials at reprocessing facilities.

In SECY–08–0059, "Rulemaking Plan: 10 CFR Part 74—Material Control and Accounting of Special Nuclear Material" (NRC, 2008) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080580273), the staff, in part, addressed the deletion of certain exemptions in the current regulations. The Commission directed the NRC staff [in Staff Requirements Memorandum (SRM)–SECY–08–0059 (ML090360473) (NRC, 2009)] to proceed with the rulemaking related to MC&A requirements in 10 CFR Part 74. The staff plans to remove the exemption of irradiated fuel reprocessing facilities in 10 CFR 74.51 as part of the 10 CFR Part 74 rulemaking. The removal of this exemption will help ensure the security of materials in any proposed Category I reprocessing facility.

The NRC plans to publish the draft 10 CFR Part 74 rule at the end of 2011 and has scheduled the final rule for completion in 2012. Accordingly, the staff will address this topic in any proposed rule for a reprocessing regulation by referring to, and incorporating, the requirements in the MC&A rulemaking directed by SRM–SECY–08–0059. Thus, the staff's resolution path for this gap is to remove the current exemption for irradiated fuel reprocessing facilities in the ongoing 10 CFR Part 74 rulemaking.

4.3 Risk Informing 10 CFR Part 73 and 10 CFR Part 74 (Gap 8)

This section addresses Gap 8. As discussed in Section 4.2, the NRC regulations for physical protection and MC&A consider the type and the quantity of SNM. The current quantity-based categorization scheme in the existing regulations may pose an undue regulatory burden in operating a reprocessing facility. Risk informing 10 CFR Parts 73 and 74 would prevent unintended consequences associated with a quantity-based material categorization scheme for potential materials resulting from a reprocessing operation.

The timeline for completing the regulatory basis for this gap is different from the rest of the gaps associated with the regulatory basis for licensing spent nuclear fuel reprocessing facilities because this gap is being addressed by a separate rulemaking effort. The NRC staff received direction from the Commission on two proposed rulemakings: 10 CFR Parts 73 and 74. In 2010, the Commission directed the staff to consider revising SNM categorization in a 10 CFR Part 73 rulemaking (SRM–SECY–09–0123 (ADAMS Accession No. ML101890711) (NRC, 2010). Conforming revisions to 10 CFR Part 74 could follow the changes to 10 CFR Part 73. The staff's current view is that a proposed regulation for a reprocessing facility (e.g., 10 CFR Part 7x) should refer to requirements in 10 CFR Parts 73 and 74 for (i) physical protection and (ii) material control and accounting, respectively, rather than detailing such requirements in the reprocessing regulation itself. Rulemaking on the security of SNM, which will include revision of the material categorization scheme, is proceeding.

The basis for existing security-related regulations for fuel cycle facilities in 10 CFR Part 73 is the NRC's categorization scheme for SNM, found in the table in 10 CFR Part 110, "Export and Import of Nuclear Equipment and Material," Appendix M, "Categorization of Nuclear Material." This material categorization scheme, which is nearly 30 years old, places uranium and plutonium into one of three risk categories (Category I, II, or III), depending on quantities and enrichment. The NRC's regulations in 10 CFR Part 73 identify requirements for physical protection of that SNM depending on the category. The risk-informed categories are based on the primary concern with SNM—the ability of an adversary to create an improvised nuclear device (IND).

While the staff believes that the current security levels are sufficiently conservative for all forms of SNM, the two factors currently used to determine the importance of SNM for making an IND—the type and quantity of material—may not provide for appropriately graded protection strategies. Consideration of the form of the SNM (e.g., whether it exists as a pure oxide, is the primary material in a solution, or is one of several materials in a solid form), as well as the weight percent of the SNM in the compound being considered, is critical to determining the value, or attractiveness, that such material holds for an adversary seeking to acquire and use the material in an IND. This concept of considering the form and weight percent of the material in the overall categorization scheme has been studied and successfully implemented for more than 20 years by the Nation's nuclear weapons complex and laboratories run by the U.S. Department of Energy (DOE).

SRM–SECY–08–0059 (NRC, 2009) stated, in part, that the staff should provide a Commission paper on material categorization, "[a]fter DOE has gained some operational experience [with its revised categorization table]." However, rather than adopt the DOE approach, the staff proposed a categorization framework that specifically addresses the types of nuclear material possessed by NRC licensees, both current and future.

Because the current NRC categorization scheme lacks specificity with regard to certain forms of SNM, the staff has not been able to use the categorization scheme alone to make risk-informed decisions with regard to security requirements for some SNM. As an example, more than 5 kilograms of highly enriched uranium (HEU) in metal form presents a greater risk than more than 5 kilograms of HEU dispersed in a gondola car filled with SNM-contaminated soil. Because of this lack of specificity, licensees have requested exemptions to allow for appropriately graded protection strategies. A more risk-informed, graded-categorization scheme that includes material attractiveness would allow the staff to identify the relative risks associated with each form of SNM more specifically, thereby reducing related exemption requests and the associated burdens of processing such requests.

10 CFR Part 74

The NRC's MC&A regulations that presently grade SNM are similar to the nuclear material categorization scheme for physical protection regulations. As defined in 10 CFR Part 74, the existing scope for MC&A is almost identical to the categorization of SNM for physical protection purposes in 10 CFR Part 73. As described previously, this scheme categorizes material by type, quantity, and enrichment, where decreasing levels of protection are required for material of high, moderate, and low strategic significance (Categories I, II, and III material quantities, respectively). The current categorization does not consider the form or attractiveness for potential material theft or diversion. The MC&A regulations in 10 CFR Part 74 would use the same material categorization scheme as 10 CFR Part 73. MC&A regulations in 10 CFR Part 74 would be revised to provide a risk-informed regulatory approach to each category of SNM based upon the material's attractiveness.

4.4 Diversion Path Analysis Requirements (Gap 17)

This section addresses Gap 17. One method of providing a more risk-informed MC&A at a licensee site is to conduct a diversion path analysis (DPA). A DPA is an analytical tool for evaluating system weaknesses against reasonable and plausible adversary scenarios involving covert internal threats. A diversion path describes the malevolent activities that might be performed by overt or covert adversaries, such as providing false information or substituting SNM with a different material. Therefore, a DPA requires a study of the facility's specific processes, knowledge of the functions and activities of facility personnel having access to nuclear material, and an understanding of the facility's safeguards and security practices. Licensees should take into account all locations with nuclear material in developing feasible adversary scenarios, both simple and sophisticated (e.g., routine operations, access control, accountability measurements, adverse activities affecting performance of material control, infrequent operations, and abnormal events). A DPA describes the MC&A and physical security measures that protect against the malevolent activities it describes. The NRC staff is proposing that the regulations in 10 CFR Part 74 be amended to require a DPA for reprocessing facilities. Establishing DPA requirements for a reprocessing facility would make MC&A requirements more risk informed and would provide an effective detection and response program to mitigate potential safeguards vulnerabilities and system weaknesses.

SRM–SECY–08–0059 (NRC, 2009a) directed the staff to consider incorporating the MC&A proposals presented in that SECY into the effort to develop the regulatory framework for reprocessing facilities. Under Option 3 of the SECY, the staff would conduct a rulemaking to add a DPA requirement to the 10 CFR Part 74 regulations that would apply to nuclear facilities possessing a Category I quantity of SNM. The DPA would be part of a detection and response program to mitigate potential safeguards vulnerabilities. Because of the nature and attractiveness of nuclear materials typically processed at reprocessing facilities, including the DPA requirement in the regulations would require affected reprocessing facilities to develop a more risk-informed, performance-based MC&A program that considers a wider range of malevolent activities and that might involve facility insiders. In addition, the Commission endorsed the provision for a DPA within a facility's MC&A program as part of the requirements in the draft rule, SECY–07–0126, "Proposed Rule: Geologic Repository Operations Area Security and Material Control and Accounting Requirements" (NRC, 2007).

10 CFR Part 74 requirements for the MC&A program at a reprocessing facility should be structured to contribute in-depth protection by mitigating the risk of loss, theft, or diversion of nuclear materials. This system capability can be achieved by an assessment of MC&A program vulnerabilities that identify and analyze diversion scenarios. The licensee should evaluate existing MC&A and other mitigating measures that could interfere with adversary actions to determine whether modifications to existing counter diversion measures are necessary to effectively protect nuclear materials within the facility.

4.5 Approaches to Material Accounting Management (Gap 18)

This section addresses Gap 18. The MC&A regulations for Category I facilities described in 10 CFR Part 74 currently list requirements for material accounting using predefined limits and timeliness factors. For example, existing predefined limits on inventory difference determinations and restriction on inventory periods could be a challenge for reprocessing facilities because of the nature of large operations and large material throughputs, various measurement methods, and associated measurement uncertainties.

The NRC staff needs to conduct further analyses to determine whether the existing 10 CFR Part 74 predefined limits and guidance documents should be revised for reprocessing facilities. In particular, the staff should evaluate approaches to meet the timeliness and quantity requirements for material inventory accounting for changes or improvements. The staff should consider the currently defined limits on inventory difference evaluations in light of limits and practices used at operating reprocessing facilities in other countries, as well as the International Atomic Energy Agency's international target values. For example, foreign reprocessing plants have improved technology, such as near-real-time accounting. Such technological improvements can facilitate meeting the current timeliness and quantity requirements in 10 CFR Part 74.

Additionally, for most process operations at a reprocessing facility, the quantities of nuclear material held up in process equipment during routine processing may significantly exceed the defined limit of the active inventory. In practice, minimizing the quantity of residual holdup that is not amenable to measurements improves the quality of a physical inventory. Consequently, to minimize both the magnitude of observed inventory differences and the combined material balance uncertainty, the facility is likely to find that incorporating a material holdup management program to minimize the impact of material holdup could facilitate more accurate inventory accounting. A future U.S. facility that processes significant quantities of nuclear materials may find that either *in-situ* measurements or more thorough process cleanout operations are necessary to satisfy current regulatory requirements.

Finally, the staff should also consider potential facility vulnerabilities as part of the design, establishment, and maintenance of performance-based MC&A systems. The NRC should confirm that requirements for facility system capabilities are sufficiently risk oriented, especially in controlling and verifying the current amount, location, and status of all SNM possessed, used, or stored at fixed sites. MC&A systems should also be able to ensure that any actual loss or attempt to divert would be detected and responded to in a timely fashion. The absence of anomalies and other indicators of loss or misuse should provide assurance of the continued secured presence of SNM under NRC regulatory requirements.

If the staff determines that, based on the evaluations described previously, the material accounting requirements in 10 CFR Part 74 should be amended for reprocessing facilities, the staff will consider appropriate changes to 10 CFR Part 74.

4.6 Financial Protection Requirements and Indemnity Agreements (Gap 12)

This section addresses Gap 12. The Price-Anderson Act (the Act) was enacted into law on September 2, 1957, as Section 170 of the Atomic Energy Act of 1954, as amended (AEA), to meet two basic objectives:

(1) Remove the deterrent to private sector participation in atomic energy presented by the threat of potentially enormous liability claims in the event of a catastrophic nuclear accident.

(2) Ensure that adequate funds were available to the public to satisfy liability claims if such an accident were to occur.

The NRC codified the provisions of the Act in 10 CFR Part 140 and established the requirements and procedures for implementing its financial protection provisions. Although the regulations in 10 CFR Part 140 specify financial protection requirements for larger commercial nuclear reactors (10 CFR 140.11, “Amounts of Financial Protection for Certain Reactors”); smaller reactors (e.g., research and test reactors [10 CFR 140.12, “Amount of Financial Protection Required for Other Reactors”)]; holders of construction permits and combined licenses under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants (10 CFR 140.13, “Amount of Financial Protection Required of Certain Holders of Construction Permits and Combined Licenses under 10 CFR Part 52); plutonium processing and fuel fabrication plants (10 CFR 140.13a); and uranium enrichment facilities (10 CFR 140.13b), the regulations do not contain specific provisions for reprocessing facilities.

In addition to financial protection requirements, 10 CFR Part 140 requires the NRC to collect a fee for executing any indemnity agreement with a licensee. These fees are established in 10 CFR 140.7, “Fees.” However, 10 CFR Part 140 currently does not specify the fees applicable to reprocessing facilities. The appendices to 10 CFR Part 140 also contain several

standard forms for nuclear liability policies and for indemnity agreements with the NRC. The appendices currently list standard forms for reactor facilities and plutonium processing plants but not for reprocessing facilities.
Therefore, to address these regulatory gaps in 10 CFR Part 140, the NRC staff recommends that the agency revise the regulations as follows:

- Extend the applicability of 10 CFR Part 140 to reprocessing facilities. If a new regulation for the licensing of production or reprocessing facilities is promulgated (i.e., 10 CFR Part 7x) and adopted, the NRC must extend the applicability of 10 CFR Part 140 to this regulation.

- Establish the specific amount of financial protection required for a production facility. Establish the appropriate fee for executing and issuing indemnity agreements for production facilities. Amend the current appendices in 10 CFR Part 140, or include a new appendix, to include a standard form for indemnity agreements for production facilities.

The staff did not consider any alternatives to rulemaking as proposed solutions to address this gap in the regulations. AEA Section 103(a) requires the NRC to issue licenses for production or utilization facilities, "[…] subject to such conditions as the Commission may by rule or regulation establish to effectuate the purposes and provisions of this Act." As AEA Section 170 requires licensees of production facilities to have specific amounts of financial protection as a condition of the license, the NRC must establish, by rule, the requirements and amounts for financial protection and indemnity agreements. Accordingly, rulemaking is the only alternative available for addressing these regulatory gaps in 10 CFR Part 140.

4.7 Schedule of Fees (Gap 13)

This section addresses Gap 13. The provisions of 10 CFR 170.2, "Scope," state that the fees specified in this section apply to "an applicant for or holder of a production or utilization facility construction permit or operating license issued under 10 CFR Part 50…." The NRC is considering establishing a new regulation for licensing a reprocessing facility (i.e., 10 CFR Part 7X).

The staff recommends that, if a new or revised chapter for licensing reprocessing facilities (i.e., 10 CFR Part 7X) is promulgated, 10 CFR Part 170 be revised to include the applicability of this section to reprocessing facilities. Additionally, minor revisions would be needed to the fee schedules to reflect this change.

The staff did not consider any alternatives to rulemaking as proposed solutions to address this gap in 10 CFR Part 170 regulations, because the Omnibus Budget Reconciliation Act of 1990 (OBRA–90), Section 6101(c)(3) requires the NRC to "[…] establish, by rule, a schedule of charges fairly and equitably allocating the aggregate amount of charges described in paragraph (2) among licensees." Stakeholders had no comments on this topic.

On the basis of the information provided in Section 4.7, the staff has identified a path to resolve Gap13.

4.8 Annual Fees (Gap 14)

This section addresses Gap 14. The regulations in 10 CFR Part 171 do not currently specify annual fees for production or reprocessing facilities. In addition, 10 CFR 171.3, “Scope,” does not specifically list reprocessing or production facilities as subject to the provisions of this part. The NRC is considering establishing a new regulation for licensing a reprocessing facility (i.e., 10 CFR Part 7X). If it decides to do so, 10 CFR 171.3 would need revision to include a reference to this new regulation.

The staff recommends the following: (1) revise 10 CFR Part 171 to extend the applicability of 10 CFR Part 171 to production or reprocessing facilities; (2) if a new regulation applicable to reprocessing facilities (i.e., 10 CFR Part 7X) is promulgated, expand 10 CFR Part 171 to apply this new or revised chapter to reprocessing facilities; and (3) establish the annual fee for reprocessing or production facilities.

The staff did not consider any alternatives to rulemaking as proposed solutions to address this gap in 10 CFR Part 171 regulations, because OBRA–90, Section 6101(c)(3) requires the NRC to “…establish, by rule, a schedule of charges fairly and equitably allocating the aggregate amount of charges described in paragraph (2) among licensees.” Stakeholders had no comments on this topic.

On the basis of the information provided in Section 4.8, the staff has identified a path to resolve Gap 14.

4.9 References

NRC (2007). U.S. Nuclear Regulatory Commission, “Proposed Rule: Geologic Repository Operations Area Security and Material Control and Accounting Requirements,” SECY–07–0126, July 31, 2007.

NRC (2008). U.S. Nuclear Regulatory Commission, “Rulemaking Plan: Part 74—Material Control and Accounting of Special Nuclear Material,” SECY–08–0059, April 25, 2008.

NRC (2009). U.S. Nuclear Regulatory Commission, “Staff Requirements—SECY–08–0059—Rulemaking Plan: Part 74—Material Control and Accounting of Special Nuclear Material,” SRM–SECY–08–0059, February 5, 2009.

NRC (2010). U.S. Nuclear Regulatory Commission, “Staff Requirements—SECY–09–0123—Material Categorization and Future Fuel Cycle Facility Security-Related Rulemaking,” SRM–SECY–09–0123, July 8, 2010.

OBRA, Omnibus Budget Reconciliation Act of 1990, P.L. 101-508, 104 Stat. 1388 (1990).

U.S. Code of Federal Regulations, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” Part 52, Chapter I, Title 10, “Energy.”

*U.S. Code of Federal Regulations, “*Physical Protection of Plants and Materials,” Part 73, Chapter I, Title 10, “Energy.”

*U.S. Code of Federal Regulations, “*Material Control and Accounting of Special Nuclear Material,” Part 74, Chapter I, Title 10, “Energy.”

*U.S. Code of Federal Regulations, "*Financial Protection Requirements and Indemnity Agreements," Part 140, Chapter I, Title 10, "Energy."

*U.S. Code of Federal Regulations, "*Fees for Facilities, Materials, Import and Export Licenses, And Other Regulatory Services Under the Atomic Energy Act of 1954, as Amended," Part 170, Chapter I, Title 10, "Energy."

*U.S. Code of Federal Regulations, "*Annual Fees for Reactor Licenses and Fuel Cycle Licenses and Materials Licenses, Including Holders of Certificates of Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licenses by the NRC," Part 171, Chapter I, Title 10, "Energy."

APPENDIX A

List of Paragraphs in 10 CFR 50.54 and Their Relevancy to Reprocessing Facilities

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List of Paragraphs in 10 CFR 50.54 and Their Relevancy to Reprocessing Facilities		
Paragraph	**Topic**	**Relevant?**
(a)(1), (2), (3)	QA criteria, SAR	y
(b)	Special nuclear material	y
(c)	License transfer	y
(d)	AEA Section 108	y
(e)	Revoke license	y
(f)	Commission request for licensing docs	y
(g)	Antitrust provisions, Sec. 105a AEA	y
(h)	License subject to Act, regs	y
(i)	Requirement for licensed operators (exemptions 55.13(a)(1)); operator requalification program	y
(j)	Operating apparatus and mechanisms	y
(k)	Operator pursuant to Part 55 at controls	y
(l)	Senior operators	y
(m)(1)	Presence of senior operator	Partly—specific to reactors but an equivalent rule would be needed for FRP.
(m)(2)	Number of operators required	Partly—specific to reactors but an equivalent rule would be needed for FRP.
(n)	Tech spec modifications	y
(o)	Primary reactor containment	n
(p)	Safeguards contingency plans	y
(q)	Emergency plans; App. E	y
(r)	Power level of reactor	n
(s)	Radiological emergency response plans; Emergency Planning Zone (EPZ)	y
(t)	Emergency preparedness program	y
(u)	Submission of emergency preparedness plans to NRC	y
(v)	Protection of safeguards info	y
(w)	Insurance/financial protection	y
(x)	Deviation from licensed procedure during emergency	y
(y)	Approval for (x)	y
(z)	Notification of 10 CFR 50.72 event	n
(aa)	Federal water pollution control act	y
(bb)	Plans for management of irradiated fuel at a reactor	n
(cc)	Bankruptcy	y
(dd)	Deviation from licensed procedure during national security emergency	y
(ee)	Possession of SNM, irradiated fuel Receipt of materials	n—paragraph explicitly "…does not authorize the receipt of any material recovered from the reprocessing of irradiated fuel."
(ff)	Earthquake engineering criteria; App. S	Partly—Appendix S specific to reactors. Equivalent rule needed for FRP.
(gg)	Power levels in an emergency	n
(hh)	Plans to address aircraft threat	y

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APPENDIX B

Fire Protection Standards: National Fire Protection Association

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Fire Protection Standards: National Fire Protection Association

Publication date of current edition is shown in parentheses.

NFPA 10, “Portable Fire Extinguishers” (2010)

NFPA 11, “Low Expansion Foam and Combined Agent Systems” (2010)

NFPA 11A, “Medium and High Expansion Foam Systems” (1999)

NFPA 12, “Carbon Dioxide Extinguishing Systems” (2011)

NFPA 12A, “Halon 1301 Fire Extinguishing Systems” (2009)

NFPA 13, “Sprinkler Systems” (2010)

NFPA 14, “Standpipe and Hose Systems” (2010)

NFPA 15, “Water Spray Fixed Systems for Fire Protection” (2007)

NFPA 16, “Deluge Foam Water Sprinkler and Foam Water Spray Systems” (2011)

NFPA 17, “Standard for Dry Chemical Extinguishing Systems” (2009)

NFPA 17A, “Standard for Wet Chemical Extinguishing Systems” (2009)

NFPA 20, “Centrifugal Fire Pumps” (2010)

NFPA 22, “Standard for Water Tanks for Private Fire Protection” (2008)

NFPA 24, “Private Fire Service Mains and Their Appurtenances” (2010)

NFPA 25, “Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems” (2011)

NFPA 30, “Flammable and Combustible Liquids Code” (2008)

NFPA 45, “Laboratories Using Chemicals” (2011)

NFPA 54, “National Fuel Gas Code” (2009)

NFPA 55, “Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks” (2010)

NFPA 70, “National Electrical Code” (2011)

NFPA 72, “National Fire Alarm Code” (2010)

NFPA 75, “Electronic Computer/Data Processing Equipment” (2009)

NFPA 80, “Fire Doors and Windows” (2010)

NFPA 82, “Standard on Incinerators and Waste and Linen Handling Systems and Equipment” (2009)

NFPA 86, “Standard for Ovens and Furnaces” (2011)

NFPA 90A, “Air Conditioning and Ventilating Systems” (2009)

NFPA 90B, “Warm Air Heating and Air Conditioning Systems” (2009)

NFPA 91, “Standard for Exhaust Systems for Air Conveying of Vapors, Gases, Mists, and Noncombustible Particulate Solids” (2010)

NFPA 101, “Life Safety Code” (2009)

NFPA 241, “Standard for Safeguarding Construction, Alteration, and Demolition Operations” (2009)

NFPA 253, “Standard Method of Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source” (2011)

NFPA 255, “Standard Method of Test of Surface Burning Characteristics of Building Materials” (2006)

NFPA 484, “Standard for Combustible Metals” (2009)

NFPA 600, “Standard on Industrial Fire Brigades” (2010)

NFPA 701, “Standard Methods of Fire Tests for Flame Propagation of Textiles and Films” (2010)

NFPA 750, “Standard on Water Mist Fire Protection Systems” (2010)

NFPA 780, “Standard for the Installation of Lightning Protection Systems” (2011)

NFPA 801, “Facilities Handling Radioactive Materials” (2008)

NFPA 804, “Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants” (2010)

NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants” (2010)

NFPA 1500, “Standard on Fire Department Occupational Safety and Health Program” (2007)

NFPA 2001, “Standard on Clean Agent Fire Extinguishing Systems” (2008)

APPENDIX C

List of Regulatory Guides Related to Reprocessing

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List of Regulatory Guides Related to Reprocessing		
Guide Number	**Title**	**Publish Date**
3.3	Quality Assurance Program Requirements for Fuel Reprocessing Plants and for Plutonium Processing and Fuel Fabrication Plants	03/1974 (Rev 1)
3.6	Content of Technical Specifications for Fuel Reprocessing Plants	04/1973
3.17	Earthquake Instrumentation for Fuel Reprocessing Plants	02/1974
3.18	Confinement Barriers and Systems for Fuel Reprocessing Plants	02/1974
3.19	Reporting of Operating Information for Fuel Reprocessing Plants	02/1974
3.20	Process Offgas Systems for Fuel Reprocessing Plants	02/1974
3.21	Quality Assurance Requirements for Protective Coatings Applied to Fuel Reprocessing and to Plutonium Processing and Fuel Fabrication Plants	03/1974
3.22	Periodic Testing of Fuel Reprocessing Plant Protection System Actuation Functions	06/1974
3.26	Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants	02/1975
3.27	Nondestructive Examination of Welds in the Liners of Concrete Barriers in Fuel Reprocessing Plants	05/1977 (Rev 1)
3.28	Welder Qualification for Welding in Areas of Limited Accessibility in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants	05/1975
3.29	Preheat and Interpass Temperature Control for the Welding of Low-Alloy Steel for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants	05/1975
3.30	Selection, Application, and Inspection of Protective Coatings (Paints) for Fuel Reprocessing Plants	05/1977
3.31	Emergency Water Supply Systems for Fuel Reprocessing Plants	05/1977
3.32	General Design Guide for Ventilation Systems for Fuel Reprocessing Plants (for Comment)	09/1975
3.37	Guidance for Avoiding Intergranular Corrosion and Stress Corrosion in Austenitic Stainless Steel Components of Fuel Reprocessing Plants (for Comment)	09/1975
3.39	Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants	01/1976
3.40	Design Basis Floods for Fuel Reprocessing Plants and for Plutonium Processing and Fuel Fabrication Plants	12/1977

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APPENDIX D

Review of Applicability of Current Regulatory Guides for Reprocessing Facilities

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Review of Applicability of Current Regulatory Guides for Reprocessing Facilities

The following table contains the list of regulatory guides (RGs) that the staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed during the development of the regulatory basis document for the reprocessing rulemaking. The table contains three columns that identify the RGs, the gap they are associated with, and the decision the NRC staff made about the applicability of the RG and recommended future action.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
3.14	Seismic Design Classification for Plutonium Processing and Fuel Fabrication Plants	1	**Do not use**. RG 1.29 might also be useful, but needs clarification in the SRP.
3.17	Earthquake Instrumentation for Fuel Reprocessing Plants	1	**Withdraw**. RG 1.12 may be used instead, with clarification in the SRP.
3.73	Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations	1	**Do not use**. RG 3.73 discusses how to satisfy the seismic-related requirements (e.g., site characteristics, determination of design earthquake ground motion (DE), uncertainties using PSHA method). In determining the DE, RG 3.73 still uses "Reference Probability," which was described in RG 1.165. The new approach for DE is to use the performance-based approach in RG 1.208.
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	1, 5, 9	**Use**, but provide guidance in the SRP on modification at high frequency for Central and Eastern United States (CEUS).
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	1, 5, 9	**Use** this RG to satisfy 10 CFR Part 50, Appendix A, GDC 2.
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1, 5	**Use** to meet the seismic design requirements of GDC 2.
1.165	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion	1, 5	**Do not use**. Replaced with RG 1.208 and withdrawn.
1.12	Nuclear Power Plant Instrumentation for Earthquakes	1, 9	**Use** for guidance on modern instrumentation. May need to combine with guidance in SRP on instrumentation placement.
1.29	Seismic Design Classification	1, 5, 10, 9	**Use** to inform the update to 3.14 or with clarification in the SRP.
1.61	Damping Values for Seismic Design of Nuclear Power Plants	1, 9	**Use** this RG to satisfy 10 CFR Part 50, Appendix A, GDC 2.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants	1, 9	**Use** if reprocessing facilities have to be designed with high seismic demand like power plants.
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions	1	**Use** with RG 1.12.
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event	1	**Do not use;** not directly applicable.
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	1, 9	**Use** for facility siting.
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	1, 5	**Use**. RG 1.208 uses the performance-based approach of ASCE/SEI Standard 43-05, which applies to all seismic design categories, including SDC5 (which includes reactors and reprocessing facilities).
3.40	Design Basis Floods for Fuel Reprocessing Plants and for Plutonium Processing and Fuel Fabrication Plants	9	**Withdraw**. RG 1.59 would be better. The technical justification for withdrawal will be submitted to the ACRS at the appropriate time as per the normal RG withdraw process.
1.59	Design Basis Floods for Nuclear Power Plants	9	**Use** this RG to satisfy 10 CFR Part 50, Appendix A, GDC 2.
1.102	Flood Protection for Nuclear Power Plants	9	**Use** this RG to satisfy 10 CFR Part 50, Appendix A, GDC 2.
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	9, 11	**Use**, if needed.
3.21	Quality Assurance Requirements for Protective Coatings Applied to Fuel Reprocessing and to Plutonium Processing and Fuel Fabrication Plants	1	**Withdraw**. Use RG 1.54 "to the extent applicable." Also use 10 CFR Part 50, Appendix B, and NUREG–0800, as appropriate. The technical justification for withdrawal will be submitted to the ACRS at the appropriate time as per the normal RG withdraw process.
3.27	Nondestructive Examination of Welds in the Liners of Concrete Barriers in Fuel Reprocessing Plants	NA	**Update,** based on information from NUREG–0800, NDE sections, and RG 1.65.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
3.29	Preheat and Interpass Temperature Control for the Welding of Low-Alloy Steel for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants	NA	**Withdraw**. Use RG 1.50. The technical justification for withdrawal will be submitted to the ACRS at the appropriate time as per the normal RG withdraw process.
3.30	Selection, Application, and Inspection of Protective Coatings (Paints) for Fuel Reprocessing Plants	NA	**Use** with RG 1.54, as applicable. The coating used in reprocessing should be more protective, because the solution chemistry is different from the reactor; pH values can be lower, but the radiation level can also be lower. Consider updating RG 3.30 to be technology specific.
3.37	Guidance for Avoiding Intergranular Corrosion and Stress Corrosion in Austenitic Stainless Steel Components of Fuel Reprocessing Plants	NA	**Use**, with the guidance in NUREG–0800. (May update later)
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb	NA	**Use** updated information in NUREG–0800, Section 5.3.1, as providing an acceptable methodology to evaluate vessel materials with low upper shelf energy (below 50 ft-lb).
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	NA	**Use**, with information in NUREG–0800, Sections 5.2.3 and 5.3.1, in connection with welding of austenitic stainless steel (specifically the welder filler metal).
1.34	Control of Electroslag Weld Properties	NA	**Use**.
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	NA	**Use**, as applicable, with the guidance from NUREG–0800.
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	NA	**Use**.
1.44	Control of the Processing and Use of Stainless Steel	NA	**Use**.
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	NA	**Use**.
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants	NA	**Use**. Need clarification of the service levels and DBA for reprocessing.
1.65	Materials and Inspections for Reactor Vessel Closure Studs	NA	**Use**, but for low-pressure operation, this is not as significant.
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	NA	**Do not use**. A detailed review of NUREG–0800 and the standards would be needed to determine which are applicable to reprocessing (technology specific).

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
3.18	Confinement Barriers and Systems for Fuel Reprocessing Plants	9, 19	**Update** or use other guidance.
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	9	**Use** this RG to satisfy 10 CFR Part 50, Appendix A, GDC 2.
1.91	Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants	9	**Use** this RG to satisfy 10 CFR Part 50, Appendix A, GDC 2.
1.117	Tornado Design Classification	9	**Do not use;** too reactor specific.
1.132	Site Investigations for Foundations of Nuclear Power Plants	NA	**Use** as general guidance for facility structural design.
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	NA	**Use** as general guidance.
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	5	**Do not use**. Concepts can be used in future guidance, as needed.
1.175	An Approach for Plant-Specific, Risk-Informed Decision-making: In-service Testing	5	**Do not use**. Concepts can be used in future guidance, as needed.
1.177	An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications	11, 5	**Do not use** directly. Consider content when updating RG 3.6.
1.217	Guidance for the Assessment of Beyond Design-Basis Aircraft Impacts	9	**Do not use**. Similar technology-specific guidance will be needed later.
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	5	**Do not use**. Concepts can be used in future guidance, as needed.
1.204	Guidelines for Lightning Protection of Nuclear Power Plants	9	**Use** this RG to satisfy 10 CFR Part 50, Appendix A, GDC 2.
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	5, 10	**Do not use**. Concepts can be used in future guidance, as needed.
3.3	Quality Assurance Program Requirements for Fuel Reprocessing Plants and for Plutonium Processing and Fuel Fabrication Plants	1	**Withdraw**. Use RGs 1.28 and 1.33 instead. The technical justification for withdrawal will be submitted to the ACRS at the appropriate time as per the normal RG withdraw process.
3.22	Periodic Testing of Fuel Reprocessing Plant Protection System Actuation Functions	1, 22	**Update**, if it is considered useful.
3.28	Welder Qualification for Welding in Areas of Limited Accessibility in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants	NA	**Do not use**. Content duplicates RG 1.71.
1.28	Quality Assurance Program Criteria (Design and Construction)	1	**Use** with NQA-1 to meet the Appendix B criteria.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
1.33	Quality Assurance Program Requirements (Operation)	1	**Use** with NQA-1 to meet the Appendix B criteria.
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	1, 9, 11	**Do not use**. 10 CFR 50.65 is not being incorporated into 10 CFR Part 7x.
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	9	**Do not use**. Concepts can be used in future technology-specific guidance, as needed.
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)	9	**Do not use**. Can refer to modern standards in the SRP.
1.71	Welder Qualification for Areas of Limited Accessibility	9	**Use;** can be used in the SRP.
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	9	**Do not use**. Concepts and content can be used in a reprocessing-specific RG.
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	9	**Do not use**. Concepts and content can be used in a reprocessing-specific RG.
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	9	**Do not use**. Concepts and content can be used in a reprocessing-specific RG.
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants	9	**Do not use**. Concepts and content can be used in a reprocessing-specific RG.
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	9	**Do not use**. Concepts and content can be used in a reprocessing-specific RG.
1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	9	**Do not use**. Concepts and content can be used in a reprocessing-specific RG.
1.211	Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants	9	**Do not use**. Concepts and content can be used in a reprocessing-specific RG.
1.213	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	9	**Do not use**. Concepts and content can be used in a reprocessing-specific RG.
3.20	Process Offgas Systems for Fuel Reprocessing Plants	9, 19	**Update**. Much of this RG still seems applicable. May want to consider management of other fission off gases (e.g., Kr), and technology-specific issues.
3.33	Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant	9	**Do not use**. This RG has been withdrawn. This would relate to source-term development.
3.71	Nuclear Criticality Safety Standards for Fuels and Material Facilities	1, 9	**Update, or qualified use** to remove the endorsement of ANS-8.10.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste	19	**Do not use;** issue reprocessing-specific RG on this topic.
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)	9	**Use** when handling burnt fuel assemblies or to provide input to a reprocessing-specific RG.
1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants	5, 9	**Use**.
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Demonstrating Compliance with 10 CFR Part 50, Appendix I	19	**Do not use**.
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	19	**Do not use**, except as guidance for general methods.
4.1	Radiological Environmental Monitoring for Nuclear Power Plants	19, 11	**Use**, “to the extent applicable.”
4.2	Preparation of Environmental Reports for Nuclear Power Stations	19	**Do not use**. Probably good as a basis if guidance on the preparation of environmental reports for reprocessing facilities is needed.
4.5	Measurements of Radionuclides in the Environment—Sampling and Analysis of Plutonium in Soil	19	**Do not use**. It has been withdrawn.
4.8	Environmental Technical Specifications for Nuclear Power Plants	11, 19	**Do not use**. It has been withdrawn and is not applicable.
4.15	Quality Assurance for Radiological Monitoring Programs (Inception Through Normal Operations to License Termination)—Effluent Streams and the Environment	19	**Do not use**. Provide input to reprocessing-specific RG.
4.16	Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants	19	**Do not use**. It is being revised.
4.16 (Rev 2)	Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants	19	**Do not use**. Issue reprocessing-specific RG.
4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors	19	**Do not use**. It is being revised.
4.20 (Rev 1)	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors	19	**Do not use** if RG 1.21 or NMSS reprocessing guidance is referenced.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	19, 9	**Use**.
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters	NA	**Do not use**. It is being revised.
8.4 (Rev 1)	Personnel Monitoring Device—Direct-Reading Pocket Dosimeters	NA	**Use** for general guidance.
8.7	Instructions for Recording and Reporting Occupational Radiation Exposure Data	NA	**Use.** 10 CFR Part 20 reporting requirements apply to reprocessing.
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable	NA	**Use** if an ALARA RG for reprocessing is not developed.
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	NA	**Use** for general guidance.
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	NA	**Use** for general guidance.
8.13	Instruction Concerning Prenatal Radiation Exposure	NA	**Use** for general guidance.
8.15	Acceptable Programs for Respiratory Protection	NA	**Use** for general guidance.
8.19	Occupational Radiation Dose Assessment in Light-Water Reactor Power Plant—Design Stage Man-Rem Estimates	5	**Do not use**. Is reprocessing-specific guidance needed?
8.20	Applications of Bioassay for I-125 and I-131	19	**Do not use**. See also RG 8.26.
8.21	Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants	19	**Use** if not addressed by reprocessing-specific guidance.
8.24	Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication	19	**Do not use;** not applicable to fission products.
8.25	Air Sampling in the Workplace	NA	**Use** for general guidance.
8.26	Applications of Bioassay for Fission and Activation Products	19	**Use** if no bioassay RG is developed for reprocessing. See also RG 8.20.
8.28	Audible-Alarm Dosimeters	NA	**Use** for general guidance.
8.29	Instruction Concerning Risks from Occupational Radiation Exposure	NA	**Use** for general guidance.
8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses	NA	**Use** for general guidance.
8.35	Planned Special Exposures	NA	**Use** for general guidance.
8.36	Radiation Dose to the Embryo/Fetus	NA	**Use** for general guidance.
8.37	ALARA Levels for Effluents from Materials Facilities	19	**Use** if an ALARA RG for reprocessing is not developed.
3.7	Monitoring of Combustible Gases and Vapors in Plutonium Processing and Fuel Fabrication Plants	1, 19	**Withdraw**. Outdated. See new guidance in NUREG–1718, Section 7.4.3.2.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
3.16	General Fire Protection Guide for Plutonium Processing and Fuel Fabrication Plants	1	**Do not use**. Only refers to NUREG–1718. If we include Appendix D material in the reprocessing SRP, we do not need to use this RG.
3.38	General Fire Protection Guide for Fuel Reprocessing Plants (for Comment)	1	**Do not use;** outdated and withdrawn.
1.189	Fire Protection for Nuclear Power Plants	1, 9	**Do not use;** too reactor specific.
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	1, 5, 9	**Do not use;** too reactor specific.
5.44	Perimeter Intrusion Alarm Systems	8	**Do not use**, being replaced by a NUREG.
5.52	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (Other than Nuclear Power Plants)	8	**Use**. Caution: Will likely need to be replaced during 10 CFR Part 73 fuel cycle security rulemaking.
5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities	8	**Use**. Caution: Will likely need to be replaced during 10 CFR Part 73 fuel cycle security rulemaking.
5.59	Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance	8	**Use**. Caution: Will likely need to be replaced during 10 CFR Part 73 fuel cycle security rulemaking.
5.70	Design Basis Threat (C)	8	**Use**.
5.8	Design Considerations for Minimizing Residual Hold-up of SNM in Drying and Fluidized-Bed Operations	18	**Do not use**, as RGs 5.8, 5.25, and 5.42 are being revised and combined. Use the new residual holdup RG.
5.23	In Situ Assay of Plutonium Residual Holdup	18	**Do not use;** being revised. Use the new nondestructive assay RG.
5.25	Design Considerations for Minimizing Residual Hold-up of SNM in Equipment for Wet Process Operations	18	**Do not use**, as RGs 5.8, 5.25, and 5.42 are being revised and combined. Use the new residual holdup RG.
5.37	In Situ Assay of Enriched Uranium Residual Hold-up	18	**Do not use;** being revised.
5.42	Design Considerations for Minimizing Residual Hold-up of SNM in Equipment for Dry Process Operations	18	**Do not use**, as RGs 5.8, 5.25, and 5.42 are being revised and combined. Use the new residual holdup RG.
5.80	Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material	NA	**Use**.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
5.Z	This new RG is being developed for Nondestructive Assay Techniques (combining RGs 5.9, 5.11, 5.21, 5.23, 5.34, 5.37, 5.38, and 5.53).	18	**Use** with clarification in the SRP.
5.Y	This new RG is being developed for Destructive Assay Techniques (combining RGs 5.4, 5.5, 5.39, 5.48, and 5.58).	18	**Use** with clarification in the SRP.
5.X	This new RG is being developed for Residual Holdup (combining RGs 5.8, 5.25, and 5.42).	18	**Use** with clarification in the SRP.
5.W	This new RG is being developed for Statistics (combining RGs 5.3, 5.18, 5.22, and 5.36).	18	**Use** with clarification in the SRP.
5.V	This new RG is being developed for Inventory (combining RGs 5.13, and 5.33).	18	**Use** with clarification in the SRP.
5.U	This new RG is being developed for Shipping, Receiving, and Transferring SNM (combining RGs 5.28, 5.49, and 5.57).	18	**Use** with clarification in the SRP.
5.26	Selection of Material Balance Areas and Item Control Areas	18	**Use** the updated version, with clarification in the SRP.
5.27	Special Nuclear Material Doorway Monitors	8, 18	**Use** the updated version, once the revision is complete.
5.51	Management Review of Nuclear Material Control and Accounting Systems (for Comment)	18	**Use** the updated version.
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	1	**Do not use;** too reactor specific.
1.184	Decommissioning of Nuclear Power Reactors	1	**Do not use**. If a PSDAR is required for decommissioning reprocessing facilities, consider incorporating some of Section 4 (on PSDAR) into specific guidance on reprocessing, either by inclusion or reference.
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	1	**Do not use**. If a PSDAR is required, consider making a similar RG applicable to reprocessing facilities.
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	1	**Do not use;** too reactor specific.
3.65	Standard Format and Content of Decommissioning Plans for Materials Licensees	1	**Do not use**. This RG endorses NUREG–1757, Vol. 1.
3.66	Standard Format and Content of Financial Assurance Mechanisms	1	**Do not use**. This RG endorses NUREG–1757, Vol. 3.
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors	1	**Do not use**. Not applicable. Provides guidance to colocated licensees or applicants.
2.6	Emergency Planning for Research and Test Reactors	1	**Do not use**. It is being revised, and is not applicable.

Regulatory Guides That the NRC Staff Reviewed			
RGs	**Title**	**Gap(s)**	**Decision/Action Required**
DG-2004	Emergency Planning for Research and Test Reactors	1	**Do not use**. It is not applicable.
DG-1237	Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors	1	**Use** if the emergency planning requirements for reprocessing are based on 10 CFR Part 50
3.31	Emergency Water Supply Systems for Fuel Reprocessing Plants	1	**Withdraw**. Not risk informed or performance based.
3.67	Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities	1	**Use** if the emergency planning requirements for reprocessing are based on 10 CFR Part 70.
3.6	Content of Technical Specifications for Fuel Reprocessing Plants	11	**Update;** consider the material in RG 1.177 and DG-1227.
3.10	Liquid Waste Treatment System Design Guide for Plutonium Processing and Fuel Fabrication Plants	19	**Do not use;** somewhat applicable to PUREX but insufficient.
3.12	General Design Guide for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants	19	**Use**. This is applicable to aqueous reprocessing facilities.
3.19	Reporting of Operating Information for Fuel Reprocessing Plants	1, 11	**Update**. Outdated, but low priority.
3.26	Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants	7	**Withdraw** and replace with SRP, or **update**.
3.32	General Design Guide for Ventilation Systems for Fuel Reprocessing Plants	19	**Withdraw**. Content is outdated and mostly duplicated by RG 3.12, which is more detailed.
1.8	Qualification and Training of Personnel for Nuclear Power Plants	7	**Use**. Addresses how to meet 10 CFR 50.120 and 50.34(b)(6)(i). Relates to RGs 1.28 and 1.33. The consensus standards it references are somewhat applicable. Some clarification on how to use it may be needed in the SRP.
1.105	Setpoints for Safety-Related Instrumentation	9, 11	**Use**.
1.114	Guidance to Operators at the Controls and Senior Operators in the Control Room of a Nuclear Power Unit	7	**Use** “to the extent applicable.” The SRP will likely need to explain its use.
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants	7	**Use**.
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements	7	**Do not use**. Issue new RG.
1.215	Guidance for ITAAC Closure Under 10 CFR Part 52	10	**Use**. This could be adapted for reprocessing facilities depending on the level of ITAAC detail.
1.206	Combined License Applications for Nuclear Power Plants (LWR Edition)	10	**Use**. This is applicable with modifications to reprocessing facilities.

APPENDIX E

List of NUREGs Pertaining to Nuclear Power Plants and Fuel Cycle Facilities

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List of NUREGs Pertaining to Nuclear Power Plants and Fuel Cycle Facilities

Nuclear Power Plants

NUREG–75/014: "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (WASH–1400)"

NUREG–75–087: "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"

NUREG–0396: "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants"

NUREG–0654: "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (FEMA–REP–1)"

NUREG–0713: "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities"

NUREG–0800: "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"

NUREG–1437: "Generic Environmental Impact Statement for License Renewal of Nuclear Plants"

NUREG–1521: "Technical Review of Risk-Informed, Performance-Based Methods for Nuclear Power Plant Fire Protection Analyses"

NUREG–1537: "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors"

NUREG–1555: "Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan (with Supplement 1 for Operating Reactor License Renewal)"

NUREG–1577: "Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance"

NUREG–1700: "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans"

NUREG–1713: "Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors"

NUREG–1774: "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 Through 2002"

NUREG–1800: “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants”

Fuel Cycle Facilities

NUREG–1520: “Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility—Final Report”

NUREG–1567: “Standard Review Plan for Spent Fuel Dry Storage Facilities”

NUREG–1617: “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel”

NUREG–1718: “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility”

NUREG–1757: “Consolidated Decommissioning Guidance”

NUREG–1767: “Environmental Impact Statement on the Construction and Operation of a Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina”

NUREG–1821: “Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina”

NUREG–1927: “Standard Review Plan for Renewal of Independent Spent Fuel Storage Installation Licenses and Dry Cask Storage System Certificates of Compliance”

APPENDIX F

Existing Definitions for Inclusion in 10 CFR Part 7x

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Existing Definitions for Inclusion in 10 CFR Part 7x

Definitions from existing parts of the *Code of Federal Regulations* (CFR) (with appropriate modifications highlighted) are being considered for inclusion in a new 10 CFR Part 7x.

10 CFR Part 50

Act means the Atomic Energy Act of 1954 (68 Stat. 919) including any amendments thereto.

Applicant means a person or an entity applying for a license, permit, or other form of Commission permission or approval under this part of this chapter.

Combined license means a combined construction permit and operating license with conditions for a reprocessing facility issued under this part.

Commission means the U.S. Nuclear Regulatory Commission or its duly authorized representatives.

Department and Department of Energy means the Department of Energy established by the Department of Energy Organization Act (Pub. L. 95-91, 91 Stat. 565, 42 U.S.C. 7101 et seq.), to the extent that the department, or its duly authorized representatives, exercises functions formerly vested in the Atomic Energy Commission, its Chairman, members, officers and components and transferred to the U.S. Energy Research and Development Administration and to the Administrator thereof pursuant to Sections 104 (b), (c) and (d) of the Energy Reorganization Act of 1974 (Pub. L. 93-438, 88 Stat. 1233 at 1237, 42 U.S.C. 5814) and retransferred to the Secretary of Energy pursuant to Section 301(a) of the Department of Energy Organization Act (Pub. L. 95-91, 91 Stat. 565 at 577-578, 42 U.S.C. 7151).

License means a license, including a construction permit or operating license under this part, an early site permit, combined license, or manufacturing license under Part 52 of this chapter, or a renewed license issued by the Commission under this part, Part 52, or Part 54 of this chapter.

Licensee means a person who is authorized to conduct activities under a license issued by the Commission.

Production facility means (1) Any nuclear reactor designed or used primarily for the formation of plutonium or uranium-233; (2) Any facility designed or used for the separation of the isotopes of plutonium, except laboratory-scale facilities designed or used for experimental or analytical purposes only; or (3) Any facility designed or used for the processing of irradiated materials containing special nuclear material, except

(i) Laboratory-scale facilities designed or used for experimental or analytical purposes,

(ii) Facilities in which the only special nuclear materials contained in the irradiated material to be processed are uranium enriched in the isotope U-235 and plutonium produced by the irradiation, if the material processed contains not more than 10^{-6} grams of plutonium per gram of U-235 and has fission product activity not in excess of 0.25 millicuries of fission products per gram of U-235, and

(iii) Facilities in which processing is conducted pursuant to a license issued under Parts 30 and 70 of this chapter, or equivalent regulations of an Agreement State, for the receipt, possession, use, and transfer of irradiated special nuclear material, which authorizes the processing of the irradiated material on a batch basis for the separation of selected fission products and limits the process batch to not more than 100 grams of uranium enriched in the isotope-235 and not more than 15 grams of any other special nuclear material.

Note: Pursuant to Subsections 11v. and 11cc., respectively, of the Act, the Commission may from time to time add to, or otherwise alter, the foregoing definitions of production and utilization facility. It may also include as a facility an important component part especially designed for a facility, but has not at this time included any component parts in the definitions.

Safe shutdown means bringing the reprocessing plant to shutdown conditions preceding an accident according to plant technical specifications.

Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.

Special nuclear material means (i) plutonium, uranium-233, uranium enriched in the isotope-233 or in the isotope-235, and any other material which the Commission, pursuant to the provisions of Section 51 of the act, determines to be special nuclear material, but does not include source material or (ii) any material artificially enriched by any of the foregoing, but that does not include source material.

10 CFR Part 52

Decommission means to remove a facility or site safely from service and reduce residual radioactivity to a level that permits (i) release of the property for unrestricted use and termination of the license or (ii) release of the property under restricted conditions and termination of the license.

Early site permit means a Commission approval, issued under this part, for a site or sites for reprocessing facilities. An early site permit is a partial construction permit.

Limited work authorization means the authorization provided by the Director of Nuclear Material Safety and Safeguards under § 7x.yy of this chapter.

Site characteristics are the actual physical, environmental, and demographic features of a site. Site characteristics are specified in an early site permit or in a final safety analysis report for a combined license.

10 CFR Part 70

Available and reliable to perform their function when needed, as used in this part, means that, based on the analyzed, credible conditions in the integrated safety analysis, items relied on for safety will perform their intended safety function when needed, and management measures will be implemented that ensure compliance with the performance requirements in this part, considering factors such as necessary maintenance, operating limits, common-cause failures, and the likelihood and consequences of failure or degradation of the items and measures.

Contiguous sites means licensee-controlled locations, deemed by the Commission to be in close enough proximity to each other that the special nuclear material must be considered in the aggregate for the purpose of physical protection.

Critical mass of special nuclear material (SNM) means special nuclear material in a quantity exceeding 700 grams of contained uranium-235; 520 grams of uranium-233; 450 grams of plutonium; 1,500 grams of contained uranium-235, if no uranium enriched to more than 4 percent by weight of uranium-235 is present; 450 grams of any combination thereof; or one-half such quantities if massive moderators or reflectors made of graphite, heavy water, or beryllium may be present.

Integrated safety analysis (ISA) means a systematic analysis to identify facility and external hazards and their potential for initiating accident sequences, the potential accident sequences, their likelihood and consequences, and the items relied on for safety. As used here, integrated means joint consideration of, and protection from, all relevant hazards, including radiological, nuclear criticality, fire, and chemical. However, with respect to compliance with the regulations of this part, the NRC requirement is limited to consideration of the effects of all relevant hazards on radiological safety, prevention of nuclear criticality accidents, or chemical hazards directly associated with NRC-licensed radioactive material. An ISA can be performed process by process, but all processes must be integrated and process interactions considered.

Integrated safety analysis summary means a document or documents submitted with the license application, license amendment application, or license renewal application, that provides a synopsis of the results of the integrated safety analysis and contains the information specified in § 7x.yy. The ISA Summary can be submitted as one document for the entire facility, or as multiple documents that cover all portions and processes of the facility.

Items relied on for safety mean structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility that could exceed the performance requirements in § 7x.yy or to mitigate their potential consequences. This does not limit the licensee from identifying additional structures, systems, equipment, components, or activities of personnel (i.e., beyond those in the minimum set necessary for compliance with the performance requirements) as items relied on for safety.

Plutonium processing and fuel fabrication plant means a plant in which the following operations or activities are conducted: (1) operations for manufacture of reactor fuel containing plutonium, including any of the following: (i) preparation of fuel material; (ii) formation of fuel material into desired shapes; (iii) application of protective cladding; (iv) recovery of scrap material; and (v) storage associated with such operations; or (2) research and development activities

involving any of the operations described in paragraph (1) of this definition except for research and development activities utilizing unsubstantial amounts of plutonium.

Site area emergency means events may occur, are in progress, or have occurred that could lead to a significant release of radioactive material and that could require a response by offsite response organizations to protect persons offsite.

Source material means source material as defined in Section 11.z of the Act and in the regulations contained in Part 40 of this chapter. (According to AEA, the term “source material” means (i) uranium, thorium, or any other material that is determined by the Commission pursuant to the provisions of Section 61 to be source material or (ii) ores containing one or more of the foregoing materials, in such concentration as the Commission may, by regulation, determine from time to time.)

10 CFR Part 72

As low as is reasonably achievable (ALARA) means as low as is reasonably achievable taking into account the state of technology and the economics of improvement in relation to

(1) Benefits to the public health and safety,
(2) Other societal and socioeconomic considerations, and
(3) The utilization of atomic energy in the public interest.

Greater than Class C waste (GTCC waste) means low-level radioactive waste that exceeds the concentration limits of radionuclides established for Class C waste in Section 61.55 of this chapter.

High-level radioactive waste (HLW) means (i) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations and (ii) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.

Spent fuel storage cask or *cask* means all the components and systems associated with the container in which spent fuel or other radioactive materials associated with spent fuel are stored in an independent spent fuel storage installation.