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Regulatory Cross-Cutting Topics for Fuel Cycle Facilities

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Risk and Reliability, Org. 6231 and Severe Accident Analysis, Org. 6232

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ABSTRACT

This report overviews crosscutting regulatory topics for nuclear fuel cycle facilities for use in the Fuel Cycle Research & Development Nuclear Fuel Cycle Evaluation and Screening study. In particular, the regulatory infrastructure and analysis capability is assessed for the following topical areas:

- Fire Regulations (i.e., how applicable are current Nuclear Regulatory Commission (NRC) and/or International Atomic Energy Agency (IAEA) fire regulations to advance fuel cycle facilities)
- Consequence Assessment (i.e., how applicable are current radionuclide transportation tools to support risk-informed regulations and Level 2 and/or 3 PRA)

While not addressed in detail, the following regulatory topic is also discussed:

- Integrated Security, Safeguard and Safety Requirement (i.e., how applicable are current Nuclear Regulatory Commission (NRC) regulations to future fuel cycle facilities which will likely be required to balance the sometimes conflicting Material Accountability, Security, and Safety requirements.)

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NOMENCLATURE

ANS	American Nuclear Society
ANSI	American National Standards Institute
ARF	Airborne Release Fraction
CFR	Code of Federal Regulations
DOE	Department of Energy
DR	Damage Ratio
FHR	Fluoride Salt Reactor
HTGR	High Temperature Gas Cooled Reactor
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
EPRI	Electric Power Research Institute
LFR	Lead Fast Reactor
LPF	Leak Path Factor
MAAP	Modular Accident Analysis Progression
MAR	Material At Risk
MELCOR	Not an acronym. NRC Severe Accident Analysis Code Supported by SNL
MC&A	Material Control and Accounting
MSR	Molten Salt Reactor
NFPA	National Fire Protection Agency
NRC	Nuclear Regulatory Agency
ORNL	Oak Ridge National Laboratory
PUREX	Plutonium and Uranium Recovery by Extraction
RF	Reparable Fraction
RG	Regulatory Guide (Reg. Guide)
RES	Office of Regulatory Research
SCWR	Super Critical Water Reactor
SDO	Standards Development Organization
SECY	Commission Papers
SFR	Sodium Fast Reactor
SNL	Sandia National Laboratories

SSC	Structures, Systems, and Components
Gen IV	Generation IV

1. INTRODUCTION

1.1. Purpose

This report overviews crosscutting regulatory activities for nuclear fuel cycle facilities for use in the Fuel Cycle Research & Development Nuclear Fuel Cycle Evaluation and Screening study [1]. In particular, the applicability of current fire regulations on an advanced fuel cycle and the ability to assign and assess safety limits are assessed.

1.2. Approach

This report limits its evaluation to reactor and reprocessing fuel cycle facilities. Discussion will focus on both the maturity of the existing regulatory infrastructure and the ability to demonstrate compliance with said structure. Due to limited resources, a surrogate modeling approach was taken to analyze crosscutting regulatory issues (i.e., regulatory issues that will apply across the fuel cycle). This approach requires that representative regulatory spheres will be analyzed in detail. The results from these detailed investigations will then be extrapolated to the greater regulatory arena. While this extrapolation does impart a large degree of uncertainty in the overall results, the value of this report will truly be found in the surrogate analysis.

This report considers the following reprocessing options:

- Aqueous Reprocessing and
- Electrochemical.

This report considers the following reactor options:

- High Temperature Gas Reactor,
- Supercritical-Water-Cooled Reactor,
- Molten-Salt Reactor (only fire regulatory implications are considered),
- Fluoride Salt Reactor (only consequence assessment implications are considered),
- Sodium-Cooled Fast Reactor, and
- Lead-Cooled Fast Reactor.

While rankings were assigned to the regulatory readiness of the various fuel cycle options, these rankings were viewed by the authors as being of secondary importance to the review provided in this report. For consistency, the authors attempted to utilize the symbolic rating method used in greater evaluation and screening study [--,-,0-,0,0+,+,++]. A more substantial regulatory review is recommended before utilizing the rating values in this report in a screening process.

1.3. Structure of the Report

The following report reviews the surrogate regulatory topics. The structure of the report will be as follows:

Chapter 1 – Introduction and Approach

Chapter 2 – Fire Regulations (i.e., how applicable are current Nuclear Regulatory Commission (NRC) and/or International Atomic Energy Agency (IAEA) fire regulations to advance fuel cycle facilities)

Chapter 3 – Consequence Assessment (i.e., how applicable are current radionuclide transportation tools to support risk-informed regulations and Level 2 and/or 3 PRA)

Chapter 4 – Overall assessment and conclusions.

Appendix A – Safeguards Security and Safety

1.4. References

1. "A Screening Method for Guiding R&D Decisions: Pilot Application to Nuclear Fuel Cycle Options," Department of Energy, http://energy.gov/sites/prod/files/4.3_DOE_NE_Screening%20Brochure_web.pdf, accessed on April 2nd, 2013.

2. FIRE REGULATION READINESS FOR ADVANCED FUEL CYCLES

In order to advance nuclear energy within the United States, existing fire regulations must be reviewed, amended, and developed for applicability towards the differing reactor and reprocessing technologies. For the existing fleet of US nuclear facilities, fire protection regulations and guidance are derived from the Code of Federal Regulations (CFR), Regulatory Guides, Commission Papers (SECY), and standards published by Standards Development Organizations (SDO). Although these various documents exist for the current fleet of nuclear facilities, they will need to be evaluated for applicability, completeness, and thoroughness in order to be implemented into the regulatory framework for advanced fuel cycles.

Before the advanced fuel cycle technologies can be commercially instituted in the US, Generation IV (Gen IV) reactors as well as reprocessing plants will have to go through a regulatory licensing process to ensure safe operation. There are regulations that are approved for the existing fleet of nuclear facilities; however, they may have varying relevance to advanced fuel cycle technologies. Although there has been a transition to more risk-informed decision making, there are areas to improve areas to improve upon the current set of data required to conduct more detailed risk analyses. The following sections will seek to review the existing regulations for applicability to Gen IV reactors and reprocessing plants.

2.1. Status of Current Fire Regulations and General Fire Hazards Associated with Fuel Cycle Technologies and Generation IV Reactors

The current fire regulations are most applicable to light water reactor (LWR) designs based on years of commercial operation experience. To provide some limited background on the regulations, the following discussion seeks to introduce applicable regulations, guidance documents, and standards for the current nuclear industry.

2.1.1. Code of Federal Regulations

The CFR annual edition is the categorization of the general and permanent rules published in the Federal Register by the departments and agencies of the Federal Government. Each title is divided into chapters which are further subdivided into parts that cover specific regulatory areas. Large parts may be subdivided into subparts and all parts are organized into sections. For the purposes of this section of the report, the following parts of Title 10 of the CFR will be reviewed in greater detail.

- Part 50 of Title 10 of the Code of Federal Regulations (written, 10 CFR 50), Domestic Licensing of Production and Utilization Facilities
- Appendix A to 10 CFR 50, General Design Criterion for Nuclear Power Plants
- Appendix R to 10 CFR 50, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

- Part 70 of Title 10 of the Code of Federal Regulations (written, 10 CFR 70), Domestic Licensing of Special Nuclear Material

2.1.2. Regulatory Guides

The Regulatory Guide (RG) series provides guidance to licensees and applicants on implementing specific parts of the NRC's regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. For the purposes of this section of the report, the following RG will be reviewed in greater detail.

- RG 1.174, Revision 1, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis
- Regulatory Guide 1.75, Revision 3 (written RG 1.75, Revision 3) Criteria for Independence of Electrical Safety Systems
- RG 1.189, Revision 2, Fire Protection for Nuclear Power Plants
- RG 1.191 Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown
- RG 1.205, Revision 1, Risk-Informed, Performance Based Fire Protection for Existing Light-Water Nuclear Power Plants
- RG 3.16, General Fire Protection Guide for Plutonium Processing and Fuel Fabrication Plants
- RG 3.38, General Fire Protection Guide for Fuel Reprocessing Plants[1]

2.1.3. Commission Papers

Commission papers (SECY) are submitted by NRC staff to the Commission to inform them about policy, rulemaking, and adjudicatory matters. These papers are publically available unless they contain information warranting protection (e.g., security related information). For the purposes of this report section, the following Commission papers may provide additional details to the necessary fire related issues for current and advanced fuel cycle initiatives while and additional list of SECY documents may be found in the Appendix.

- SECY 01-0188, Future Licensing and Inspection Readiness Assessment
- SECY 03-0047, Policy Issues Related to Licensing Non-Light-Water Reactor Designs
- SECY 03-0059, NRC’s Advanced Reactor Research Program
- SECY 05-0006, Second Status Paper on the Staff’s Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing
- SECY 06-0007, Staff Plan to Make a Risk-Informed and Performance-Based Revision to 10 CFR Part 50

[1] RG 3.38 was formally withdrawn November 14, 2008

- SECY 06-0066, Regulatory Resource Implications of a Department of Energy Spent Nuclear Fuel Recycling Program
- SECY 06-0217, Improvement to and Update of the Risk-Informed Regulation Implementation Plan
- SECY 07-0074, Update on the Improvement to the Risk-Informed Regulation Implementation Plan
- SECY 07-0081, Regulatory Options for Licensing Facilities Associated with the Global Nuclear Energy Partnership
- SECY 07-0101, Staff Recommendations Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50
- SECY 07-0167, Revision of Policy Statement on Regulation of Advanced Reactors
- SECY 08-0130, Updated Policy Statement on Regulation of Advanced Reactors
- SECY 08-0134, Regulatory Structure for Spent Fuel Reprocessing
- SECY 09-0082, Update On Reprocessing Regulatory Framework-Summary of Gap Analysis
- SECY 10-0034, Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs
- SECY 11-0156, Feasibility of Including Risk Information in Categorizing Structures, Systems, and Components as Safety-Related or Nonsafety-Related
- SECY 11-0163, Reprocessing Rulemaking: Draft Regulatory Basis and Path Forward
- SECY 12-0081, Risk-Informed Regulatory Framework for New Reactors

2.1.4. Fire Protection Standards

Technical insights are provided by SDO and developed into standards of practice for various applications. For the purposes of this section, the National Fire Protection Association (NFPA) will be discussed in greater detail as it pertains to the current status of regulations and the projected regulatory requirements. In particular, the section will focus on the following standards.

- NFPA 801, Standard for Fire Protection for Facilities Handling Radioactive Materials
- NFPA 804, Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants
- NFPA 805, Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants
- NFPA 806, Performance-Based Standard for Fire Protection for Advanced Nuclear Reactor Electric Generating Plants Change Process

2.1.5. NUREG Series Documents

The NRC governs all nuclear related activities in the US. To help address the technical complexities of regulations, NUREG documents (reports developed or sponsored by the NRC) are written to provide guidance for meeting requirements listed in the CFR. The following

NUREG documents will be discussed throughout the report as they relate to meeting fire protection requirements.

- NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants
- NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility
- NUREG-1718, Standard Review Plan for the Review of an Application for Mixed Oxide Fuel Fabrication Facility
- NUREG-1860, Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing
- NUREG/CR-6850, Fire PRA Methodology for Nuclear Power Facilities

2.1.6. Fuel Cycle Facilities

For the purposes of this work, the fuel cycle technologies discussion is focused on reprocessing. The two primary processes that will be discussed are the mature aqueous reprocessing and the developing electrochemical technologies. In general, a brief discussion will be followed by some potential fire risks. Understanding these risks will be important as licensing regulations are developed for their commercial use.

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

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

This regulation indicates that a fire protection plan must describe the overall fire protection program for the facility, identify the various positions and personnel within the licensee's organization that are responsible for the program, and outline the plans for fire protection, fire detection, and suppression capability, as well as the limitation of fire damage. Additionally, the fire protection plan must describe specific features necessary to implement the referenced fire protection program, such as: administrative controls and personnel requirements for fire prevention and manual fire suppression activities, automatic and manually operated fire detection and suppression systems, and the means to limit fire damage to Structures Systems and Components (SSCs) important to safety so that the capability to shut down the plant safely is ensured.

In an analysis found in [10], the authors had an interesting perspective on the applicability of the regulations to reprocessing facilities. “Although the NRC developed these regulations primarily to address NPPs, it is staff’s opinion that they are generically written and therefore can be viewed as establishing a minimum level of fire protection for nuclear fuel reprocessing facilities. However, as described previously, portions of these regulations are not directly applicable to nuclear fuel reprocessing, and there are several hazards specific to reprocessing that these regulations do not address”

In licensing fuel cycle facilities, requirements for fire protection are derived from 10 CFR 70, specifically Subpart H – Additional Requirements for Certain Licensee Authorized to Possess a Critical Mass of Special Nuclear Material [1] and will need to be satisfied. NUREG-1520 and NUREG-1718 provide guidance on meeting the criteria listed in the regulation. As a requirement from the CFR, an Integrated Safety Analysis (ISA) needs to be performed and details are presented in both documents. An ISA is defined as a systematic analysis used to identify facility and external hazards and their potential for initiating accident sequences, the potential for accident sequences, their likelihood and consequences, and the items relied on for safety . Currently, there are no approved NFPA standards for reprocessing facilities listed in the CFR. What follows are some brief descriptions of fuel cycle technologies along with associated fire-risk phenomena that should be investigated in greater detail in support of a performance-based regulatory structure.

Aqueous Reprocessing

Aqueous reprocessing is a method of nuclear processing that relies on the use of an acidic solution to dissolve nuclear fuel to which it is chemically processed to separate uranium and plutonium. The Plutonium URanium EXtraction (PUREX) Process is the dominant technology in this particular section based on its superior maturity. In this process, nitric acid is used to dissolve the irradiated fuel and solids are removed from the liquid solution. An organic solvent composed of 20 - 40% tributyl phosphate (TBP) in a hydrocarbon solvent, such as kerosene, is used to separate the uranium and plutonium from the other fission products [12]. The kerosene solution is then treated to separately remove the uranium and the plutonium. The fissile material may then be recycled as mixed oxide (MOX) fuel. This aqueous process is widely used in Europe.

Given the use of various acids and solvents in the aqueous reprocessing process, there are fire considerations which should be considered as a part of risk-informed decision making. As described by Robinson et al., there have been known explosion incidents when TBP and its diluents (e.g., kerosene) and nitric acid come in contact at temperatures above 120°C. Commonly referred to as red oil, the mixture can rapidly decompose when its temperature rises above 130°C. Keeping the material below this threshold is critical to maintaining stability.

Besides the creation of red oil material, the quality and proportions of solutions being mixed could possess another hazard which should be considered. During plutonium reclamation activities at both the Hanford and Savannah River Sites, solutions of hydroxylamine nitrate mixed with nitric acid resulted in an explosive exothermic reaction causing significant damage to

SSC within the general vicinity of the incident [13]. When developing a risk-informed regulatory framework, the analysis should consider accident scenarios involving 1) highly exothermic material byproducts and 2) highly exothermic reactions from improperly proportioned chemical mixtures.

Electrochemical

Electrochemical uses high-temperature electrorefining and is considered as an alternative reprocessing technology to the more common aqueous reprocessing; however, it has not been implemented on a large scale [12]. In this alternative process, molten salt electrolytes are used in the solution rather than the acidic aqueous solutions and organic solvents. The use of electrolytes supports the electrochemical separations such as uranium electrorefining and electrolytic reduction of oxide fuel. Simpson and Law indicated that although this technology is yet to be commercially viable, extensive worldwide research and development funding has been targeted towards its advancement. Continuing, they state that the use of the electrolytes allow for "more stable process liquids in the presence of high radiation fields, improved criticality safety due to the lack of neutron moderators, and waste processing that is integrated with the separations flowsheet."

The use of molten salt electrolyte provides additional challenges to fire risk assessments. Since it is a pyrophoric material, any breach of process equipment and components resulting in an electrolyte leak in air will cause a fire reaction that could impact surrounding systems. Additionally, dust may be created when the Zircaloy cladded fuel is cut and prepared for electrochemical. This dust, if it accumulates, may pose a dust explosion hazard. These fire related issues should be considered when developing risk-informed analysis.

2.1.7. Reactors

The fire at Browns Ferry Nuclear Power Plant fundamentally changed the concept of fire protection and associated regulatory requirements for nuclear power plants (NPP) in the United States. Fire protection regulations were revised to ensure plant safety by reducing the likelihood of fire and their associated consequences. Under Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50), licensees are required to maintain a fire protection plan and also ensure safe shutdown in the event of fire [2]. In general, the fire protection plan must satisfy the following criteria found in 10 CFR 50.48:

1. Describe the overall fire protection program for the facility;
2. Identify the various positions within the licensee's organization that are responsible for the program;
3. State the authorities that are delegated to each of these position to implement those responsibilities; and
4. Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage.

For units operating prior to January 1, 1979, the NRC published Appendix R which requires plants to implement design features that are intended to protect a redundant system of electrical cables and equipment necessary to safely shut down the NPP during a fire emergency. This appendix is more deterministic and prescriptive in nature and details acceptable methods for protecting these systems; however, compliance with these regulations does not uniformly enhance the level of fire safety [3]. For these instances, an exemption process exists which allows for demonstration that alternative fire protection features can provide a level of equivalent safety.

In an effort to address this concern, a more risk informed approach was necessary for addressing the level of fire safety at nuclear facilities. Effective July 16, 2004, NRC modified the fire protection regulations to allow licensees to voluntarily adopt the National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants" [4] thereby broadening the use of risk information for decision making.

Since 2002, to satisfy the NFPA 805 criteria, NRC Office of Regulatory Research (RES) and the Electric Power Research Institute (EPRI) have been developing state-of-the-art methods for conducting fire probabilistic risk assessment (PRA) at nuclear plants. The methods, detailed in EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities (NUREG/CR-6850) [17], are now the subject of training modules which focus on PRA, Electrical Analysis, Fire Analysis, Human Reliability Analysis (HRA), and Advanced Fire Modeling [5]. This effort, coupled with that of integrating NFPA 805 into the regulatory framework, clearly show the trend towards risk informed decision making in fire protection. The following sections describe some of the Gen IV reactors associated fire-risk related phenomena.

Thermal Reactors

Very-High Temperature Reactor (VHTR) or High Temperature Gas-Cooled Reactor (HGTR)

The use of a graphite moderator poses a different fire and explosion challenges than LWR designs. Generally speaking, graphite is combustible and can ignite with enough heat and air present. In the event of a breach of containment, the solid prismatic block of graphite could ignite when exposed to air, causing fission products to be dispersed as aerosols and vapors into the air/coolant mixture exiting the core region. It should be noted that the pebble bed reactor core could pose a different hazard. The production of graphite dust during normal processes could accumulate over time. If there was a disturbance of the dust in the presence of an ignition source, the aerosolized product could result in a dust explosion. Given the trend towards more risk-informed decision making, fire risk assessments on VHTR/HGTR would have to characterize these two fire scenarios and the consequences appropriately.

The VHTR/HGTR concepts are designed to produce hydrogen which itself poses a different challenge in assessing the fire and explosion risks. Hydrogen has a lower and upper explosivity limit from 4 – 75% in air concentration with minimum ignition energy of 0.01 mJ [6]. This wide

range of explosivity must be addressed in the fire risk assessment since the consequences would be dramatically different.

Supercritical-Water-Cooled Reactor (SCWR)

The supercritical-water design produces high temperatures and pressures during energy production which may affect the material performance in radiation environments. The high temperature steam corrosion issue may also lead to additional failures, such as explosive ruptures, of SSC which were otherwise unaccounted for in LWR designs. If breaks are not catastrophic in nature and only result in localized failure in associated equipment, secondary incidents could occur. As an example, energized electrical cabling can fail under thermal exposure duress.

Molten-Salt Reactor (MSR)

Molten-salt has a boiling point of 1400°C and is pyrophoric, which means it will ignite when exposed to air [8]. A breach in containment or associated SSC will result in an exposure of the molten-salt to air. In designs when the molten-salt is the fuel, products of combustion may be aerosolized with the fissile materials.

Fast Reactors

Sodium-Cooled Fast Reactor (SFR)

Fire protection measures, such as water-based sprinklers, are normally used to contain fires; however, the sodium will react with water to produce sodium hydroxide and hydrogen gas. The exothermic reaction is sufficient enough to result in hydrogen explosions. In the event of a coolant or fuel leak, firefighting intervention has to be carefully considered, and most SFRs have employed some version of inerted containment or non-reactive gas system to contain sodium leaks. As the regulations for licensing Gen IV reactors shifts to a more performance-based, risk-informed methodology, these phenomenological issues will have to be considered in order to provide reasonable assurances to the risk analysis. Recent reviews of sodium technology have shown that gaps still exist in the SFR community's ability to model sodium fires [9]. From that report:

> *For this Sodium Technology Gap Analysis, the panel experts gathered and reviewed the available pertinent information, prioritized the dominant phenomena, identified and ranked gaps in the database, and proposed approaches to close these gaps. A summary table of our gap analysis is provided in the Appendix. We summarize here is key actions as a path forward, for each of the general sodium technology phenomena areas:*
>
> • *Sodium spray and jet dynamics*
>
> • *Sodium-fluid interactions*
>
> • *Sodium-pool fire on an inert substrate*
>
> • *Aerosol dynamics*
>
> • *Sodium-Cavity interactions (Sodium-Liner, Sodium-Concrete, plus melt)*

Lead-Cooled Fast Reactor (LFR)

Although sources indicate that molten lead poses no considerable fire risk, it should be noted that coolant leaks could impact secondary combustible materials, such as energized cables, causing thermally induced failures or fires. Additionally, it has been shown experimentally that liquid water and molten lead interactions could lead to steam explosions [11]. The effects of steam explosions could impact the functionality of SSC. Given these possible concerns, risk-informed decisions would need to analyze the possible consequences associated with these accident scenarios.

2.2. Ability to Comply with Projected Regulatory Requirements

2.3.1. Fuel Cycle Facilities

In a draft document posted in ADAMS [15], NRC staff produced a critical evaluation and comparison of ISA and PRA methodologies. In this review, staff concluded that, where approved in 10 CFR 70, ISAs would adequately satisfy the regulatory requirements. However, for risk significant determination, caution should be exercises in the use of ISA results. In this paper it was acknowledged that PRA can provide more fidelity (uncertainty analysis, for example) in the final analysis, which in some circumstances is necessary or perfered. Considering the new technologies and associated fire phenomenology in reprocessing, as previously discussed, PRA methodologies may provide more appropriate quantification of overall risks in the fuel cycle facility.

The incorporation of risk-based decision analyses for fuel cycle facilities is not permissible under the current CFR. Additionally, the fire standard for fuel cycle facilities, NFPA 801, is prescriptive in nature. Specific guidelines, such as interior finish or emergency lighting, are defined clearly and are without consideration to a fire-risk methodology. Although there is no performance-based fire protection standard for reprocessing plants, one could potentially be modeled and structured off of NFPA 805; however, the new standard would still have to be incorporated into the CFR. Similar to NUREG/CR-6850, a guideline for generic reprocessing plants would be necessary to detail the process for conducting fire PRA. Given some of the phenomena discussed in previous sections, additional research should be conducted to provide more accurate risk profiles.

In a draft NUREG, RGs pertaining to fuel cycle facilities were identified and general assessments on their usability were determined based on SECY 09-0082. From Table 1, RG 3.16, 3.38, 1.189, and 1.205 were deemed unusable based on gap 1, which can be found in Table 2. RG 1.205 should not be used based on the gap presented in Table 3. Both RG 1.189 and 1.205 were not recommended for reprocessing facilities based on gap 9 which may be found in Table 4.

Table 1: Regulatory Guides That the NRC Staff Reviewed [10]

RGs	*Title*	*Gap(s)*[2]	*Decision/Action Required*
3.16	General Fire Protection Guide for Plutonium Processing and Fuel Fabrication Plants	1	Do not use; only refers to NUREG-1718.
3.38	General Fire Protection Guide for Fuel Reprocessing Plants	1	Do not use; outdated and withdrawn
1.189	Fire Protection for Nuclear Power Plants	1, 9	Do not use; too reactor specific
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	1, 5, 9	Do not use; too reactor specific

[2] Gaps reference SECY 09–0082, "Update on Reprocessing Regulatory Framework—Summary of Gap Analysis," dated May 28, 2009 (ADAMS Accession No. ML09152024)

Table 2: Gap 1, Regulatory Framework Options, Part 50 or Part 70 [16]

Gap	*Regulations currently exist to license reprocessing facilities under Title 10 of the* Code of Federal Regulations *(10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," but the regulations in 10 CFR Part 50 have always focused, for the most part, on reactors. The regulations in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," do not currently provide the necessary framework for licensing production facilities, including reprocessing facilities.*
Gap Type	Existing regulations pose a significant hindrance to effective and efficient licensing.
Basis for Gap	An application for a reprocessing facility would have to be licensed today under 10 CFR Part 50 or by order. Licensing the facility under 10 CFR Part 50 could require many exemptions, especially given the interest in pursuing one-step licensing for reprocessing facilities. The regulations in 10 CFR Part 70 apply to licensing special nuclear material (SNM), not production facilities.
Implications for Regulating and/or Licensing Reprocessing Facilities	Currently, licensing a reprocessing facility under 10 CFR Part 50 would pose a significant hindrance to effective and efficient licensing. 10 CFR Part 70, as currently written, does not provide a regulatory framework to license a production facility.
Priority for Resolution	High
Path Forward	Currently, licensing a reprocessing facility under 10 CFR Part 50 would pose a significant hindrance to effective and efficient licensing. 10 CFR Part 70, as currently written, does not provide a regulatory framework to license a production facility.

Table 3: Gap 5, Risk considerations for a production facility licensed under 10 CFR 70 [16]

Gap	*The risk assessment required by 10 CFR Part 70 involves an integrated safety analysis (ISA), and a qualitative characterization of the consequences and likelihoods of credible accident sequences. Currently, these existing requirements do not adequately address the increased risk a reprocessing facility poses relative to that of other fuel cycle facilities.*
Gap Type	Gap resulting from potentially licensing a production facility under 10 CFR Part 70.
Basis for Gap	Reprocessing facilities will have a higher source term than that of other fuel cycle facilities. The higher source term increases the relative risk of these facilities. The NRC revised 10 CFR Part 70 in 2000 based on a limited number of lower risk fuel cycle facilities, and the revision did not consider higher risk reprocessing facilities. These higher risks are not adequately addressed in the methodology established in 10 CFR Part 70.
Implications for Regulating and/or Licensing Reprocessing Facilities	The requirements for reprocessing facilities licensed under 10 CFR Part 70 will be the same as those for the lower risk fuel cycle facilities, if revisions are not made to consider the risk of these facilities.
Priority for Resolution	High
Path Forward	The staff will address the gap through the technical basis development and consider various qualitative and quantitative approaches.

Table 4: Gap 9, Baseline Design Criteria [16]

Gap	The staff will address the gap through the technical basis development and consider various qualitative and quantitative approaches.
Gap Type	Gap resulting from potentially licensing a production facility under 10 CFR Part 70.
Basis for Gap	Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides general design criteria (GDC) for nuclear power plants. Currently, no GDC are specific to reprocessing facilities. The regulations in 10 CFR Part 70 have a few BDC directed more toward lower risk fuel cycle facilities.
Implications for Regulating and/or Licensing Reprocessing Facilities	Appropriate BDC for reprocessing facilities should be established.
Priority for Resolution	High
Path Forward	The regulations in 10 CFR Parts 50, 70, and 72 and the BDC described in the NEI white paper can be used as a foundation for establishing these criteria.

2.3.2. Reactors

As discussed above, 10 CFR 50 and 10 CFR 70 both provide regulations for the licensing of domestic reactors and these may, at a minimum, provide a guideline for Gen IV reactors. Although they may be applicable, there are still fire considerations that need to be addressed further. NFPA 806 provides a performance-based methodology for assessing the risk at advanced reactors. For this to be effective, additional data may be required for the quantification of fire-risk in the new facilities. Also, as of the writing of this report, NFPA 806 was not approved in the CFRs or by NRC for use in Gen IV reactor analyses.

2.4. Overall Readiness Grade

As reflected in previous sections, numerous sources were used in the readiness assessment for both reprocessing facilities and advanced reactor technologies. The following table summarizes the readiness of CFR, RG, SDO, and NUREG documents for the licensing of advanced reactors as well as reprocessing plans. The table is further divided into the specific type of facility (i.e., Gen IV reactor or fuel cycle plant).

The general results from the analysis indicate deficiencies in licensing for each of the categories (see relative rankings in Table 5). It was found that reprocessing facilities had the least amount of available regulatory structure in place, which is not necessarily unanticipated. The historical/available literature is not entirely technology specific; however, it would appear that the electrochemical-type plants have much less available information. With respects to reprocessing, the aqueous-type facility have been explored and discussed in some great amount

of detail in NUREG documents (e.g., NUREG-1718). Although there is the NUREG document tailored to MOX facilities, documentation supporting its execution at a facility could not be found and was therefore not deemed as fully ready for implementation.

In general, an SDO standard was completed for reprocessing facilities (i.e., NFPA 801) and advanced reactors (i.e., NFPA 804 and NFPA 806). For reprocessing facilities, the standard was written as more prescriptive in nature, mirroring NFPA 804's nominal organization and structure. NFPA 806 was similarly structured to NFPA 805 and was indeed more performance-based. However, unlike NFPA 805, NFPA 806 has not gone through the same rigors of implementation throughout the US nuclear fleet.

Table 5. Fire Protection Regulatory Readiness Evaluation Metrics

	Facility Type / Guidance Type[3]	CFR	RG	SDO	NUREG	Overall
Fuel Cycle Facilities	Aqueous	0-	0-	0	0+	0
	Electrochemical	0-	0-	0	0	0-
Advanced Nuclear Reactors	High Temperature Gas Reactor	0	0	0+	0	0
	Supercritical-Water-Cooled Reactor	0	0	0+	0	0
	Molten-Salt Reactor	0	0	0+	0	0
	Sodium-Cooled Reactor	0	0	0+	0	0
	Lead-Cooled Fast Reactor	0	0	0+	0	0

2.5. References

1. "Additional Requirements for Certain licensee Authorized to Possess a Critical Mass of Special Nuclear Material," 10 CFR 70, Subpart H, 2010.
2. "Fire Protection," 10 CFR 50.48, 2007
3. "NRC's Oversight of Fire Protection at U.S. Commercial Nuclear Reactor Units Could Be Strengthened." United States Government Accountability Office, GAO-08-747, June 2008.
4. "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants." National Fire Protection Association, Standard 805, Edition 2001.
5. "2012 NRC-RES/EPRI Fire PRA Course." http://www.nrc.gov/public-involve/conference-symposia/epri-fire-pra-course/epri-fire-pra-course-info.html, US Nuclear Regulatory Commission, last update September 21, 2012.
6. Drysdale, Dougal, "An Introduction to Fire Dynamics," John Wiley & Sons, Second Edition, 1998.
7. "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, Volumes 1 and 2," NUREG-1860, December 2007.
8. "Molten Salt Reactors (MSRs)," https://inlportal.inl.gov/portal/, Idaho National Laboratory, Date accessed April 3, 2013.

[3] CFR – Code of Federal Regulations, RG – Regulatory Guides, SDO – Standards Development Organizations, NUREG - Reports or brochures on regulatory decisions, results of research, results of incident investigations, and other technical and administrative information.

9. LaChance, J. et al, “Sodium Fast Reactor Safety and Licensing Research Plant – Volume II,” Sandia National Laboratories, SAND-2012-4259, May 2012.
10. “Draft Regulatory Basis for Licensing and Regulating Reprocessing Facilities,” US Nuclear Regulatory Commission, NUREG-XXXX, November 2011.
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12. Simpson, M.F. and Law, J.D, “Nuclear Fuel Reprocessing,” Idaho National Laboratory, INL/EXT-10-17753, February 2010.
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14. Robinson, R.N., Gutowski, D.M., Yeniscavich, W., “Control of Red Oil Explosions at Defense Nuclear Facilities,” DNFSB/TECH-33, November 2003.
15. “NRC Staff Considerations on a Comparison of Integrated Safety Analysis to Probabilistic Risk Assessment,” Draft, ML102870157
16. SECY 09-0082, Update On Reprocessing Regulatory Framework-Summary of Gap Analysis
17. Nowlen, S. et al, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Summary & Overview,” US Nuclear Regulatory Commission, NUREG/CR-6850, September, 2013.

3. CONCEQUENCE ASSESSMENT READINESS FOR ADVANCED FUEL CYCLES

The ability for advanced fuel cycles to have regulatory guidance and consequence modeling capabilities is an important component in the ability to deploy an advanced fuel cycle. This section will review the existing safety regulations and the computational capabilities for various fuel cycle options to meet those regulations.

3.1. Status of Current Consequence Regulations

3.1.1. Fuel Cycle Facilities

Current regulations directly affecting releases fuel cycle facilities are 10 CFR Part 70[1], and in 10 CFR 50 Appendix F [2]. Unfortunately, 10 Part 70 does not currently provide the framework for licensing production facilities. As a result, the NRC has conducted a gap analysis (SECY-09-0082) of the current regulatory structure for licensing of production facilities [3]. Appendix F to Part 50, which specifically calls out reprocessing facilities, includes the following:

> *"Public health and safety considerations relating to licensed fuel reprocessing plants do not require that such facilities be located on land owned and controlled by the Federal Government. Such plants, including the facilities for the temporary storage of high-level radioactive wastes, may be located on privately owned property."*

Although there is wording that reprocessing facilities may be located on privately owned property, there is currently no language that sets requirements about where the facilities must be located (e.g. flood zones, earthquake zones, etc.) or the distance at which the facility must be located from a population zone.

> *"A fuel reprocessing plant's inventory of high-level liquid radioactive wastes will be limited to that produced in the prior 5 years… High-level liquid radioactive wastes shall be converted to a dry solid as required to comply with this inventory limitation, and placed in a sealed container prior to transfer to a Federal repository in a shipping cask meeting the requirements of 10 CFR part 71… All of these high-level radioactive wastes shall be transferred to a Federal Repository no later than 10 years following separation of fission products from the irradiated fuel."*

Additionally, there are no limits defined in Appendix F [2] for the capacity of a reprocessing plant. This could significantly affect the severity of accident scenarios (based on available inventory). The regulation also states that there may be two types of waste forms – liquid and dry solid waste. This could also affect postulated releases during an accident scenario.

> *"A design objective for fuel reprocessing plants shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned."*

In the following excerpt, SECY-09-0082 [3] states that many limits for fuel cycle facilities may be imposed by the US EPA, not the NRC:

> *"The U.S. Environmental Protection Agency (EPA) regulations at 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," establishes dose and effluent limits for "uranium fuel cycle" operations, which includes the "reprocessing of spent uranium fuel" (40 CFR 190.02(b), 40 CFR 190.10). These EPA dose and effluent limits could pose a challenge for reprocessing facilities. However, as these limits are established by the EPA, they are not considered in the NRC regulatory gap analysis."*

40 CFR 190.02(b) simply defines "Uranium Fuel Cycle" for the regulation, in which reprocessing of spent uranium fuel is part of the definition. 40 CFR 190.10 [4] are "Standards for normal operation" which states:

> *"The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations… The total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle, contains less than 50,000 curies of krypton-85, 5 millicuries of iodine-129, and 0.5 millicuries combined of plutonium-239 and other alpha-emitting transuranic radionuclides with half-lives greater than one year."*

40 CFR 190.11 does allow for a variance for unusual operations, although no value is defined:

> *"The regulatory agency has granted a variance based upon its determination that a temporary and unusual operating condition exists and continued operation is in the public interest."*

Important gaps discussed in SECY-09-0082 are quoted below [3]:

> *Gap 1: Existing regulations pose a significant hindrance to effective and efficient licensing. Current regulations are mainly focused on power reactors (10 CFR Part 50). Additionally, 10 CFR Part 70 "Domestic Licensing of Special Nuclear Material" does not provide the necessary framework for licensing a reprocessing plant.*
>
> *Gap 2: There is no regulation under 10 CFR Part 72 for accommodating interim (commercial independent) storage for solidified high-level waste (HLW) from reprocessing facilities.*
>
> *Gap 3: There is a lack of regulations defining waste streams resulting from spent fuel reprocessing. Examples of incidental wastes from reprocessing streams include*

HLW tank residues, chopped and leached fuel hulls, irradiated fuel hardware, reprocessing facility equipment, personnel protection equipment, etc.

Gap 4: 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. It is necessary to remove this exclusion to ensure the security of material in any proposed Category I reprocessing facility.

Gap 5: Gap resulting from potentially licensing a production facility under 10 CFR Part 70. Reprocessing facilities will have a higher source term than other fuel cycle facilities, which increases the relative risk of the facility. The main issue/argument is that consequences at a reprocessing facility are somewhere between those of a fuel cycle facility and a reactor facility.

Gap 6: There are currently no definitions for the terms "reprocessing", "recycling", and "vitrification".

Gap 7: There are needs to develop criteria for "operators' licenses," for reprocessing plants.

Gap 8: The current quantity-based categorization scheme in the existing regulations may pose an undue regulatory burden on operating a reprocessing facility. Risk-informing 10 CFR Part 73 and 10 CFR Part 74 is needed to prevent unintended consequences associated with a quantity-based material categorization scheme for potential materials resulting from a reprocessing operation.

Gap 9: Existing baseline design criteria in 10 CFR Part 70 do not comprehensively address hazards posed by the operation of a reprocessing facility. The baseline design criteria cover five main categories: overall requirements, radiological protection, chemical and hazardous materials protection, equipment services protection, and facility confinement protection.

Gap 10: Clarity is needed in 10 CFR Part 70 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 license, the AEA, and the Commission's rules and regulations.

Gap 11: 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. Technical specifications control important processes and protect against the uncontrolled release of radioactivity. They include safety limits, limiting safety system settings, and limiting control settings.

Gap 12: Price Anderson protection and indemnity fees and amounts for reprocessing facilities are currently not included in 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements." Additionally, several appendices to 10 CFR Part 140 do not include forms for reprocessing facilities.

Gap 13: The scope of 10 CFR Part 170, "Fees for Facilities, Materials, Import and Export Licenses, and Other Regulatory Services under the Atomic Energy Act of

1954, as Amended," does not include a production facility licensed outside 10 CFR Part 50.

Gap 14: The annual fees for licenses issued are established in 10 CFR Part 171, but the regulation sets no such annual fees for licenses of reprocessing facilities, and these facilities are not included in the scope of the regulation.

Gap 15: The waste confidence decision published in the Federal Register on August 31, 1984 (49 FR 34658) discusses waste from reprocessing facilities in the first and third finding. The generic waste confidence rule in § 51.23 applies only to waste from reactor facilities. The staff will need to develop a technical basis to expand the waste confidence rule to encompass waste from a reprocessing facility.

Gap 16: The tables in 10 CFR 61.55, "Waste Classification," do not include all reprocessing-related radionuclides. As a result, some waste streams may be considered Class A but may not be generally acceptable for near surface disposal.

Gap 17: There are no existing regulations for a diversion path analysis requirement under 10 CFR Part 74. Establishing diversion path analysis requirements would make 10 CFR Part 74 more risk-informed and would provide an effective detection and response program to mitigate potential safeguards vulnerabilities and system weaknesses.

Gap 18: Approaches to meet the timeliness and goal quantities for material inventory accounting will be addressed for changes and/or improvements. Modern technology has been developed or is being developed to help reprocessing facilities meet the existing timeliness and quantity goals. Additionally, incorporating a material holdup management program requirement into 10 CFR Part 74 to minimize the impact of material holdup could facilitate more accurate inventory accounting.

Gap 19: 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.

Gap 20: Existing regulations do not address security risks for other fissile material that can be separated in more advanced fuel cycle separations. Certain fissile elements such as americium (Am), neptunium (Np), and others are currently not regulated or treated as other fissile or SNM material.

Gap 21: The current generic fuel cycle rule provides environmental impact data associated with both an open and closed fuel cycle for uranium only. The data provided in Tables S-3 and S-4 (10 CFR 51.51) do not include environmental impacts associated with recycle of plutonium, neptunium, or other actinides in mixed-oxide fuel, for example.

Gap 22: Different licensing options (e.g., one- and two-step) may necessitate additional requirements than those currently in 10 CFR Part 70. The regulation in 10 CFR 52.79 outlines the technical information required in applications for a combined operating license. In accordance with 10 CFR 52.77, this information is in addition to the requirements for these applications given in 10 CFR 50.33.

Gap 23: The illustrative list of reprocessing plant components found in Appendix I to 10 CFR Part 110 is focused mainly on aqueous methods. The list does not include equipment related to electrochemical or other related reprocessing facilities.

3.1.2. Reactors

All nuclear reactors will have to comply with regulatory requirements, but currently the regulatory structure is only in place to license LWRs (10 CFR 50 and 10 CFR 52). Even with the current regulatory structure, demonstrating the safety of a given LWR design is an extremely slow and complicated process. For example, the Advanced Passive 1000 (AP1000) submitted its license application to the NRC in March of 2002 under Part 52 and did not receive its final design certification until December of 2011, a delay of over nine years [5]. Supporters of advanced reactors, which are not explicitly supported in the regulatory structure, will likely need to invest significant effort to create the supporting regulatory infrastructure long before the design is ready to be submitted for licensing.

If a reactor designer approached the U.S. NRC to license an advanced reactor, the designer would have to demonstrate that they meet all appropriate codes and standards to ensure the safety of the public. Unfortunately, while some historical advanced reactor standards were published, most previous attempts to develop advanced reactor codes and standards were abandoned when the research programs were terminated.

Sodium Fast Reactors

For example, the ANS and ANSI are currently working on an update of the 1989 SFR industry consensus standard ANSI/ANS 54.1 - 1989, *General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant* [6] [7]. The 1989 standard was intended to provide the SFR equivalent of the LWR-specific general design criteria (GDC) that are promulgated in the U.S. NRC regulation 10 CFR 50 Appendix A. This current updated draft standard recognizes changes in SFR technology, safety, and the use of PRA in achieving a risk-informed and performance-based approach to the safe design of a reactor. In addition to the providing the a framework to risk-inform the safety of the SFR, the update to ANS/ANSI 54.1 provides SFR specific GDCs which can supplement and or replace LWR specific GDCS in 10 CFR 50 Appendix A.

High Temperature Gas Reactors

Other advanced reactor standards have been examined as well. The ANS and ANSI completed a Helium-cooled reactor industry consensus standard ANSI/ANS 53.1 - 2011, *Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants* [8]. ANS/ANSI 53.1focused on a risk-based approach to safety, classifying events requirements through a frequency/consequence curve (see Figure 1). This approach has met resistance by the ACRS [9] who would rather see a

risk-informed approach to event classification which allows for additional regulatory discretion in event categorization.

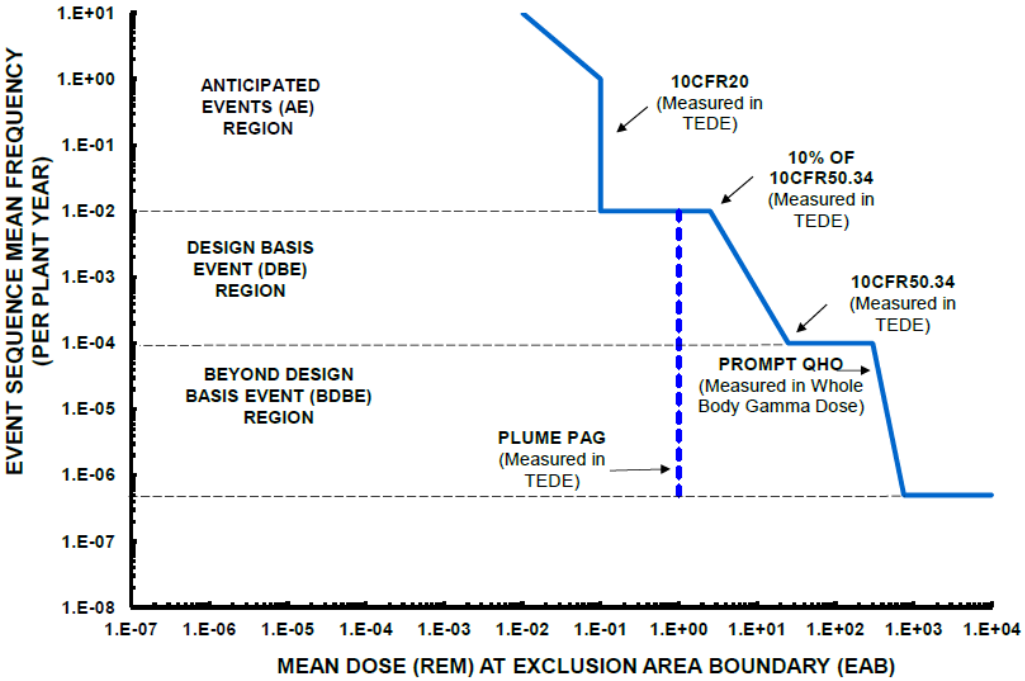

Figure 1– Next Generation Nuclear Power Frequency-Consequence Curve[8]

Fluoride Salt Reactors

The ANS and ANSI are also developing a Fluoride Salt-cooled High Temperature Reactor (FHR) industry consensus standard ANSI/ANS 20.1, *Nuclear Safety Design Criteria for Fluoride Salt-Cooled High-Temperature Reactor Nuclear Power Plants* [10]. ANS/ANSI 20.1 defines nuclear safety criteria and design requirements to enables users to design, license, construct and operate new facilities using this technology. In addition to the providing the a framework to risk-inform the safety of the FHR, ANS/ANSI follows ANSI/ANS 54.1's lead in defining FHR specific GDCs which can supplement and or replace LWR specific GDCs in 10 CFR 50 Appendix A.

Closing Remarks

A new DOE/NRC cooperative effort has been started to continue the technology specific GDC work started by the ANS/ANSI working groups [11]. Led out of Oak Ridge National Laboratory (ORNL) and Idaho National Laboratory (INL), with support from DOE and NRC, this new effort will develop generic GDC (derived from Appendix A of 10 CFR 50) and develop technology-specific GDC for at least one reactor type (TBD) to supplement the generic GDC for compliance with § § 50.34.,52.47 and 52.79. Once the generic GDCs have been created, outside experts will review the work and provide additional suggestions. This effort will greatly expedite new reactor concepts develop technology specific GDCs, as can be seen through Figure 2.

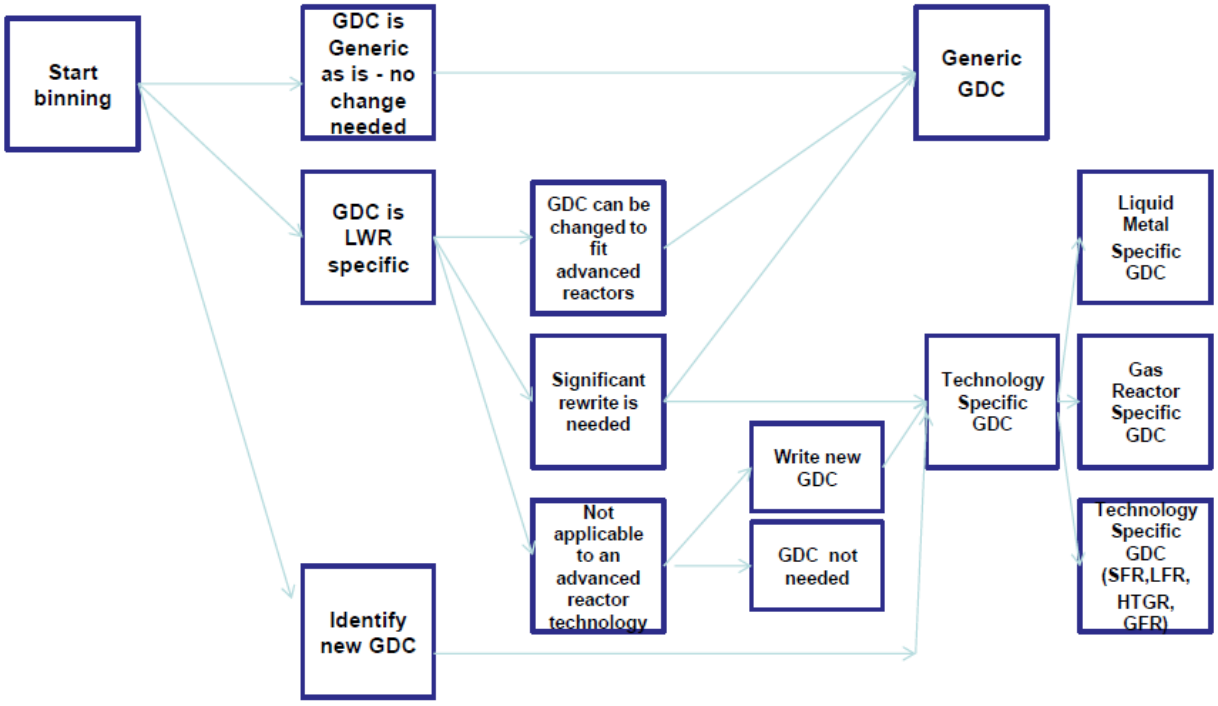

Figure 2– Categorization of existing GDCs [11]

3.2. Ability to Comply with Projected Regulatory Requirements

3.2.1. Fuel Cycle Facilities

Spent fuel reprocessing is a backend fuel cycle option for many countries today. In the U.S., there is not a single spent fuel reprocessing facility operating today. However, the U.S. had two facilities built in the mid 1960 to early 1970. The Atomic Energy Commission, AEC (the predecessor of Nuclear Regulatory Commission, NRC) used 10 CFR Part 50 to grant an operating license for the Nuclear Fuel Services reprocessing facility at West Valley, New York in 1966, and a construction permit for the Barnwell Nuclear Fuel Plant (BNFP) Separation Facility in 1970 [21]. Only the West Valley facility operated for a period of time. The BNFP was built but was never operated.

Over forty years since the West Valley facility was built and operated, the current regulation still requires that spent fuel reprocessing facility must comply with 10 CFR Part 50 to obtain a construction or operating license [21]. 10 CFR Part 50.34 which is to ensure adequate protection of public health and safety requires that applicants must demonstrate their designs meeting general design criteria and mitigating a postulated set of accidents known as “design-basis” accidents to within certain specified radiological release limits. This release is often referred to as “source term.”

To close this 40+ year gap since last licensed reprocessing facility, NRC must be prepared to review the applicant’s submittal on the accident calculations and conduct independent safety calculations. During this 40+ year gap, there were a number of accidents, ranging from minor to extensive structural damage and large radioactive contamination at the reprocessing facilities in both domestically and aboard [12, 17, 19 and 21]. The domestic accidents were mostly from the

defense reprocessing facilities rather than from the commercial reprocessing facilities. The challenge for the NRC is to formulate the regulation effectively by understanding why these accidents happen and how these accidents can be prevented and mitigated. The ability to estimate the radioactive release out of the facility is necessary for NRC to determine that the source term is within the limit identified in the regulation and to grant construction and operating licenses.

The focus in this section is to discuss the current method to deal with the modeling of the source term from a spent fuel reprocessing facility during a postulated accident as a part of consequence determination. Although there is no operating reprocessing facility in the U.S., there are many operating reprocessing facilities in the other parts of the world, such as in China, France, Great Britain, India, Japan, Russia and South Korea. The types of the reprocessing technologies varies internationally. For example, South Korea prefers the electrochemical (pyro-chemical or electrochemical) extraction technology [18, 20]. Although the concept of the electrochemical technology has been demonstrated successfully in the laboratory environment, the demonstration in the commercial scale is yet to be done.

In terms of the maturity of the reprocessing technologies, the aqueous extraction technology is well established [16, 18, 20 and 21]. In fact, many commercial spent fuel reprocessing facilities in the world are based on the aqueous extraction technology. Many countries such as France and Japan focus on advance aqueous extraction cycles that allow more refined separation of fission product types to reduce waste volumes. Additions of the extra extraction steps increase accident frequencies. Many of these advanced cycle steps are either new or do not have significant operating histories, which make it more difficult to establish modeling capability. The safety of the current state of the art of the aqueous extraction is due to years of operating experience and the lessons learned from the past accidents. Hence these safety improvements allow aqueous extraction to be ranked higher than the electrochemical technology, except for those advance aqueous extraction cycles previously described. The lessons learned allow regulators to impose stringent measures to the facility operators and designers in order to prevent and minimize such accidents from happening, and enable safety analysts to benchmark safety models.

In this section, quantification of the readiness of the modeling capabilities to deal with any potential accidents that can occur in a spent fuel reprocessing facility is discussed. These accidents are usually ranked by the magnitude of the energetic event in the accident. For example, explosions yield significant energies in a very short time interval (in milliseconds or less). The explosion may be caused by the runaway chemical reaction [13] or overpressure due to external heat. An explosion usually creates a blast wave and high temperatures that can cause significant damage to structures and vaporize materials. Another important energy yield is due to fire. The power associated with a fire can be high (from kilowatt to megawatt levels), but the duration is relatively long (minutes to hours) in comparison to the explosions and nuclear excursion. Both chemical explosions and fires can accompany a significant amount of by-product gases and solids which can cause significant pressure build-up and contamination within the facility. Primary fires and explosions can induce secondary fires and/or explosions. Although the power of the nuclear excursion can be large (kilowatts to megawatts) due to the number of fissions generated, the lack of large amount of product gases makes this accident type lower than that of explosion and fire. However, the direct radiation that nuclear excursion created may impact significantly to the workers rather than public. The potential for a critical driven excursion is higher when fissile solutions are use, due to density effect that allows several

excursions (criticality pulses) to occur [19]. The duration of nuclear excursion is in seconds to minutes. Lastly, the spill event is ranked the lowest in terms of the consequence to the public. Worker impact may be significant, depending on the materials involved.

The ability to model realistically for the consequence associated with these types of accidents is important as previously discussed. The use of the modeling approach from the nuclear power plant may not be appropriate for the spent fuel reprocessing facility because the chemical conditions and facility constructions are very different between the power plant and reprocessing facility. Power plant uses the containment concept which can withstand designed based accident conditions and external events. Reprocessing facility uses the confinement concept, which will not withstand certain energetic accidents, to prevent and mitigate accident conditions (see Figure 3). Traditionally at the Department of Energy (DOE) sites, the consequence evaluation used in DOE safety basis community to determine the safety of a nuclear facility is based on the graded and conservative approach to grant the operation or construction. Many of these facilities include waste storage, waste, plutonium and uranium processing, and nuclear explosive operations. Some of the functionalities of these facilities could be found in the spent fuel reprocessing facilities that NRC could license in the future.

DOE safety basis utilizes the five factor formula (MAR × DR × ARF × RF × LPF) to compute the source term (ST) [14]. The terms in the formula is explained as follows. Once primary confinement is breached (see Figure 3), the radioactive material (material-at-risk, MAR) is released, which can be represented by the fraction of MAR to be released as DR (damage ratio). Once releases, it can be subjected to many physical release pathways (both physical leakage, such as opening and forced convection leakage, such as ventilation and energetic event, such as fire, explosion and external wind condition). ARF represents the fraction of the release that is airborne. If the release is not airborne, it is likely to remain inside the facility and is not included in ST. RF represents the fraction of the release is respirable (aerosol size < 10 μm). For health effect, the release needs to be respirable. Leak path factor (LPF) represents the release that gets out of the facility. All DR, ARF, RF and LPF depend strongly on the accident types. If an explosion that causes all confinements to fail (see Figure 3), LPF is likely to be unity. Thus the consequence is very high. If a spill occurs, the value of DR, ARF, RF and LPF may be very small, since most of the release is localized within secondary or tertiary confinement. With active ventilation on where filtration system is functioning, LPF may be zero.

The challenge is to estimate the ST that requires the understanding of accident progression from the time when the accident initiated to the time when the release is occurred out of the facility. Thus a source term tool needs to be developed. Currently, DOE safety analysts utilize values of ARF and RF provided from a DOE handbook [14], an assumed aerosol size distribution and uses a safety system code (such as MELCOR) to estimate the LPF [15]. Although this handbook provides ARF and RF values for various accident types, these values were empirically determined from laboratory experiments and simplified approach in the 1970s and 1980s. For explosions and fires, the extrapolation from the laboratory experiments to the actual accidents in a reprocessing facility may not be appropriate. If a large scale experiment is not possible, the use of computation fluid dynamic (CFD) approach to estimate these ARF and RF, and aerosol size distribution for explosions and fires may come close to reality. The use of the past accidents for the current aqueous technology as benchmark points, this approach becomes more promising. On the other hand, both advance aqueous extraction and electrochemical technologies require

additional effort to allow this approach to become useful, because of the lacking maturity of the technology.

The readiness of the current method to estimate the ST is discussed below for each of the four major accident types.

Explosion – This accident requires a better modeling approach to capture the release profile of the radioactive materials. In addition, the use of toxic and highly combustible chemicals in the aqueous extraction makes explosion accidents are high probable events. Pryo-processing may not use toxic combustible chemicals as much, but the high temperature and the oxidation of metal possibility in the separation process may increase as a high probable event. The CFD approach previously discussed needs additional studies to establish better values for the ARF and RF, including the aerosol size distribution. The structural damage and leak path inside the facility may require safety analysts to provide explosion calculations using existing blast computer codes. With these modeling efforts, the use of the SSC, which is considered mature in terms of the system modeling capability, would enable the handling of the ST for explosions with some assumptions.

Fire – Similar to explosion above, CFD approach for fires needs additional studies. The use of the fire safety codes provides the details of the fire, energy profile and fire duration. With these modeling efforts, the use of the SSC, which is considered mature in terms of the system modeling capability, would enable the handling of the ST for fires.

Nuclear excursion (Criticality) – This accident is more likely occurred in aqueous extraction process than in the electrochemical. In the aqueous extraction, the presence of both fissile and neutron moderators (such as hydrocarbon and water) increase the potential of the inadvertent criticality. The electrochemical extraction does not contain water or hydrogen moderators. Once a criticality accident occurs, the number of fissions provides the energy of the accident to determine the failure of the vessel or container housed the fissionable materials. If it did, the ARF and RF, including the aerosol size distribution values may require CFD simulation to be performed. Similar to both explosion and fire, the use of a SSC could estimate the ST for criticality. However, if the release can be contained in a room and ventilation is functioning, the ST may be very small.

Spills – This accident type is usually associated with corrosion of the vessel or container that leads to a spill. The release from the primary confinement barrier is short and long duration depending on when the spill is detected. The ability to model this accident type by using a SSC is adequate.

Based on the discussions above, a ranking table (see Table 6) has been generated for each accident type and for the extraction technologies. This ranking does not take into account other accident initiators such as external events, or natural phenomena. The aqueous extraction has been separated into the current technology and advance technology. Both explosions and fire carry the same ranking based on the discussions above. Criticality only applies to the aqueous extraction technology. Because the advance aqueous and electrochemical technologies are relative new, the maturity of these technologies may be still some distances away. It is believed that spill accidents can be handled well with the current modeling capability. Both explosion and fire may need additional refinement in the modeling capability. Criticality in terms of the solution criticality, there is well established methodology to handle these types of events in the nuclear criticality safety engineer arena in the DOE nuclear facilities and commercial nuclear

community. Some improvements are still needed to estimate the source term if the criticality event is sufficient to induce airborne materials to be contributing.

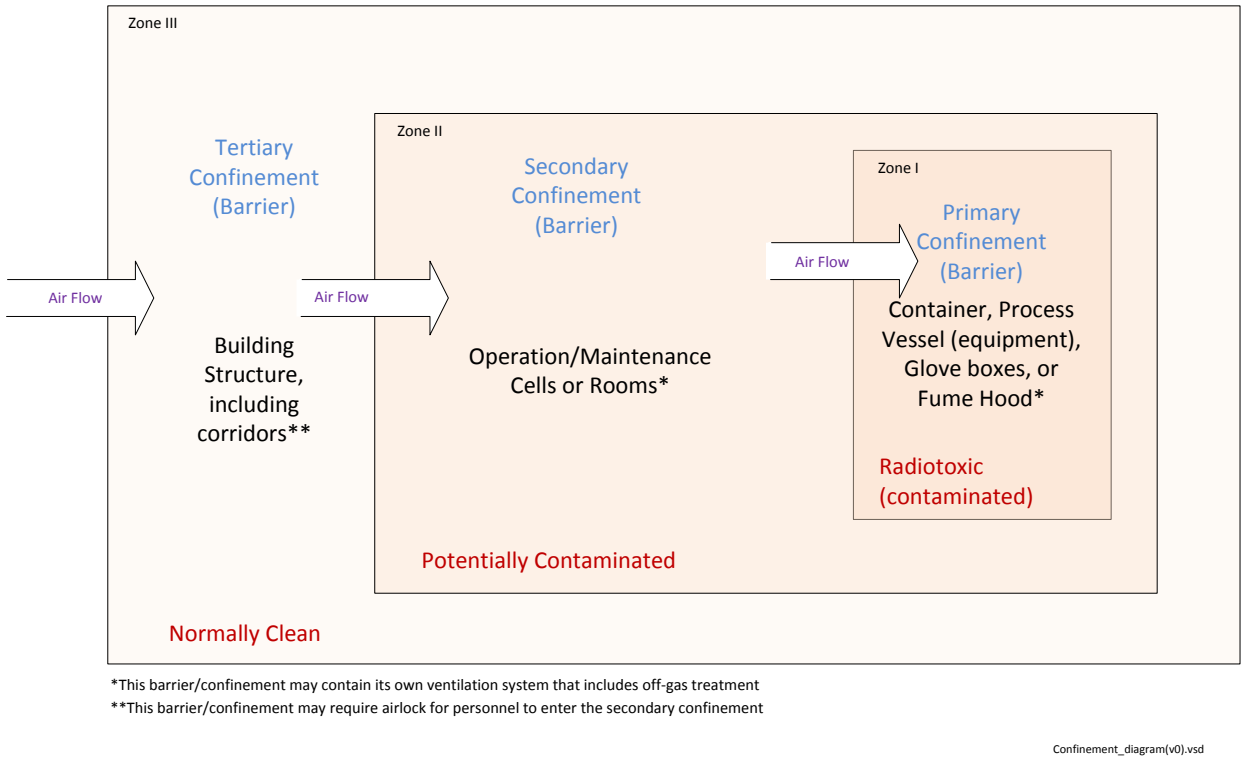

Figure 3. Typical Confinement Barriers at a Spent Fuel Reprocessing Facility

Table 6. Ranking of the Modeling Capabilities for Various Accident Types

Extraction Cycle/Accident Type	*Explosion*	*Fire*	*Criticality*	*Spill*	*Overall*
Aqueous					
Existing Technology	0-	0-	0+	+	0
Advance Technology	-	-	0+	+	-0
Electrochemical (electrochemical)	-	-	N/A	0+	-0

3.2.2. Reactors

In order to license a reactor design, the licensee is required to demonstrate that they meet the regulatory dose requirements [22]. In Light Water Reactor space, the industry has developed its own severe accident analysis code (Modular Accident Analysis Progression, MAAP) while the NRC has informed their severe accident planning through codes such as MELCOR [**Error! eference source not found.**]. This section will review the current severe accident modeling capabilities for advanced reactors in the United States.

Sodium Fast Reactors

SFRs have long sought to be licensed in the U.S. The SAS4a/SASSYS-1 computer code, and its predecessors, has been the primary best-estimate severe accident computer code supported to supplement the licensing case of SFRs. Recent reviews have shown that:

> *It was clear from this activity that in the United States the SAS4A/SASSYS-1 code system would be a central tool used in the analysis of a large majority of the scenarios considered here, and that it was generally assessed as adequate to support these activities for licensing [23].*

Unfortunately, SAS4a is not applicable for modeling containment response, sodium fires, or radionuclide release. Furthermore, the same gap analysis concluded that:

> *There is currently no maintained capability for modeling of the containment response and estimation of the corresponding source term. The extension of capabilities under a severe accident code such as MELCOR for SFRs will be important to support any licensing effort. The passive safety of current SFR designs may not avoid the need for a containment code.*

To that end, the DOE has initiated a multi-year program to incorporate sodium fire and containment modeling capabilities currently residing in the dormant code CONTAIN/LMR [25] into MELCOR computer code. It is expected that by 2016, incorporation of sodium equations of state and fire models will be integrated in the officially supported version of MELCOR.

Gas Reactors

In order to support HTGR licensing, industry developed reactor specific fission product transport codes to predict radionuclide releases from pebble bed and prismatic fuel block cores. While these codes are likely sufficient to begin the licensing process, the these tools are not yet as robust as MAAP in their ability to model various reactor designs (i.e., MAAP can model multiple Boiling and Pressurized Water Reactors, while most HTGR codes are built to model only a single HTGR design) [26].

To support NRC verification efforts, the capability to model HTGRs in MELCOR was developed by SNL from 2009 to 2011, and although the thermal-hydraulic modeling additions are complete, some work remains to fully implement the all of the important severe accident phenomena for HTGRs into MELCOR [27,28]. However, MELCOR can currently model HTGR thermal-hydraulics, core heatup and oxidation (with air-ingress), steady state buildup and transport of fission products with nominal reactor operation, and the release of radionuclides from failed TRistructural-ISOtropic (TRISO) fuel particles.

3.3. Overall Readiness Grade

3.3.1. Fuel Cycle Facilities

The following table estimates our readiness level to support the differing reprocessing options. The recycle option most widely used is the PUREX method. Plutonium and Uranium Recovery by EXtraction (PUREX) is an aqueous reprocessing method (Aqueous – Pu Separation in the table below) that separates plutonium and uranium from the rest of the minor actinides from spent nuclear fuel. This would be the most attractive reprocessing option from a development

and deployment point of view, however it poses some challenges in regards to proliferation. From a safety standpoint, both aqueous reprocessing options pose a criticality risk due to the solvents used in the process.

The electrochemical options are high-temperature methods that use molten salts and molten metals as solvents. These options would probably require more development before they could be deployed. These options are mostly beneficial for use with fast breeder reactor technology.

All reprocessing options would require significant regulatory work. The overall readiness level of reprocessing options can be found in Table 7.

Table 7 – Overall Readiness Level for Consequence Assessment of Reprocessing Facilities

Reprocessing Option	***Readiness Level***
Aqueous – Pu Separation	0
Aqueous – Minor Actinide Separation	0
Electrochemical – Pu Separation	0-
Electrochemical – Minor Actinide Separation	0-

3.3.2. Reactors

While water reactors are the more advanced in standardizing safety requirements and consequences modeling, much work has been conducted in readying sodium fast reactors and gas cooled reactors for licensing. From the reviews described in Section 3.1.2 and 3.2.2, Table 8 displays following rankings were assigned to the consequence assessment licensing readiness of various reactor types.

Table 8 – Overall Readiness Level for Consequence Assessment of Reprocessing Facilities

Facility Type / Guidance Type	***Safety Standards***	***Overall***
HTGR	0+	0+
Supercritical-Water-Cooled Reactor	0--	0
Fluoride-Salt Reactor	0+	0
Molten Salt	0--	-
Sodium-Cooled Reactor	0+	0+
Lead-Cooled Fast Reactor	0--	0-

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4. CONCLUSIONS

Deploying a new fuel cycle will involve many regulatory challenges. Some of these challenges, such as fire and consequence management, have historically been experience by the existing nuclear fuel cycle and thus the existing regulatory framework can be examined for compatibility. Some portions of the fuel cycle, such as reprocessing of used nuclear fuel, has only been conducted to a limited extent by the DOE complex (and often not using the exact same approaches proposed for an advanced fuel cycle) and thus judgment is needed to determine what new challenges would be faced by private entities. Additionally, some regulatory challenges will not be identified until the new fuel cycle is deployed.

This small regulatory cross-cut study was tailored to offer an examination into select topical regulatory areas that would be encountered by reactors and reprocessing facilities in a new fuel cycle. Fire regulations and consequence assessment were analyzed in detail in Chapters 2 and 3, while integrated safeguard security and safety was briefly examined and the tentative findings can be found in Appendix A.

4.1. Fire Regulation Readiness for Advanced Fuel Cycles

As described in Chapter 2, numerous sources were used in the readiness assessment for both reprocessing facilities and advanced reactor technologies. The final rankings summarize the readiness of CFR, RG, SDO, and NUREG documents for the licensing of advanced reactors as well as reprocessing plans.

The general results from the analysis indicate deficiencies in licensing for each of the categories. It was found that reprocessing facilities had the least amount of available regulatory structure in place, which is not completely surprising. Historical literature is not entirely technology specific; however, it would appear that the electrochemical-type plants have much less available information. With respects to reprocessing, the aqueous-type facility have been explored and discussed in some great amount of detail in NUREG documents (e.g., NUREG-1718). Although NUREG-1718 that is tailored to MOX facilities, documentation supporting its execution at a facility could not be found and was therefore not deemed as fully ready for implementation.

In general, an SDO standard was completed for reprocessing facilities (i.e., NFPA 801) and advanced reactors (i.e., NFPA 804 and NFPA 806). For reprocessing facilities, the standard was written as more prescriptive in nature structured to NFPA 805 and was indeed more performance-based. However, unlike NFPA 805, NFPA 806 has not gone through the same rigors of implementation throughout the US nuclear fleet, mirroring NFPA 804's nominal organization and structure.

4.2. Consequence Assessment Readiness for Advanced Fuel Cycles

Chapter 3 examines the readiness of our regulatory infrastructure to licenses reactors and fuel cycle facilities in an advanced fuel cycle.

The reprocessing ranking in Chapter 3 estimates our readiness level to support the differing reprocessing options. The recycle option most widely used is the PUREX method. PUREX is an

aqueous reprocessing method (Aqueous – Pu Separation in the table below) that separates plutonium and uranium from the rest of the minor actinides from spent nuclear fuel. This would be the most attractive reprocessing option from a development and deployment point of view, however it poses some challenges in regards to proliferation. From a safety standpoint, both aqueous reprocessing options pose a criticality risk due to the solvents used in the process. The electrochemical options are high-temperature methods that use molten salts and molten metals as solvents. These options would probably require more development in consequence model development before they could be licensed. All reprocessing options would require significant regulatory modifications before they could be licensed.

The reactor rankings in Chapter 3 examine the regulatory infrastructure (including the existence of ANS/ANSI safety standards) and the existence of consequence analysis codes to support a license application. In general Sodium, Fluoride-Salt, and Gas reactors gained high marks due to the existence of ANS/ANSI Safety Standards in either completed or work draft stages. Gas reactors completed their safety design standard in 2011, but did not incorporate technology specific GDCs into that standard. These technology specific GDCs appear be the focus of DOE and NRC efforts to improve the license-ability of advanced reactors. Both the Sodium and Fluoride-Salt draft safety design standards include technology specific GDCs.

The capability to model radioactivity release from Sodium and Gas-cooled reactors are currently the most mature when compared to the other advanced reactor concepts. Both reactor types have developed industry radionuclide release estimates and have (or are) developed modeling capabilities in the MELCOR computer code. While more work will be needed to refine the radionuclide release models, the initial work in this area will greatly reduce licensing uncertainty if these reactor types are eventually chosen.

4.3. Compilation of Advanced Fuel Cycle Metrics

After combining the information discussed in Chapter 2, Chapter 3, and Appendix A, overall rankings were compiled for each technology type. It should be reiterated that these rankings are by no means complete and should only be viewed as the author's opinions at the end of the report. As in a PRA, the true value of this report comes from the analysis, not the final results. A more substantial regulatory review is recommended before utilizing the rating values in this report in a screening process.

Table 9 – Overall Regulatory Fuel Cycle Rankings

	Facility Type	***Overall***
Fuel Cycle Facilities	Aqueous	0
	Electrochemical	0-
Advanced Nuclear Reactors	High Temperature Gas Reactor	0+
	Supercritical-Water-Cooled Reactor	0
	Molten Salt	0-
	Fluoride-Salt Reactor	0
	Gas-Cooled Fast Reactor	0-
	Sodium-Cooled Reactor	0+
	Lead-Cooled Fast Reactor	0-

4.4. Final Remarks

As expected, implementation of advanced fuel cycles will require significant investment in advancing the regulatory structure. Due to prior experience, Aqueous processing with SFR and/or HTGRs would be the easiest designs to license.

Regulatory and Licensing issues will be one of the primary obstacles faced by a transitioning fuel cycle. If the DOE is to promote a new generation of nuclear power production, significant effort is needed to:

- Transition Standards from prescriptive to risk-informed and performance-based to allow for greater applicability
- Develop tools (codes and analysis methods) which can convince a regulator to accept a fuel cycle change.
- Probabilistic Risk Assessment is the decision support tool currently being utilized by the NRC to help evaluate proposed changes plant designs.

A more substantial regulatory review is recommended before utilizing the rating values in this report in a screening process.

APPENDIX A - BRIEF DISCUSSION OF INTEGRATED SECURITY, SAFEGUARD AND SAFETY REQUIREMENTS

While some effort was devoted to studying cross-cutting issues associated with integrated security, safeguard and safety requirements, the underlying analysis was not complete enough to warrant its own chapter. Thus, a summary of the security, safeguard and safety (3S) review is provide here.

This section provides background on the regulations that are relevant to integrated security, safeguard, and safety of fuel cycle facilities and advanced reactor designs. NRC has documented a draft regulatory basis for reprocessing [1] – this proposes a new Part of 10 CFR (Code of Federal Regulations) for regulating reprocessing. However, the current framework does not discuss 3S.

Note: Large amount of regulations developed for security or safeguards or safety. However, integration of these concepts is relatively new.

A.1 Code of Federal Regulations and Other Guidance

The CFR annual edition is the categorization of the general and permanent rules published in the Federal Register by the departments and agencies of the Federal Government. Each title is divided into chapters which are further subdivided into parts that cover specific regulatory areas. Large parts may be subdivided into subparts and all parts are organized into sections. As can be seen from Table 10, each part of the CFR handles one (sometimes two) of the components to Security, Safeguards, and Safety. Future advanced fuel cycles will likely require adaptations to the current regulatory structure to account for the tradeoffs and synergies provided by the 3Ss.

Table 10 – CFRs as they apply to Security, Safeguards, and Safety.

CFR	*Security*	*Safeguard*	*Safety*
10 CFR 20			X
10 CFR 50	X (part34, 54, 150)		X
10 CFR 52	X		X
10 CFR 70		X	
10 CFR 73	X		
10 CFR 74		X	
10 CFR 75		X	

Many additional documents provide guidance to licensees on acceptable ways to meet regulations. These guidance documents into the Standard Review Plan [1], Regulatory Guides, Interim Staff Guidance, and NUREG reports. However, due to the volume of documents, it is not possible to review these documents within the scope of this report.

Ref. [2] provides a comprehensive discussion of the applicability of current regulatory requirements and additional sub-tier guidance for Advanced Small Modular Reactors, including both Light Water Reactor (LWR) and Liquid Metal Reactor (LMR) designs. The authors recommend completing a similar, comprehensive activity for other fuel cycle facilities.

A.2 Fuel Cycle Facilities

The radiation protection requirements in 10 CFR 20 "Standards for Protection Against Radiation" apply to all types of facilities involved in the nuclear fuel cycle. The requirements are technology neutral.

The Material Control & Accounting (MC&A) requirements of 10 CFR 74"Material control and accounting of special nuclear material" apply to any facility that possesses more than one gram of contained 235U, 233U, or Pu. These requirements are applicable to all NRC facilities except for certain independent storage facilities. The MC&A requirements appear generically applicable to advanced reactor designs, but the NRC has stated that HTGR and LMR fuels require further evaluation of the requirements due to the expected higher fuel enrichment. From [1]:

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

A.3 Preliminary Conclusions

Current regulations treat the problems of safety, security, and safeguards as separate regulatory issues. Having different regulations and organizations responsible for the three aspects introduces regulatory complexity and masks the synergies and conflicts between the three areas. The IAEA is responsible for safeguards worldwide, but there is no worldwide agency that governs safety and security of nuclear facilities.

Fuel cycle facilities regulations are more advanced in the safeguards arena, but are less robust in safety. Reactors regulations are more advanced in safety, but less in safeguards. Security is probably still evolving for both.

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