Draft Regulatory Basis for a Potential Rulemaking on Spent Nuclear Fuel Reprocessing Facilities

NRC Staff Proposals

This document summarizes the NRC staff's proposals to resolve the regulatory gaps for licensing a reprocessing facility. The gaps that staff identified are documented in SECY-09-0082 (ML091520280 and ML091520365). The proposals to resolve the gaps will be used to develop a draft regulatory basis document for a potential rulemaking for licensing a spent nuclear fuel reprocessing facility. To facilitate stakeholder involvement and to obtain comments on the NRC staff's approach and rationale for resolving the regulatory gaps, this document contains the summaries of the initial draft text for each gap. These gap summaries, as appropriate, include questions where the NRC staff is seeking input that will assist in completing the draft regulatory basis document.

I. Regulatory Framework and Definitions

Gap 1—Developing a Regulatory Framework for Fuel Reprocessing Plants

Description:

A reprocessing facility meets the definition of a “production facility”, as defined in Section 11 of the Atomic Energy Act of 1954 (as amended). Current regulations under Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” applies to the construction and operation of both production and utilization (nuclear power reactors) facilities. These regulations have evolved over time to be mostly applicable to the licensing of reactors. Because there has been limited interest in developing commercial reprocessing facilities since the 1970’s, there was no need to update the Part 50 regulations to include specific requirements for reprocessing. Because Part 50 is focused on reactor safety, using Part 50 to license a reprocessing facility would require many exemptions from Part 50 requirements that do not apply to reprocessing. These exemptions might result in a protracted licensing process. In addition, the regulations in 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” do not currently provide the necessary framework for licensing production facilities. When the last major revision of Part 70 was undertaken in 2000, the licensing of a facility with a large and varied radionuclide inventory—such as a fuel reprocessing plant —was not envisioned.

NRC Staff’s Proposed Approach:

NRC staff proposes that effective and efficient licensing of reprocessing facilities is best achieved through the development of a new part of the Code of Federal Regulations. NRC staff has carried out a major analysis of current regulations in Part 50 and Part 70 to identify those requirements that would be appropriate for incorporation, with some possible modifications, into a new Part (see Table 1). The regulations should, to the extent possible, be aligned with NRC’s risk-informed and performance-based philosophy towards rulemaking [SECY-98-144, “White Paper on Risk-Informed and Performance-Based Regulation” (March 1, 1999)]. Development of a new rule (referred to as Part 7x) would be accompanied by modifications to the existing Part 50 requirements to remove many of the references to a reprocessing facility. Relevant sections of Part 50 Appendix F, “Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities” also would be carried over to the new Part 7x. Staff is in the process of identifying the requirements in Appendix F that would need to be carried over to a new rule.

The new Part 7x is planned to contain many existing requirements for NRC licensed facilities, including those pertaining to decommissioning, accident criticality, fire protection, emergency planning and seismic siting criteria.

Background:

In 2007, NRC staff sent a paper to the Commission outlining regulatory options for developing the conceptual framework for licensing major Global Nuclear Energy Partnership (GNEP) facilities, including reprocessing plants (SECY-07-0081; May 15, 2007). These options included revising the existing Part 70 regulations to allow for the licensing of reprocessing facilities and appropriate Part 50 revisions. The Commission instructed NRC staff (SRM-SECY-08-0081;

June 27, 2007) to proceed with this option, specifically developing the regulatory basis documentation to support rulemaking for Part 70. Staff was also asked to provide a gap analysis for the current regulations. In SECY-08-0134 (September 12, 2008), NRC staff discussed the possibility of developing a new regulation specific to reprocessing. The Commission continues to consider the options outlined in the NRC staff's paper, and is provided with periodic updates of NRC staff's progress.

Industry Perspective:

In response to staff's initial work, the Nuclear Energy Institute (NEI) submitted their views on the development of specific regulations for reprocessing facilities in their White Paper titled "Regulatory Framework for an NRC Licensed Recycling Facility", dated December 24, 2008 (ML083590114). The paper details NEI's view that a reprocessing facility is more like a complex fuel cycle facility than a reactor, and consequently, NEI supports development of a new regulation, Part 7x, which is specific to reprocessing facilities. This is similar to NRC staff proposals. The paper states that Part 7x should provide flexibility for the licensing of facilities associated with reprocessing operations, such as a vitrification plant or fuel fabrication and storage, and that Part 7x should contain a provision that would allow a licensee to obtain a combined license similar to Part 52. NEI concludes this approach is consistent with the Atomic Energy Act of 1954. NEI views their proposed framework as technology neutral and, therefore, applicable to reprocessing methods using either aqueous solvent extraction or electrochemical processing. Many important aspects of NEI's proposed approach are incorporated into staff's proposed approach for a new Part 7x. Nevertheless, the most important difference is that NRC staff's approach incorporates many safety-significant requirements from 10 CFR Part 50, in addition to those from 10 CFR Part 70.

Rationale:

Results of the NRC staff's regulatory gap analysis indicated that Part 70 currently does not address specific hazards and potentially larger source terms associated with the reprocessing of spent nuclear fuel (SECY 09-0082; ML091520280). These hazards include an increase in potential for radiological risks and different types of industrial processes than occur in typical fresh nuclear fuel processing facilities that are licensed under Part 70.

A fuel reprocessing facility will have a larger and more varied radionuclide inventory than a facility that manufactures fresh nuclear fuel. However, in terms of severe accident consequences, a potential reprocessing facility has a lower hazard potential than a nuclear power plant. Consequently, from a safety perspective, a reprocessing facility can be considered as being intermediate of a nuclear power plant and fresh fuel processing facility. As a result, staff is identifying and modifying current requirements based on common safety functions between reprocessing plants and those found in nuclear power plants (i.e., 10 CFR Part 50 and 52) and currently licensed fuel cycle facilities (i.e., 10 CFR Part 70).

Questions or Topics for Which Public Feedback is Requested:

Feedback is requested on the following preliminary staff positions:

1. *Emergency Planning.* NUREG-1140, "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees" (1988) details a staff analysis which led to the Commission position that emergency planning requirements at reprocessing facilities did not need to be akin to those for nuclear power plants, primarily

because the most serious accidents (fires, explosions, and criticality accidents) are similar to industrial accidents at non-reactor facilities. The Federal Register notice which published the final rule (54 FR 14051; April 7, 1989) stated that the fast-moving nature of accidents of concern meant that formal evacuation planning like for power reactors was not considered necessary, appropriate or feasible. However, NUREG-1140 was not informed by the subsequent use of high burnup fuels (essentially double those envisioned in the mid-1980s, with correspondingly greater radiotoxic inventories per unit mass), the likely presence of a large spent nuclear fuel inventory at reprocessing and recycling facilities (based upon commercial reprocessing facility experience from 1995 to the present), the 1994 Tomsk event, and the emergency response to the recent fuel pool accident in Japan. Consequently, staff is requesting input on a planned update of NUREG-1140 and how the emergency planning requirements for reprocessing and recycling facilities will need to be enhanced beyond those of existing fuel cycle facilities.

2. *Fire Protection.* In keeping with a performance-based regulatory approach, staff suggests developing new requirements based upon (in part) 10 CFR 50.48(c) and allowing NFPA 801, "Standard for Fire Protection for Facilities Handling Radioactive Materials" to be incorporated into the new regulations. NFPA 801 is a standard that describes the methodology for applying fundamental fire protection program design and elements, determination of fire protection systems and facility features, and evaluation of special nuclear hazards, including those at fuel reprocessing plants. Staff recommends supplementing the standard with acceptance criteria in NUREG-1718, "Standard Review Plan for the Review of an application for a Mixed Oxide (MOX) Fuel Facility", which discusses additional hazards that are similar to those found in a reprocessing facility. Staff is requesting input on whether this approach is appropriate for a reprocessing facility or if other approaches should be considered.

3. *Seismic Design Requirements*. Process vessels and their connecting pipes, or electrochemical cells, which contain highly radioactive materials in the form of gases, aqueous solutions or molten salts and metals must be designed to prevent major releases of radionuclides under conditions deemed to be credible. They must provide containment integrity for naturally occurring events such as earthquakes and tornadoes. Therefore, staff recommends that reprocessing facilities are designed at the highest seismic demanding level similar to nuclear power plants (Seismic Category I). Staff is requesting input on whether this approach is appropriate for a reprocessing facility or if other approaches, such as a lower seismic design requirement, should be considered.

Questions for the Public:

1. *10 CFR Part 50, Appendix F, "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities".* What requirements within Appendix F should be adapted for inclusion in a new reprocessing regulation?

2. *Decommissioning.* What should the NRC require in terms of financial assurances and planning requirements?

3. Should a potential reprocessing plant be licensed under the current regulatory framework in Part 50, or should NRC continue to develop a new regulation that is specific to the safety requirements of a reprocessing facility?

4. What does the NRC need to consider when updating the NUREG-1140 analysis?

5. Should emergency planning facilities have an emergency planning zone, and, if so, how large should it be? Also, if there should be an emergency planning zone, should there be *more* than one (e.g., one for agricultural products and one for people)?

6. Are there emergency planning aspects that are unique to reprocessing and recycling facilities?

The table lists the regulations that will be considered for inclusion in a new Part to effectively and efficiently regulate a fuel reprocessing plant and its associated facilities.

Table 1: Applicable Regulations from Part 50 and Part 70

Regulation Needed	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	50.8, 70.8
Interpretations	50.3, 70.6
Completeness and accuracy of information	50.9, 70.9
Communications	50.4, 70.5
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, 70.32
Issuance of construction permits	50.35
Environmental conditions	50.36b
Agreement limiting access to classified information	50.37, 70.32 (see 10 CFR Part 25 and/or 95)
Ineligibility of certain applicants	50.38, 70.40
Public inspection of applications	50.39, 70.21(d)
Common standards	50.40, 70.31
Additional standard for class 103 licenses	50.42
Standards for construction permits, operating licenses, and combined licenses	50.45
Emergency plans	50.47, 70.22(i)
Fire protection	50.48, 70.64 (BDC)
Issuance of licenses and construction permits	50.50, 70.31
Continuation (renewal) of licenses	50.51, 70.33
Jurisdictional limitations	50.53
Hearings and report of the Advisory Committee on Reactor Safeguards	50.58, 70.23a
Changes, tests and experiments	50.59, 70.72
Accident source term	50.67
Criticality accident requirements	50.68, 70.24

Table 1: Applicable Regulations from Part 50 and Part 70 (continued)

Regulation Needed	Location in Current 10 CFR
Risk-informed categorization and treatment of structures, systems and components	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 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, 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
Violations	50.110, 70.91
Criminal penalties	50.111, 70.92
Aircraft impact assessment	50.150
Quality assurance criteria	Part 50 App B, 70.22(f)
Fire protection program	Part 50 App R
Earthquake engineering criteria	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	70.20(a)
Disclaimer of warranties	70.37
Reports of accidental criticality	70.52
Performance requirements	70.61
Safety program and integrated safety analysis	70.62

Gap 6—Definitions

Description:

Although several sections of the Atomic Energy Act of 1954 refers to "reprocessing," this term is not defined. Existing regulations 10 CFR Parts 20, 50, 51, 60, 63, 70 and 72 also use the term "reprocessing" without a definition. Definitions for "reprocessing" and related terms will need to be developed to describe both reprocessing and reprocessing facilities for 10 CFR Chapter I. SECY-08-0134, "Regulatory Structure for Spent Fuel Reprocessing," dated September 12, 2008 (ML082110363), identifies the need to develop regulatory definitions, particularly for "reprocessing" and "recycling," as an issue related to the regulation of a reprocessing facility. Clear definitions are needed to establish the meaning and significance of terms related to licensing reprocessing facilities, decrease regulatory uncertainty, and provide boundaries for acceptable practice and action by NRC.

NRC Staff's Proposed Approach:

NRC staff is considering several variations of the term "reprocessing" for potential use in a new part of the Code of Federal Regulations regarding fuel reprocessing facilities. Amongst these are a definition that appears in the IAEA Safety Glossary (2007 Edition) version. A separate definition of "recycling" and a clarification of the term "high level waste" are being considered. NRC staff has also surveyed current definitions that exist in the Code of Federal Regulations and associated laws and have identified those definitions that may have applicability in a new Part 7x regulation.

Alternative Approaches:

In its White Paper ("Regulatory Framework for an NRC Licensed Recycling Facility", December 24, 2008 (ML083590114)), NEI proposes that all instances of the term "reprocessing" in the relevant regulations be replaced with "recycling". NRC staff disagrees that these two terms are interchangeable. NRC staff views "recycling" as a term that describes an integrated lifecycle process that results in the production of new reactor fuel from spent nuclear fuel. In contrast, reprocessing is a stage in this lifecycle process that refers to the actual mechanics of removing unwanted components from spent nuclear fuel. Reprocessing, though not defined in any applicable laws, has been used in these laws to describe the separation of spent fuel components for many decades. Therefore, staff does not see a compelling need to change the term "reprocessing" to "recycling" in regulations.

Questions or Topics for Which Public Feedback is Requested:

1. Does the public believe that there are important differences between the terms "reprocessing" and "recycling?" If so, how should those differences be expressed in a new regulation for a reprocessing facility?

II. Safety, Risk and Licensing Considerations

Gap 5—Safety and Risk Assessment Methodology

Description:

The NRC's regulations require licensed facilities to demonstrate adequate assurances of safety and limiting risk to acceptable levels. The analysis of risk involves the interactions between regulated activities, their potential hazards, the potential consequences if something unanticipated occurs, and the probabilities of occurrence. Usually, risk is defined as the product of consequence and probability. In addition to assuring acceptable levels of safety and risk, the NRC is authorized by the Atomic Energy Act (AEA Section 53e(7) of 1954, as amended) to not only protect but also minimize danger to life and property. This minimization may require measures that increase safety and reduce risk below acceptable levels. The existing regulations in 10 CFR Part 70 do not adequately address the potential hazards, consequences, and risks of reprocessing and recycling (R&R) facilities, including distinguishing potentially life-threatening events from lesser ones, minimization of risks, and property and environmental damage. As discussed further in the NRC's gap analysis (SECY-09-0082/ML091520243), 10 CFR Part 50 focuses primarily on light water reactors (LWRs), while the existing requirements and integrated safety analysis (ISA) approach of Part 70 do not adequately address the potentially larger source terms, greater number of scenarios (total, aggregated risk and/or accumulative impact), and more sequence consequences much higher than 70.61 thresholds, of R&R facilities as compared to existing fuel cycle facilities.

NRC Staff's Proposed Approach:

The NRC staff concludes approaches that incorporate more quantitative risk assessment, including probabilistic risk analysis (PRA), are needed to adequately address safety and risk at R&R facilities. The staff is considering two basic approaches: a hybrid ISA-PRA approach and a PRA approach based upon recommendations by the ACRS (Advisory Committee on Reactor Safeguards). The staff also recognizes there may be other options that incorporate more PRA methodologies into the analysis than the hybrid approach. The hybrid ISA - PRA approach has four main themes:

- quantify to the extent practical,
- use the ISA to identify accident sequences and categorize them by consequence,
- apply PRA to very high consequence events (VHCEs) and calculate risk,
- apply safety controls to reduce total risk from the R&R facility.

The hybrid processes include the following:

- Quantify all analyses to the extent practical and as supported by the state of the art.

- Use a quantified ISA to identify all credible accident sequences that, when uncontrolled, could exceed the consequence thresholds (attached), in a manner analogous to 10 CFR 70.61. The quantified ISA may use some conservative values as part of the binning process.

- Identify a subset of "high consequence events" (HCE) based upon attributes that significantly increase consequences above the high consequence thresholds in 10 CFR

70.61 and designate this subset as very high consequence events (VHCE). Currently identified attributes include the presence of reactor grade plutonium, other TRU isotopes, and/or fission products, or other characteristics (e.g., multiple receptors, significant contamination/loss of property/use or environmental degradation) that potentially significantly increase consequences above Part 70.61 thresholds. Likely examples of VHCEs include holding and storage tank failures/overflows leading to criticality scenarios, separator problems and fires, and SNF pool fires.

- Apply controls (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).

- Conduct probabilistic (i.e., quantitative) risk analyses (PRA) on HCEs and VHCE to the extent practical and consistent with the state of the art, based upon more realistic consequence and frequency information.

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

- Adjust (reduce) risk as needed to meet the appropriate NRC risk limits/criteria (these risk limits/criteria would need to be developed) by the application of additional controls (IROFS) and analyzing their effect on PRA results. The PRA may be used to rank IROFS.

- Minimize total risk to receptors by applying as low as reasonably achievable (ALARA) /as low as reasonably practicable (ALARP) to accident sequences.

- Prioritize IROFS based upon their importance to safety.

- Identify general design criteria and/or other controls (e.g., defense in depth measures) that are needed to meet ALARA/ALARP as items supporting safety (ISS) for accident situations.

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

The PRA approach recommended by the ACRS is discussed in NUREG-1909 (ML081550505). This would apply PRA methodologies to identified accident sequences at R&R facilities, perhaps using existing staff safety evaluation reports on process analogues and accident analysis handbooks as sources of potential accident sequences or for comparison with a license application. The staff is also planning activities to develop generic R&R facility flow sheets and accident source terms that will further identify sequences and provide parameters for PRAs.

The staff also anticipates recommending thresholds for environmental releases, environmental contamination, economic/schedule/availability impacts, and loss or property/land use for HCEs and VHCEs (e.g., analogous to the requirements in 70.23(a)(3) and (a)(4)). Guidance will also be needed to support application of quantitative risk analysis approaches to R&R facilities. The staff also anticipates criticality safety will follow an approach similar to Part 70 (e.g., double contingency).

Figures 1 and 2 summarize the hybrid ISA-PRA approach.

Alternative Approaches:

The staff found most safety/risk limits (i.e., aggregated or total risk) correspond to circa 1E-6/yr and this is consistent with NRC documents and policy on risk. Staff concludes an aggregate risk limit would need to be developed for R&R facilities but it would most likely be similar to these existing criteria. Staff considered and reviewed several different assessment methodologies for safety and risk:

- Option 1 considered qualitative approaches using multiple consequence and likelihood bins. This approach was one of several suggested at the public workshops in Fall 2010. Staff found more quantification was needed to avoid differences in qualitative judgement, improve consistency, and provide a reasonable basis for regulatory decisions involving R&R facilities, and, thus, did not consider this further.

- Option 2 evaluated semi-quantitative methods, such as using indices. It was also found to rely heavily on judgments and did not provide an adequate calculational continuum for R&R facilities, and was not considered further.

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

- Option 4 evaluated a hybrid ISA-PRA approach, where ISA is used for some accident sequence categorizations and PRA approaches are used for other accident sequence categories (e.g., VHCEs). This approach was one of several suggested at the public workshops last Fall.

- Option 5 considered a full PRA approach. This approach was one of several suggested at the public workshops last Fall.

Option 5 is the methodology recommended by the advisory committees (ACRS and the ACNW&M – Advisory Committee on Nuclear Waste and Materials). In particular, in NUREG-1909, the Advisory Committee on Reactor Safeguards provided important insights on risk. The ACRS stated that:

"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 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 NRC staff concurs with the ACRS's recommendations if data and methods are available to support a PRA. However, the staff review found that methods relying extensively on PRAs usually could address VHCEs and HCEs, but were cumbersome in screening lower risk sequences and non-binary logic events, such as chemical reactions, at R&R facilities, and sufficient data may not be available for PRA analyses of all potential events. R&R facilities will probably have significant chemical hazards which could be adequately addressed by quantitative ISA methods. The staff intends to continue its evaluation of PRAs throughout the rulemaking process. In addition, the staff notes that accident event sequences for R&R facilities would likely be different for different designs and technologies, and, thus, a method for identifying and screening event sequences is also needed. Any proposed regulation would be risk-informed, not risk-based, per Commission's long standing policy (e.g., the Policy Statement on Risk-Informed, Performance Based Regulation [1997]). Consequently, at this time, the staff envisions a rule that encompasses either Option 4 or Option 5 - a hybrid ISA-PRA methodology for identifying and screening accident events, with PRA applied to VHCE for adequately addressing risk and providing risk insights; or a full PRA approach based upon the recommendations of the ACRS that would be applied to all significant event sequences. The staff will assess both options (pros and cons) during the rulemaking process and make a recommendation at that time.

Rationale:

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 and recycling facilities have potential hazards and characteristics more similar to reactor facilities, while other areas and processes are more similar to uranium fuel cycle facilities. The NRC staff analyses found that areas and processes at reprocessing and recycling facilities with hazards and characteristics more similar to reactor (10 CFR Parts 50 and 52) facilities need to be analyzed with the same degree of scrutiny and rigor as is applied in addressing hazards and characteristics associated with reactor facilities, while areas at reprocessing and recycling facilities with hazards and characteristics more similar to other fuel cycle (10 CFR Part 70) facilities need to 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, consistent with their hazards and risks.

The NRC staff considers the ISA method required by 10 CFR Part 70 is 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 transuranic isotopes at R&R facilities have the potential to greatly increase consequences far above the 10 CFR Part 70 high consequence thresholds for some accident sequences. These very high consequence events require more rigorous analyses and controls to reduce their probability (e.g., to "very highly unlikely/incredible"), and ultimately reduce their risks to acceptable levels. Staff also agrees with the recommendation made by the ACRS in their February 17th, 2011 letter to the Chairman 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."

Staff also notes that there is the potential for a large number of high consequence events and VHCE. The number of these events for R&R facilities is likely to far exceed the number at existing fuel cycle facilities, and, consequently, this would increase risk above that intended for 10 CFR Part 70. Staff concludes aggregation of the risk from these accident sequences and the requirement to meet a risk limit is necessary to reduce total risk to levels commensurate with other NRC-licensed facilities, such as fuel cycle facilities.

In addition, an IROFS prioritization scheme is needed because of the potential for a large number of IROFS. Prioritization facilitates the NRC licensing and inspection activities, as well as ensuring the applicant's management measures are commensurate with the level of safety provided by the IROFS.

Questions or Topics for Which Public Feedback is Requested:

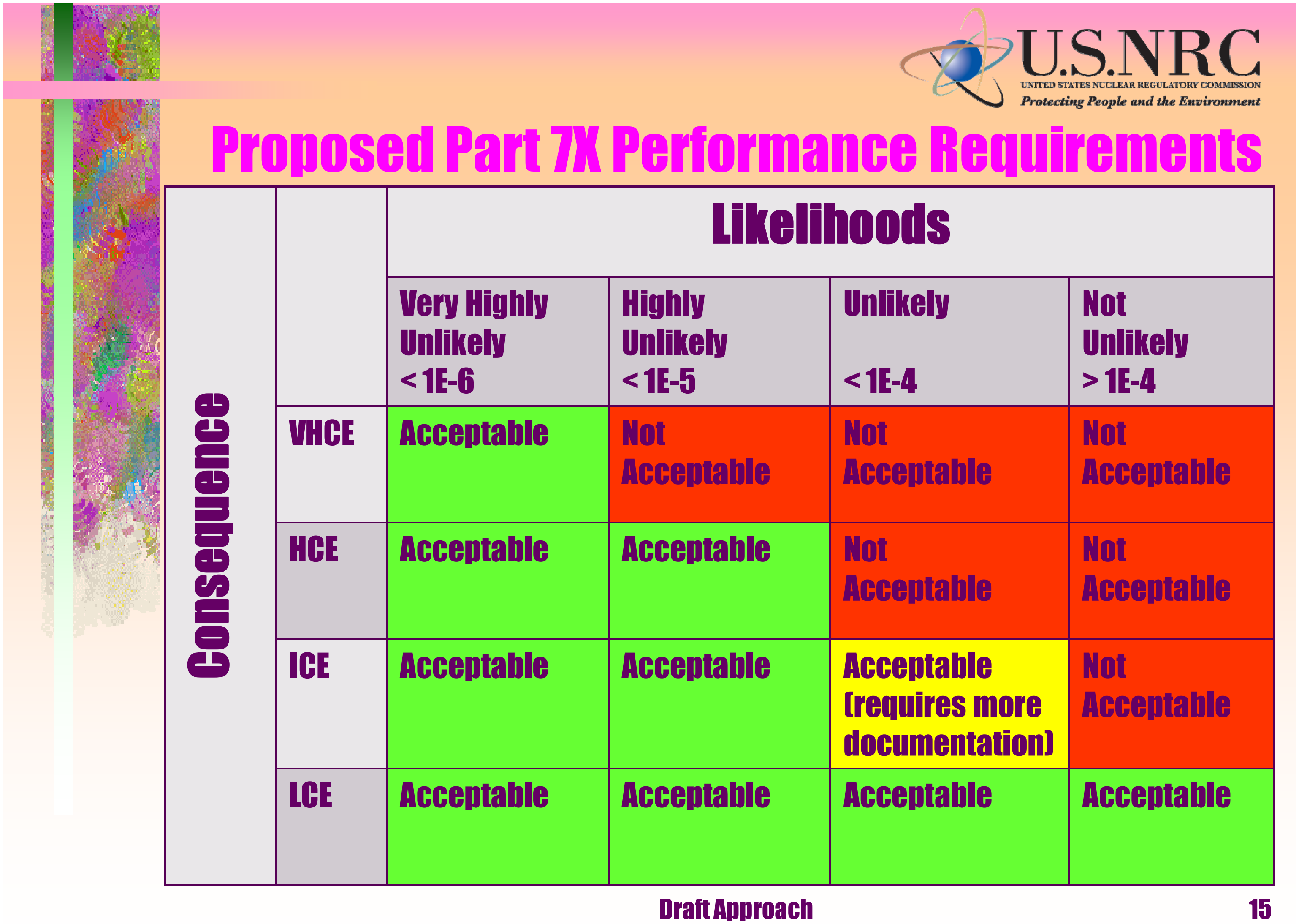
The NRC has a Commission Policy Statement on safety and total risk from nuclear power plant operations. In a similar manner, should there be a safety and total risk limit to members of the public developed for reprocessing and recycling facilities? If so, should this safety and total risk limit apply only to VHCEs, VHCEs and HCEs, or some other grouping of accident sequences and categories?

In a similar manner, should the total risk to workers or member of the work force be assessed? If so, which accident sequence categories should be included in the calculations?

Should the total risk goal, for a worker and a member of the public, for R&R facilities be the same as the goal for commercial nuclear power plants?

Should the staff consider a total risk limit, or a total risk limit and a total risk aspirational goal?

The NRC has a Commission Policy Statement that recommends the use of PRA techniques to the extent practical and supported by the state of the art. The ACRS and ACNM&W also recommend the use of PRA for PRA for complex facilities, such as R&R. Consequently, should PRA methods be used more or less extensively than the hybrid approach mentioned above?



Consequence		Likelihoods			
		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 (requires more documentation)	Not Acceptable
	LCE	Acceptable	Acceptable	Acceptable	Acceptable

Figure 1: Proposed Performance Requirements

Receptor Event	Worker	Individual Outside Controlled Area (IOC) (aka General Public)
VHCE - Very High Consequence Event **- Prevent to very highly unlikely** **- PRA required**	**- >> 100 rem (TEDE)** **- > endanger life** **- presence of FPs, RG Pu, TRUs, >> 1 receptor, unique chem** **- aggregate, limit on total risk** **-- ALARA/ALARP**	**- > 100 rem** **- chemical – endanger life** **-- presence of FPs, RG Pu, TRUs, >> 1 receptor, unique chem** **-- aggregate, limit on total risk** **-- ALARA/ALARP** **-- > 500,000 Part 20, App B** **-- > PAG** **-- >$1B**
HCE - High Consequence Event: **- Prevent to highly unlikely** **- Prevent or mitigate to intermediate or low**	- > 100 rem (TEDE) - Endanger life of worker (chemical) -**aggregate, limit on total risk** **-- ALARA/ALARP**	- > 25 rem - > 30 mg soluble U - Irreversible or serious, long-lasting health effects (chemical) -**aggregate, limit on total risk** **-- ALARA/ALARP** **-- > 50,000 Part 20, App B** **-00 > PAG** **-- > $100M**
ICE - Intermediate Consequence Event: **- Prevent to unlikely** **- Mitigate to "low"**	-> 25 rem - Irreversible or serious long-lasting effect (chemical)	- > 5 rem - Mild transient health effects (chemical) - > 5000x Part 20, App B
(Low Consequence)	**Mild transient health effects or less**	**Lesser effects** **Draft Approach**

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Proposed Part 7X Receptor And Consequence Matrix

Changes are in red and purple, Bold, underline

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Figure 2: Proposed Consequence Limits

Gap 7—Licensed Operators and Criteria for Testing and Licensing Operators

Issue:

The Atomic Energy Act of 1954, as amended, requires production facilities to have licensed operators (42 USC 2137). The current regulations for operator licensing are in 10 CFR Part 55, "Operators' Licenses," which are not applicable, in whole, to operators of reprocessing facilities. Thus, the NRC staff is developing a regulatory basis on who should be licensed operators and what criteria should be used for testing and licensing operators.

NRC Staff's Proposed Approach:

The NRC staff reviewed 10 CFR Part 55, previous versions of 10 CFR Part 55 which applied to reprocessing, and 10 CFR Parts 50, 70, and 72, and is proposing that requirements for operator licensing for a reprocessing facility be derived primarily from 10 CFR Part 55. In addition, the NRC staff is proposing to apply a risk-informed and performance-based approach to determine which personnel need to be licensed and the requirements for their licensure.

Consistent with the approach outlined in Gap 5 (risk considerations for a production facility licensed under 10 CFR Part 70), the NRC staff is proposing that personnel whose actions are clearly related to safety, such as being relied on to control the important parameters of systems, if not controlled, could lead to accident sequences with very high consequences events will be licensed by the NRC. The licensed operators would be personnel 'in charge' of the controls and systems such that: they operate the controls or directly oversee the operation of the controls; are responsible for bringing the system back to a safe configuration following violation of a limiting condition of operation; and whose knowledge and consent is needed before other personnel manipulate apparatus and mechanisms which may affect controlled parameters (similar to 10 CFR 50.54(j)) or items relied on for safety to prevent a very high consequences event. The NRC staff is proposing to allow the reprocessing facility licensee the ability to develop and run the certification program that trains and qualifies licensed operators. The NRC staff would review and audit the facility licensee's certification program to ensure that it adequately trained and tested the licensed operator candidates. A candidate that graduated from an NRC approved certification program would be eligible to be a licensed operator (the candidate would also be subject to other requirements such as those in 10 CFR 55, Subpart C–Medical Requirements).

10 CFR 55 includes requirements for written examinations of operators in 10 CFR 50.41, written examinations of senior operators in 10 CFR 55.43, operating tests in 10 CFR 55.45, and simulation facilities in 10 CFR 55.46. The NRC staff is proposing to use these sections as the basis for the requirements for areas of training and testing of operators of a reprocessing facility.

Simulation facilities are used to perform operating tests, and can be used to meet experience and training requirements. In 10 CFR 55.46 a simulation facility is: 1) a plant referenced simulator, or 2) using the facility as a simulator, or 3) a Commission approved simulator. The NRC staff is proposing to adopt a similar approach for use of simulation facilities at a reprocessing facility. The acceptability of a simulation facility would be based on demonstrating fidelity during normal and accident sequences such that negative training is avoided.

The NRC staff is proposing that the operator and senior operator approach in 10 CFR 55 be applied to a reprocessing facility. The NRC staff provides alternative approaches for applying requirements for senior operators in the next section.

Alternative Approaches:

The Nuclear Energy Institute (NEI) White Paper on the “Regulatory Framework for Recycling Nuclear Fuel” (ADAMS ML083590115 and ML083590129) addressed operating licensing. The NRC staff has identified a range of alternate roles and responsibilities that the NRC staff could have in testing the licensed operator and senior operator candidates. 10 CFR Part 55 contains requirements for testing candidates with both written examinations and operating tests. 10 CFR Part 55 requires that each test must be prepared, proctored, and graded. However, the roles and responsibilities can vary as shown in Table 1.

The NEI-like approach is similar to what NEI proposed in its white paper. The 10 CFR Part 55-like approach would simply apply the existing 10 CFR Part 55 approach to reprocessing. These two approaches bound a range of other possible variations. The NRC staff’s proposed approach would allow the facility licensee to develop and run the certification program that trains and qualifies licensed operators; but includes provision for NRC review and approval to ensure that the tests used to support granting the operator’s license are acceptable.

The NRC staff identified the following approaches for addressing the regulatory approach for senior operators: (i) apply a 10 CFR Part 55 approach (senior operators with additional training and oversight role); (ii) remove senior operator oversight role (senior operators with additional training, may act as advisors); and (iii) increase operator training (all operators have senior operator training).

Table 1. NRC and Licensee Roles and Responsibilities Using Different Approaches for Testing Licensed Operator and Senior Operator Candidates

10 CFR Part 55–Like Approach			
	Prepares	**Proctors**	**Grades**
Written Examinations	Licensee; the NRC staff can elect to prepare the exam, but the NRC staff must review and approve licensee-prepared exams	Licensee and an NRC staff contact is available during exam	Licensee (usually for licensee-developed exams); the NRC staff for NRC-developed and the NRC staff must review and approve the facility-recommended grades
Operating Tests	Licensee, the NRC staff can elect to prepare the test, but must review and approve licensee-prepared tests	The NRC staff administers the operating test	The NRC staff grades the operating test
NEI–Like Approach			
Written Examinations	Licensee, the NRC staff audits	Licensee, the NRC staff audits and/or monitors	Licensee, the NRC staff audits
Operating Tests	Licensee, the NRC staff audits	Licensee, the NRC staff audits and/or monitors	Licensee, the NRC staff audits

Table 1. NRC and Licensee Roles and Responsibilities Using Different Approaches for Testing Licensed Operator and Senior Operator Candidates (continued)

NRC Staff's Proposed Approach			
	Prepares	**Proctors**	**Grades**
Written Examinations	Licensee, NRC can elect to prepare but must review and approve the test (The examination standards would require some examinations to be NRC-developed)	Licensee, NRC Contact Available during Exam	Licensee (usually for licensee-developed exams), NRC for NRC-developed, NRC reviews and approves the licensee recommended grades
Operating Tests	Licensee, NRC can elect to prepare but must review and approve the test (The examination standards would require some tests to be NRC-developed)	Licensee administers with NRC co-evaluation; or the NRC administers	Licensee grades licensee administered tests with comparison to NRC evaluation grades, NRC grades NRC administered operating test
Note: For all alternatives, the NRC staff proposes to retain the authority to step in and take over if requested or under its own discretion [see 10 CFR 55.40(c)].			

Rationale:

Years of industry and NRC experience with simulators at reactor units has proven the value of simulators in the training of licensed operators at nuclear power plants. The use of simulators to train for accident conditions is especially important as such training could not be easily and safely accommodated through use of the facility. The acceptability of a simulation facility is based primarily on demonstrating fidelity such that negative training is avoided.

NRC staff considers the Part 55 approach to the roles and responsibilities of the NRC and licensee during the testing of the candidate to be an excellent starting point; as this framework has been successfully applied for many years at reactors. As in the staff's proposed approach any problems or disagreements in the exams, tests, and grades must be resolved before the exams, and tests are given, or the grades used in operator licensing. If such problem's in exams or tests were discovered later it would be extremely difficult to take corrective action. The staff's proposed approach would allow the staff to co-evaluate operating tests as this approach still allows staff to ensure that the test is administered properly, while reducing the use of staff resources and increasing flexibility.

The NRC staff has preliminarily concluded that the distinction between an operator and a senior operator, in 10 CFR Part 55, is applicable to a reprocessing facility because of the size and complexity of a proposed reprocessing facility, and because distinct facilities, operated by different operators in different control rooms, could have hypothetical very high consequence event sequences. The NRC staff considers that there are two parts to addressing the senior operator issue. The first is the supervisory role; the second is the greater training and testing that senior operators receive. The NRC staff considers that the personnel who direct license operators should at least be licensed operators themselves. The NRC staff also considers that personnel with the additional training that senior operators receive under the existing framework [10 CFR 55.43(b)] would be required at the facility to ensure an integrated site-wide approach to safety.

Questions or Topics for Which Public Feedback is Requested:

On each of the following topics the NRC staff is seeking stakeholder input on the data that may be used to support the NRC's proposed approach (e.g. is using DOE information on its requirements for operators at comparable facilities, such as vitrification facilities, and its testing approach a reasonable technical basis, and is there other information that the NRC staff should consider). Also the NRC staff is seeking input on whether there are additional alternative approaches that the NRC staff should consider in each of the topic areas.

Roles and Responsibilities:

1. What role should the NRC play in testing candidates?

2. An auditing approach may be problematic because tests and grading used in operator licensing may later be found to be deficient; how could this issue be addressed?

Simulation Facilities:

1. Should simulation facilities be required for training and testing?

2. What requirements should there be for an acceptable simulation facility (e.g., fidelity, plant referenced, facility-as-simulator)?

3. What areas of training and testing should take place using a simulation facility?

Licensed Personnel

1. Which personnel should be licensed?

Senior Operators

1. Should the NRC staff's proposed approach include requirements for senior operators?

2. If the NRC staff includes requirements for senior operators, should the supervisory requirements be removed?

3. If the NRC staff does not include requirements for senior operators, should training for operators be increased?

Gap 9—General Design Criteria

Description:

The NRC establishes minimum requirements for proposed facilities or applications of licensed radioactive materials that provide:

- assurance that structures, systems, and components that are important to safety 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

NRC regulations frequently identify these minimum requirements by terminology such as general design criteria (used in 10 CFR Part 50) or baseline design criteria (used in 10 CFR Part 70). These terms are essentially synonymous in NRC regulations, although the use of general design criteria is much more prevalent and general design criteria tend to be more specifically stated. The baseline design criteria for fuel cycle facilities in 10 CFR part 70 are very general and do not comprehensively address hazards posed by the operation of reprocessing and recycling facilities. Whereas the more detailed general design criteria in 10 CFR Part 50 only apply to nuclear power plants. Additionally, 10 CFR 50.34(a)(3)(i) states that general design criteria for chemical processing facilities are being developed. Thus, currently, general design criteria do not exist for reprocessing and recycling facilities, and, thus, a regulatory gap exists.

NRC Staff's Proposed Approach:

The NRC staff evaluated several different sources of information on potential design criteria. The NRC staff reviewed existing regulations (10 CFR Parts 20, 50, 70, and 72), proposed regulations [proposed 10 CFR Part 50 Appendices P (39 FR 26293, July 18, 1974) and Q (39 FR 26296, July 18, 1974)], and stakeholder information [Nuclear Energy Institute White Paper on the "Regulatory Framework for Recycling Nuclear Fuel" (ADAMS ML083590115 and ML083590129) and Advisory Committee on Nuclear Waste and Materials (NUREG-1909, "Background, Status, and Issues Related to the Regulation of Advanced Spent Nuclear Fuel Facilities")].

On the basis of its review, the NRC staff proposes ten categories for general design criteria: Overall, Confinement and Containment, Process Safety, Criticality Safety, Radiological Protection, Physical Security, Material Control and Accounting, Fuel and Radioactive Waste, Siting, and Decommissioning. The NRC staff identified 78 potential general design criteria within these ten categories (Table 1).

Alternative Approaches:

The NRC staff considered using the general design criteria in each of following separate sources of information for the proposed general design criteria: (i) existing 10 CFR Part 50 general design criteria; (ii) general design criteria proposed for 10 CFR Part 50 for reprocessing facilities (proposed Appendices P and Q); (iii) 10 CFR Part 70 baseline design criteria; (iv) 10 CFR Part 72 general design criteria; and (v) the NEI white paper proposed baseline design criteria. For example, the NRC staff considered, as one option, just using the 10 CFR Part 70 baseline design criteria for the potential general design criteria for a recycling and reprocessing facility.

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

Rationale:

The NRC staff has preliminarily concluded that reprocessing and recycling facilities have many design characteristics similar to 10 CFR Part 50 and 10 CFR Part 72 facilities that handle spent nuclear fuel and irradiated materials and thus the general design criteria requirements should reflect those identified in 10 CFR Part 50 and 10 CFR Part 72. Consequently, the NRC staff proposes general design criteria that largely follow the general design criteria in 10 CFR Part 50 and 10 CFR Part 72. This initial list of proposed general design criteria were modified to address the pertinent characteristics of reprocessing and recycling facilities, and those general design criteria proposed, decades ago, for reprocessing facilities (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 thresholds for applying general design criteria and that current reprocessing and recycling facilities are integrated, with only nominal physical and process separation between areas. Thus, the NRC staff concludes a basis for a threshold does not exist.

Specific Questions or Topics for Which Public Feedback is Requested:

1. What additional general design criteria categories, issues, or criteria should be added to ensure public health and safety from reprocessing and recycling facilities?

2. What general design criteria categories, issues, or criteria could be removed, if any, without reducing public health and safety from reprocessing and recycling facilities?

3. Are there general design criteria categories, issues, or criteria that should not be included because the topic could be addressed more effectively and efficiently by relying on an existing different regulatory concept (for example, instead of a general design criteria for design, construction and operation of the facility to facilitate decontamination and decommissioning could the regulatory basis for this topic rely on specific a regulatory citation such as 10 CFR 20.1406)?

4. Given that the 1974 proposed general design criteria in 10 CFR Part 50 Appendices P and Q were not technological neutral, and were not implemented, how should the NRC treat these proposed technology-specific general design criteria within the regulatory basis for a technology-neutral approach?

Table 1. Summary of Proposed General Design Criteria for Reprocessing and Recycling Facilities

Draft General Design Criteria Categories with Associated Proposed General Design Criteria									
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

Gap 10—One-Step Licensing and Inspection, Testing and Acceptance Criteria (ITAAC)

Issue:

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, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” The one-step licensing process combines the authorization of a construction permit and a license for the operation of the nuclear power plant into a one-step licensing process called a combined license (COL). In the one-step licensing process, the NRC verifies the licensing process through the use of inspections, testing, analyses, and acceptance criteria to ensure the plant operates as designed and constructed before the combined operating license is issued.

The NRC’s 10 CFR Part 52 regulations do not apply to spent nuclear fuel reprocessing facilities. In addition, the requirements for approval of applications for Domestic Licensing of Special Nuclear Material in 10 CFR Part 70.23, “Requirements for the Approval of Applications,’’ do not address reprocessing facilities. As a result, there are no regulations for one-step licensing of a reprocessing facility. Gap 10 in SECY-09-0082, “Update of Reprocessing Regulatory Framework-Summary of Gap Analysis” dated May 29, 2009 describes a one-step licensing approach for fuel reprocessing facilities.

NRC Staff’s Proposed Approach:

The NRC staff is proposing to implement licensing requirements for a reprocessing facility that are similar to the 10 CFR Part 52 requirements for a combined license for the construction and operations of a nuclear power plant. This one-step licensing process for reprocessing facility will include inspection, testing, analyses, and acceptance criteria (ITAAC) to confirm the facility meets the design, construction and licensing requirements.

The proposed regulation will provide the requirements and procedures for the Commission to issue an early site permit for approval of a site for a reprocessing facility separate from the filing of an application for a combined license for the facility. Thus, a one-step license application could reference an early site permit. If an early site permit is not referenced, the applicant would be required to provide an equivalent level of information in the one-step licensee application.

10 CFR Part 52 also includes options for standard design certifications, standard design approvals, and manufacturing licenses. The NRC staff’s proposal does not include these options. Instead, information on confirming reprocessing designs, requirements for design approvals by the NRC staff, and requirements and approvals for manufacturing of spent fuel reprocessing facilities components would be incorporated in the one-step licensing application processes.

Separate appendices may be added to describe unique design requirements to be addressed for liquid-liquid aqueous separation processes and the electrochemical separation process.

The NRC staff’s regulatory basis will also address general and technical information to be included in the contents of a combined license application. The NRC staff will also identify the technical information that must be incorporated in the applicant’s final safety analysis report describing the spent nuclear fuel reprocessing facility.

Alternate Approaches:

A traditional two-step license approach could be adopted where 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. This approach may introduce additional costs and increase the time to construct and operate the facility.

In a second alternative, NRC would not permit an applicant for a reprocessing facility to reference an early site permit, and the applicant would be required to address all siting issues in the combined license application.

A third alternative would allow an applicant for a combined license application to obtain a pre-approval of the design and obtain standard design certifications.

Rationale:

The NRC staff concludes it is appropriate that a one-step licensing approach with ITAAC be implemented for licensing spent nuclear fuel 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 construction and licensing of the facility in a one-step licensing process.

In the Staff Requirements Memorandum to SECY-06-0066, the Commission required the staff to consider the most effective and efficient elements of the NRC's licensing process to develop a licensing process for spent nuclear fuel reprocessing facilities, including a review of the one-step licensing provisions for enrichment facilities as described in Section 193 of the Atomic Energy Act, and features of nuclear power plant combined licensing under 10 CFR Part 52 (i.e., construction authorization and operating license hearing process, design certification process, and early site permitting process). In SECY-09-0082, the NRC staff identified that clarity is needed to provide reasonable assurance that a reprocessing facility, undergoing a one-step licensing process, will have been constructed and will operate in conformity with the Atomic Energy Act, 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 ITAACs.

In a letter of interest in construction a reprocessing facility, one company indicated that a one-step licensing process is needed for the same reasons that support the reactor COL process (ML081280528). The Nuclear Energy Institute (NEI) has issued a white paper that includes a framework that could allow an applicant the flexibility for either a two-step or one-step licensing process. NEI also supports the developments of ITAACs to ensure a reprocessing facility will be built and operated safely.

Specific Questions or Topics for Which Public Feedback is Requested:

1. What specific inspection, testing, analyses and acceptance criteria should NRC require of an applicant for an aqueous separation reprocessing facility? For an electrochemical separation reprocessing facility?

2. Should different design criteria be requested for aqueous an electrochemical separation processes?

3. Because there are no standard designs for reprocessing facilities, what design information should be required in a combined license application so that NRC staff can effectively conduct its licensing review within the one-step licensing and ITAAC framework?

4. What information should be included in the applicant's final safety analysis report?

Gap 11—Technical Specifications

Issue:

Technical specifications for reprocessing facilities in 10 CFR Part 50 require modification to reflect the risk basis for safe operation of production facilities under 10 CFR Part 70. Requirements for technical specifications for reprocessing facilities currently exist in 10 CFR Part 50. 10 CFR Part 70, which requires items relied on for safety, does not require technical specifications. Therefore, for incorporation of technical specification requirements into 10 CFR Part 70 or a new Part 7x, revisions will be needed to clarify the division between items relied on for safety in the safety analysis and technical specifications.

NRC Staff's Proposed Approach:

Technical specification requirements for production facilities in 10 CFR 50.36 include items in the categories: (i) safety limits and limiting control settings; (ii) limiting conditions of operation; (iii) surveillance requirements; (iv) design requirements; and (v) administrative controls. The NRC staff is proposing to modify and update these technical specification requirements and place them in Part 7x. The NRC staff is proposing that the safety limits and limiting control settings, limiting conditions of operation, surveillance requirements, and design requirements technical specifications would apply to accident sequences that can endanger the life of a member of the public, result in "high consequences" as defined in Part 70 to large groups of individuals, or result in widespread contamination of land and property. The NRC staff considers these accident sequences as a subset of high consequence accident sequences and the NRC staff is referring to these as "very high consequence" accident sequences. The NRC staff is also proposing to include a requirement that overall facility technical specifications, such as a spent fuel burn-up limit, operational technical specifications to address natural phenomena hazards, and administrative technical specifications, be developed by the applicant.

Reprocessing of spent fuel would involve de-encapsulation, and release into process vessels, large quantities of radioactive materials including fission product gases such as krypton-85 and particulates such as iodine-129 and transuranic radionuclides that could become airborne. The NRC staff is considering requiring technical specifications for effluents at reprocessing facilities, as is required for power reactors in 10 CFR 50.36a, to make releases of such gases and particulates to the environment as low as is reasonably achievable (ALARA).

Alternative Approaches:

The Nuclear Energy Institute (NEI) White Paper on the "Regulatory Framework for Recycling Nuclear Fuel" (ADAMS ML083590115 and ML083590129) recommended developing technical specifications for those items relied on for safety which will be applied to protect against or mitigate the potential accident consequences that could result in a high consequence event involving fission product releases to an individual located outside the controlled area. Because very high consequence accident sequences are a subset of high consequence accident sequences referred to in NEI's White Paper, NEI's recommendation is comparable 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 do not directly address safety of workers and protection of property and environment from very high consequence accident sequences.

Rationale:

The NRC staff is including technical specifications in the regulatory basis because Section 182a. of the Atomic Energy Act of 1954, as amended, mandates the inclusion of technical specifications for production facilities, and a reprocessing facility is a production facility. The NRC staff concludes technical specifications are appropriate for areas and processes at reprocessing and recycling facilities with hazards and characteristics more similar to reactor (Parts 50 and 52) facilities, while areas at reprocessing and recycling facilities with hazards and characteristics more similar to other fuel cycle (Part 70) facilities would not be subject to technical specifications. The NRC staff's proposal for where technical specification would be required is consistent with a risk-informed and performance-based regulatory approach.

10 CFR Part 70 requires, for certain licensees authorized to possess a critical mass of special nuclear material, an integrated safety analysis and implementing and maintaining items relied on for safety identified in the integrated safety analysis to ensure safety from potential radiological and certain chemical accidents. The integrated safety analysis method required by 10 CFR Part 70 is considered appropriate to address the types of hazards and accident sequences associated with fuel cycle facilities. However, the presence and processing of large quantities of fission products and transuranic 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 being addressed in the regulatory basis for Gap 5, very high consequence accident sequences would be made "very highly unlikely." Specific 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, because reprocessing processes would involve large quantities of highly radioactive and other hazardous material, the NRC staff considers it reasonable to establish, as in the case of power reactors, general technical specifications that may not necessarily trip the very high consequence accident sequence criteria but would still have a clear and important nexus to public health and safety. Examples of such technical specifications may be a burn-up limit and applying the as low as is reasonably achievable principle to environmental effluents.

Specific Questions or Topics for Which Public Feedback is Requested:

1. Should effluent technical specification requirements be established for reprocessing facilities?

2. Are the NRC staff's proposed overall facility technical specifications reasonable and complete?

III. Waste Management & Environmental Considerations

Gap 2—Independent Storage of High Level Waste

Issue:

No independent waste storage options are available under 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," to accommodate interim, commercial independent storage of solidified high-level waste (HLW) from reprocessing facilities.

NRC Staff's Proposed Approach:

NRC staff is considering incorporating the requirements for safe and secure HLW storage within the general license for a potential reprocessing facility. This approach would allow for the regulation of HLW storage at a potential reprocessing facility similar to how spent nuclear fuel (SNF) storage is regulated at nuclear power plants. Authority to issue this general license, however, would require modification of 10 CFR Part 72 Subpart K to include the storage of both SNF and HLW at a licensed reprocessing facility. Currently, this general license authority extends only to SNF at nuclear power reactors licensed under Part 50. Similar to the regulatory approach given in 10 CFR Part 50 Appendix A, NRC staff propose the inclusion of general design criteria in the new Part 7X to address safety considerations for (i) SNF and HLW storage, handling and radioactivity control, (ii) prevention of criticality in storage and handling, and (iii) monitoring conditions. NRC staff assumes that solidified reprocessing HLW would likely be stored in the same types of canister systems that currently are used to store commercial SNF. Nevertheless, NRC staff recognizes that additional licensing review might be needed to confirm that canister systems licensed to store SNF (i.e., § 72.214) can safely and securely store solidified HLW from reprocessing. In addition, NRC staff is considering that reasonable limits might need to be established on the amount of SNF stored at a reprocessing facility, to distinguish the proposed facility from an independent spent fuel storage installation.

Alternative Approaches:

A specific license for a separate, independent spent fuel storage installation for solidified HLW from reprocessing could potentially be constructed and licensed using 10 CFR Part 72, if additional regulatory requirements for HLW storage are developed.

Rationale:

Currently, Subpart K of 10 CFR Part 72 permits the issuance of a general license for SNF storage at the site of a nuclear power reactor licensed under 10 CFR Part 50. Regulatory requirements for issuing this general license also are developed in Subpart K. To issue a general license for HLW storage at a reprocessing facility, rulemaking would be needed to amend Subpart K to authorize this action. This rulemaking also would need to allow issuance of a general license for SNF storage at a reprocessing facility licensed under new Part 7x. Similarly, Subpart L of 10 CFR Part 72 provides regulatory requirements for the approval of SNF storage casks, but has no provisions for approving casks for HLW storage. Rulemaking would be needed to amend Subpart L to allow for cask certification for HLW storage, and to identify technical requirements that are needed for safe HLW storage (e.g., § 72.236).

If a general license is not available for the storage of SNF and HLW at a potential reprocessing facility, a specific license for a separate storage facility would be needed. Although provisions exist in Part 72 for licensing a U.S. Department of Energy (DOE)-operated monitored retrievable storage (MRS) facility for solidified HLW from reprocessing, there is no national program for development of an MRS. Although Part 72 allows the storage of SNF at a commercial interim spent-fuel storage installation, this regulation does not authorize commercial storage of HLW. Significant revision to Part 72 would be needed in order to develop an appropriate regulatory framework for the licensing of a commercial facility for the storage of HLW. These revisions would be in addition to the revisions already identified for Subparts K and L. NRC staff concludes that although revisions to Part 72 could be developed to support a specific license for commercial HLW and SNF storage at an independent waste storage installation associated with a reprocessing facility, detailed analyses would be needed to consider the full range of potential effects on existing holders of Part 72 specific licenses.

<u>Specific Questions or Topics for Which Public Feedback is Requested:</u>

1. From a technical standpoint, the NRC staff considers that the storage of solidified HLW from reprocessing is not significantly different from the storage of SNF (e.g., 51 FR 19106; May 27, 1986). Are there technical issues related to the storage of solidified HLW from reprocessing that are not sufficiently represented by the storage of SNF?

2. Should storage of solidified HLW from reprocessing be regulated as part of the general license for a potential reprocessing facility (similar to a nuclear power plant), or should NRC develop some other approach for safe and secure storage of solidified HLW from reprocessing at a separate facility?

3. Should limits be placed on the amount of SNF that could be stored at a reprocessing facility, if a general license for storage is issued?

Gap 3—Waste Incidental to Reprocessing

Issue:

Radioactive wastes from a commercial reprocessing facility will need to be disposed of at an appropriate waste disposal facility. Distinguishing between high-level and low-level radioactive waste (HLW and LLW) associated with the reprocessing of spent nuclear fuel (SNF) is necessary to ensure that appropriate safety requirements are met for both interim storage and ultimate disposal.

In 1969, the Atomic Energy Commission (AEC) published a draft *Policy Statement* entitled "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities (Appendix D)" to 10 CFR Part 50 ("Domestic Licensing of Production and Utilization Facilities"). At the time, the AEC assumed that SNF would be reprocessed and the residual uranium and plutonium would be recycled as fuel. The proposed Appendix D to Part 50 proposed that certain reprocessing wastes did not have to be disposed of (geologically) 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 the requirements of § 20.302 (the predecessor of today's 10 CFR Part 61) could be met. *[published in the Federal Register on June 3, 1969, Volume 34, page 8712 (34 FR 8712)]* Paragraphs 6 and 7 of the proposed Appendix D 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 land owned by the Federal or State governments. In particular, Paragraph 7 of the draft *Policy Statement* said that:

> *"… other 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 Part 50 that provided an alternative definition of HLW 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 ...." (35 FR 17533)*

Those portions of the earlier proposed *Policy Statement* concerning references to 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 (35 FR 17530). Given this policy, HLW became whatever material was left after fuel reprocessing and recycling. 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, Congress passed the Nuclear Waste Policy Act (NWPA), Public Law 97-425, which provided further clarification regarding the earlier definition of HLW. This clarification included specific legislative reference to SNF. Section 2.(12) of the act defined the term "HLW" 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 "in 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) 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…." This criterion was not included, however, in the site-specific disposal regulation for geologic disposal at Yucca Mountain, which maintained the Nuclear Waste Policy Act criterion "A" in the HLW definition (10 CFR Part 63.) Additionally, some noncommercial reprocessing facilities operated by U.S. Department of Energy (DOE) have used the term "waste incidental to reprocessing," or WIR, to determine what material could be safely disposed of *in situ*, in near-surface systems, subject to certain conditions being met. Although wastes could be stored temporarily at a reprocessing facility, permanent waste disposal would not be permitted at an NRC-licensed reprocessing facility.

The NRC staff is considering various options to classify certain types of wastes resulting from reprocessing as LLW instead of HLW. The NRC staff believes wastes that are not "highly radioactive" can be safely disposed of in a near-surface disposal facility as long as the waste streams in question could meet the requirements for disposal specified in 10 CFR Part 61. The NRC staff believes 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, requires deep geologic disposal, in contrast to those lower activity wastes that could be safely disposed in a near-surface facility that met the radioactive disposal requirements of Part 61.

NRC Staff's Suggested Approaches (Not Rank Ordered):

1. The NRC staff would seek relief from Congress. The legislative proposal would request the adoption of exceptions to the definition of HLW similar to those implemented in the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005, Public Law 108-375, which would effectively remove much of this non-highly radioactive waste from the definition of HLW.
2. The NRC staff would promulgate a regulation to clarify the meaning of "highly radioactive" and "in sufficient concentrations" in the context of high-level waste. This rulemaking would allow for the 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. No action would be taken. All highly radioactive waste streams associated with the reprocessing of SNF would be considered HLW.

Specific Questions or Topics for Which Public Feedback is Requested:

1. What waste disposal options should NRC consider for the management of waste generated by a commercial SNF reprocessing facility?

Gap 15—Waste Confidence for Reprocessing Facilities

Issue:

The NRC's Waste Confidence Rule (10 CFR § 51.23) applies only to spent fuel generated in a reactor. Under the current regulations, applicants for a reprocessing facility license would need to address the potential environmental impacts from long-term waste storage in the environmental reports submitted as part of a license application. Similarly, 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 application.

NRC Staff's Proposed Approach:

NRC 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 the long-term storage of reprocessing waste in its environmental assessment or environmental impact statement.

Alternative Approach:

NRC could expand the existing Waste Confidence Rule in 10 CFR § 51.23 to encompass the solidified high-level waste resulting from reprocessing at any facility licensed under the requirements of new 10 CFR 7X.

Rationale:

In the original 1984 Waste Confidence Rule and subsequent updates in 1990 and 2010, the NRC examined available information and determined that the safe disposal in a mined geologic repository of either spent nuclear fuel (SNF) or high level waste (HLW), including solidified HLW resulting from reprocessing is technically feasible. In addition, Section 302 of the Nuclear Waste Policy Act of 1982, as amended, establishes that DOE is authorized to enter into contracts with any domestic producer of HLW for its disposal. Section 302 also requires that a contract, or active negotiations for a contract, for disposal of HLW is entered into with the DOE before NRC can issue a license. Thus, NRC's confidence that safe disposal can occur for HLW from reprocessing is founded on both technical information and an established legal framework. NRC also established confidence that both SNF and HLW from reprocessing could be managed safely until disposal occurred. The basis for this finding was that HLW and SNF management would occur at a licensee's site, and that compliance with applicable NRC regulations and specific license conditions would provide the assurance of safety (49 FR 34680, August 31, 1984).

In contrast, only SNF was evaluated in the Waste Confidence finding to establish that safe storage could occur for at least 60 years after the licensed life of the reactor. This finding was supported by technical information derived from several decades of operating and licensing experience in nuclear power plant operation and associated SNF storage. Because storage of commercial HLW from reprocessing was not occurring in the United States, the types of information needed to support a generic finding of confidence for long-term HLW storage were not developed.

NRC 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 preclude NRC staff from recommending expansion of 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 has not licensed a commercial reprocessing facility. 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. Limited technical information is available on some HLW forms to support analyses of potential long-term processes that might affect safety. Additionally, casks used to store SNF have not undergone a licensing certification to identify technical requirements for safe storage of HLW. While no single factor precludes NRC staff from concluding that the environmental impacts from long-term storage of HLW from reprocessing are small or low, the scope and magnitude of existing knowledge gaps currently prevents NRC staff from having reasonable assurance of such a conclusion.

To meet applicable National Environmental Protection Act requirements, the NRC staff would evaluate all potential environmental impacts associated with the storage of HLW produced at commercial SNF reprocessing facility, including the post-licensed life. The NRC staff believes that the applicant's environmental report should consider all environmental impacts associated with the storage of HLW resulting from reprocessing, including the post-licensed life.

Specific Questions for Which Public Feedback is Requested:

1. Is there technical information and operating experience to demonstrate that safe long-term storage of solidified HLW from reprocessing can occur safely and with small to low environmental impacts?

2. Does the information supporting safe storage of SNF in dry casks at reactor sites bound the types of concerns associated with storage of solidified HLW from reprocessing in similar dry cask systems? In other words, if storage of SNF in dry casks is safe and secure, is storage of HLW in similar casks in the same types of storage facilities equally, more, or less, safe and secure?

Gap 16—LLW Waste Classification

Issue:

Development of Part 61 in the early 1980s was based on several assumptions regarding the types of wastes likely to be disposed of in a commercial low-level radioactive waste (LLW) disposal facility. To better understand what the likely inventory of wastes available for disposal might be, the U.S. Nuclear Regulatory Commission (NRC) conducted a survey of existing LLW generators. The survey, documented in Chapter 3 of NUREG-0782 ─ the Part 61 Draft Environmental Impact Statement (DEIS) ─ 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. Waste streams associated with U.S. Department of Energy's (DOE's) nuclear defense complex were not considered as part of the survey, since those wastes were to be disposed of at the DOE-operated sites.

The suite of 24 radionuclides were 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 tables in Tables 1 and 2 of § 61.55. Table 1 in § 61.55 provides the limiting concentrations for certain long-lived radionuclides; Table 2 in § 61.55 provides the limiting concentrations for certain short-lived radionuclides.

The licensing and operation of any commercial reprocessing facility will produce several radioactive waste streams. Gaseous effluents will be regulated under Part 20, using standards similar to EPA's NESHAP regulations. The disposal of spent nuclear fuel (SNF) and other (aqueous) high-level radioactive wastes will be subject to regulation under Part 60. The disposal of any waste streams determined to be LLW would be regulated under Part 61.

The LLW classification tables in Tables 1 and 2 of § 61.55 include many radionuclides that may be associated with reprocessing of 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.

Besides reprocessing-related LLW, other unevaluated waste streams have been identified for possible disposal in a near-surface disposal facility licensed under Part 61. They include large quantities of highly-concentrated depleted uranium, large-scale blended LLW, and possibly certain defense-related LLW streams generated by DOE.

NRC Staff's Proposed Approach:

To address the potential impact of the disposal of large quantities of depleted uranium in a Part 61 disposal facility, the Commission directed the staff to undertake a limited rulemaking that would require Part 61 licensees to conduct a site-specific analysis prior to the disposal of large quantities of depleted uranium and other unique waste streams. See SRM-SECY-08-0147. 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 facilitate the disposal of radionuclides on the basis of their risk.

Consistent with that approach, the rulemaking evolved into an analysis to evaluate low-level waste streams disposed at a Part 61 disposal facility under a performance-based, risk informed framework. The analysis would ensure that the low-level wastes streams met the Part 61 Subpart C performance objectives, and would identify any additional measures that would enhance the adequate protection of public health and safety. This proposed rulemaking is expected to be finalized in 2011.

In the LLW blending SRM (SECY-10-0043; SRM M100617B), the Commission also directed the staff to incorporate large-scale LLW blending into the limited Part 61 rulemaking. The staff recommended engaging stakeholders and soliciting their views on whether there should be amendments to the current Part 61 and if so, what the nature of those amendments should be before 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 Part 61, recommendations for specific changes to the existing rule, or suggestions for possible new approaches to commercial LLW management.

Alternative Approaches:

None

Rationale:

These approaches have been undertaken in response to earlier Commission direction.

Specific Questions or Topics for Which Public Feedback is Requested:

Staff will be seeking stakeholder comments on both the proposed rule on radionuclides not considered in Tables 1 and 2 to § 61.55 and the comprehensive revision to Part 61.

Gap 19—Effluent Controls and Monitoring

Issue:

The requirements of 10 CFR Part 70 do not sufficiently address effluent controls and monitoring for reprocessing facilities. Some requirements for effluent controls and monitoring releases from production and utilization facilities are codified in 10 CFR Part 50. Requirements for effluent controls and monitoring may be needed for reprocessing facilities because of their increased source term and greater potential for emissions.

NRC Staff's Proposed Approach:

The NRC staff is proposing to use the existing regulations in 10 CFR Part 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors", as a basis for developing specific design and operating requirements to minimize radiation exposures from radioactivity released in effluents at reprocessing and recycling facilities. 10 CFR Part 50.34a requires a licensee to describe the methods and equipment that will be used to control effluents, and stipulate the quantities of radioactive isotopes that will be released under normal operating conditions. The requirements in 10 CFR Part 50.34a are not specific to a type of technology, and thus would be consistent with a technology-neutral regulatory framework.

The NRC staff is also proposing to develop general design criteria, based upon those found in 10 CFR Part 50, Appendix A, specifically criteria # 60 (Control of Releases of Radioactive Materials to the Environment) and # 64 (Monitoring Radioactivity Releases). The NRC will need to develop additional general design criteria relating to holdup capabilities of both waste and effluents.

10 CFR Part 20, "Standards for Protection against Radiation" contains many of the ALARA (as low as is reasonably achievable) requirements. For example, 10 CFR 20.1101(b) requires "the licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable." 10 CFR 50.34a requires certain licensees to "identify the design objectives for, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the use of atomic energy in the public interest." 10 CFR 50.34a identifies that the guides set out in 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") provide numerical guidance on design objectives to meet the requirements that radioactive material in effluents released to unrestricted areas be kept ALARA. The NRC staff is considering developing similar criteria to those in 10 CFR Part 50, Appendix I for reprocessing facilities.

The Advisory Committee on Nuclear Waste and Materials (ACNWM) wrote to Chairman Klein on regulation of advanced spent nuclear fuel reprocessing and refabrication facilities (ADAMS No. ML072840119). The ACNWM recommended that NRC should hold interagency discussions with EPA on whether (i) existing release limits for Kr-85 and I-129 need to be reexamined to reflect current technology and (ii) release limits need to be established for tritium and carbon-14. Additionally, ACNWM recommended that methodologies based on concepts other than collective dose should be used as a basis for revised release limits. The NRC staff is considering how best to respond to ACNWM's recommendations on release limits.

Rationale:

Operations at reprocessing facilities will generate a larger and more varied source term, with radionuclides in potentially very mobile forms (e.g. liquids and gases), than other fuel cycle facilities and nuclear power plants. For example, in a nuclear power plant, volatile fission products are contained, to a large extent, in the fuel assembly by the fuel's cladding. In a fuel reprocessing plant, the fuel is dismantled, releasing volatile fission products to the environment of the process vessel. Therefore, reprocessing plants could release considerable quantities of radionuclides into the environment if steps are not taken to mitigate the release so that it meets 10 CFR Part 20 dose limits. Thus, regulatory requirements for reprocessing facilities must include adequate effluent controls and monitoring to ensure protection of the health and safety of people and the environment.

Alternative Approaches:

The Nuclear Energy Institute (NEI) white paper (ADAMS ML083590129) addressed the issue of effluent monitoring and control in their proposed baseline design criteria. There are many similarities between the NEI and NRC approaches. The NEI requirements are mostly derived from the equivalent requirements in 10 CFR 50.34a, and Criteria 60, 63 and 64 in 10 CFR Part 50, Appendix A. NEI provided additional requirements in their proposed framework on holdup capabilities that would be needed to control the release of gaseous and liquid effluents. NEI's proposed Criterion 13, "Control of releases of radioactive materials to the environment," addressed holdup capabilities.

NEI did not include any of the requirements in 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors" in its regulatory framework for recycling nuclear fuel. NEI's reporting requirements on effluent releases were derived from 10 CFR Part 70.

Specific Questions or Topics for Which Public Feedback is Requested:

1. Should 10 CFR Part 50, Appendix I type regulations regarding ALARA be developed?

2. Should the NRC, in coordination with the EPA, develop release limits for carbon-14 and tritium?

IV. Material Control and Accounting, Security, and Financial Considerations

Gap 8—Risk-Informing 10 CFR Part 73 and 10 CFR Part 74

Issue:

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 Part 73, "Physical Protection of Plants and Materials," and 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material," is needed to prevent unintended consequences associated with a quantity-based material categorization scheme for potential materials resulting from a reprocessing operation.

NRC Staff's Proposed Approach:

The regulatory basis for this gap will be completed on a different timeline than the rest of the gaps associated with the regulatory basis for licensing spent nuclear fuel reprocessing facilities. Because the NRC staff has received direction from the Commission on two proposed rulemakings on 10 CFR Part 73 and 10 CFR Part 74, the regulatory bases for risk-informing 10 CFR Part 73 and 10 CFR Part 74 will be included in separate regulatory basis documents supporting the potential 10 CFR Part 73 and 10 CFR Part 74 rules.

The Commission provided direction to the NRC staff in Staff Requirements Memoranda (SRM) – SECY-09-0123 – Material Categorization and Future Fuel Cycle Facility Security-Related Rulemaking, dated July 8, 2010. The Commission stated that the rulemaking should not focus on the categorization of material associated with reprocessing. The NRC staff was directed, as a separate effort, and not on the same priority of this rulemaking, to provide a detailed analysis and recommendations for the use of a material categorization approach for potential reprocessing facilities. The Commission stated that the staff should consider such analysis to be of low priority until such time that substantive plans are presented to deploy a reprocessing technology in United States.

The above SRM was a response to the NRC staff proposal, in SECY-09-0123, "Material Categorization and Future Fuel Cycle Facility Security Related Rulemaking," for a categorization framework that specifically addresses the types of nuclear material possessed by NRC licensees, both current and future. SECY-09-0123 requested permission from the Commission to pursue a revised categorization scheme for special nuclear material, which includes a material attractiveness approach, as part of an upcoming fuel cycle facility security rulemaking. The NRC staff is awaiting completion of a study by Los Alamos National Laboratory that will assist in defining the attractiveness of a number of forms of special nuclear material.

The Commission provided directions to the NRC staff in SRM-SECY-08-0059 – Rulemaking Plan: Part 74 – Material Control and Accounting of Special Nuclear Material, dated February 9, 2009. The Commission approved rulemaking limited to revising and consolidating the current material control and accounting regulations in Part 74. However, the Commission stated that the staff should consider integrating the material control and accounting proposals presented in SECY-08-0059 into the effort to develop the regulatory framework for reprocessing facilities. Two proposals in SECY-08-0059 are related to the regulatory framework for reprocessing facilities. First, SECY-08-0059 described adding a new NRC special nuclear material categorization table to 10 CFR Part 74, that would (i) modify the special nuclear material Category I – III threshold values; (ii) add subcategories for Categories I, II, and III to reflect attractiveness levels; and (iii) grade the requirements within each Category to reflect the attractiveness levels. Second, SECY-08-0059 proposed that a diversion pathway analysis would be required (this analysis is discussed in Gap 17- diversion path analysis requirements).

Because the NRC's material control and accounting regulations for special nuclear material are presently graded similarly to the nuclear material categorization scheme for physical protection regulations, and the Commission has directed NRC staff in SRM-SECY-09-0123 to consider the material categorization approach for potential reprocessing facilities to be a low priority, the regulatory bases for risk-informing of 10 CFR Part 73 and 10 CFR Part 74 are being developed, but are not anticipated to be included in the regulatory basis document developed for a potential rulemaking for licensing of reprocessing facilities.

Alternative Approaches:

No alternatives are considered because the NRC staff is implementing the Commission's directions.

Rationale:

The NRC staff is implementing the Commission's requirements in SRM-SECY-08-0059 and SRM-SECY-09-0123.

Specific Questions or Topics for Which Public Feedback is Requested:

1. What problems, if any, are created by development of the regulatory basis for risk-informing 10 CFR Part 73 and 10 CFR Part 74 separately from the regulatory basis for a potential rulemaking for licensing of reprocessing facilities?

Gap 4—Exclusion of Irradiated Fuel Reprocessing Facilities in 10 CFR 74.51

Issue:

The regulation in 10 CFR 74.51, "Nuclear Material Control and Accounting for Strategic Special Nuclear Material," currently excludes irradiated fuel reprocessing facilities from Category I material control and accounting (MC&A) requirements. A reprocessing facility would possess Category I quantities of special nuclear materials.

NRC Staff's Proposed Approach:

The NRC staff was directed by the Commission (in Staff Requirements Memoranda [SRM)-SECY-08-0059] to remove this exemption from 10 CFR Part 74 in the proposed 10 CFR Part 74 rulemaking effort (SECY-08-0059). The draft 10 CFR Part 74 rule is being prepared and this exemption will be removed.

Alternative Approaches:

As this exemption is already being removed per Commission direction, no alternative approaches are considered.

Rationale:

The NRC staff is implementing the Commission's requirements in SRM-SECY-08-0059.

Specific Questions or Topics for Which Public Feedback is Requested:

None.

Gap 17—Diversion Path Analysis Requirements

Issue:

There are no existing regulations for a diversion path analysis requirement under 10 CFR Part 74.

NRC Staff's Proposed Approach:

The NRC staff is proposing that the regulations for a reprocessing facility require that a diversion path analysis be conducted by the licensee. The diversion path analysis is a systematic process for generating, documenting, and analyzing diversion paths throughout a facility as a measure of the overall effectiveness of the safeguards system. The NRC staff will develop rule language and a guidance document to assist applicants in conducting a diversion path analysis.

Alternative Approaches:

The NRC could require other measures to mitigate the risk of loss, theft, or diversion of special nuclear materials. For example, the NRC could consider a "safeguards by design" approach in the rulemaking effort.

Rationale:

The Commission directed the NRC staff (in Staff Requirements Memoranda [SRM-SECY-08-0059, February 5, 2009]) to consider integrating the material control and accounting (MC&A) proposals presented in SECY-08-0059 into the effort to develop the regulatory framework for reprocessing facilities. In SRM-SECY-07-0126 (November 27, 2007) the Commission endorsed diversion path analysis as part of a facility's MC&A requirements in the geological repository operations area security and MC&A proposed draft rule (SECY-07-0126). Establishing diversion path analysis requirements 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.

Specific Questions or Topics for Which Public Feedback is Requested:

1. What should a diversion path analysis include?

2. Which documents should NRC staff consider in developing the rule language and guidance for conducting a diversion path analysis?

Gap 18—Approaches Toward Material Accounting Management

Issue:

10 CFR Part 74 currently lists requirements for material accounting in predefined limits and timeliness factors. Existing predefined limits could pose a challenge for reprocessing facilities due to large material throughputs, measurement uncertainties, and limitation of various measurement methods.

NRC Staff's Proposed Approach:

The NRC staff is considering modifying the inventory goal quantities and timeliness requirements for reprocessing facilities in a rulemaking for 10 CFR Part 74 specific to reprocessing. The NRC staff is also considering adding requirements with respect to a material holdup management program to facilitate more accurate accountability measurements. The NRC staff is proposing that a separate 10 CFR Part 74 rulemaking would be conducted in parallel with the proposed safety rulemaking for reprocessing.

Alternative Approaches:

If the potential licensee community provides information that indicates that new requirements would not be needed, then the NRC staff would identify in its regulatory basis that no change in the regulations would be pursued.

In addition to the proposed approach, the NRC staff could consider a "safeguards by design" approach in the potential rulemaking effort.

Rationale:

The large throughput of a commercial scale reprocessing plant, the limitations of certain measurement methods, the quantities of nuclear material holdup in process equipment, and the associated uncertainties of applicable measurement methods will likely result in a reprocessing facility having accounting limits such as inventory differences, shipper-receiver differences, and other statistical measurements that exceed current regulatory limits. The frequency of inventory and other activities may need to be adjusted to ensure timeliness goals for detecting potential loss, theft, or diversion of special nuclear materials can be met.

Specific Questions or Topics for Which Public Feedback is Requested:

1. The NRC staff is currently collecting data from foreign regulators concerning material accounting management limits for reprocessing plants. The NRC staff is seeking input from the public and potential applicants on what new requirements might be proposed related to these statistical limits.

Gap 12—Financial Protection Requirements and Indemnity Agreements (10 CFR Part 140)

Issue:

Section 170 of the Atomic Energy Act of 1954, as amended, also known as the Price Anderson Act, addresses indemnification and limitation of liability for each license issued under Section 103 of Atomic Energy Act of 1954, as amended. Because a reprocessing facility is a production facility and the NRC would license the facility under the authority provided in Section 103 of Atomic Energy Act of 1954, as amended, a reprocessing facility would be subject to Price Anderson framework. The NRC's requirements for financial protection requirements and indemnity agreements are codified in 10 CFR Part 140.

10 CFR Part 140 does not establish the required amount of private liability insurance for production facilities. The regulations do establish a specific amount of private liability insurance for other types of NRC-licensed facilities, such as reactors, plutonium processing plants, and uranium enrichment facilities. However, Section 170(b) of the Atomic Energy Act of 1954, as amended establishes the required amount of primary financial protection for production facilities as the maximum amount of liability insurance available from private sources. Furthermore, Section 170(c) states that the NRC will indemnify the license holder for the amount of liability in excess of the amount of private liability insurance obtained, up to a maximum amount of $560,000,000.

10 CFR Part 140 does not establish the fees payable for executing an indemnity agreement with the licensee of a production facility.

The appendices to 10 CFR Part 140 establish standard forms for nuclear liability policies and for indemnity agreements with the NRC. The appendices currently list standard forms for reactor facilities and for plutonium processing plants. The appendices do not include a standard form for production facilities.

NRC Staff's Proposed Approach:

The NRC staff is proposing that 10 CFR Part 140 be revised to: (i) extend the applicability of 10 CFR Part 140 to reprocessing facilities; (ii) establish the specific amount of primary liability insurance required for a production or reprocessing facilities; (iii) establish the appropriate fee for executing and issuing indemnity agreements for production or reprocessing facilities; and (iv) amend current appendices or include a new appendix to include a standard form for indemnity agreements for production or reprocessing facilities.

Alternative Approaches:

The NRC staff did not consider alternative approaches to this alternative because financial protection and liability insurance for reprocessing facilities are required by statute.

Rationale:

Section 170 of the Atomic Energy Act of 1954, as amended, also known as the Price Anderson Act, requires production facilities (and therefore also reprocessing facilities) licensed by NRC to have specific amounts of financial protection and liability insurance for nuclear accidents.

In accordance with these provisions, the holder of a license to operate a reprocessing facility is required to obtain the maximum amount of private insurance available in the private market. The NRC would then indemnify the license holder for the difference between the amount of private insurance obtained and the maximum amount of liability for this type of facility, established by statute at $560,000,000.

Specific Questions or Topics for Which Public Feedback is Requested:

None

Gap 13—Schedule of Fees (10 CFR Part 170)

Issue:

- The provisions of 10 CFR 170.2 state that the fees specified in this section are applicable to "an applicant for or holder of a production or utilization facility construction permit or operating license issued under 10 CFR part 50…"

- The staff is considering establishing a new regulation for licensing a reprocessing facility. If NRC decides to establish a new regulation for the licensing of reprocessing facilities, 10 CFR 170.2 needs to be revised to include the applicability of this new section to reprocessing facilities. Additionally, minor revisions to the fee schedules must be made to reflect this change.

NRC Staff's Proposed Approach:

The staff recommends that 10 CFR Part 170 be revised to include the applicability of this section to production or reprocessing facilities. If a new regulation for the licensing of production or reprocessing facilities is necessary and adopted, the applicability of 10 CFR Part 170 must be extended to this new or revised regulation chapter that may be applicable to reprocessing facilities.

Alternatives Approaches:

The staff did not consider any alternatives to rulemaking as proposed solutions to address this gap in the regulations. Section 6101(c)(3) of Omnibus Budget Reconciliation Act of 1990 (OBRA-90) requires NRC to "[…] establish, by rule, a schedule of charges fairly and equitably allocating the aggregate amount of charges described in paragraph (2) among licensees." Accordingly, rulemaking is the only alternative available for addressing these regulatory gaps in 10 CFR Part 170.

Rationale:

- Section 6101(c)(3) of Omnibus Budget Reconciliation Act of 1990 (OBRA-90) requires NRC to "[…] establish, by rule, a schedule of charges fairly and equitably allocating the aggregate amount of charges described in paragraph (2) among licensees." This statute also has, in recent years, required NRC to recover 90% of its budget authority through fees.

- The requested changes will ensure there is clarity in the applicability of this regulation to reprocessing facilities.

Specific Questions or Topics for Which Public Feedback is Requested:

None

Gap 14—Annual Fees (10 CFR Part 171)

Issue:

- The regulations in 10 CFR Part 171 do not currently specify annual fees for production or reprocessing facilities. In addition, the scope section of Part 171 (10 CFR 171.3) does not specifically list reprocessing or production facilities as subject to the provisions of this part.

- The staff is considering developing a new regulation (10 CFR Part 7X) to license reprocessing facilities. If NRC decides to establish such a new regulation for the licensing of reprocessing facilities, then 10 CFR 171 needs to be revised to identify that reprocessing facilities are subject to 10 CFR Part 171 and revised to include the fees for licensed reprocessing facilities.

NRC Staff's Proposed Approach:

The staff recommends that 10 CFR Part 171 be revised as follows:

- Add reprocessing facilities to the scope of 10 CFR Part 171 by revising 10 CFR 171.3 to include reprocessing facilities.

- Establish the annual fee for reprocessing facilities.

Alternative Approaches:

The staff did not consider any alternatives to rulemaking as proposed solutions to address this gap in the regulations. Section 6101(c)(3) of Omnibus Budget Reconciliation Act of 1990 (OBRA-90) requires NRC to "[…] establish, by rule, a schedule of charges fairly and equitably allocating the aggregate amount of charges described in paragraph (2) among licensees." Accordingly, rulemaking is the only alternative available for addressing these regulatory gaps in 10 CFR Part 171.

Rationale:

- Section 6101(c)(3) of Omnibus Budget Reconciliation Act of 1990 (OBRA-90) requires NRC to "[…] establish, by rule, a schedule of charges fairly and equitably allocating the aggregate amount of charges described in paragraph (2) among licensees." This statute also has, in recent years, required NRC to recover 90% of its budget authority through fees.

- The requested changes to 10 CFR Part 171 will ensure there is clarity in the applicability of 10 CFR Part 171 to reprocessing facilities.

Specific Questions or Topics for Which Public Feedback is Requested:

None

Gap 14—Annual Fees (10 CFR Part 171)

Issue:

- The regulations in 10 CFR Part 171 do not currently specify annual fees for production or reprocessing facilities. In addition, the scope section of Part 171 (10 CFR 171.3) does not specifically list reprocessing or production facilities as subject to the provisions of this part.

- The staff is considering developing a new regulation (10 CFR Part 7X) to license reprocessing facilities. If NRC decides to establish such a new regulation for the licensing of reprocessing facilities, then 10 CFR 171 needs to be revised to identify that reprocessing facilities are subject to 10 CFR Part 171 and revised to include the fees for licensed reprocessing facilities.

NRC Staff's Proposed Approach:

The staff recommends that 10 CFR Part 171 be revised as follows:

- Add reprocessing facilities to the scope of 10 CFR Part 171 by revising 10 CFR 171.3 to include reprocessing facilities.

- Establish the annual fee for reprocessing facilities.

Alternative Approaches:

The staff did not consider any alternatives to rulemaking as proposed solutions to address this gap in the regulations. Section 6101(c)(3) of Omnibus Budget Reconciliation Act of 1990 (OBRA-90) requires NRC to "[…] establish, by rule, a schedule of charges fairly and equitably allocating the aggregate amount of charges described in paragraph (2) among licensees." Accordingly, rulemaking is the only alternative available for addressing these regulatory gaps in 10 CFR Part 171.

Rationale:

- Section 6101(c)(3) of Omnibus Budget Reconciliation Act of 1990 (OBRA-90) requires NRC to "[…] establish, by rule, a schedule of charges fairly and equitably allocating the aggregate amount of charges described in paragraph (2) among licensees." This statute also has, in recent years, required NRC to recover 90% of its budget authority through fees.

- The requested changes to 10 CFR Part 171 will ensure there is clarity in the applicability of 10 CFR Part 171 to reprocessing facilities.

Specific Questions or Topics for Which Public Feedback is Requested:

None

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Attachment 1: Potentially Pertinent Guidance Documents

Reg Guide	Title	Related Gap(s)
3.14	Seismic Design Classification for *Plutonium Processing* and Fuel Fabrication Plants	1
3.17	Earthquake Instrumentation for Fuel **Reprocessing** Plants	1
3.73	Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations	1
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	1, 5, 9
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	1, 5, 9
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1, 5
1.165	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion	1, 5
1.12	Nuclear Power Plant Instrumentation for Earthquakes	1, 9
1.29	Seismic Design Classification	1, 5, 10, 9
1.61	Damping Values for Seismic Design of Nuclear Power Plants	1, 9
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants	1, 9
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions	1
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event	1
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	1, 9
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	1, 5
3.40	Design Basis Floods for Fuel **Reprocessing** Plants and for *Plutonium Processing* and Fuel Fabrication Plants	9
1.59	Design Basis Floods for Nuclear Power Plants	9
1.102	Flood Protection for Nuclear Power Plants	9
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	9, 11
3.21	Quality Assurance Requirements for Protective Coatings Applied to Fuel **Reprocessing** and to *Plutonium Processing* and Fuel Fabrication Plants	1
3.27	Nondestructive Examination of Welds in the Liners	NA

	of Concrete Barriers in Fuel **Reprocessing** Plants	
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
3.30	Selection, Application, and Inspection of Protective Coatings (Paints) for Fuel **Reprocessing** Plants	NA
3.37	Guidance for Avoiding Intergranular Corrosion and Stress Corrosion in Austenitic Stainless Steel Components of Fuel **Reprocessing** Plants	NA
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb	NA
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	NA
1.34	Control of Electroslag Weld Properties	NA
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	NA
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	NA
1.44	Control of the Processing and Use of Stainless Steel	NA
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	NA
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants	NA
1.65	Materials and Inspections for Reactor Vessel Closure Studs	NA
1.147	In service Inspection Code Case Acceptability, ASME Section XI, Division 1	NA
3.18	Confinement Barriers and Systems for Fuel **Reprocessing** Plants	9, 19
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	9
1.91	Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants	9
1.117	Tornado Design Classification	9
1.132	Site Investigations for Foundations of Nuclear Power Plants	NA
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	NA
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	5
1.175	An Approach for Plant-Specific, Risk-Informed Decision making: In service Testing	5
1.177	An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications	11, 5
DG-1227	An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications	11, 5
DG-1176	Guidance for the Assessment of Beyond Design-Basis Aircraft Impacts	9
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	5

1.204	Guidelines for Lightning Protection of Nuclear Power Plants	9
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	5, 10
3.3	Quality Assurance Program Requirements for Fuel **Reprocessing** Plants and for Plutonium Processing and Fuel Fabrication Plants	1
3.22	Periodic Testing of Fuel **Reprocessing** Plant Protection System Actuation Functions	1, 22
3.28	Welder Qualification for Welding in Areas of Limited Accessibility in Fuel **Reprocessing** Plants and in *Plutonium Processing* and Fuel Fabrication Plants	NA
1.28	Quality Assurance Program Criteria (Design and Construction)	1
1.33	Quality Assurance Program Requirements (Operation)	1
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	1, 9, 11
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	9
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)	9
1.71	Welder Qualification for Areas of Limited Accessibility	9
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	9
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	9
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	9
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants	9
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	9
1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	9
1.211	Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants	9
1.213	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	9
3.20	Process Offgas Systems for Fuel **Reprocessing** Plants	9, 19
3.33	Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel **Reprocessing** Plant	9
3.71	Nuclear Criticality Safety Standards for Fuels and Material Facilities	1, 9
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid	19

	Waste	
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
1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants Concrete Radiation Shields for Nuclear Power Plants	5, 9
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
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	19
4.1	Radiological Environmental Monitoring for Nuclear Power Plants	19, 11
4.2	Preparation of Environmental Reports for Nuclear Power Stations	19, 22
4.5	Measurements of Radionuclides in the Environment--Sampling and Analysis of Plutonium in Soil	19
4.8	Environmental Technical Specifications for Nuclear Power Plants	11, 19
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) -- Effluent Streams and the Environment	19
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
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
4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors	19
4.20 (Rev 1)	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors	19
4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	19, 9
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters	NA
8.4 (Rev 1)	Personnel Monitoring Device—Direct-Reading Pocket Dosimeters	NA
8.7	Instructions for Recording and Reporting	NA

	Occupational Radiation Exposure Data	
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable	NA
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	NA
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	NA
8.13	Instruction Concerning Prenatal Radiation Exposure	NA
8.15	Acceptable Programs for Respiratory Protection	NA
8.19	Occupational Radiation Dose Assessment in Light-Water Reactor Power Plant — Design Stage Man-Rem Estimates	5
8.20	Applications of Bioassay for I-125 and I-131	19
8.21	Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants	19
8.24	Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication	19
8.25	Air Sampling in the Workplace	NA
8.26	Applications of Bioassay for Fission and Activation Products	19
8.28	Audible-Alarm Dosimeters	NA
8.29	Instruction Concerning Risks from Occupational Radiation Exposure	NA
8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses	NA
8.35	Planned Special Exposures	NA
8.36	Radiation Dose to the Embryo/Fetus	NA
8.37	ALARA Levels for Effluents from Materials Facilities	19
3.7	Monitoring of Combustible Gases and Vapors in *Plutonium Processing* and Fuel Fabrication Plants	1, 19
3.16 (Rev 1)	General Fire Protection Guide for *Plutonium Processing* and Fuel Fabrication Plants	1
3.16	General Fire Protection Guide for *Plutonium Processing* and Fuel Fabrication Plants	1
3.38	General Fire Protection Guide for Fuel **Reprocessing** Plants (for Comment)	1
1.189	Fire Protection for Nuclear Power Plants	1, 9
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	1, 5, 9
5.44	Perimeter Intrusion Alarm Systems	8
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
5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities	8
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
5.70	Design Basis Threat (C)	8

5.8	Design Considerations for Minimizing Residual Hold-up of SNM in Drying and Fluidized-Bed Operations	18
5.23	In Situ Assay of Plutonium Residual Holdup	18
5.25	Design Considerations for Minimizing Residual Hold-up of SNM in Equipment for Wet Process Operations	18
5.37	In Situ Assay of Enriched Uranium Residual Hold-up	18
5.42	Design Considerations for Minimizing Residual Hold-up of SNM in Equipment for Dry Process Operations	18
5.80	Pressure-Sensitive And Tamper-Indicating Device Seals For Material Control And Accounting Of Special Nuclear Material	NA
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
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
5.X	This new RG is being developed for Residual Holdup. (Combining RGs 5.8, 5.25, and 5.42)	18
5.W	This new RG is being developed for Statistics. (Combining RGs 5.3, 5.18, 5.22, and 5.36)	18
5.V	This new RG is being developed for Inventory. (Combining RGs 5.13, and 5.33)	18
5.U	This new RG is being developed for Shipping, Receiving, and Transferring SNM. (Combining RGs 5.28, 5.49, and 5.57)	18
5.26	Selection of Material Balance Areas and Item Control Areas	18
5.27	Special Nuclear Material Doorway Monitors	8, 18
5.51	Management Review of Nuclear Material Control and Accounting Systems (for Comment)	18
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	1
1.184	Decommissioning of Nuclear Power Reactors	1
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	1
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	1
3.65	Standard Format and Content of Decommissioning Plans for Materials Licensees	1
3.66	Standard Format and Content of Financial Assurance Mechanisms	1
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors	1
2.6	Emergency Planning for Research and Test Reactors	1
DG-2004	Emergency Planning for Research and Test Reactors	1
DG-1237	Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors	1

3.31	Emergency Water Supply Systems for Fuel **Reprocessing** Plants	1
3.67	Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities	1
3.6	Content of Technical Specifications for Fuel **Reprocessing** Plants	11, 22
3.10	Liquid Waste Treatment System Design Guide for *Plutonium Processing* and Fuel Fabrication Plants	19, 22
3.12 (Rev 1)	General Design Guide for Ventilation Systems of *Plutonium Processing* and Fuel Fabrication Plants	19
3.12	General Design Guide for Ventilation Systems of *Plutonium Processing* and Fuel Fabrication Plants	19
3.19	Reporting of Operating Information for Fuel **Reprocessing** Plants	1, 11
3.26	Standard Format and Content of Safety Analysis Reports for Fuel **Reprocessing** Plants	7, 22
3.32	General Design Guide for Ventilation Systems for Fuel **Reprocessing** Plants	19
1.8	Qualification and Training of Personnel for Nuclear Power Plants	7
1.105	Setpoints for Safety-Related Instrumentation	9, 11
1.114	Guidance to Operators at the Controls and senior Operators in the Control Room of a Nuclear Power Unit	7
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants	7
1.149	Nuclear Power Plant Simulation Facilities For Use In Operator Training, License Examinations, And Applicant Experience Requirements	7
1.215	Guidance for ITAAC Closure Under 10 CFR Part 52.	10
1.206	Combined License Applications for Nuclear Power Plants (LWR Edition)	10

Attachment 2: List of Reprocessing Regulatory Guides

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	04/1937
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

Attachment 3: List of NUREG's 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”

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”