ORNL/TM-2022/2671

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September 2022

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Nuclear Energy and Fuel Cycle Division

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managed by
UT-BATTELLE LLC
for the
US DEPARTMENT OF ENERGY
under contract DE-AC05-00OR22725

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ABBREVIATIONS

AEA	Atomic Energy Act
AGR	advanced gas-cooled reactor
ALARA	as low as reasonably achievable
ANSTO	Australian Nuclear Science and Technology Organization
AOO	anticipated operational occurrence
ARDC	advanced reactor design criteria
ASTM	American Society for Testing and Materials
BDBA	beyond design basis accident
BDBE	beyond design basis event
BLEU	Blended Low Enriched Uranium
CANDU	Canada Deuterium Uranium
CDC	complementary design criteria
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
COL	combined license
CP	construction permit
DBA	design basis accident
DC	design criterion
DG	Draft Guide
DOE	US Department of Energy
DSF	dry storage facility
DSS	dry storage system
EAB	exclusion area boundary
EIS	environmental impact statement
EP	emergency planning
EPA	US Environmental Protection Agency
EPZ	emergency planning zone
FP	fission product
FSF	fundamental safety function
GCD	general design criteria
GTCC	greater than Class C
HLW	high-level waste
IE	initiating event
IROFS	item relied on for safety
ISA	integrated safety analysis
ISFSI	independent spent fuel storage installation
ISG	interim staff guidance
IU	irradiation unit
LFTR	lithium fluoride thorium reactor
LLRW	low-level radioactive waste
LTA	lead test assembly
LWR	light-water reactor
MC&A	material control and accounting
MCFR	molten chloride fast reactor
MOX	mixed oxide
MRS	monitored retrievable storage installation
MSBR	molten salt breeder reactor
MSR	molten salt reactor

MSRE	Molten Salt Reactor Experiment
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NFPA	National Fire Protection Association
NFS	Nuclear Fuel Services
NMAC	nuclear material accountability and control
NRC	US Nuclear Regulatory Commission
OL	operating license
ONT	other new technology
ORNL	Oak Ridge National Laboratory
PAG	protective action guideline
PB	performance-based
PDC	principal design criteria
PRA	probabilistic risk assessment
PSAR	preliminary safety evaluation report
PWR	pressurized water reactor
QA	quality assurance
RFDC	required functional design criteria
RG	Regulatory Guide
RISC	Risk-Informed Safety Class
RPF	radioisotope production facility
SAFDL	specified acceptable fuel design limit
SBO	station blackout
SMR	small modular reactor
SNF	spent nuclear fuel
SNM	special nuclear material
SRP	standard review plan
SSC	structure, system, and component
SSR	Stable Salt Reactor
TVA	Tennessee Valley Authority

ABSTRACT

Molten salt reactor (MSR) sites may include additional elements of the nuclear fuel cycle beyond those of the existing fleet. In the existing fleet, the individual elements of the fuel cycle typically have been located on different sites and licensed separately. Providing robust separation between hazards remains a useful safety practice for MSRs. Although the different elements of the fuel cycle at a nuclear site that includes MSRs may transfer material between processes more frequently than prior practices, providing adequate separation between distinct facilities avoids the potential for adverse interactions. Additionally, some elements of the MSR fuel cycle, such as fuel salt preparation or waste stabilization, may be more efficient to share among multiple nearby reactors, and nuclear sites that include MSRs may also include other reactor classes. Hence, discrete MSR fuel cycle facilities located at a common site could be physically separated with robust barriers—albeit potentially connected by piping—and licensed individually. This report describes the hazards of individual elements of representative MSR fuel cycle facilities, including their relationship to overall site level hazards. The report maps the regulatory compliance aspects of the individual MSR fuel cycle elements (e.g., fuel salt preparation, reactor, waste stabilization) to existing and developing regulations, as well as describes current and developing site-level regulations from an MSR perspective.

1. INTRODUCTION

Existing regulations are tailored to efficiently evaluate the safety issues raised by the once-through fuel cycle implemented with solid fuel rods at large, light-water reactors (LWRs). Multiple presumptions, which are embedded into the regulatory process, regarding how fuel will be fabricated, used, and stored do not match the characteristics of molten salt reactors (MSRs), which will incorporate additional on-site elements of the fuel cycle beyond those of large LWRs. These may include fuel salt preparation, online fuel conditioning, and stabilization of used fuel salt and waste materials. Lacking an efficient and effective regulatory process could significantly increase MSR deployment cost and time without a corresponding safety benefit. Additionally, MSRs are only a portion of a fuel cycle system and will exist alongside other elements. An efficient regulatory process would enable an evaluation of risks from multiple nuclear facilities located on one site to be integrated, including clear requirements for a site license and individual facility licenses.

The particular ensemble of fuel cycle elements Implemented at any MSR is an important aspect of plant design. Additionally, MSR designs tend to be sufficiently flexible that owners may elect to change fuel cycle elements over the course of the plant life. Consequently, an efficient, flexible regulatory process that includes guidance on the required information needed to reach a regulatory conclusion about adequate safety and provides clear lines of review authority would minimize the regulatory portion of the investment risk for MSR deployment.

One crucial regulatory alignment issue is the division of regulations that govern the use of special nuclear material (SNM) to generate power (Title 10 of the *Code of Federal Regulations* [CFR] Part 50, "Domestic Licensing of Production and Utilization Facilities") from those governing performing chemical processes on SNM (10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"). Although this regulatory division aligns with the current LWR fleet and fuel cycle facilities, the separation does not align well with the characteristics of liquid salt–fueled plants and fuel cycle facilities. LWR fuel is fabricated at separate, dedicated fuel fabrication facilities. The regulatory separation matches the physical facility separation. Additionally, used LWR fuel does not require chemical stabilization for adequately safe, long-term storage but does require liquid cooling for several years.

MSR fuel cycle elements that lack a direct analogy to current practices for solid fuel also lack an efficient regulatory process. Laws and regulations governing the processing of used fuel reflect the characteristics of solid fuel rods. For example, the Atomic Energy Act (AEA) of 1954 [4], as amended, Section 123 a (7) indicates that prior approval is required for any international cooperation that includes changing the form or content of irradiated nuclear material. Because allowing fuel salt to solidify changes its form and either adding cementing agents or removing halogens changes its content, both interim storage and waste stabilization of used fuel salt would require prior authorization for international cooperation. Increasing the cost and time for US MSR designers to gain permission to access internationally developed fuel cycle technologies or for US developers to partner with allies to develop tailored fuel cycle technologies could delay MSR deployment and add unnecessary cost. Efficiently applied regulations would accelerate the necessary approvals for beneficial international cooperation on MSR fuel cycle activities.

The review of an MSR's fuel cycle by different parts of the US Nuclear Regulatory Commission (NRC) may generate conflicts that arise from the multiple safety assessments and review methods for the same MSR structures, systems, and components (SSCs). For instance, the additional fuel cycle activities anticipated to be performed at MSR plant sites involve hazards, but these are primarily chemical hazards. The recommended method for evaluating fuel cycle facility chemical hazards employs process hazard assessment, as described in NUREG-1513, *Integrated Safety Analysis Guidance Document Format and Content*, whereas reactor safety adequacy review originates from compliance with principal design criteria (PDC) required by 10 CFR 50 and 10 CFR 52[1]. The adequacy review is described for LWRs in NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition* [5], and probabilistic risk assessment (PRA). Effective regulations and guidance provide assurance that all identified hazards do not constitute an unacceptable risk to the public or the environment.

MSRs are a broad and diverse reactor class. The challenge of developing effective and efficient regulation is increased by the wide design and operational variance in MSRs. Relevant differences in risks between MSRs include the following.

- Fuel salt might be synthesized on-site from varying feedstocks.
- Fuel salt might be synthesized into a concentrate or dilute form.
- Fission product (FP) separations will occur during operation. Both solid and gaseous FPs will inherently separate from the liquid fuel salt, but some designs will reincorporate daughter products (e.g., ^{137}Cs resulting from ^{137}Xe decay) into their fuel salt, and others will process them as a separated waste stream.
- Some designs might employ additional chemical separations to remove neutron absorbers or chemical contaminants.
- Some designs might include a breeder blanket as part of fuel salt synthesis. Different designs would incorporate different types of separations from the breeding blanket. Separations could include extracting fissile or fissile precursor materials from the breeding blanket.
- Most designs must implement waste stabilization to enable long-term fuel salt storage due to radiolysis following salt cooling. Waste stabilization requirements are significantly affected by design details, such as choice of halide (fluoride, chloride, or mixed) and salt power density. The continued storage rule requires the ability to safely store used fuel on-site indefinitely following plant shutdown (NUREG-2157, *Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel* [32]).

This report describes fuel cycle features that are potentially relevant at MSRs and identifies the pertinent set of regulations for each, emphasizing the issues arising from colocation. The report then develops the technical bases for integrated MSR and fuel cycle facility safety evaluation reflective of the hazards of each fuel cycle feature. The goal of the report is to provide the technical information needed to develop an effective and efficient regulatory process reflective of the risks of MSRs, including their fuel cycle. The envisioned process would be effective because it provides sufficient evidence to reach a conclusion of reasonable assurance of adequate protection of people and environment. The process would also be efficient because it provides clear expectations of the evidence needed to support regulatory conclusions.

Dose at the site boundary under both normal and accident conditions is the regulatory concept underlying the safety analysis. The potential dose from one MSR at an isolated site would inadequately represent deployments that include multiple facilities on one site. One potential deployment scenario would be a single-fuel salt waste stabilization facility located on a site along with several MSRs. Each MSR would individually contribute to the potential site dose along with the single-waste salt preparation facility. Multiple MSRs could also be located at an existing LWR site, potentially with a fuel salt preparation facility. MSRs and other associated fuel cycle facilities would not generally be vulnerable to loss of common services because most designs do not depend on external services (e.g., electrical power, cooling

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[1] As noted in 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, "under the provisions of 10 CFR 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. Under the provisions of 10 CFR 52.47, 52.79, 52.137, and 52.157, an application for a design certification, combined license, design approval, or manufacturing license, respectively, must include the principal design criteria for a proposed facility."

water) to achieve adequate safety. However, plants located on a common site can be often exposed to external events. For example, ashfall from a fire could foul heat rejection surfaces from multiple units simultaneously. If the applicant seeks a collective license for all collocated MSR fuel cycle facilities and such a license is granted by the NRC, the site license and overall environmental impact analysis must integrate potential releases from each individual facility on-site. Issues arising from integrating risks from multiple, colocated nuclear (i.e., potentially non-MSR related) facilities are largely site specific and thus are beyond the scope of the current analysis.

Liquid-fueled MSR facilities are expected to be significantly different from previously licensed nuclear fuel and reactor facilities by the NRC. The types of radiological systems and where fissile material is present vary significantly by MSR concept. Therefore, it is impractical to attempt to identify one preferred or obvious licensing pathway for all liquid-fueled MSR concepts. However, the treatment of the hazards and regulatory requirements strongly depends on the adopted licensing pathway. An overview of some potential options to facilitate discussion on the regulatory analysis of the fuel cycle–related hazards is presented in Figure 1.

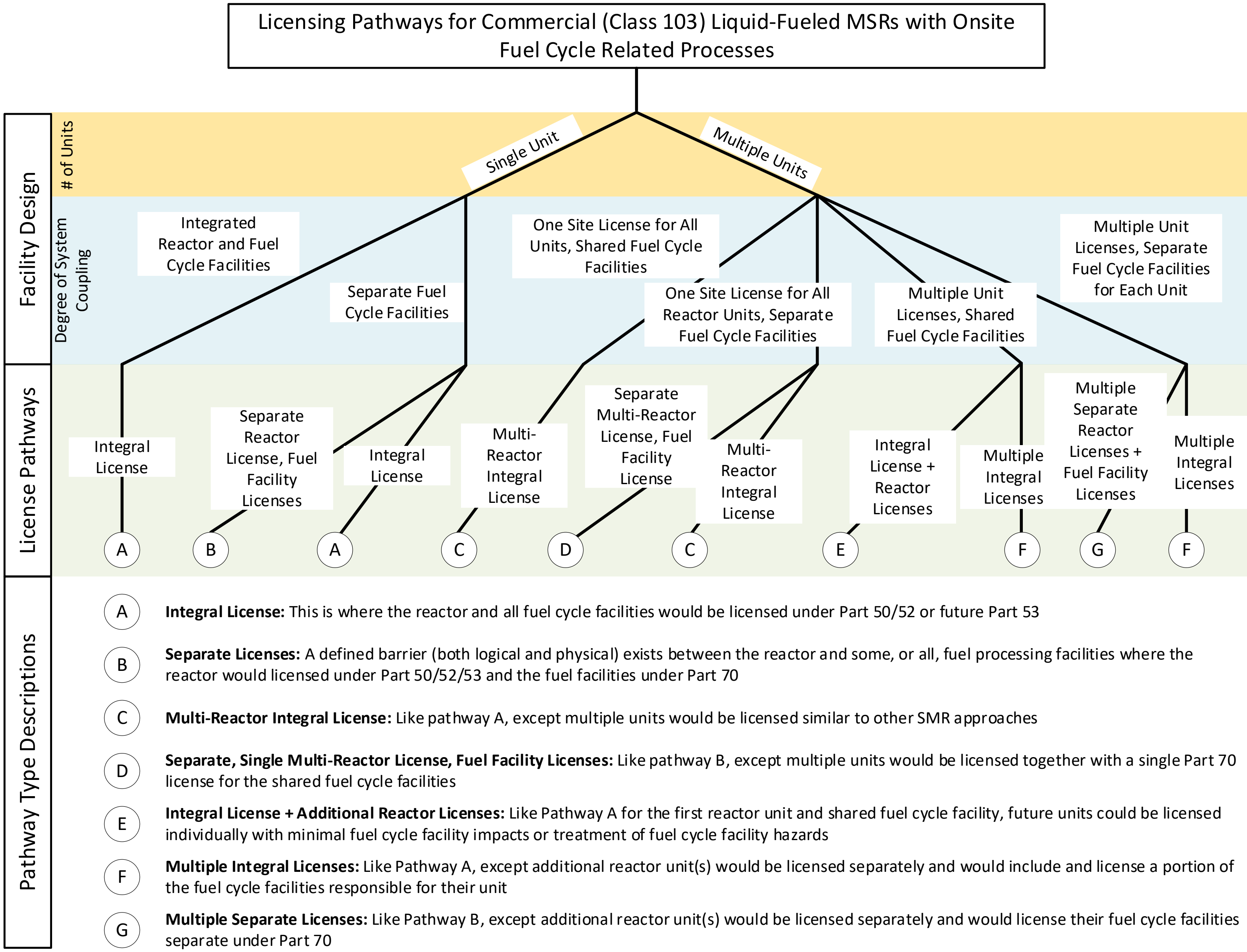

Figure 1. Licensing pathways for various liquid-fueled MSR facility designs.

In Figure 1, for one unit, the licensing pathway decision depends on the degree of system coupling between the reactor and fuel cycle facilities. For systems with a defined logical and physical barrier between the facilities, the benefits of licensing the fuel cycle facilities separate from the reactor include:

- simplification of reactor safety analysis,
- avoidance of fuel cycle SSC safety function and safety classifications for reactor safety,
- avoidance of technical specifications and limiting conditions of operation of fuel cycle SSCs with respect to reactor safety,
- fuel cycle facility operations and trained operator qualifications could be simplified and divorced from reactor operator training and qualifications, and
- inspections, testing, and maintenance activities are not tied to reactor safety and can be developed in accordance with 10 CFR Part 70 requirements and safe operation of the fuel cycle facilities.

However, achieving this licensing structure could be challenging, even for systems with a high degree of independence. Any postulated fuel cycle facility accident or transient should not affect reactor operation, or place the reactor in a state in which an accident is more likely or the consequence of an accident is heightened beyond anticipated or nominal values. Even for the aforementioned items listed previously, it is unclear the degree to which they would present an unallowable economic or other burden on the utility/owner. Licensing separate systems also has technical challenges. These include:

- demonstrating a strong technical basis for the logical and physical separation between the reactor and fuel cycle facilities,
- developing a separate integrated safety analysis (ISA) for the fuel cycle facility and license application in which some duplication may be necessary,
- operating and maintaining two separate facilities where the site and total facility design may require some duplication (e.g., control rooms, electrical distribution areas), and,
- needing potential additional interfacing system components (e.g., two pumps may then be necessary—one for the reactor and one for the fuel cycle facility—instead of just one pump).

Ultimately, the developer and utility must decide on the licensing pathway for MSRs with one or multiple integrated fuel facilities. The intent of Figure 1 is to visualize the various possible combinations and options for licensing these types of facilities.

For any of the licensing pathways shown, following 10 CFR Part 50; 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants”; or a future 10 CFR Part 53, “Risk Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors,” could be demonstrated successfully. No inherent roadblocks are identified for any option. Pathways A and F are similar because each reactor unit and fuel cycle facility would be licensed together. Similarly, pathways B and G are similar because each reactor unit and fuel cycle facility would be licensed separately. Pathway C is a variant on pathway A but with multiple reactor units being licensed together with a fuel cycle facility. Pathway D is a variant on pathway B in which the reactor license application includes multiple units but still keeps the fuel cycle facilities separate. Pathway E is similar to pathway A for the first reactor unit, then subsequent reactor units would be licensed separately (e.g., pathway B).

The choice between 10 CFR Parts 50, 52, and 53 is a complex decision that involves many different factors and stakeholders. Multi-unit sites may lead toward preferring a design certification approach. However, industry opinions vary significantly between organizations. As shown in Figure 1, only pathway E is less likely to favor a design certification approach for the first unit because the first unit would be an integral license with additional units, not including the fuel facility features. However, after completing initial licensing following pathway E, additional units could follow a design certification approach, adopting the licensing experience from the first unit. In any pathway in which additional units are licensed separately (i.e., pathways E, F, or G), a hybrid approach could be followed in which a construction permit (CP) and an operating license (OL) are issued separately for the initial reactor unit (10 CFR Part 50 approach) followed by the issuance of a combined license (COL) for any additional units adopting the licensing experience of the first unit (10 CFR Part 52 approach). However, without relying upon a design certification, issues arising in predecessor reviews would be re-evaluated with the information that exists at the time of the new filing.

In addition to the reactor and fuel cycle facility licensing options, siting options may also exist. Siting requirements are specified in 10 CFR 100, "Reactor Site Criteria," and NRC guidance is published in Regulatory Guide (RG) 4.7. Additionally, siting guidance for advanced reactors is changing to have more flexibility in terms of allowable nearby population centers and population density. SECY-20-0045, *Population Related Siting Considerations for Advanced Reactors*, describes some of the options and recommendations for changing advanced reactor siting requirements relative to population density. Recently, the Commission approved option 3 of SECY-20-0045, which allows for a risk-informed calculation of doses to be included in RG 4.7 for advanced non-LWRs. Figure 2 defines various siting requirements and guidance relative to dose and population density.

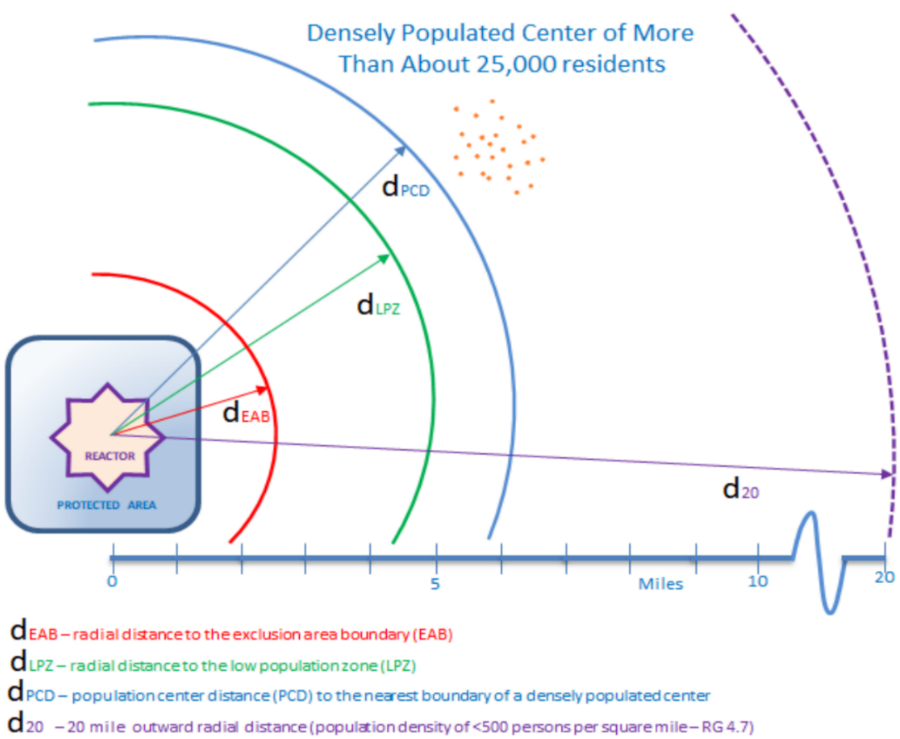

Figure 2. Visualization of distances associated with population density from 10 CFR 100 and RG 4.7.

One challenge when licensing the fuel cycle facility independently from the reactor is in evaluating accident consequences for siting and limiting emergency planning (EP) actions. In the case of a radiological emergency, which could initiate from any sources on-site, protective action guidelines (PAGs) are specified by the US Environmental Protection Agency (EPA) [1]. The desire of many advanced reactor developers is to avoid needing to develop protective actions for the public. A proposed rule change published in the Federal Register, 85 FR 28436, for Small Modular Reactors and Other New Technologies places a limit on the greatest accidental release or dose consequence to the public at the exclusion area boundary (EAB) or closest distance to the EAB (<1 rem over four days) The rule change is expected to be finalized in January, 2023.

Once a preferred site is selected, environmental reviews are largely unaffected by the decision to license one or more reactors independently from their fuel cycle facilities. Typically, the content of an environmental impact statement (EIS) follows the following outline.

1. Introduction (e.g., purpose and need, alternatives, report contents)
2. Affected environment (e.g., site, water, ecology, socioeconomics, environmental justice)
3. Site layout and plant description
4. Construction impacts
5. Operational impacts
6. Fuel cycle, transportation, and decommissioning
7. Cumulative impacts
8. Need for power
9. Environmental impacts of alternatives, including the no-build option alternative
10. Conclusions (e.g., impacts including those unavoidable impacts, cost benefits, recommendation)
11. References
12. Index

Both operational transients and potential accidents, design basis accidents (DBAs) and beyond DBAs (BDBAs), are considered in the section on potential radiological impacts during operation. This implies that both fuel cycle and reactor events must be considered. However, the reactor and fuel cycle facilities are licensed, regardless of whether any services are shared, and an evaluation of potential accident consequences of BDBAs may be expected.

For MSRs, some environmental benefit may exist with fuel cycle facilities being colocated with the reactor. Normally, fresh U is mined, converted, enriched, and then fabricated all at separate sites. For Th-conversion MSRs, this lessens the need for U enrichment facilities. This should be presented as a benefit in the comparison with other nuclear alternatives. Much of the remainder of the environmental assessment and siting aspects would be similar for MSRs when compared to other nuclear power designs.

An overview of licensing requirements as they might apply to MSRs is discussed in Section 2, and other regulatory compliance considerations are discussed in Section 3. A more complete discussion of these regulations is provided in Appendix A.

2. LICENSES AND CERTIFICATES

The NRC licenses all commercial nuclear power plants that produce electricity or provide other energy services in the United States [2]. Currently, nuclear power plants can be licensed under 10 CFR Parts 50 or 52. A 10 CFR Part 53 license process is under development for advanced non-LWRs. Nuclear power plants licensed under 10 CFR Part 50 undergo a two-step licensing process that first grants a CP and, after further review, an OL. An alternative licensing process is available under 10 CFR Part 52 that combines a CP and an OL, with certain conditions, into one license known as a COL [3]. 10 CFR Part 70 regulates the licensing required to receive title to, own, acquire, deliver, receive, possess, use, and transfer SNM. However, a separate 10 CFR Part 70 license is not required for possession or use of SNM for the operation of a nuclear reactor licensed under 10 CFR Parts 50 or 52 (10 CFR 70.22[b]). Clustered facilities supporting an MSR or a fleet of MSRs at a collocated site may need a separate 10 CFR Part 70 license. Additionally, any MSR facility contemplating the sale, distribution, or recycling of byproduct material resulting from expanded fuel cycle activities at its must consider the licensing requirements found in 10 CFR Part 30.

This section summarizes the various portions of Title 10 of the CFR that may be applicable to an OL for an MSR technology that may also intend to incorporate some portion of the frontend or backend of the traditional fuel cycle as part of normal reactor site management. Such facilities are referred to in this section as *broad-spectrum fuel cycle MSR facilities or technologies*. Expanded information on licenses and certificates is provided in Appendix A.

2.1 10 CFR PART 50, "DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES"

The regulations in 10 CFR Part 50 are promulgated by the NRC pursuant to the AEA of 1954 [4], as amended (68 Stat. 919) [4], and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242) [2] to provide for the licensing of production and utilization facilities.

2.1.1 10 CFR 50.22, "Class 103 Licenses; for Commercial and Industrial Facilities"

Chapter 10 of the AEA, as amended, addresses licensing. Sections 103 and 104 of the AEA outline the two types of licenses that can be granted to an applicant. Section 103 discusses commercial licenses, and Section 104 discusses medical therapy and research and development licenses. The licensing requirements of the AEA are reflected in 10 CFR 50.21, "Class 104 Licenses: for medical therapy and research and development facilities," and in 10 CFR 50.22, "Class 103 Licenses; for commercial and industrial facilities." This licensing classification will be the same for MSR facilities.

The standard review plan (SRP) for LWR applications is found in NUREG-0800 [5]. NUREG-0800 is updated by chapter, and several chapters have been updated based on lessons learned from the review of the Westinghouse AP-1000 and other advanced LWRs. Unique review plans have also been developed for some small modular reactors (SMRs). Advanced reactor reviews (e.g., SMRs, non-LWRs, and microreactors) may be based on NUREG-0800 and the regulatory bases found in 10 CFR Parts 50, 52, or 53. An alternative review plan, such as NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* [6], [7], (see SHINE in Appendix B 1.1) may also be considered because many of the proposed SMRs and advanced non-LWRs may have small source terms similar to the source terms associated with some non-power reactors. A Licensing Modernization Project, outlined in NEI 18-04 [105], *Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development*, provides and alternate approach to licensing within 10 CFR 50 and 10 CFR 52. This content is also expected to change the required content of advanced reactor applications, which is currently being addressed by the industry-led Technology

Inclusive Content of Application Project (TICAP). Additionally, various nonreactor fuel cycle facilities have separate review plans.

2.1.2 Licensing Requirements with Potential Significance for MSR Technologies, Including Additional Fuel Cycle Activities

As noted in SECY-09-0082, *Update on Reprocessing Regulatory Framework – Summary of Gap Analysis* [8], 10 CFR Part 50 provides the licensing framework for production and utilization facilities. This includes reprocessing facilities and certain frontend fuel cycle functions that may be associated with some MSR technologies because they are a type of production facility [8]. However, as noted in SECY-13-0093, *Reprocessing Regulatory Framework – Status and Next Steps* [9], the staff concluded that the regulatory framework for licensing a reprocessing facility under 10 CFR Part 50 may not be efficient or effective. The licensing requirements located in 10 CFR Part 50 are principally based on the long NRC regulatory history with LWR technology. This will limit the ease with which 10 CFR Part 50 can be applied to other production facility designs and technologies [9].

A facility employing a broad-spectrum fuel cycle MSR technology must consider the intent of the regulations and make the case for an alternative method to meet a given requirement or seek a waiver. The following regulatory sections in 10 CFR Part 50 are specifically noted by the authors of this report for nuanced application to a broad-spectrum fuel cycle MSR facility applicant.

2.1.2.1 Standards for Licenses, Certifications, and Regulatory Approvals

The regulations in 10 CFR Parts 50.40, 50.42, and 50.43 provide for common standards and standards specific to the issuance of a Class 103 license, including the development and operation of a prototype plant to gather additional data. A prototype power reactor would be licensed similarly with specific license conditions associated with the prototypic features. The use of a prototype power reactor may not accelerate the licensing process. as a result, many MSR developers will likely consider pursuing an extensive test program that may include a test reactor. The review process for a test reactor application would be under the performance-based (PB) criteria found in NUREG-1537 [6].

2.1.2.2 10 CFR 50.34, "Content of Applications; Technical Information"

Each application for a production or utilization facility CP shall include a preliminary safety analysis report. This includes the development of PDC. The Nuclear Energy Institute (NEI) incorporated consideration for PDC in some of its advanced reactor guidance documents. NEI issued technical report NEI 21-07 [106], *Technology Inclusive Guidance for Non-Light Water Reactors*, to inform the safety analysis report content for applicants using the methodology in NEI 18-04 [105]. NRC guidance for implementing NEI 18-04 is found in RG 1.233 [104], *Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors*. NEI 21-07 provides a systematic approach for identifying PDC as an alternative to the traditional deterministic approach of identifying PDC used for LWRs. The methodology in NEI 18-04 and implementing guidance in NEI 21-07 will likely not lead to PDC that are identical to the general design criteria (GDC) found in Appendix A of 10 CFR Part 50 or the advanced reactor design criteria (ARDC) found in RG 1.232, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors* [18]. Neither resource provides regulatory requirements for non-LWR applicants. However, the GDC and the ARDC do provide guidance for developing the scope of proposed PDC to be developed by a non-LWR applicant under 10 CFR Parts 50 and 52.

If 10 CFR Part 53 were enacted as currently proposed [10], PDC for the reactor portion of a broad-spectrum fuel cycle MSR facility would need to consider the technology functional design criteria for safety-related SSCs and SSC special treatments for defense in depth or the performance of risk-significant functions. Non-LWR applicants would be required to provide PDC using the GDC or other generally accepted consensus codes and standards to inform PDC development.

2.1.2.3 10 CFR 50.36, "Technical Specifications"

Each applicant for a license authorizing operation of a production or utilization facility shall include proposed SSC technical specifications for the facility. Design certifications under 10 CFR Part 52 also require SSC technical specifications for the design. A broad-spectrum fuel cycle MSR technology must propose technical specifications for all aspects of the fuel cycle represented at the plant, including the reactor and any proposed frontend or backend facility. Broad-spectrum fuel cycle MSR facilities must specify appropriate technical specifications for the reactor and any other fuel cycle facilities associated with the MSR site.

2.1.2.4 10 CFR 50.47, "Emergency Plans"

A production or utilization facility applicant must provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency in accordance with the regulations in 10 CFR 47 and Appendix E of 10 CFR 50, "Emergency Planning and Preparedness for Production and Utilization Facilities." A COL issued under 10 CFR Part 52 must also meet these requirements. In the case of LWRs, this has traditionally required that a plume exposure pathway EP zone (EPZ) for nuclear power plants comprise an area of ~10 mi (16 km) in radius and an ingestion pathway EPZ comprise an area of ~50 mi (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular production or utilization facility can be adjusted in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.

However, a rulemaking is in process for SMRs and other new technologies (ONTs) [11]. In SECY-11-0152, *Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors* [12], the staff discussed potential changes to the EP and preparedness framework for SMRs. The staff indicated their expectation that the dose assessments could be a factor for the basis for the EPZ distances based on a spectrum of accidents using the plant design PRA, as well as including current insights on severe accident progression.

Subsequently, in SECY-15-0077 [13], the staff proposed a consequence-based approach to establishing requirements for off-site EP for SMRs and ONTs. In the related SRM [14], the NRC approved the staff's proposal to revise NRC regulations and guidance through rulemaking to "demonstrate how their proposed facilities achieve U.S. Environmental Protection Agency (EPA) Protective Action Guide (PAG) dose limits at specified EPZ distances, which may include the site boundary." The PAG manual was updated in 2017 [1].

Essentially, the proposed rule will provide [11]:

- a new alternative PB EP framework,
- a hazard analysis of any NRC-licensed or nonlicensed facility contiguous or nearby to an SMR or ONT that considers any hazard that would adversely affect the implementation of emergency plans,
- a scalable approach for determining the size of the plume exposure pathway EPZ, and
- a requirement to describe ingestion response planning in the EP, including the capabilities and resources available to prevent contaminated food and water from entering the ingestion pathway.

The proposed rule [11] offers a new 10 CFR 50.160 requirement for EP for SMRs and ONTs as an option to the current requirements in 10 CFR 47. The proposed rule will benefit MSR technologies because the radionuclide driving force following any accident is greatly reduced for low-pressure MSRs. The subsequent reduction in the off-site source term will result in smaller EPZs, which is an economic consideration. The rulemaking also provides for appropriate references to the new rule in 10 CFR 50.2, "Definitions," to provide definitions for non-LWRs, nonpower production or utilization facilities, and SMRs. Revisions are also included in 10 CFR 50.33, "Contents of applications; general information"; 10 CFR 50.34, "Contents of applications; technical information, 10 CFR 50.47"; 10 CFR 50.54, "Conditions of licenses;" and Appendix E of 10 CFR Part 50 to include appropriate references and options for the new rule. Rule publication is targeted for 2022. However, there is no schedule update as of this report's publication.

2.1.2.5 10 CFR 50.48, "Fire Protection"

Any reactor licensed under 10 CFR Parts 50 or 52 must have a fire protection plan. The fire protection requirements for older LWRs are prescribed in Appendix R of 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." Appendix R requires that two separate water supplies be provided to furnish the necessary water volume and pressure to the fire main loop. A water-based fire suppression system is not the best technology choice for fire suppression at a broad-spectrum fuel cycle MSR facility because one advantage of MSR technologies is that they operate at low pressure with a functional containment that reflects the low risk for an accident-induced pressure spike spread of contamination. Introducing water into the high-heat MSR containment environment would provide an unnecessary potential for a steam explosion and subsequent spread of contamination.

MSR technologies may need to rely on combustible controls; controls for graphite applications outside the core, such as in an off-gas system; and chemical fire suppressants to meet the requirements of Appendix R of 10 CFR 50.48; National Fire Protection Association (NFPA) 805, *Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants* [16], and RG 1.205, *Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants* [17]. The various aspects of the frontend or backend fuel cycle applications may require segmented cells for processing to limit the risk of fire or to mitigate fire damage.

The draft language for 10 CFR 53.875, "Fire Protection," promotes PB requirements and does not specify a water-based fire suppression system [10].

2.1.2.6 10 CFR 50.55a, "Codes and Standards"

This section lists codes and standards that are approved for incorporation by reference into an application. Because 10 CFR Part 50 is biased toward LWR technology, as discussed in Section 4, MSR designs must make the case for newer standards that support their technologies.

2.1.2.7 10 CFR 50.63, "Loss of All Alternating Current Power"

Although 10 CFR 50.63 is LWR specific, the underlying safety basis is reflected in the advanced reactor design criterion (DC) discussed in RG 1.232 [18]. ARDC 17 shifts the safety emphasis from the fuel to the FP barriers and associated safety functions.

An additional aspect of 10 CFR 50.63 is station blackout (SBO) coping time. Coping time for active safety systems is discussed in RG 1.155 [19], *Station Blackout*, and is largely based on the operating status of emergency diesel generators. Because passive safety systems do not rely on the operating status of emergency diesel generators, a different approach to coping time is needed for advanced LWRs and non-LWRs. NUREG-0800 Chapter 8.4, "Station Blackout" [20], indicates that passive LWR technologies do not need to evaluate SBO coping duration if they can demonstrate that the technology being considered can perform safety-related functions for 72 h. The 72 h approach is consistent with the duration approved by the NRC for the AP 1000 design and has become a basic passive safety standard tenet. Thus, MSRs must include a discussion of their passive safety capabilities and consideration of SBO coping time within their technology license applications.

2.1.2.8 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

The requirements found in 10 CFR 50.65 require that licensees monitor the performance or condition of SSC against licensee-established goals (i.e., technical specifications). Licensees shall provide reasonable assurance that these SSC can fulfill their intended functions. Industry-wide operating experience should be factored into SSC monitoring, evaluation, and maintenance.

Because most MSR SSCs are always subjected to high-radiation environments, maintenance is more difficult and often must be performed remotely. Therefore, SSC monitoring, maintenance planning, and maintenance execution must be discussed in an MSR technology license application. Additionally, passive SSCs are used more extensively in advanced reactor applications, including MSRs. Passive SSCs more commonly exhibit performance degradation rather than outright failure. Such degradation may not constitute a system failure and may not compromise overall plant safety. MSR developers must provide a discussion of degraded passive SSC performance and work with NRC staff to develop advanced reactor guidance that would indicate that passive SSCs must remain capable of adequately fulfilling their intended functions so that overall plant operation continues to achieve the fundamental safety functions (FSFs).

2.1.2.9 10 CFR 50.68, "Criticality Accident Requirements"

The requirement in 10 CFR 50.68 provides two options for preventing a criticality accident. A Class 103 license holder shall comply with either 10 CFR 70.24, "Criticality Accident Requirements," or the requirements in paragraph (b) of 10 CFR 50.68.

Paragraph (b) applies to solid LWR fuel and is not directly applicable to MSR technologies. However, the underlying safety goal of 10 CFR 50.68(b) is to prevent a criticality condition away from the reactor core. To meet the paragraph (b) safety goal, MSR technologies must specify SSCs, design parameters, and technical specifications that are intended to prevent a criticality accident.

MSR technologies can also look to 10 CFR 70.24 for an alternate set of criticality accident requirements. The requirements in 10 CFR 70.24 are for detecting a criticality accident. Per 10 CFR 70.24, each licensee authorized to possess SNM in quantities necessary for reactor operations shall maintain a neutron and gamma radiation monitoring system. Two detectors shall provide coverage of all areas. This applies to the portions of an MSR fuel cycle at a plant beyond actual reactor operation. However, the underlying safety goal is still to prevent a criticality condition away from the reactor core. Therefore, it is still incumbent upon MSR designers to specify how they will meet this safety goal.

2.1.2.10 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

The requirements of 10 CFR 50.69 provide a voluntary approach for a risk-informed process to evaluate the safety significance of SSCs and establish the appropriate level of special treatment requirements for SSCs [21]. The grouping and integration of risk-informed requirements within this requirement provide the capability to apply a risk-informed approach more broadly within 10 CFR Parts 50 and 52. Under this approach, licensees and NRC staff can focus regulatory resources on SSCs with significant contributions to plant safety.

This voluntary requirement applies only to LWRs. However, because 10 CFR Part 50 applies to all production and utilization facilities, MSR SSCs associated with the frontend or backend of the fuel cycle production or utilization facility can make the case to treat their respective SSCs in accordance with 10 CFR 50.69. MSR technologies must specify safety-related SSCs and SSCs important to safety for the entire MSR facility to support technical specification development and accident analyses.

2.1.2.11 10 CFR 50.150, "Aircraft Impact Assessment"

The Aircraft Impact Rule [22] requires that design and license applicants for new LWRs perform a rigorous assessment of their designs to identify design features and functional capabilities that could provide additional inherent protection to avoid or mitigate the effects of an aircraft impact. Applicants for new nuclear power reactors must perform a realistic design-specific assessment of their designs to identify design features and functional capabilities that could provide additional inherent protection to avoid or mitigate the effects of the impact of a large commercial aircraft.

Applicants are required to identify and incorporate into the design those design features and functional capabilities that avoid or mitigate—to the extent practical and with reduced reliance on operator actions—the effects of the aircraft impact on key safety functions [23]. The applicant must show that with reduced operator actions, the reactor core remains cooled or the containment remains intact and spent fuel pool cooling or spent fuel pool integrity is maintained. An MSR applicant will need to translate these functions to the attributes of their specific technology and safety systems.

2.1.2.12 10 CFR 50.155, "Mitigation of Beyond Design Basis Events"

In 10 CFR Parts 50 and 52, beyond design basis events (BDBE) are identified and assessed on an as-needed basis using a prescriptive approach in which uncertainties are addressed by exercising conservative assumptions. The requirement in 10 CFR 50.155 is heavily influenced by experience with LWRs, but a similar accident response discussion will be required for MSRs per 10 CFR 34. Under this regulation, licensees must (1) develop, implement, and maintain mitigation strategies for beyond-design basis external events and (2) develop extensive damage mitigation guidelines associated with the loss of large areas of the plant.

If Part 53 is enacted as currently proposed, then BDBEs must be identified and supporting SSCs must analyzed to confirm that the collective SSC provide an adequate response to mitigate the BDBE and maintain an adequate level of defense-in-depth. supporting broad-spectrum fuel cycle MSR facility applications under 10 CFR Parts 50, 52, or 53.

2.1.2.13 Waste Management

The requirements in 10 CFR 50.36a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors," states that an applicant shall "keep levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable [ALARA], taking into account the state of technology."

Additionally, the requirements in 10 CFR 50.34, "Contents of Applications; Technical Information," indicate that the kinds and quantities of radioactive materials expected to be produced during operations, including anticipated operational occurrences (AOOs), and the means to control and limit radioactive effluent releases and radiation exposures within the limits of 10 CFR Part 20 for members of the public must be included in the safety analysis report.

10 CFR Part 50 Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," and NUREG-0800 Chapter 11 [24], [25], [26] provide LWR requirements and review standards for liquid waste management systems, gaseous waste management system, and solid waste management systems. As noted in NUREG-0800, the acceptance criteria for these systems are largely based on requirements found in 10 CFR Part 20 and 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," which are technology neutral.

Liquid Waste

According to Chapter 11.2 of NUREG-0800 [24], the application and review of an LWR liquid waste management system should encompass all tanks, piping, pumps, valves, filters, demineralizers, mobile equipment connected to permanently installed systems, and any additional equipment that may be needed to process and treat liquid wastes and route them to the point of discharge from the system. All SSCs included in a broad-spectrum fuel cycle MSR technology must also include an evaluation of any liquid waste streams. Since acceptance criteria are noted to be essentially technology-neutral, an MSR applicant must demonstrate compliance with regulatory limits on liquid effluent discharges and associated doses to members of the public to ensure that releases and doses are ALARA. Exceptions and additional limits will need to be negotiated with the NRC based on experience, which will come largely from test loops and test reactors.

Gaseous Waste

According to Chapter 11.3 of NUREG-0800 [25], the application and review of an LWR gaseous waste management system should provide for the management of radioactive gases generated by a gaseous radwaste system or the off-gas system, which may include waste gas storage tanks, waste gas decay tanks, and charcoal delay beds, depending on the type of plant and design features. Apart from a gaseous waste handling system, all MSR technologies will employ a separate off-gas system to allow the removal of gaseous FPs in order to avoid pressurization of the reactor vessel. Therefore, all SSCs included in a broad-spectrum fuel cycle MSR technology must include an evaluation of gaseous waste streams and other gas handling systems while demonstrating compliance with regulatory limits on gaseous effluent discharges and associated doses to members of the public in ensuring that releases and doses are ALARA. As noted above, acceptance criteria are noted to be essentially technology-neutral. Exceptions and additional limits will need to be negotiated with the NRC based on experience, which will come largely from test loops and test reactors.

Solid Waste

According to Chapter 11.4 of NUREG-0800 [26], the application and review of an LWR solid waste management system should encompass design features that are necessary for collecting, handling, processing, and storing wastes in buildings that are part of the overall nuclear facility. Many MSR technologies plan for the periodic replacement of large, contaminated SSCs. An application for a broad-spectrum fuel cycle MSR technology must demonstrate that the collection, handling, storage, and off-site shipment of such contaminated SSCs have been considered within the plant life cycle. Compliance with limits on gaseous and liquid effluent discharges from the operation of the solid waste processing equipment and associated doses to members of the public must be evaluated to ensure that any releases and doses are ALARA. As noted above, acceptance criteria are noted to be essentially technology-neutral.

Exceptions and additional limits will need to be negotiated with the NRC based on experience, which will come largely from test loops and test reactors.

2.1.2.14 Decommissioning

A decommissioning plan is not required by an applicant for a 10 CFR Parts 50 or 52 license. However, 10 CFR 50.33, "Contents of Applications; General Information," specifies that "reasonable assurance will be provided that funds will be available to decommission the facility" as described in 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning."

The requirements in 10 CFR 50.75 include a table that provides a decommissioning fund formula for LWRs based on the long history with that technology [27]. Because of the uncertainty associated with new technologies, other advanced reactor technologies, including MSRs, must provide some background information on expected decommissioning tasks for that specific technology to satisfy regulatory staff that adequate decommissioning funds are being set aside. Draft language proposed for 10 CFR Part 53 indicates that cost estimates for decommissioning advanced non-LWR technologies must be site specific and account for the engineering, labor, equipment, transportation, disposal, and related charges needed to support license termination.

2.1.2.15 Siting of Fuel Reprocessing Plants and Related Waste Management Facilities

A broad-spectrum fuel cycle MSR technology may include fuel reprocessing on-site with the reactor. 10 CFR Part 50 Appendix F, "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities," states that 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. Thus, siting a reprocessing facility near a reactor is permissible. However, fuel reprocessing will generate high-level waste (HLW), and the disposal of HLW material is permitted only on land that the federal government owns and controls. Thus, the logistics for waste processing associated with MSR fuel reprocessing must be considered by a broad-spectrum fuel cycle MSR technology.

If separated products from fuel reprocessing have a value or purpose for other aspects of the fuel cycle or other industrial uses, then an exception to Appendix F can be sought for those materials.

2.2 10 CFR PART 52, "LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS"

10 CFR Part 52 compliments the licensing requirements in 10 CFR Part 50. Specifically, 10 CFR Part 52 governs the issuance of early site permits, standard design certifications, COLs, standard design approvals, and manufacturing licenses for nuclear power facilities licensed under Section 103 of the AEA of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242) [2].

Most production and utilization facility requirements in 10 CFR Part 50 include a reference to the COL process in 10 CFR Part 52. COL applicants can reference an approved standard design and/or an early site permit. An MSR standard design or early site permit can include broad-spectrum fuel cycle MSR technology.

However, some additional requirements in 10 CFR Part 52 are not present in 10 CFR Part 50. Principally, these include the requirements for performing inspections, tests, analyses, and acceptance criteria that define when a reactor may transition from being in a construction state to being in an operating state. Additionally, 10 CFR Part 52.47(a)(27) requires a design-specific PRA, which is not a requirement under

10 CFR Part 50. The developer must thoroughly consider these and other differences before proceeding down a specific licensing path.

2.3 10 CFR PART 53, "RISK INFORMED, TECHNOLOGY-INCLUSIVE REGULATORY FRAMEWORK FOR ADVANCED REACTORS"

The Nuclear Energy Innovation and Modernization Act (NEIMA) [28] directs the NRC to develop the regulatory infrastructure to support the development and commercialization of advanced nuclear reactors. In response to NEIMA, the NRC is preparing a rulemaking for a new 10 CFR Part 53, which is intended to establish a technology-inclusive regulatory framework for optional use by applicants for new commercial advanced nuclear reactors. The regulatory requirements developed in this rulemaking would use methods of evaluation, including risk-informed and PB methods, that are flexible and practicable for application to a variety of advanced reactor technologies [29].

As shown in Figure 3, the NRC has proposed a general structure for Framework A and Framework B of the 10 CFR Part 53 rulemaking [30]. Draft language has been proposed for each subpart and is actively being debated among various advanced reactor technology stakeholders. Where appropriate, the 10 CFR Part 53 draft language is compared with existing regulations within this report to provide insight on what may be required for a broad-spectrum fuel cycle MSR technology license application. However, no 10 CFR Part 53 language has been finalized. Part 53 is scheduled to be finalized and available for use in 2025.

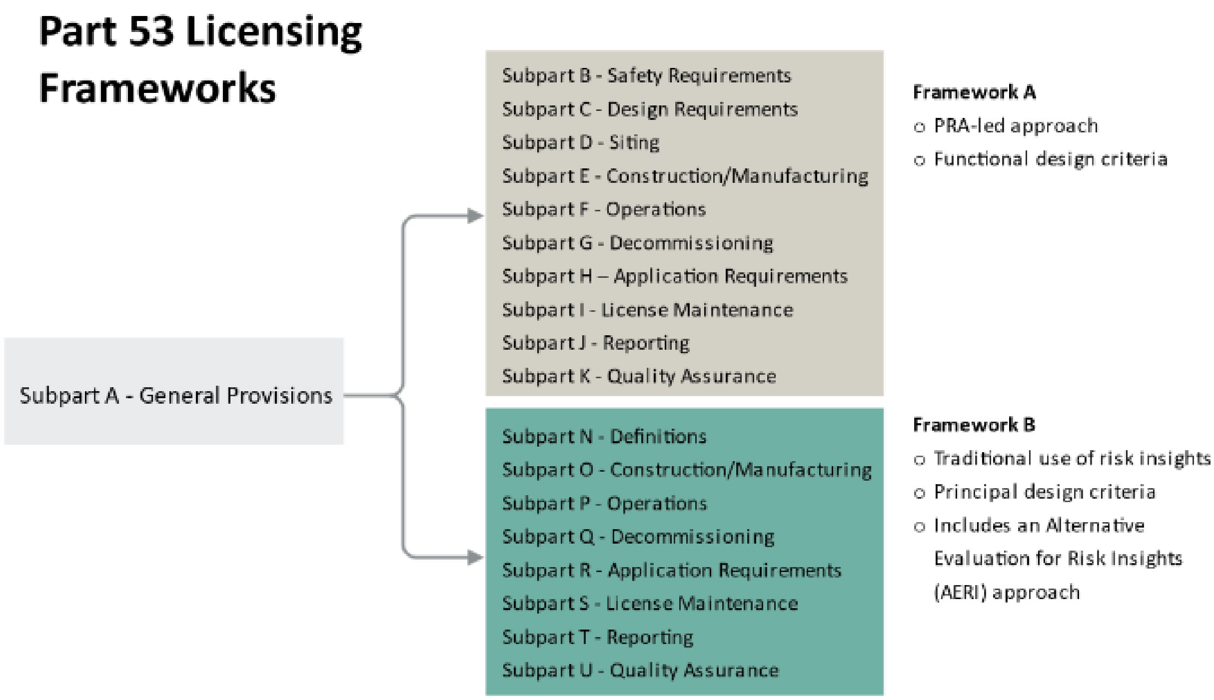

Figure 3. Proposed 10 CFR 53 subparts [33].

2.4 10 CFR PART 70, "DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL"

10 CFR Part 70 regulates the licensing required to receive title to, own, acquire, deliver, receive, possess, use, and transfer SNM. Although a separate 10 CFR Part 70 license is not required for possession or use of SNM for the operation of a nuclear reactor licensed under 10 CFR Parts 50 or 52, a broad-spectrum

fuel cycle MSR facility may require a separate license to possess and use SNM in an on-site fuel fabrication plant. SNM includes the fresh fuel for a reactor. Fuel for LWRs is in the form of fuel rods incorporated into fuel assemblies. Such heterogeneous fuel is relatively easy to inspect upon receipt, account for the individual assemblies, and provide for security on-site. New fuel for an MSR could be in solid or liquid from, could be fresh or recycled material, and could be ready to insert into the reactor core or require further batch processing. Fuel scrap recovery and conversion is also considered under 10 CFR Part 70. Applicants must make the case for whether any similar activities at a broad-spectrum fuel cycle MSR facility rise to the level of fuel fabrication.

2.5 10 CFR PART 72, "LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL, HIGH-LEVEL RADIOACTIVE WASTE, AND REACTOR-RELATED GREATER THAN CLASS C WASTE"

10 CFR Part 72 regulates the licensing required to receive, transfer, and possess reactor spent fuel, reactor-related greater than Class C (GTCC) waste and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI). The regulations in 10 CFR Part 72 also establish the requirements, procedures, and criteria for issuing a spent fuel storage cask Certificate of Compliance (CoC).

MSR technologies must consider spent fuel handling and processing over the plant lifetime. If the ultimate disposition of the spent fuel will include an on-site ISFSI facility, then an applicant for a broad-spectrum fuel cycle MSR facility must describe "the receipt, handling, packaging, and storage of spent fuel, high-level radioactive waste, and/or reactor related GTCC waste as appropriate, including how the ISFSI or MRS will be operated" (10 CFR 72.24). Dose evaluations and a discussion of SSCs must be included. A separate 10 CFR Part 72 license will be required for a collocated ISFSI. EP (10 CFR 72.32) must be coordinated with the emergency plan described in 10 CFR 50.47.

2.6 10 CFR PART 30, "RULES OF GENERAL APPLICABILITY TO DOMESTIC LICENSING OF BYPRODUCT MATERIAL"

Any radioactive material—except SNM—produced or made radioactive by exposure to the radiation incident to the process of producing or using SNM is defined in 10 CFR Part 50 as byproduct material. MSRs may strip actinides from spent fuel for waste stabilization. Although the actinides may be recycled or held for disposal, the remaining radioactive waste is technically a byproduct of the MSR operation. This could provide an MSR licensee with a byproduct material that is useful as a sealed industrial heat source. Any MSR facility contemplating the sale, distribution, or recycling of byproduct material resulting from expanded fuel cycle activities at its site must consider the licensing requirements found in 10 CFR Part 30.

3. OTHER COMPLIANCE CONSIDERATIONS

3.1 10 CFR PART 20, "STANDARDS FOR PROTECTION AGAINST RADIATION"

The requirements in 10 CFR Part 20 "establish standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the NRC". 10 CFR Part 20 controls the receipt, possession, use, transfer, and disposal of licensed material by any licensee to ensure that the total dose to an individual does not exceed the standards for protection against radiation.

The requirements in 10 CFR 20.1101, "Radiation Protection Programs," compel each licensee to "develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities". Such programs must provide procedures and engineering controls to "achieve occupational doses and doses to members of the public that are ALARA". This will encompass all activities at a broad-spectrum fuel cycle MSR facility.

3.2 10 CFR PART 51, "ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS"

10 CFR Part 51 contains environmental protection regulations applicable to the NRC's domestic licensing and related regulatory functions. 10 CFR Part 51 implements the National Environmental Policy Act of 1969, as amended [31].

3.2.1 10 CFR 51.23, "Environmental Impacts of Continued Storage of Spent Nuclear Fuel Beyond the Licensed Life for Operation of a Reactor"

NUREG-2157 [32] provides the environmental impacts of continued storage of spent nuclear fuel (SNF) beyond the licensed life for reactor operation.

High-temperature gas-cooled reactors and liquid metal fast reactors are outside the scope of NUREG-2157 because of the differences in the fuel form compared with those of LWRs. Although MSR fuel is not specifically mentioned, it is implied that MSR fuel will also not be covered by NUREG-2157. Nonetheless, the environmental impact time frames stated in NUREG-2157 must still be considered.

Unlike LWR fuel, MSR fuel salt tends to become less stable with time after irradiation. There is no cladding, and the radiolytic release of gas from the fuel salt will occur if fuel stabilization is not initiated. Some of the FP halide vapors can include radioactive materials, and ^{36}Cl is a long-lived beta emitter. Additionally, fluoride fuel salt can also release UF_6 or UO_2F_2 when cooled if the salt is not handled properly. MSR technologies must address the necessary controls for the indefinite storage of fuel on-site. These will likely be more complex than comparable LWR indefinite storage systems.

3.3 10 CFR PART 61, "LICENSING REQUIREMENTS FOR LAND DISPOSAL OF RADIOACTIVE WASTE"

10 CFR Part 61 regulates the procedures, criteria, and terms and conditions upon which the NRC issues licenses for the disposal of radioactive wastes containing byproduct, source, and SNM in a land disposal facility. Some important considerations for a broad-spectrum fuel cycle MSR facility include waste classification and waste characteristics [26].

3.3.1 10 CFR 61.55, "Waste Classification"

The Low-Level Radioactive Waste Policy Amendments Act of 1985 [33] and the Nuclear Waste Policy Act of 1982 [34] divide most radioactive waste into two categories: HLW and low-level radioactive waste (LLRW). HLW includes highly radioactive waste material associated with fuel reprocessing or other waste that requires permanent isolation. LLRW is material that is not HLW, spent fuel, or certain byproduct materials [35]. The 1985 Low-Level Radioactive Waste Policy Amendments Act provides for the classification of LLRW and gives responsibility for disposal to individual states, except for waste generated by the US Department of Energy (DOE) or the US Navy. The federal government is responsible for disposing of LLRW that is not defined in 10 CFR 61.55 or is classified as HLW.

The following waste classes are specified by 10 CFR 61.55.

- Class A waste is usually segregated from other waste classes at the disposal site.
- Class B waste must meet more rigorous requirements on waste form to ensure stability after disposal.
- Class C waste must meet more rigorous requirements on waste form to ensure stability but also requires additional measures at the disposal facility to protect against inadvertent intrusion.
- LLRW streams that contain radionuclide concentrations exceeding the limits for Class C waste are referred to as *GTCC waste*.

Classes A, B, and C waste can be disposed of using near-surface retention. Near-surface disposal can include surface disposal, trench disposal within 30 m of the surface, below ground vaults, earth mounds, and boreholes. Currently, GTCC waste is generally not acceptable for near-surface disposal because the form and disposal methods must be different—and, generally, more stringent—than those specified for Class C and lower waste. Under current regulations, GTCC waste must be held on-site to be disposed of in a geologic repository when available, except when allowed by the NRC on a case-by-case basis. GTCC waste typically includes neutron-activated components, in-core neutron detectors, activated metals, radioactive sources, and alpha emitting transuranics [26], [35].

Waste classification and disposal options are important to an MSR facility because of the potential for significant quantities of waste associated with associated fuel cycle activities and solid wastes associated with component replacements.

3.3.2 10 CFR 61.56, "Waste Characteristics"

Appropriate waste characteristics are intended to facilitate handling at the disposal site and protect the health and safety of personnel at the disposal site.

Broad-spectrum fuel cycle MSR facilities or technologies are expected to generate significant solid waste. MSR LLRW must meet the requirements in 10 CFR Part 61 for disposal. Additionally, MSR LLRW must be in a stable form to ensure that the waste does not structurally degrade and affect the overall stability of the site through slumping, collapse, or other failure of the disposal unit leading to water infiltration.

Waste characteristics are important to an MSR facility because many byproduct materials can exist as a solid or a liquid, depending on material temperature and long-term heat removal paths.

3.3.3 Proposed Rulemaking with Implications for MSR waste

In SECY-20-0098, *Path Forward and Recommendations for Certain Low-Level Radioactive Waste Disposal Rulemakings* [36], the staff propose combining two NRC-directed activities that could result in amendments to 10 CFR Part 61.

- An LLRW disposal rulemaking to address the disposal of waste streams (e.g., depleted U) that were not envisioned to be disposed of in significant quantities when 10 CFR Part 61 was originally enacted in 1982.
- A GTCC waste rulemaking that would allow near-surface disposal of GTCC waste beyond the case-by-case approval currently authorized in 10 CFR Part 61. Additionally, the definition of *waste* in 10 CFR Part 61 is proposed for revision so that LLRW that is acceptable for disposal under 10 CFR Part 61 no longer excludes transuranic waste.

As part of SRM SECY-20-0098 [37], the commissioners directed the NRC staff to reexamine the technical basis for the performance objectives in 10 CFR Part 61 and ensure that the compliance period following the closure of a disposal facility is performance based using scientific data. This rulemaking could have significant implications for MSRs with expanded fuel cycle facilities on-site. With this rulemaking, used MSR equipment that would currently be classified as GTCC waste requiring on-site storage until disposal in a geologic waste repository may gain a faster site removal path using near-surface disposal. This is discussed further in Section 4.1.1.6.

3.4 10 CFR PART 73, "PHYSICAL PROTECTION OF PLANTS AND MATERIALS"

10 CFR Part 73 regulates the establishment and maintenance of a physical protection system that will have capabilities for protecting SNM at production and utilization sites. The entire broad-spectrum fuel cycle MSR facility must be capable of defending against the design basis threat identified in 10 CFR Part 73, including appropriate safeguards systems to "protect against acts of radiological sabotage and to prevent the theft or diversion of special nuclear material".

The typical sabotage threats for MSR technologies may be different or on a different scale than the threats associated with LWRs. These threats must be noted when discussing MSR plant security. For example, MSRs will use more automation for plant maintenance and other activities because of the high radiation fields associated with plant operation. More plant automation can lead to greater cyber-security concerns. Additionally, the low-pressure MSR containment structure will be thinner, perhaps leading to different opportunities to penetrate containment. Fuel cycle activities that involve separation processes can invite the diversion of radiological material if not properly protected.

Proposed rule changes for 10 CFR Part 73 for advanced reactors include a PB approach to physical security and a graded approach to cyber security and access authorization [38], [39], [40].

3.5 10 CFR PART 74, "MATERIAL CONTROL AND ACCOUNTING OF SPECIAL NUCLEAR MATERIAL"

10 CFR Part 74 regulates the control and accounting of SNM at nuclear facility sites. This is traditionally identified as nuclear material accountability and control (NMAC). Requirements for the control and accounting of source material at enrichment facilities are also included. The general conditions and procedures for submitting a license application for the activities covered in 10 CFR Part 74 are detailed 10 CFR Part 70.

10 CFR Parts 50 and 70 and the proposed draft text for 10 CFR Part 53 all define SNM as:

- Plutonium, U^{233}, U enriched in the isotope U^{233} or in the isotope U^{235}, and any other material that the NRC pursuant to the provisions of Section 51 of the AEA, as amended, determines to be SNM but does not include source material or
- Any material artificially enriched by any of the foregoing but does not include source material.

10 CFR Part 53 proposes to define *fuel* as SNM, discrete elements that physically contain SNM, and homogeneous mixtures that contain SNM, intended to or used to create thermal power in a commercial nuclear plant.

10 CFR Part 74 requires that all nuclear facilities, including broad-spectrum fuel cycle MSR facilities, provide for NMAC for all SNM in the facility. Homogenous MSR fuel would be included under the traditional definition of SNM. However, under the proposed definition for fuel in 10 CFR Part 53, the use of homogenous fuel forms is specifically included.

The requirements of 10 CFR 74.17, "Special Nuclear Material Physical Inventory Summary Report," specify that an inventory of SNM be taken annually and the results reported to the NRC by plant and total facility. NMAC techniques must be innovative for MSR liquid fuel forms.

3.6 10 CFR PART 110, "EXPORT AND IMPORT OF NUCLEAR EQUIPMENT AND MATERIAL"

10 CFR Part 110 regulates the export and import of nuclear equipment and material. The definitions included in 10 CFR 110.2 define high-enriched uranium (HEU), low-enriched uranium (LEU), and natural U.

- *HEU* means U enriched to 20% or greater in the isotope ^{235}U.
- *LEU* means U enriched below 20% in the isotope ^{235}U.
- *Natural U* means U as found in nature, containing about 0.711% of ^{235}U, 99.283% of ^{238}U, and a trace (0.006%) of ^{234}U.

Definitions for HEU and LEU are similar in 10 CFR Part 50. These definitions are problematic for MSR technologies contemplating the use of a Th-U fuel cycle.

A Th-U breeder reactor can remain critical based on the production of ^{233}U. Additionally, fuel salt containing ^{233}U could be separated and processed for use in other reactors. However, 10 CFR Parts 110 and 50 do not include ^{233}U in the current definition of LEU. Therefore, the existing definitions of LEU in 10 CFR 110.2 and 10 CFR 50.2 must be updated to enable future licensees to develop Th-U plants that meet regulatory expectations.

3.7 INTERNATIONAL COOPERATION

International cooperation on specific MSR technologies is beyond the scope of this report. However, international cooperation on basic MSR research is an acceptable activity. Conversely, international cooperation on specific technology applications is likely prohibited by the AEA [4] under Section 123(a)(7).

4. DEVELOPMENT OF THE TECHNICAL BASIS FOR SAFETY ADEQUACY ASSESSMENT OF ADDITIONAL ELEMENTS OF FUEL CYCLE AT MSRS

4.1 MSR DESIGN DIVERSITY

Liquid salt fuel provides substantial design and operational flexibility. Different designs are currently under development to achieve different performance objectives. For example, some designs are focused on consuming the actinides in spent LWR fuel to reduce long-term radiotoxicity, whereas others seek to implement a modernized version of the Molten Salt Reactor Experiment (MSRE) technology to minimize their development risks. One consequence of the design diversity is that technology-specific regulatory structures for any one MSR design will not match the characteristics of all MSRs. One important example of the design diversity within MSRs is the location within the fuel cycle of separations processes. If used fuel is being recycled, then FPs can be removed from fuel salt before reactor operations as part of reactor operations—potentially on a side stream—or following reactor operations. The technologies, proliferation resistance issues, and hazards of each option have significant differences. Some separations result from inherent physical processes, such as fission gas release from fuel salt, whereas other fuel cycle options include physical or chemical separation of parasitic neutron absorbers or actinides. Process hazard assessment is an established method for developing the technical basis for safety adequacy assessment. To illustrate the difference in hazards of different MSR configurations, this section includes high-level descriptions of the following:

- Thermal-spectrum, fluoride salt MSRs operating on a Th-U fuel cycle and designed to be proliferation resistant
- A fast spectrum, chloride salt MSR operating on a U-Pu fuel cycle
- A thermal-spectrum, fluoride salt MSR employing a once-through LEU fuel path

The reactor concepts outlined here are intended to be representative of the broader potential set of MSRs. Additionally, this section discusses process steps for converting used LWR fuel into MSR fuel salt. Although the current US government responsibility to dispose of used LWR fuel provides substantial disincentives to reuse LWR fuel materials, the relative ease of transferring the actinides from used LWR fuel into MSR fuel salt has encouraged commercial interest in recycling [41], [42]. The overviews provided include the main fuel cycle and operational process steps with a particular focus on describing the hazards of locating the processes on a common site.

A PB safety-adequacy evaluation process would be especially useful for MSRs to accommodate their widely varying designs. PB regulations are based upon verifying the accomplishment of a safety objective rather than requiring specific technology to accomplish the objective. Technology-specific regulations also inevitably focus on the technology available at the time they were written, potentially inhibiting the introduction of improved technology.

Although MSRs are already developing highly diverse designs, multiple additional design options with substantial potential safety, regulatory, and proliferation implications would become possible with reasonably anticipated technology developments and/or maturation. Based on the recent rapid evolution of MSR technology, this report includes low technology readiness elements within its discussion of plant hazards. For example, an MSR core configuration with the fuel salt within tubes connected to inlet and outlet manifolds and cooled externally, similar to a shell-and-tube heat exchanger, would become possible with improved materials. This configuration resembles the dual-fluid MSR historic molten salt breeder reactor (MSBR) called the MOSEL reactor concept [43] and has been proposed as a fast spectrum, dual fluid reactor with Na [44] or Pb [45] as the coolant fluid. Alternatively, remote, automated fuel salt processing technology is crucial to one method for enabling breeding gain in the Th-U fuel cycle while avoiding direct access to separated fissile materials.

4.1.1 Hazards of Close Coupling Fuel Cycle Processes

The colocation of fuel cycle processes can introduce additional hazards. For example, using water to cool waste streams provides a potential mechanism for using steam production to pressurize one or more containment layers, which would provide an energy source capable of dispersing radionuclides into the environment. Alternatively, materials used to trap fission gases may be combustible, increasing the plant fire hazard. Additionally, locating fuel salt processing facilities near a reactor means that the facilities must contribute to overall plant achievement of reactor requirements. For example, 10 CFR 50.150 requires that containment remain intact following a large civilian plane impact. Separate fuel cycle facilities not located under commercial flight paths do not have an equivalent requirement. Individual steps in fuel salt processing themselves pose hazards, which were recently reviewed [46]. Colocating fuel processing facilities can compound the hazards because of interconnections between nearby processes. One anticipated technique for minimizing the hazard of locating multiple hazardous processes within a common outer containment is to segment the containment into a set of cells with robust barriers between the cells. A segmented configuration is conceptually similar to a hot cell processing line in which the individual processing steps are performed in a sequence of individual cells with controlled interconnections.

The diversity and low technical maturity of potential processing operations prevents detailed hazard analyses. However, NUREG-1520, *Standard Review Plant for Fuel Cycle Facilities License Applications* [47], recognizes that fuel processing facilities can be licensed without complete detail or final design. NUREG-1520 instead indicates that sufficient information must be provided to describe the operation and function of each item relied upon for safety and provide reasonable assurance that all credible accident sequences have been identified, safety-related SSCs have been identified, and management measures will be sufficient to ensure that safety-related SSCs will be both capable and adequately reliable.

4.1.1.1 Fuel Salt Preparation from Used LWR Fuel

MSRs may be deployed at existing nuclear plant sites to take advantage of existing site environmental evaluations, the approved EPZ, existing site security, access to skilled staff, and access to used LWR fuel without requiring off-site transportation. Utilities may elect to employ used LWR fuel as the source for some or all fissile material in MSR fuel salt. The proposed requirements in 10 CFR Part 53 for the licensee to be responsible for the storage costs of used fuel increase the incentives to reuse the actinides.

Some of the steps employed to convert used water reactor fuel to fuel salt would be the same as those used to clean and manage the actinide content of used fuel salt and thus would more likely be integrated into MSR operations. However, solid fuel processing steps, such as decladding and oxide removal, are sufficiently distinctive so that they are unlikely to be integrated into an MSR's facilities. However, decladding and oxide removal may be located in an adjacent facility and thus could contribute to overall site hazards.

Very limited commercial experience exists for actinide recovery from used fuel. Most prior used fuel processing technology evaluations have focused on large-scale facilities intended to process a substantial fraction of the accumulated US used fuel waste inventory. Individual MSRs would require much smaller amounts of production and would put substantial additional value on minimizing hazardous chemicals and conditions near the reactor, as well as on avoiding the generation of radioactive waste streams that would not be efficient to process at small scale. Traditional technologies for converting actinide oxides into fluorides involve significant quantities of fluorinating chemicals (e.g., HF, F_2, NF_3, NH_4HF_2) at elevated temperatures and can result in substantial volumes of contaminated waste. Moreover, these technologies do not scale efficiently to the smaller size of an individual MSR. One recent evaluation of the costs and technologies for converting used LWR fuel into fluoride fuel salt cautions that the cost variance may be large with a substantial risk of cost escalation [48]. Given the low overall technical maturity, a detailed evaluation of specific plant hazards is not yet possible.

Multiple alternative technologies for processing used LWR fuel are currently in early phase development. Low-temperature chlorination of Zr shows promise to remove the fuel cladding into a low radioactivity stream [49]. Lower-temperature, less toxic oxide-to-fluoride conversion techniques based on fluorinated ionic liquids are in early-stage development [50]. More established pyroprocessing technologies are also applicable for converting actinide oxides to chlorides. Although the main steps in pyroprocessing and mitigating their associated hazards have been known for decades, advanced technologies such as the use of $ZrCl_4$ to convert actinide and rare earth oxides into chlorides are more recent developments [51].

Nonaqueous organic U chemistry continues to develop rapidly [52]. The efficient transfer of actinides from fluoride salts into an Al alloy, leaving behind the lanthanides and other FPs, was demonstrated under laboratory conditions [53] along with technology to extract the actinides into chlorides from the resultant actinide-Al alloy [54], [55], [56], [57]. Subsequent conversion of the actinide chlorides into fluorides using fluorinated ionic liquids under mild conditions is also in early-phase development [58]. Novel methods for synthesizing both UCl_3 [59] and UCl_4 [60] under mild conditions was recently demonstrated under laboratory conditions. Chlorinating agents and methods for converting UO_2 to chloride in a chloride salt bath were also demonstrated under laboratory conditions [61]. General precursor U compounds for chloride fuel salt synthesis were also demonstrated [62], and a start-up company was formed to commercialize molten chloride salt fuel production [63].

4.1.1.2 Thorium-Uranium Fuel Cycle Technology Hazards

The following sections describe two examples of design concepts that feature breeding gains using the Th-U fuel cycle while maintaining strong proliferation resistance. The different designs have significantly different hazards because of the different processing steps.

First Th-U Example

Increased proliferation resistance in the first design concept is provided by a combination of isotopic dilution and keeping all fissile and nonfissile actinides together so that a separated stream of fissile material does not exist in the plant and the plant does not include the technology needed to separate the fissile materials from nonfissile actinides or isotopes. Several processes and SSCs associated with a co-separation–based Th-U breeder MSR remain at low technology readiness levels. One crucial attribute for any MSR, which increases the difficulty to create a separated stream of fissile material, is the inherent fuel homogenization provided by the liquid mixing. All fissile material production occurs within the fuel salt (i.e., no breeding blanket or fertile salt loop). The effective moderation and low excess reactivity also combine to efficiently fission the fissile isotopes, resulting in a high-burnup composition. Plant modifications near the highly radioactive fuel salt, which would be needed to install fissile material separations SSCs, would be both technically difficult (i.e., proliferation resistant) and readily apparent (i.e., safeguardable).

A co-separation Th-U breeder reactor would be started on a mixed Th and $^{235}U/^{238}U$ (LEU) fuel salt employing a ^{7}Li enriched FLiBe salt as the carrier. Thorium would be included in the fuel salt to produce ^{233}U. The ^{235}U provides the startup fissile material, and the ^{238}U denatures the fuel salt to contain only LEU. Thorium and a sufficient amount of natural U for the U in the fuel salt to remain low enrichment would be used to refuel. However, the United States does not currently have a definition of what constitutes LEU that includes mixtures of ^{233}U, ^{235}U, and ^{238}U, so it is not yet possible to conclusively determine whether the plant will serve a useful purpose proportionate to the quantities of SNM or source material to be used as required by Section 103 of the AEA. The NRC's existing definitions of LEU in 10 CFR 110.2 and 50.2 must be updated to enable future licensees to develop plants that meet regulatory expectations. Although it would be possible not to add the natural U to the fuel salt, causing the fuel salt to gradually increase its fissile U fraction, the LEU loading necessary for initial criticality will result in a substantial residual nonfissile U content for decades and a buildup of other nonfissile actinides that increase the technical challenge for separating fissile isotopes of U from the remainder of the fuel salt. Neutron capture in ^{238}U will produce ^{239}Pu and other heavy actinides through subsequent capture reactions. The plant would not include any technologies to separate Pu from the other actinides. The overall plant neutronic efficiency coupled with semicontinuous and continuous on-line refueling will prevent the accumulation of substantial excess quantities of fissile materials as the fissile isotopes are preferentially consumed to generate power.

A thermal-spectrum Th-U breeding fuel cycle requires efficient moderation and low parasitic absorption because of the relatively low neutron yield per absorption in ^{233}U (η ~2.3 for thermal energy neutrons). The need for neutronic efficiency has substantial reactor system design and operational implications. Minimizing parasitic neutron capture requires minimizing neutron capture in ^{233}Pa before its decay into ^{233}U, minimizing parasitic capture in FPs, and minimizing neutron capture in ^{238}U. Neutron capture in ^{233}Pa is minimized by transferring the fuel salt after a limited term use (~1 week) in the active fuel salt circuit to a low neutron flux area for several half-lives ($t_{½} \approx 27$ days). Neutron capture in ^{238}U is minimized by maximizing the Th-U ratio in the fuel salt, providing a very soft (thermal peak < 300 meV) neutron spectrum and employing a heterogeneous moderator and fuel configuration that minimizes actinide resonance capture. The fuel salt processing steps are shown in Figure 4. All actinides would be stripped (i.e., co-separation) from the fuel salt once removed from the active circuit using the Al metal transfer process [54]. The FPs would then be stripped from the residual carrier salt—minimizing parasitic neutron capture—via conversion to insoluble oxides or oxyfluorides [64], reductive extraction into liquid bismuth [65], or multistage vacuum evaporation [66]. The cleaned carrier salt would then be reused. The first stage in converting the actinide-Al alloy back into fluoride fuel salt is converting it to a chloride salt [55]–[58]. The actinide chloride salt would then be converted to a fluoride salt using a fluorinated ionic liquid [59]. Fuel salt concentrate in excess of that needed to fuel the local reactor (i.e., bred fuel) would be stored separately as startup fuel for progeny reactors.

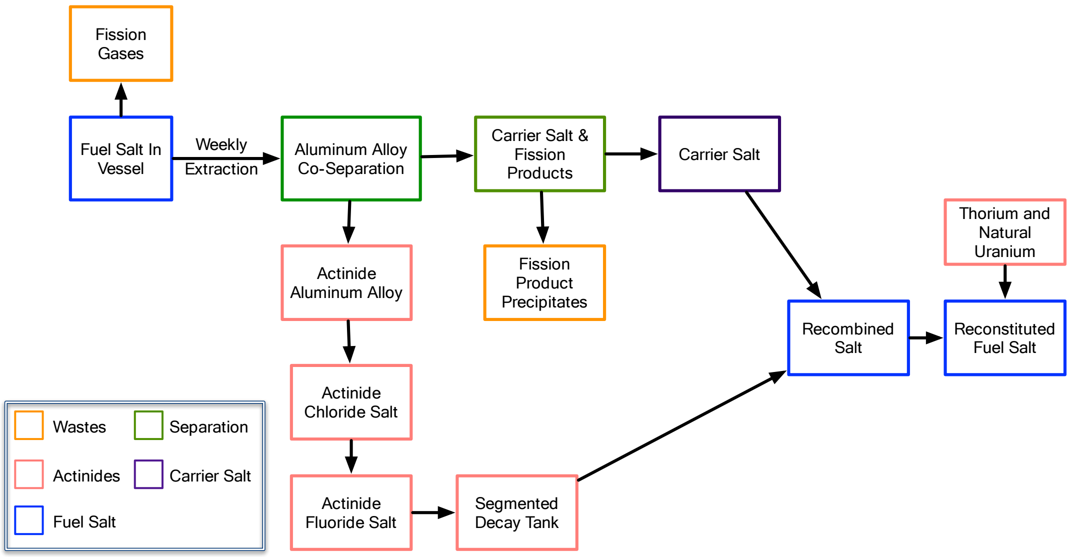

Figure 4. Fuel salt processing flowchart.

The required active fuel salt volume would be substantially reduced by configuring the fuel salt into tubes in the critical region that are connected in parallel to input and output manifolds. Fuel salt would flow slowly (a few centimeters per second) through the core under natural circulation with a total fuel salt circuit residence time of ~1 week. A conceptual layout of the vessel components and flow paths is shown in Figure 5. Most of the fission heat would be transferred via forced coolant flow in the core. The liquid fuel salt would be separated from the fission gases in a liquid-gas separator tank submerged in coolant salt in the upper portion (i.e., noncritical region) of the reactor vessel. Stripping ^{135}Xe from the fuel salt would significantly decrease parasitic neutron absorption. Stripping fission gases from all the fuel salt each circulation pass through the core avoids the need for a sweep gas, substantially reducing the gas volumetric flow. Fission gases would be drawn out of the liquid-gas separator via suction through extended heat transfer tubing (e.g., via a helical coil), allowing most of the fission gas decay heat to be transferred to the coolant salt before exiting the reactor vessel. The fuel salt would flow down through tubes located outside the critical region within the reactor vessel (i.e., downcomer) to the core entrance plenum. Coolant salt would enter the vessel above the top of the core and be directed to a downcomer region around the reactor vessel periphery. Exterior fuel salt tubes would be cooled by the coolant salt in the downcomer. The coolant salt would exit the reactor vessel near its top after passing through the core and around the fission gas separation tank.

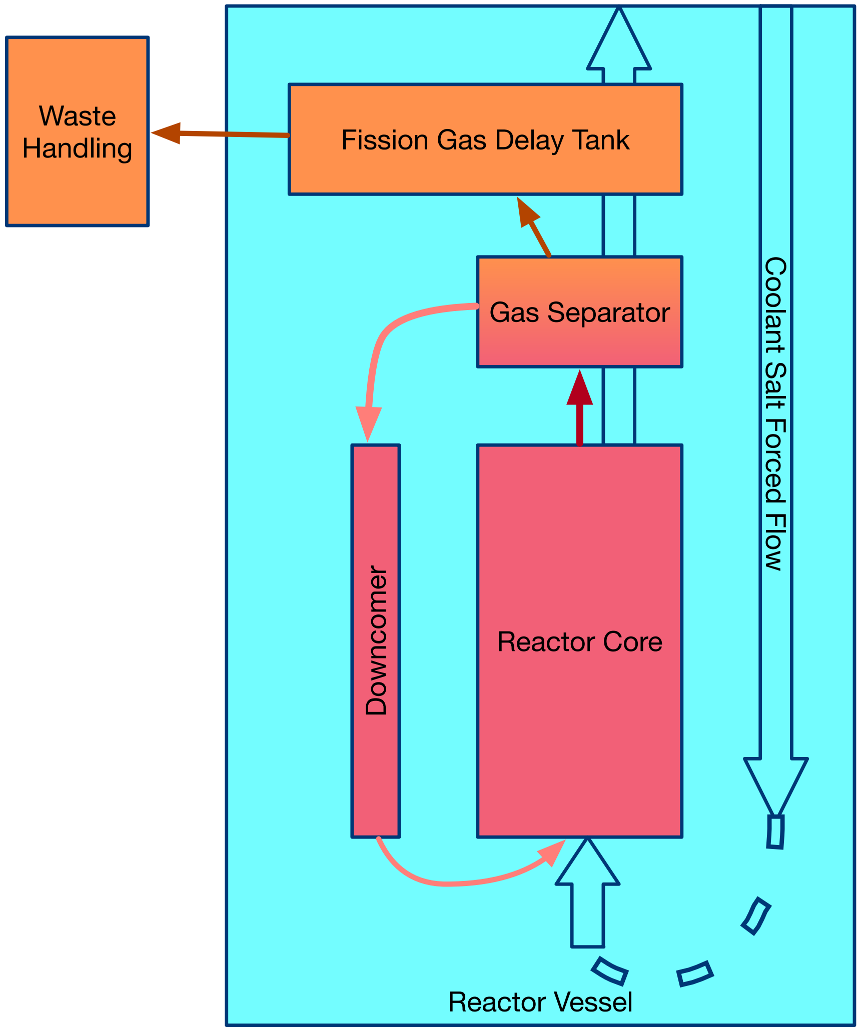

Figure 5. In-vessel components and flow path.

All actinides would be recycled within the plant to avoid producing an actinide waste stream. FPs would be converted to a stable waste form, as described in Section 4.1.1.6. The specific plant hazards depend on the particular technologies selected to perform each step. For example, reductive extraction of the FPs into liquid bismuth could carry over the liquid bismuth into the active fuel salt circuit, which would rapidly corrode any Ni alloys that it encountered.

Second Th-U Example

A second fuel cycle concept that would provide Th-U breeding in an MSR while maintaining strong proliferation resistance would be to implement the historic, single-fluid MSBR fuel cycle separations [67] fully remotely within a permanently inaccessible citadel. In this concept, the proliferation resistance would be achieved through the difficulty in accessing the fissile materials. Inaccessibility as a proliferation resistance feature is also employed by LWRs with their Pu sealed within the pressurized reactor vessel and fuel cladding and protected by the high fuel radiation dose. In this MSR concept, all fuel salt processing steps would be performed within the citadel with only mixed fuel salt entering or leaving. The citadel would be a permanently sealed (e.g., via reinforced concrete) unit that is very difficult to open and not intended to be opened as part of operations or maintenance. The citadel would be replaced as a unit when interior components break or wear out. The cell level replacement concept is functionally equivalent to the black cells currently employed at waste processing facilities. Although the individual fuel salt processing steps were demonstrated during the historic MSBR program, they have never been integrated into a system. Moreover, the technology for fully remote operations has substantial development remaining. More detailed system-level designs and operational plans would be needed to construct a remote operations technology maturation plan. Additionally, the historic MSBR program intended to employ a pure Th-U fuel cycle. Although the high radiation dose in the mixed fuel salt would provide substantial proliferation resistance, the U in the fuel salt would be a greater proliferation concern than mixed U isotope fuel salt. Additional proliferation resistance could be provided by including sufficient ^{238}U in the fuel salt. One principal objection to including ^{238}U in the fuel has been the resultant production of a long-lived, heavy actinide waste stream. Modern designs would fully recycle all actinides internally to avoid producing actinide waste until the termination of MSRs as a reactor class.

Fuel Salt Transfer

Both Th-U fuel cycle concepts require transferring fuel salt into and out of the active circuit. Multiple different design options are possible in accomplishing the fuel salt transition while minimizing power generation disruption and stress on plant components. One candidate method is to employ a gooseneck-type pipe connection at the bottom of the lower fuel salt plenum. Fuel salt would be held in the critical loop by gas pressure above the gooseneck during normal operations. Fuel salt would be removed from the critical loop by applying positive gas pressure from the top of the fuel salt loop. Each transfer methodology has its own distinctive hazards. Designs that employ batch fuel salt exchange instead of continuous side stream fuel salt exchange would not generate power during the transfer. Batch and continuous fuel salt transfer designs will require passively safe cooled storage of recently used salt because the extremely radioactive fresh fuel salt would be more difficult to chemically process. The need to store fuel salt concentrate in a critically safe configuration while its ^{233}Pa decays to ^{233}U is another distinctive hazard of the Th-U fuel cycle.

4.1.1.3 Fast Spectrum U-Pu Fuel Cycle Technology Hazards

MSRs can be designed to operate on a fast neutron spectrum. The optimal fast spectrum fuel cycle depends on the reactor mission. One beneficial characteristic of fast spectrum MSRs is their ability to consume minor actinides. Thus, one anticipated role for fast spectrum MSRs is as a waste burner. Most of the distinctive waste-burning hazards in fast spectrum MSRs are in fuel salt preparation, as mentioned in Section 4.1.1.1. This section focuses on the hazards—especially the fuel cycle process colocation hazards—of fast spectrum MSRs operating on a U-Pu fuel cycle.

Because the neutron yield per fission increases substantially with increasing energy, one important design objective for a fast spectrum MSR is to harden the neutron spectrum. The neutron yield per fission that results from fast neutron interaction is substantially larger for ^{239}Pu than for ^{233}U. Consequently, most fast spectrum MSRs are anticipated to operate predominately on the U-Pu fuel cycle. The fast neutrons breed (primarily through neutron capture in ^{238}U) and consume (primarily through ^{239}Pu or ^{235}U fissions) actinides. Representative components of a fast spectrum chloride fuel salt would be NaCl and UCl_3 along with those bred in $PuCl_3$ and FPs; $MgCl_2$ and/or KCl would be other typical constituents. Some proposed designs jointly employ the Th-U fuel cycle along with the U-Pu fuel cycle because of the higher solubility of Th in some carrier salts. With a sufficiently hard neutron spectrum, fast spectrum MSRs can operate in breed-and-burn mode in which they are refueled with U_{nat} fuel, which provides the means to produce and fission ^{239}Pu.

Most fast spectrum MSR conceptual designs have employed chloride fuel salt because some chloride salts can include substantial quantities of actinides while maintaining an adequately low melting point and because Cl has decreased moderating power relative to F. However, chloride salts can be highly corrosive, especially if contaminated with moisture. Chlorine has two stable isotopes: ^{35}Cl and ^{37}Cl. The lighter isotope has a somewhat higher parasitic neutron absorption, although recent evaluations indicated that the cross section difference may be smaller than previously anticipated [68]. Also, one Cl activation product, ^{36}Cl, is a long half-life energetic beta emitter. Mixed chloride/fluoride salt can have advantageous properties [69], so mixed halide melts have also been considered [70]. UF_4-UCl_3 salt mixtures have high actinide loadings, and some compositions have a melting point below 600°C, allowing the use of engineering alloys in direct contact with the fuel salt. Fluorine is more electronegative than Cl, shifting the overall melt redox condition. However, much less is known about the thermochemical and thermophysical properties of mixed chloride/fluoride salts. Fluoride fuel salts have also been considered for fast spectrum MSRs. The European molten salt fast reactor is a fluoride salt fast spectrum design [71]. One limitation of using fluoride salts in fast spectrum MSRs is that fluoride salts have limited actinide solubility [72], which necessitates higher operating temperatures and results in a more thermal neutron spectrum. The higher solubility of ThF_4 in some fuel salts provides an incentive to employ a Th-U fuel cycle despite the fact that ^{233}U has a lower fission neutron yield for fast neutrons than ^{239}Pu.

Neutronic efficiency is also important to fast spectrum reactors. Fast spectrum reactors require approximately eight to 10 times more fissionable material to maintain criticality than their thermal-spectrum counterparts because of the substantially lower fast spectrum fission cross sections. In a fast reactor neutron spectrum, the neutron yield per absorption increases substantially with increasing neutron energy. The actinide atoms are only a portion of the fuel salt. The need to include a carrier salt to lower the fuel salt melting point decreases the actinide density of the fuel salt. Increasing the amount of non-actinide materials in the reactor core softens the neutron spectrum, decreasing the plant neutronic efficiency. Apart from UF_4-UCl_3 mixtures, the actinide density of fast spectrum fuel salt tends to be a quarter or less of that of solid UO_2 fuel, necessitating cores that are substantially larger than solid actinide oxide fuel fast reactors. The large fissile material requirements and low actinide density can result in large critical regions that contain substantial fissile resources, especially for unreflected cores [73]. The melting point of useful, single-component actinide halide salts is too high to be compatible with conventional alloys (e.g., the melting point of UCl_3 is 837°C). Ceramic or refractory fuel containers or directly cooled containers (e.g., fuel tubes) could enable higher fuel salt temperatures and eliminate the need for a carrier salt, substantially increasing the actinide density. The large fissile material requirements to maintain criticality place additional value on minimizing the amount of fuel salt outside the critical region. Consequently, integral or in-core heat transfer (e.g., fuel tubes) designs are advantageous for fast spectrum systems.

Mechanical fuel damage can eventually necessitate replacing solid fuels, so one important advantage of liquid-fueled MSRs is that they avoid this damage. With a sufficiently hard neutron spectrum, MSR fuel salt could continue to be used indefinitely without chemical processing. However, both insoluble and gaseous FPs inherently separate from the fuel salt during use. Current-generation engineering materials do not support sufficiently high temperatures without container cooling to enable breed-and-burn strategies with reasonable core sizes with the possible exception of mixed chloride-fluoride salts. Designs in which the fuel salt circuit is immersed in coolant (i.e., directly cooled) can maintain their container materials at lower temperatures while keeping the fuel salt hot. However, direct cooling designs incur additional startup and shutdown complexity because of the need to avoid freezing the fuel salt. One potential hazard of directly cooled containers is vulnerability to loss of forced cooling. Under normal operating conditions, the container material temperature would be near that of the coolant because of the effective heat transfer provided by forced coolant flow. However, if forced coolant circulation ceases without fuel salt draining, then the resultant container temperature profile could result in significant stress.

Fast spectrum MSRs will likely employ a sufficiently high atomic mass, low-absorption neutron reflector to minimize neutron leakage and decrease the required amount of fuel salt. The sufficiently high atomic mass causes energetic neutrons to take many collisions to thermalize. The neutron reflector can be internal or external to the critical region. Fuel salt in tubes cooled and reflected by liquid Pb is a representative in-core reflector configuration, whereas a reactor vessel submerged in a pool of high atomic mass coolant would be a typical external reflector configuration. Lead has been considered a useful MSR reflector, shield, and coolant material for decades [74], [75] because of its high atomic mass and low neutron absorption. However, lack of an adequately corrosion and radiation damage–resistant wall material between the fuel salt and Pb has limited the rate of concept development. Sodium's relatively low boiling point limits its use as a fast spectrum MSR reflector or coolant.

The high actinide fuel salt density results in higher power density. Higher power density results in increased radiation damage to nearby SSCs, as well as increased decay heat and fission gas production. Increased radiation damage decreases SSC lifetimes. Cladding material radiation damage remains a crucial technical issue for sodium-cooled fast reactors [76]. Radiation damage will similarly be a significant issue for SSCs in the high radiation field produced by a fast spectrum MSR. Fast spectrum MSRs will likely employ a component replacement strategy to minimize the potential for fuel salt containment-layer breaches. Additionally, fast spectrum MSRs will likely operate the fuel salt at a pressure lower than its external environment to minimize the consequences of barrier failure. Fast spectrum MSRs must be able to remove the fuel salt from its container and replace and store the fuel salt container materials. Colocated mechanical manipulation components and radioactive waste handling systems will result in additional plant hazards. However, component replacement will likely be performed during a maintenance outage, significantly reducing the potential accident consequences. Nevertheless, the inadvertent or inappropriate use of mechanical manipulators could damage other SSCs, including safety-related SSCs. Also, fuel salt–wetted SSCs will be highly contaminated and activated and will require the ability to reject decay heat, which will require additional colocated SSCs. Removing degraded components from containment and introducing new ones will require opening the exterior containment layers, which will also introduce additional operational hazards. Installing, connecting, and verifying replacement components also introduces additional hazards. Safety-focused design and operational planning will continue to be important aspects of maturing MSR technology.

Fast spectrum MSR fuel salt has an increased potential for inadvertent criticality compared with thermal-spectrum fuel salt because of its increased fissile material loading. The potential to increase the reactivity of the fuel salt through increased moderation is a distinctive risk for fast reactor fuel salt. The potential for inadvertent criticality exists throughout the fuel salt life cycle both in transport and storage. Moreover, accidents such as spilling the fuel salt during processing can be aggravated by the moderation provided by shielding materials, such as concrete. The higher fuel salt power density that results from the increased actinide loading also increases the decay heat removal requirements, necessitating substantially larger decay heat removal components. Used fuel salt processing remains vulnerable to radionuclide releases, mostly because of decay heat production and the consequent potential for radioactive vapor or aerosol generation. Used fuel salt processing will require adequate radionuclide containment but would not necessarily be performed within the same containment structures as the reactor. The liquid state of the fuel decreases the difficulty of transferring the fuel to an adjacent facility that incorporates barriers from the reactor. Fuel salt will likely be allowed to cool for a few days before processing to minimize the decay heat management requirements during processing and additional gas generation from precursor decay. The used fuel storage tank may be within the reactor containment or the fuel processing facility containment. Reactors that employ fuel salt in tubes may keep the used fuel tubes in a noncritical region of the coolant pool while their decay heat production decreases sufficiently to facilitate manipulation. Used fuel salt processing will increase the salt's nuclear reactivity. Adequate defense against inadvertent criticality must be provided with the fuel salt in its most reactive configuration.

Fast spectrum MSRs that are not operating on the Th-U fuel cycle lack the requirement to remove ^{233}Pa before neutron capture, and the neutron absorption cross sections of FPs are much larger for thermal neutrons, which are much less prominent in fast spectrum reactors. Consequently, although some fast spectrum MSRs may incorporate chemical fuel salt processing, the processing would be much less intensive than for an equivalent power thermal-spectrum MSR. The goal of the fuel salt processing would be to remove elements that raise the fuel salt melting point or decrease the solubility of actinides, corrode the container, soften the neutron spectrum, or parasitically absorb neutrons.

Fast spectrum MSRs will generate larger quantities of fission gases per unit fuel volume because of their higher power densities. Fission gases produce significant quantities of decay heat. Fast spectrum MSRs are less likely to employ a sparge gas to extract fission gases from the fuel salt because of their much lower sensitivity to thermal neutron capture in ^{135}Xe. Because the fission gases initially produce a few percent of fission power generation, the plant thermal efficiency would be increased by allowing the fission gases to initially transfer their decay heat to the fuel salt. Consequently, most plants are anticipated to incorporate a tortuous escape route for the fission gases (e.g., coiled tubes) to facilitate decay heat transfer. Krypton-85 ($t_{½} \approx 10.8$ years) and ^{133}Xe ($t_{½} \approx 5.25$ days) are the longest half-life fission gases and will consequently be the largest fraction of the gases emerging from the fission gas delay tubing. The primary requirements for the emerging fission gases are to provide adequate cooling and maintain containment. The cooling and containment will likely be provided in an cell isolated from the fuel salt system to reduce the potential for accidents in one system to affect the other. Once adequately separated from the fuel salt, the fission gases will likely be placed in a sequence of cooled tanks for storage for a few months. After storage, the nearly pure ^{85}Kr would likely be placed in pressurized cylinders and then sold or allowed to decay for several decades. The primary associated hazard of the fission gas storage occurs when water coolant interacts with hot fuel salt, causing a pressure spike that challenges containment. The primary mitigation method is to provide substantial separation between the fission gas storage and liquid fuel salt.

One challenge for transferring radioactive materials between containment cells is the additional risk posed by the piping, which provides a pathway across containment layers. Criteria 54 and 57 of 10 CFR Part 50 Appendix A directly address requirements for piping that penetrate reactor containment, providing guidance on issues such as leak detection, testability, and adequate system isolation. Although MSRs require somewhat modified language to capture the technical differences between LWR pressure boundaries and the multiple layers of functional containment at MSRs, the safety element of the piping penetrating containment GDCs are applicable to MSRs.

4.1.1.4 Once-Through, U-Pu Thermal-Spectrum MSR Technology Hazards

Thermal-spectrum reactors require moderation. Graphite is the only known fuel salt compatible, high-temperature moderator. MSRs can operate on a once-through U-Pu fuel cycle similar to that of the existing LWR fleet. The inherent characteristics of MSRs influence core design optimization. Because fissile material and fuel salt are expensive, MSR designers are motivated to maximize fissile material use. Additionally, radiation damage to graphite is the core material lifetime-limiting phenomenon in graphite-moderated MSRs, so minimizing radioactive waste costs is another design objective. Fuel salt is anticipated to be reused until it accumulates so much contamination (i.e., absorbing FPs) that it no longer supports criticality. The fission to absorption ratio in fuel salt increases as the neutron energy decreases, providing incentive to employ the lowest energy of a neutron spectrum possible. Larger amounts of graphite decrease the mean neutron energy. The core volume for an optimized MSR will be much larger than that of an equivalent power water-cooled reactor. The neutron migration length in graphite is roughly 10 times greater in graphite than in water. Consequently, the graphite moderator volume needed to provide equivalent thermalization is roughly 10 times larger than that of water. The graphite volume in a graphite-moderated MSR will be further increased by the need to employ a thick reflector to minimize the neutron flux (i.e., activation and radiation embrittlement) of the reactor vessel. Alternately, a graphite-moderated reactor could employ higher enrichment fuel to achieve higher burnup. However, the larger amounts of graphite enable a longer graphite lifetime by spreading the radiation damage over a larger volume.

Radiation damage to graphite was mentioned as a specific limitation in the US Atomic Energy Commission's evaluation of MSR technology in 1972 [77]. Radiation damage to graphite was also central to ORNL's decision in 1967 to shift from a dual-fluid to a single-fluid MSR and was a central rationale in decreasing the power density of the MSBR's conceptual design [78], [79]. Excessive radiation damage to graphite presents a potential safety issue because fuel salt can penetrate into graphite via cracks with sufficient radiation damage. Increasing the amount of fuel salt in the core changes the system reactivity, potentially affecting the controlling reactivity FSF. Graphite radiation damage also causes mechanical distortion and eventually mechanical damage to the graphite, defining the fuel salt flow channels. Damage that impedes the ability of the fuel salt to provide adequate cooling adversely affects the achievement of the adequate cooling FSF.

Although it may be technically possible to refabricate used graphite locally, used MSR graphite reprocessing technology requires a significant amount of complex processing with highly radioactive materials, as well as high-temperature graphitization [80]. The technology seems to add so much cost and potential risk to an MSR that it is unlikely to be performed locally but may eventually be performed at a centralized facility that supports multiple plants.

Some FPs penetrate into graphite surfaces, especially those with gaseous precursors [81]. Additionally, a fraction of the noble metal FPs deposit onto the graphite surface [82]. Also, the neutron flux results in ^{14}C production within the graphite both from neutron capture by ^{13}C and $^{14}N(n.p)^{14}C$ reactions with trapped N. The combination of contamination and activation results in used graphite that exceeds allowable contamination levels for near-surface disposition (i.e., used MSR graphite is anticipated to be GTCC waste based upon 10 CFR 61.55 limits) under existing regulations. The NRC is currently in the process of promulgating requirements for the near-surface disposal of GTCC waste [36], which is anticipated to enable the near-surface disposal of MSR graphite once packaged into an adequately durable waste form. This is discussed further in Section 4.1.1.6.

Moderator graphite may be replaced one or more times during a thermal-spectrum MSR plant lifetime. Removing graphite from the reactor vessel would entail the mechanical manipulation of highly contaminated and activated material. However, because replacement would be performed within containment with the reactor defueled, the manipulation would not have significant risk for radionuclide release outside the plant but would entail a significant potential for spreading contamination within the plant and for exposing plant staff. Alternatively, the graphite and reactor vessel could be replaced as a unit with the reactor vessel serving as the container for the contamination. Although substantially contaminated, the used reactor graphite would not be anticipated to generate significant quantities of heat a few months after fuel salt removal.

MSRs operating on a once-through fuel cycle must store used fuel salt and salt-wetted SSCs until they can be transferred to an independent storage facility. Used fuel salt may be transferred to a separate containment cell to minimize the potential for interaction with the active fuel salt and facilitate cooling. Used reactor vessels that contain used graphite would be large enough that opening containment sufficiently to move them to a radioactive waste handling facility and subsequently resealing containment would be a significant plant design element. For the first few years, the reactor vessel would likely be stored in a water pool or a capped trench. Additional local processing of the used reactor vessel will depend on the amount of shielding necessary for transport to a final disposal site. The fuel salt space within the otherwise graphite-filled reactor vessel may be backfilled with grout to produce a long-term, stable, non-water–soluble waste form [83]. Overall, the used reactor vessel would be anticipated to be prepared to meet the requirements for near-surface disposal under the ongoing updating of 10 CFR 61. However, colocating the waste preparation and handling operations would not significantly affect an adjacent reactor apart from needing to shut down the reactor when the containment is opened to introduce or remove large components.

4.1.1.5 Additional radioactive material handling

Operating most MSRs will involve additional radioactive material handling compared with solid-fueled reactors. The additional radioactive material handling derives from the extended distribution of radionuclides at MSRs combined with the usefulness of manipulating the chemical content of the fuel salt (e.g., filtering suspended solids, refueling, FP separation). Many MSR radioactive material handling hazards are analogous to those within the existing fleet. For example, the hazards of MSR fuel salt filter changeout are closely related to those of resin bead replacement at LWRs. Most radioactive material handling (e.g., salt-wetted component replacement) will be performed with fuel salt removed. Consequently, the material handling has limited potential for off-site consequences.

The release of fission gases from the fuel salt is an especially important MSR property. The fission gases initially contain a significant fraction of the decay heat and are more mobile than the liquid salt fuel. The amount of gas emerging from the fuel salt depends on the reactor design. Although all fissions generate roughly the same amount of fission gas, one method for reducing the quantity of neutron-absorbing ^{135}Xe within the core is via sparging with inert gas. Sparging significantly increases the amount of gas emerging from the fuel salt. However, sparging is only one method for disengaging dissolved gases from the fuel salt. Reactor designs that employ alternate mechanisms or elect to leave the fission gases in the fuel salt (e.g., fast spectrum designs) will have much lower quantities of gases emerging from the fuel salt. The emerging gas stream also includes aerosols and vapors (e.g., CsI) that can deposit onto surfaces. Also, the fission gases decay into isotopes that are solid at relevant temperatures and that will initially be suspended in the gas mixture. The suspended solids can subsequently deposit onto surfaces, progressively building up material and eventually inhibiting flow and requiring cooling. One option for filtering suspended materials from the fission gases is to pass the emerging gases through a filter bed. Both solid (C particle) and liquid hydroxide filter beds have been proposed. Each filtering technique has distinctive hazards for reactor operations. Liquid hydroxide backflowing into the fuel salt would result in adverse chemical reactions. A C filter bed would employ high surface area C, which is combustible. Furthermore, the intense radioactivity of the trapped FPs provides an inherent ignition source. Exposing the hot C filters to flowing air could result in a fire. Hence, preventing O access to the hot C filters will be an important design consideration.

The first stages of the filter bed will contain substantial amounts of decay energy and thus must be cooled. The cooling process may introduce additional hazards into plant operations. For example, if cooling ceases, the filter may reach a sufficiently high temperature to melt its container, challenging the radionuclide containment FSF. Alternately, if water is employed as a cooling fluid for the filter media, insufficient cooling may result in steam generation, increasing the system pressure and damaging the fuel salt off-gas containment. Solid filter change-out may also introduce hazards to reactor operations. Although multiple parallel filter bank lines would likely be employed to enable change-out while at power, mechanical operations introduce the potential for error by allowing fission gas leaks into the next containment layer or plugging the reactor gas vent line and allowing the fuel salt container to pressurize.

The gas stream emerging from the filter bed will still require containment because some fission gases have longer half-lives. If sparging was used to strip the ^{135}Xe, then gases must be held for decay for at least a couple of days to allow for a decay before recycling for sparging. Following filtering and a few days of decay, the emerging gas stream will mostly comprise ^{133}Xe, ^{85}Kr, and any carrier gas—likely

He—employed in the stripping. In designs that do not employ a carrier gas, the emerging fission gases will likely be bottled and maintained in a cooled environment for a couple of months and then potentially recompressed for longer term ^{85}Kr storage. Designs that employ a carrier gas would divert most of the mixed gas back to sparging and hold the remainder in a cooled environment for ^{133}Xe decay. The remainder of the gas contains a mix of carrier gas, and ^{85}Kr would then likely be bottled for longer term storage. Although the first few bottles of stored gas would contain a significant fraction of carrier gas, over time, dilution would cause the sparge gas stream to become largely ^{85}Kr.

4.1.1.6 Integrated Waste Processing

The requirement for indefinitely storing used fuel salt and the long-term radiolytic instability of fuel salt also increase the amount of required radioactive material handling. All US nuclear plants require the ability to store used fuel indefinitely following the end of plant operations to accommodate the possibility that a repository never becomes available [32]. Additionally, the limited lifetime of fuel salt–wetted components and the resultant need to replace the main components multiple times during plant lifetime also result in additional radioactive material handling. Used liquid fuel can be transferred to a nearby facility for processing. Processing operations are anticipated to be performed in separated facilities to minimize the potential for adverse interactions between the reactor and fuel salt processing. Fuel salt intended for reuse may be processed shortly after removal from the active circuit. However, fuel salt intended for disposal will likely be held for a few years before processing to minimize the heat generation from the waste and radiation damage to salt processing equipment. In both cases, ensuring adequate decay heat removal will be a significant concern for the first few years following removal from use. Used fuel salt and its cooling mechanisms will likely be substantially separated from the active fuel salt during storage to minimize the potential for adverse interactions.

Used fuel salt will require decay heat removal, radionuclide containment, and shielding. Fully passive plants will require the ability to passively reject decay heat following shutdown from full-power operation. The high radiation and heat production from recently used fuel salt increase the technical difficulty of maintaining adequate cooling under accident conditions during fuel salt transfers. Hence, recently used fuel salt will likely be processed or stored relatively near the active fuel salt circuit and perhaps within a common exterior containment structure. Although fuel salt processing or storage could adversely affect reactor safety, it will be governed by reactor regulations (e.g., 10 CFR Part 50). The reactor portions of nuclear power regulations are the most restrictive because of the higher consequences of reactor accidents. The ability to adequately separate the storage or processing of used fuel salt from the reactor will be crucial to shifting the safety requirements to nonreactor regulations.

Although the mechanisms employed to provide used fuel salt cooling, containment, and shielding will vary substantially from existing LWR practices, many aspects of the regulations have been PB for decades. For example, GDC 61, “Fuel Storage and Handling and Radioactivity Control,” requires suitable shielding for radiation protection and residual heat removal systems with reliability and testability that reflects their importance to safety. However, some aspects of the regulatory language are prescriptive based upon historic LWR fuel characteristics. For example, because the fuel salt can also serve the heat transfer medium while it remains liquid, language such as “prevent significant reduction in fuel storage coolant inventory under accident conditions” in GDC 61 is confusing and inappropriate. One potential liquid salt decay heat transfer configuration would be to employ a parallel set of heat pipes to reject decay heat to the air. In such a design, the heat pipe hot-end would be thermally connected to the hot salt, and the cold-end would be thermally connected to the atmosphere. The NRC has already endorsed a PB alternative to the prior language through its endorsement of RG 1.232 [18], which includes ARDC 61. Instead of requiring avoiding loss of coolant inventory, RG 1.232 [18] requires the prevention of significant reduction in the cooling function.

Fuel salt–wetted SSCs and those within a high-neutron flux environment will likely degrade over time and may need to be replaced one or more times during plant lifetime. Other SSCs may acquire significant contamination during plant lifetime. Part of a decommissioning plan will determine whether and how to decontaminate surface-contaminated SSCs. Existing regulations, such as 10 CFR 50.33(k), for a power reactor license indicate that an application must contain reasonable assurance that sufficient funds will be available to decommission the facility. The technical basis for developing the reasonable assurance is provided in 10 CFR 50.75(a)(4), which indicates that the determination of sufficient funds may be based upon a cost estimate. However, only guidance for developing a decommissioning cost estimate for LWRs is provided. The decommissioning costs for an MSR will be different and will likely be higher than those for an equivalent power LWR, especially if the MSR includes additional fuel cycle facilities as part of a single license. As discussed in Section 2.1.2.14, the proposed text for 10 CFR Part 53 will require a site-specific in-depth decommissioning cost estimate.

Used fuel salt–wetted materials will constitute a significant waste stream from MSRs. The waste stream will include the container materials (e.g., piping and vessels) and other salt-contacting SSCs (e.g., graphite and pump impeller blades). Materials located near the reactor core will become activated because of neutron exposure. Used fuel salt–wetted SSCs will likely be classified as GTCC waste because of the FPs deposited onto their surfaces and, in the case of graphite, because of the penetration of radionuclides into pores and cracks. Also, MSR graphite and structural materials could become activated to GTCC levels during use. Most fuel salt hydraulic components will have radioactive materials on their inner surfaces. However, their outer surfaces will likely be uncontaminated, providing impetus for interior radionuclide stabilization.

The NRC is currently engaged in rulemaking to promulgate requirements for the near-surface land disposal of GTCC and transuranic waste (SRM-SECY-20-0098 [37]). Waste disposal pathways for MSRs could be substantially affected by the ongoing rulemaking. Used MSR reactor vessels are similar to the emptied radioactive waste tanks on DOE sites in that they may have significant quantities of deposited, difficult to remove radionuclides on their inner surfaces. DOE has selected tank backfilling with grout as an acceptable process for creating mechanically and chemically stable and robust waste forms at multiple sites. Although further technical and regulatory evaluation is recommended, backfilling MSR salt hydraulic components with grout will likely create waste forms capable of meeting the technical objectives necessary for near-surface disposition. The primary technical difficulty with a grout-in and shallow burial approach for MSR hydraulic components is that reactor vessels may be large and, once filled with grout, sufficiently heavy to make subsequent transportation difficult. Hence, backfilling with grout may be delayed until the final disposition site. One crucial but unresolved cost question is the amount of subsequent monitoring that will be required for GTCC waste forms in shallow (i.e., at least 8 m below grade) burial sites.

The actinides could be separated from the used fuel salt as a precursor element of some waste form preparation pathways. Actinide separation from fuel salt would likely be based on the Al metal transfer process described in Section 4.1.1.1 for fluoride salts or an electrochemically driven process similar to that employed in sodium-cooled fast reactor fuel processing for chloride salts. Separating the actinides from the remainder of the used fuel salt enables actinide reuse and changes the regulatory requirements for the remainder of the fuel salt.

The halogen salts have high solubility in water and thus are generally unsuitable as a direct waste form in most geological repositories. Known glasses with characteristics suitable for serving as waste forms cannot incorporate high loadings of halide atoms. Because waste volumes are the primary cost driver for geological repositories, multiple strategies have been proposed to provide higher used fuel salt loading into the waste form [84], [85]. Overall, the two general approaches for incorporating higher halogen loading in an adequately stable and durable waste form are to (1) encapsulate halogen salts as a crystalline phase within a robust matrix (e.g., waste glass or metal) or (2) strip the halogen atoms from the used fuel salt and replace them with O before creating the waste form. With sufficiently high halide particle loadings in either the glass or metal, the soluble halide particles interconnect and can be dissolved by water, limiting the overall density achievable with halide particle-level encapsulation. Cermet waste forms have a long history with oxide-based HLWs [86]. However, prior cermet waste evaluations have employed nonleachable oxide ceramics [87].

Dehalogenation will inherently create halogen-bearing secondary waste streams. Fluorine does not have a long-lived radioisotope, so the main technical issue with the separated F is ensuring that it is adequately free from radioactive contaminants so that it can be reused as a low-radioactivity material. Separated Cl contains ^{36}Cl, which is a long-lived radioisotope produced primarily by neutron reactions with Cl. Disposal requirements for the Cl depend on the radionuclide concentration. Chlorine-36 is not included in the 10 CFR 61.55 long-lived radioisotope table, so no clear guidance is currently available as to what *sufficient concentrations* means in the Nuclear Waste Policy Act test regarding whether the material would be considered HLW and thus require deep geologic disposal. Reusing the Cl in the creation of additional actinide chloride fuel salt would avoid needing to dispose of the Cl.

Halide fuel salt freezes into a polycrystalline mass. Once fuel salt cools following use to well below the solidus, recombination ceases to dominate radiolysis and then generates halogen gases [88]. Although most gases remain trapped within the crystals, gases that form at or near the surfaces of the many small salt crystals can migrate along their interconnected grain boundaries. Particle-level encapsulation is advantageous for gas-forming materials because gas pressure will build up over time around polycrystalline materials due to gas migration along the interstices between the crystals. Encapsulating the halogen-bearing crystalline phase within a durable oxide glass prevents the gas migration [89].

Four representative used fuel salt treatment processes are briefly outlined to illustrate the potential hazards of processing adjacent to the reactor building: (1) dehalogenation and conversion to a borosilicate glass, (2) dehalogenation and conversion to an $FePO_4$ glass, (3) conversion to a glass ceramic composite with an embedded fluorite ceramic phase within a Na aluminoborosilicate glass, and (4) conversion to a ceramic metal composite (i.e., cermet).

Dehalogenation and conversion to a borosilicate glass are applicable to chloride, fluoride, and mixed halide salts [90]. Dehalogenation accompanied by oxidation has been central to creating stable waste forms from halide-bearing materials for decades [91] and has been considered as an alternative for the stabilization of MSRE fuel and flush salts [92]. In the glass material oxidation and dissolution system, the used fuel salt is introduced into a Pb borate glass melter. The halides interact with the lead oxide to form PbF_2, which has a boiling point of 1,293°C, or $PbCl_2$, which has a boiling point of 950°C. These compounds volatilize from the melt and are subsequently recombined with NaOH to form $Pb(OH)_2$ and either PbF_2 or $PbCl_2$. Silicon dioxide and other glass-forming materials are then added to the melt to create a durable glass form.

Dehalogenation and conversion to an $FePO_4$ glass is also applicable to chloride, fluoride, and mixed halide salts. However, different process steps are involved in removing the chloride and fluoride ions. One route for removing F is to first convert the fluoride salt to a nitrate by dissolving the used fuel salt in dilute HNO_3 and then boiling off the F as HF [93]. The resultant nitrate salt then follows the same path as a chloride salt being introduced into a stirred glass melter along with H_3PO_4 and FeO to form an $FePO_4$ glass [94]. Alternately, $NH_4H_2PO_4$ could be used to remove Cl from chloride salts as NH_4Cl [95], [96].

One additional option for creating a stable waste form from halide salts is to segregate the halide into a crystalline halide phase embedded within a high-durability glass phase that contains FP and potentially actinide oxides. Trapping the F atoms in a stable alkali fluoride phase within an aluminoborosilicate glass is the basis for the "synroc" process developed at the Australian Nuclear Science and Technology Organization (ANSTO) [97]. The ANSTO process grinds the fuel salt into a powder and mixes it with glass, forming oxides, calcining, and then hot isostatically pressing the material. Important advantages of the ANSTO process include its higher maturity and lower amounts of secondary waste generation. The fuel salt stabilization would likely be performed in an adjacent facility once the salt has cooled for a few years. Consequently, the facility would not have significant colocation-based hazards because of its proximity to the active reactor or used fuel salt storage.

Halide salt particles could alternatively be embedded as a ceramic phase within a metal matrix. Copper is a likely candidate material for the metal matrix because of its chemical robustness, high-thermal conductivity, and suitable mechanical processing parameters. The basic process steps for creating a Cu metal matrix composite with halide salt with or without actinide halides are as follows: (1) mill the Cu along with the fuel salt to form a powder mixture, (2) place the mixture within a Cu shell, (3) compact and weld shut under vacuum, and (4) extrude or hot press. Extrusion has the advantage of developing a mechanically derived chemical bond at a lower temperature than required for hot pressing. However, optimal mass fractions of the Cu powder and halide salt, particle sizes, and processing parameters have yet to be determined. An alternate method for preventing radiolytic gas migration from the chloride salt is to first recrystallize the melt to produce homogenous phases (i.e., zone-refine the melt) and then grow each phase of the chloride salt into one large crystal, which would subsequently be encapsulated to prevent moisture access.

Although the dehalogenation, mechanical alloying, and glass-forming waste form preparation technologies are based on known chemical reactions and known processing technologies, none are adequately technically mature for a detailed hazard assessment. Each technology involves high-temperature processes, some with caustic chemicals, and the potential for airborne release of radionuclides. Adequate separation between the activities must be maintained to minimizing colocation hazards of fuel salt waste form preparation and reactor operations. Waste preparation would likely be performed in a facility adjacent to the reactor with separate containment and limited shared services.

All the envisioned fuel salt waste forms are stable and durable. The very high durability of stabilized MSR waste forms, both metal and glass, may change them from immediately being a waste to being a useful heat and irradiation source. Although used LWR fuel could not reasonably be used to provide process heat because of the security requirements of its fissile content and the potential for contamination spread, actinide stripped and chemically stabilized used MSR fuel lacks either hazard. Significant heat production will continue for decades to centuries. Stabilized MSR salt forms could provide water sterilization and heating for municipal or industrial applications.

Although isotopic separation processes for ^{7}Li and ^{37}Cl currently remain either environmentally unacceptable (e.g., Hg amalgam process), expensive, or immature, the effectiveness of multiple potential liquid-phase isotope separation technologies that range from crystallization under magnetic fields [98] to anion exchange chromatography [99], [100] suggests that acceptable cost, large-scale isotope sources will become available with adequate market demand. Hence, bulk isotope recovery from used fuel salt is not considered likely in the near term. One potential caveat to the lack of separation is that if Cl isotope separation does become inexpensive, then the Cl emerging from de-halogenating the fuel salt could be isotopically separated to remove the ^{36}Cl, which could reduce the hazard category.

4.1.2 Advantages of Close Coupling Fuel Cycle Processes

For the operating LWR fleet and most other reactor concepts, fuel cycle operations are performed at separate sites significantly away from the reactor site. This is done primarily to service a larger industry with multiple reactor units and to centralize operations, and it has led to separate hazard and safety assessments for each reactor and fuel cycle facility. For multiple operational and physical reasons, liquid-fueled MSRs are anticipated to have their frontend, operations, and backend fuel cycle processes connected to and at the reactor site. This report focuses on the additional regulatory and safety considerations of combining these at one site. However, combining these processes at one site eliminates many safety and security risks that result from having two distinct sites.

One option for MSRs is to use fissile materials recycled from used water reactor fuel as their fuel source. This would require a facility and equipment to remove cladding, fittings, and other hardware to obtain the nuclear material and FPs. This facility could be adjacent to or at the water reactor site, but this is not a requirement, whereas having online salt processing systems at the MSR site is a requirement. For MSRs that derive their nuclear fuel from used water-cooled reactor fuel, additional hazards are introduced through the new facility operations and potential transportation between sites. However, these processes reduce the total volume of used LWR fuel, so some potential benefit toward the reduction of this hazard should also be considered. Reducing used LWR fuel reduces the economic burden on the reactor site currently housing the used fuel.

Used fuel salt processing could be performed at a central facility with the stabilized waste, then returned to the reactor site if alternate disposal facilities are unavailable. A centralized facility model optimizes the use of capital assets because fuel salt waste processing would be infrequent at a plant site. However, transporting used fuel that has only been stored for a few years also incurs risks and expenses. Another alternative would be to perform fuel salt waste form preparation in one facility on a multi-unit site.

One advantage that significantly reduces the risk associated with transporting used fuel and/or reactor byproducts is that the used fuel currently stored at US LWR sites is expected to be transported to either a permanent repository or reprocessing facility. Internationally, this is mostly already performed. Although shipping packages for used fuel undergo thorough safety assessments, tests, and inspections, the probability of an accident is never eliminated. The consequences of a transportation accident are potentially more complex because the environment and surrounding populations are dynamically changing. Liquid-fueled MSRs will generate many long-lived fission and activation products. However, the total volume and hazard magnitude (e.g., energy content) is expected to be significantly lower for designs with salt processing.

Another advantage for colocating used LWR fuel conversion with the MSR is that this reduces the security and proliferation risk associated with the transportation of fresh fuel. Fresh fuel for LWRs and other thermal-spectrum concepts require a certain enrichment of fissile material. For some fast spectrum designs, significant quantities of fissile material may be needed for the first core.

Finally, combining fuel cycle activities with reactor operations also reduces the total MSR-related footprint in comparison with the separate facility case, assuming that the separate facilities could operate in such a configuration. The probability and hypothetical consequences of a reactor or fuel cycle process accident could be reduced compared with the separate facilities case. A combined facility would also require only one EPZ.

5. FUEL CYCLE REACTOR OPERATION

For most reactor operation phases, the reactor core and its fuel represent the most significant hazard in terms of source activity and the need for confinement and safety controls. For liquid-fueled MSRs, the distribution of radionuclides is more complicated than reactors with solid fuel in a static position. For example, the fuel salt normally flows between a critical core location and a primary heat exchanger to transport the nuclear heat to power conversion and/or other industrial process applications. These two locations could be in an integral configuration (i.e., the same vessel) or in a modular configuration connected by piping. Additionally, for liquid-fueled MSRs, the distribution of radionuclides is naturally arranged between the molten salt and gas volumes, the latter generally of which generally has a lower source activity but is more transportable in the event of a system boundary breach.

In the United States, the evaluation of core and fuel-related radionuclide hazards is expected to be documented in accordance with those applicable regulations found in 10 CFR 50. Some discussion of relevance to the latest consolidated draft of 10 CFR Part 53 is also included. These safety evaluation processes have been well established for LWRs and are being reconsidered now for advanced SMRs and non-LWRs. Similarly, separate safety evaluation processes have been well established for fuel cycle facilities with pertinent regulations existing mainly in 10 CFR 70. This section identifies and discusses the hazards associated with fuel cycle operations that are unique to liquid-fueled MSRs and are cross-cutting between the regulations and thus are prime candidates for improving regulatory efficiency.

Fuel cycle operations associated with reactor operation typically include online refueling and fuel salt purification, conditioning, or polishing, and polishing could include proliferation resistance processes. Different reactor concepts with different fuel salts and neutron spectrum approach online refueling and fuel salt purification differently. Therefore, the hazards associated with technology could be substantially different. This in turn could result in different decisions about the degree to which safety and other regulatory requirements are satisfied.

The hazards associated with MSR fuel process operations were documented in a 2019 ORNL report [46] and are discussed in Section 4.1.1. Additionally, salt purification and waste processing hazards have been identified and discussed several other reports [101], [102]. These hazards are summarized in Table 1 with some discussion of more recent developments and their connection to the applicable regulations. A more detailed description of these MSR concepts can be found in Holcomb et al. [103].

Table 1. MSR fuel cycle operations.

MSR, company	Power	Fuel salt	Fuel cycle operations
Lithium fluoride thorium reactor (LFTR), FLiBe Energy	250 MWe (Ref. design)	Fuel salt: $LiF\text{-}BeF_2\text{-}UF_4$ (FliBeU) Blanket salt: $LiF\text{-}ThF_4$ (FLiTh)	Online refueling: • Fresh blanket salt is added, which is continuously processed to remove U, which is then added to the fuel salt. Salt purification: • Online chemical process to remove Pa (highly neutron absorbing) from the blanket salt and allow it to decay. • After decay of blanket salt, fluorination is performed to extract U. • Fuel salt is fluorinated to remove FPs and then combined with the U from the blanket stream. • H_2 reduction is then performed to produce purified fuel salt.
Stable Salt Reactor (SSR), Moltex Energy	300–500 MWe/unit	Fuel salt: Chloride based in fuel pins (fixed position) Coolant salt: sodium magnesium chloride	Online refueling: • Fuel assemblies replaced are similar to Canada Deuterium Uranium (CANDU) and UK advanced gas-cooled reactors (AGRs). • No chemical processing as spent fuel management is similar to solid-fueled reactor.
Heavy-water moderated MSR, Copenhagen Atomics	100 MW(t)	Starting fuel: $LiF\text{-}ThF_4PuF_3$	Online refueling: • Continuous chemical processing. • 3–5 year design life.
Thermal molten salt, ThorCon US Inc.	500 MWe (two cans)	$NaF\text{-}BeF_2\text{-}ThF_4\text{-}UF_4$	Online refueling: • Fresh salt is added and continually processed. Salt purification: • Redox control only.
Molten chloride fast reactor (MCFR), TerraPower	800 MWe	Fuel salt: $NaCl\text{-}UCl_3$ Coolant salt: $NaCl\text{-}MgCl_2$	Online refueling: • Natural or depleted U is added during fuel polishing. Salt purification: • No FP removal occurs because of fast spectrum.
Integral MSR, Terrestrial Energy	200 MWe	Fuel salt: Fluoride based	Online refueling: • Seven year core unit life, seal and swap approach. • Used fuel salt is partially reused in subsequent core operational units. Salt purification: • Redox control only.

5.1 REFUELING

As evidenced by the summarized MSR designs in Table 1, both batch and continuous-mode fuel cycle operations are being pursued. For continuous-mode fuel cycle operations, this is referred to as *online refueling*. Waste products must also be dispositioned, as summarized in the next section. Fresh fuel, either fissile or fertile (i.e., breeder-type designs), must be added continuously or at intervals dictated by criticality and other operational parameters. For batch-type refueling operations, this could either comprise a whole-core replacement or a salt-only replacement. For either option, core internals, graphite, or other structures may also be replaced at specified intervals.

Online refueling is easily enabled by MSRs because new fuel salt can be injected and mixed with the current fuel salt in the reactor as part of a purification/polishing system or as a blanket/breed salt. For one concept, the Moltex SSR fuel assemblies are swapped out similar to existing CANDU or UK AGR-type reactors.

Table 2 summarizes the refueling concepts and potential hazards.

Table 2. MSR refueling concepts and potential hazards.

Fuel cycle mode	Type	Description	MSRs	Potential hazards
Batch	Whole-core replacement	While shut down, the whole core vessel is removed, sealed, and swapped with a fresh core vessel	Integral MSR	• Worker and physical safety hazards associated with large component removal and transportation • Radiation hazards and shielding associated with the decay of FPs in the transportation container • Transportation-related hazards, such as environmental hazards, drops, spills, and breaks of the primary fuel volume along the transportation path • Criticality accidents
	Salt-only replacement	While shut down, the entire primary loop is drained with the used fuel salt sealed and then replaced with fresh fuel salt	Copenhagen	• Same as for whole-core replacement; however, the primary fuel salt could be divided into smaller containers, reducing the potential for criticality accidents and shielding concerns. Nevertheless, multiple containers increase the likelihood for transportation-related drops, spills, and breaks. The consequence associated with these events is then lower.
Continuous	Intravenous	Fresh fuel salt is added continuously at power to the primary fuel salt loop	LFTR, MCFR	• High-temperature molten salt processing lines, tanks, valves, and other physical equipment • Direct radiation hazards associate with primary fuel salt in proximity to potential workers • Potentially large (i.e., large relative to other MSR accidents) consequences due to pipe or other component breaks leading to radionuclide release
	Discrete	Fresh fuel comprises discrete pins or assemblies and are replaced at power	SSR	• Potential accidents during fuel handling and transfer • Direct radiation hazards • Chemical and high-temperature hazards with pool salt

5.2 FUEL SALT PURIFICATION, CONDITIONING, OR POLISHING

In addition to the hazards associated with refueling, for any MSR and refueling approach, redox control systems are expected. These comprise a monitoring element and a salt chemical–adding element to the primary fuel system. Typically, these salt chemicals include Be for fluoride salts and Mg for chloride salts.

Fuel sampling is also expected as part of these systems to test and measure the appropriate quantities of additional salts that are needed. As discussed in Section 5.3, hazards associated with these operations are considered to be potentially consequential; however, multiple mitigation strategies can be employed to protect worker health.

5.3 HAZARDS

A detailed assessment of fuel processing hazards is provided in McFarlane et al. [46]. Table 3 summarizes some of the primary hazards as they relate to the fuel cycle operations presented here.

Table 3. Example primary hazards from McFarlane et al. [46].

Physical or chemical process	Key hazards	Judgment/assessment		Mitigation strategies
		Technology Readiness Level, 1 (low), 9 (high)	Severity, 1 (low), 5 (high)	
Fuel loading	• Contamination • Air sensitive • Criticality	5	4	• Inert gas flush • Double barrier containment
Fuel sampling during loading and operations	• Contamination • Air sensitive • Criticality • Be (if present as BeF_2) • Very high radiation • Heat	4	5	• Remote handling • Inert gas flush • Gamma, densitometer, control rod measurements • Double barrier containment
Off-gas sampling and scrubbing with He sparge	• Pressure/flow restrictions • Heat • Very high radiation	2	4	• Remote handling • Rapid measurements for radiation and process variables • Purge gas pressure relief • Secondary containment
Refueling: reductant addition	• Very high radiation • Heat • Corrosive • Flow restrictions in helium sparge • Very high contamination (leakage of volatile FP, actinides in salt) • Be (if present as Be, BeF_2)	2	5	• Remote handling • Rapid measurements for radiation and process variables • Pressure relief • Secondary containment
Refueling: fissile material addition	• Very high radiation • Heat • Corrosive • Flow restrictions in helium sparge • Very high contamination (leakage of volatile FP, actinides in salt) • Accountability	2 (Cl salt), 5 (Fl salt)	5	• Remote handling • Rapid measurements for radiation and process variables • Pressure relief • Secondary containment

Physical or chemical process	Key hazards	Judgment/assessment		Mitigation strategies
		Technology Readiness Level, 1 (low), 9 (high)	**Severity, 1 (low), 5 (high)**	
Salt processing for actinide recovery, FP separation, redox control, salt returned to reactor	• Very high radiation • Heat • Corrosive • Very high contamination (leakage of volatile FP, actinides in salt) • Rad waste generation • Precipitation of U-metal for UCL_3 salts	2	5	• Remote handling • Rapid measurements for radiation and process variables • Shielded waste storage • Secondary containment
Core unloading and spent fuel removal by pressurization	• Very high radiation • Heat • Corrosive • Be (if present as Be, BeF_2) • Very high contamination (leakage of volatile FP, actinides in salt) • Halogens • Pressure	2	5	• Pressure relief • Flow monitoring • Remote handling • Secondary containment • Recycle/waste stream established before operation • Online instrumentation
Waste form preparation and processing: pyroprocessing	• Very high radiation • Heat • Very high contamination • Halogens • Corrosive	2	4	• Remote handling • Secondary containment • Online instrumentation • Recycle/waste stream established before operation
Waste form preparation and processing: hydrofluorination	• H_2/HF, Be • Very high radiation • Very high contamination • Heat • Halogens • Corrosive • Off-gas handling	5	4	• Pressure relief • Flow monitoring • Remote handling • Secondary containment • Online instrumentation • Recycle/waste stream established before operation
Waste form preparation: dehalogenation	• Be (if relevant) • Very high radiation • Very high contamination • Heat • Volatile fluorides/chlorides • Corrosive	2	5 (Fl salt), 4 (Cl salt)	• Remote handling • Secondary containment • Online instrumentation • Recycle/waste stream established before operation

5.4 REGULATORY ANALYSIS

Most of the MSR concepts discussed (i.e., those identified in Table 1) are expected to apply for a Class 103 power reactor commercial license. Generally, safety features and controls over any hazard should not depend on whether the applicant is applying for a nonpower or power reactor license. The most critical or principal regulations are outlined in Table 4, and further discussion is provided in Section 2.

Table 4. Principal Regulations for MSR Reactor Operations

Regulation	Title	Relationship to MSR reactor operation
10 CFR 50.34	*Content of Applications; Technical Information*	Any physical or chemical process during MSR operation may be involved in or initiate a potential accident. Design criteria should be established for any fuel cycle aspect that has some safety significance or is related to an FSF. Any MSR should review each hazard for design criteria development or safety criteria of DBAs if intending to follow a 10 CFR Part 53 application. At a minimum, processes that would likely be included in the design basis include salt processing, refueling/fissile material addition, and off-gas sampling and scrubbing. However, it is unclear whether all fuel cycle–related actives and hazards must automatically be included in the reactor application if a natural separation between the reactor plant and "fuel cycle plant" can be easily identified. If so, those fuel cycle activities may be licensed under 10 CFR Part 70 using an ISA. For either case, potential collocated hazards and accidents should be addressed.
10 CFR 50.36	*Technical Specifications*	A technical specification limiting condition for operation must be established if one or more of the four criteria described in 10 CFR 50.36 is met. Considering the many systems (i.e., plant components or pipes) that are responsible for controlling the release of FPs, of these four conditions, criterion 3, which states that an SSC is part of the primary success path in preventing or mitigating a DBA, is likely to be met for many SSCs associated with the hazards in Table 3.
10 CFR 50.47	*Emergency Plans*	Like 10 CFR 50.34 and 50.36, any hazard may contribute to potential off-site consequences, depending on the accident analysis. Opportunities exist for smaller EPZs, depending on the accident analysis.
10 CFR 50.48	*Fire Protection*	Fire protection must be established for many areas in which radionuclides are present and controlled. Non-water–based systems will likely be employed, and 10 CFR 53.875 may offer PB advantages over the existing requirements.
10 CFR 50.55(a)	*Codes and Standards, Documents approved for incorporation by reference*	Many of the approved codes and standards may not be applicable to MSRs. Although applicants would benefit from approved new MSR codes and standards, it is not strictly required.. This is an ongoing challenge faced by a broad range of advanced reactor concepts.
10 CFR 50.63	*Loss of All Alternating Current Power*	One benefit of MSRs is passive cooling for any loss of power or station blackout–type accidents. However, all physical and chemical processes should be evaluated for loss of power. Typical questions may be asked: Does the salt freeze? Does salt freezing challenge a barrier? How is heat/pressure relieved in gas-processing systems? Design criteria for electric power systems may be required if the fuel cycle–related processes challenge a safety function on loss of power for 72 h.

Regulation	Title	Relationship to MSR reactor operation
10 CFR 50.65	*Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*	As discussed in Section 2, SSC monitoring is not required if preventive maintenance and reliability analysis are being effectively performed. Fuel cycle–related activities that perform a safety function, such as radionuclide retention, would be required to monitor or perform preventative maintenance.
10 CFR 50.68	*Criticality Accident Requirements*	Refueling, fuel processing, fuel sampling, and other systems that contain fissile material will be designed to prevent criticality accidents. However, off-normal and accident conditions, such as pipe break/leak, should be assessed for these systems/volumes so that criticality accidents are prevented.
10 CFR 50.150	*Aircraft Impact Assessment*	Although the regulation is LWR specific, similar concepts of containment and spent fuel can be extended to MSRs. Containment and spent fuel would indicate that most, if not all, fuel cycle–related processes that have a safety function to retain radionuclides must be included in an aircraft impact assessment.
10 CFR 50.155	*Mitigation of Beyond Design Basis Events*	Fuel cycle–related processes and hazards that may not contribute to DBAs may be deemed contributory to certain BDBEs. Mitigation approaches may need to be explored for those systems, such as off-gas processing, waste form preparation, and fuel processing.
10 CFR Part 20 and ALARA	*Standards for Protection Against Radiation*	The design and operations of fuel cycle–related processes will be required to adhere to ALARA for occupational doses and doses to members of the public.
10 CFR Part 70	*Domestic Licensing of Special Nuclear Material*	As mentioned in 10 CFR 50.34, 10 CFR Part 70 is not required if all fuel cycle–related activities are included in the reactor application.
10 CFR Part 73	*Physical Protection of Plants and Materials*	Design basis threat and possible sabotage should include fuel cycle processes in which radionuclides and/or fissile material is present.
10 CFR Part 74	*Material Control and Accounting of Special Nuclear Material*	All processes that involve fuel will be subject to NMAC.

6. CONCLUSIONS AND RECOMMENDATIONS

An individual MSR will be an element of a nuclear site and fuel cycle. MSRs may be integrated into nuclear fuel cycle facilities and sites that include other reactor technologies (e.g., fuel salt produced from used LWR fuel). Consequently, establishing PB, technology-neutral language equivalent to existing LWR-centric regulations would be more useful than developing MSR-specific language.

One area for which it would be especially useful to develop focused guidance is in integrating the hazards of multiple separately licensed facilities into one site environmental and hazard evaluation. One requirement for this would be performing a site-level assessment of external events. Common vulnerabilities to large-scale external events and the potential for shared services combined with different safety bases for different facility classes (e.g., legacy reactors, new reactors, independent spent fuel storage facilities, used fuel processing facilities) complicate any site-level safety adequacy assessment. Although the individual facilities must comply with their individual license requirements, earthquakes, wildfires, derechos, and so on can often affect even widely spaced facilities. Developing clear regulatory processes for assessing adequate site-level protection from infrequent, severe external events on multifacility sites is recommended.

Safety concepts in existing regulations remain appropriate for application to new technologies. MSRs that achieve many or all the advanced reactor safety objectives—such as those in NRC–2008–0237, *Policy Statement on the Regulation of Advanced Reactors*—will be resistant to internal accidents and external events. The integrated site licensing process is recommended to reflect the safety characteristics of the individual facilities located on- the site as well as the adverse consequences of regulating nuclear site hazards beyond the levels necessary to achieve the adequate safety.

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[105] NEI 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Revision 1, August 2019.

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APPENDIX A. DETAILED REGULATION SUMMARY

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The NRC licenses all commercial nuclear power plants that produce electricity or provide other energy services in the United States [1]. Currently, nuclear power plants can be licensed under 10 CFR Parts 50 or 52. A 10 CFR Part 53 license process is under development for advanced non-LWRs. Nuclear power plants licensed under 10 CFR Part 50 undergo a two-step licensing process that first grants a CP and, after further review, an OL. An alternative licensing process is available under 10 CFR Part 52 that combines a CP and an OL, with certain conditions, into one license known as a COL [2]. 10 CFR Part 70 regulates the licensing required to receive title to, own, acquire, deliver, receive, possess, use, and transfer SNM. However, a separate 10 CFR Part 70 license is not required for possession or use of SNM for the operation of a nuclear reactor licensed under 10 CFR Parts 50 or 52 (10 CFR 70.22[b]). Clustered facilities supporting an MSR or a fleet of MSRs at a collocated site may need a separate 10 CFR Part 70 license. Additionally, any MSR facility contemplating the sale, distribution, or recycling of byproduct material resulting from expanded fuel cycle activities at its must consider the licensing requirements found in 10 CFR Part 30.

This appendix summarizes the various portions of Title 10 of the CFR that may be applicable to an OL for an MSR technology that may also intend to incorporate some portion of the frontend or backend of the traditional fuel cycle as part of normal reactor site management. Such facilities are referred to in this appendix as *broad-spectrum fuel cycle MSR facilities or technologies*. Expanded information on licenses and certificates is provided in Appendix A.

A.1 10 CFR PART 50, "DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES"

The regulations in 10 CFR Part 50 are promulgated by the NRC pursuant to the AEA of 1954 [4], as amended (68 Stat. 919) [3], and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242) [1] to provide for the licensing of production and utilization facilities.

A.1.1 10 CFR 50.22, "Class 103 Licenses; for Commercial and Industrial Facilities"

Chapter 10 of the AEA, as amended, addresses licensing. Sections 103 and 104 of the AEA outline the two types of licenses that can be granted to an applicant. Section 103 discusses commercial licenses, and Section 104 discusses medical therapy and research and development licenses. The licensing requirements of the AEA are reflected in 10 CFR 50.21, "Class 104 Licenses: for medical therapy and research and development facilities," and in 10 CFR 50.22, "Class 103 Licenses; for commercial and industrial facilities." This licensing classification will be the same for MSR facilities.

An NRC Class 103 reactor license will be issued to "an applicant qualified to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use a production or utilization facility for industrial or commercial purposes." Any production or utilization facility that could also be useful in the conduct of research and development activities is deemed to be used for industrial or commercial purposes if more than 50% of the annual cost of owning and operating the facility is devoted to (1) the production of materials, products, or energy for sale or commercial distribution or (2) the sale of services other than research and development, education, or training. The SRP for LWR applications is found in NUREG-0800 [4]. Advanced reactor reviews for SMRs, non-LWRs, and microreactors may be based on NUREG-0800 and the regulatory bases found in 10 CFR Parts 50, 52, or 53. An alternative review plan, such as NUREG-1537 [5] (see SHINE in Appendix B), may also be considered [2], [5] because many of the proposed SMRs and advanced non-LWRs may have small source terms similar to the source terms associated with some nonpower reactors. . A Licensing Modernization Project, outlined in NEI 18-04 [105], *Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light*

Water Reactor Licensing Basis Development, provides and alternate approach to licensing within 10CFR 50 and 10 CFR 52. This content is also expected to change the required content of advanced reactor applications, which is currently being addressed by the industry-led Technology Inclusive Content of Application Project (TICAP). Additionally, various nonreactor fuel cycle facilities have separate review plans.

A representation of the NRC licensing structure for a reactor technology is shown in Figure A.1.

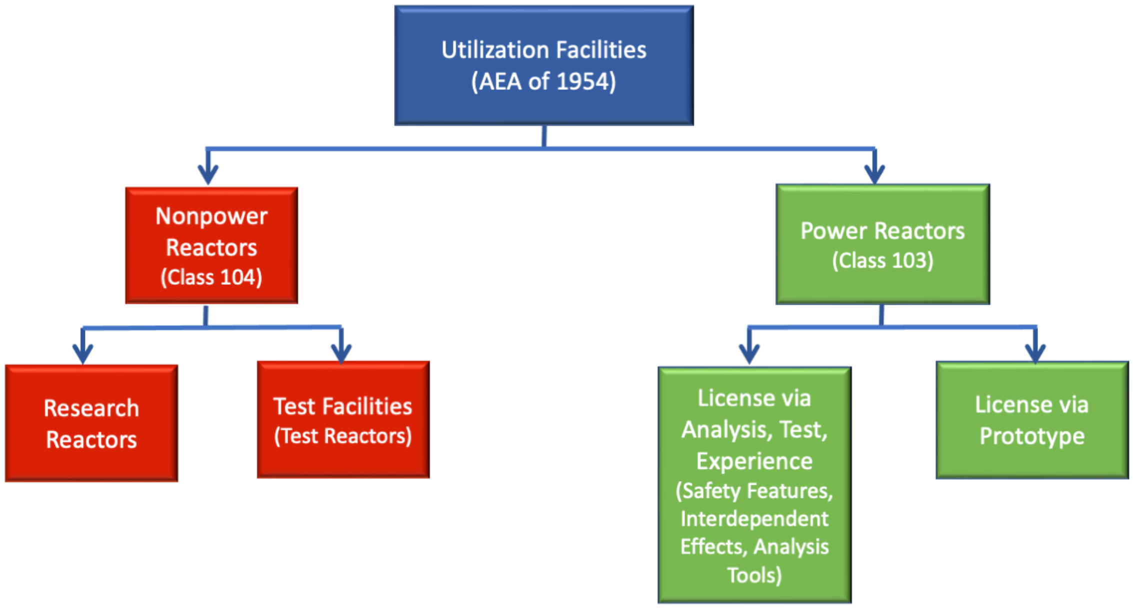

Figure A.1. NRC licensing structure.

A.1.2 10 CFR Part 50 Licensing Process

As noted previously, 10 CFR Part 50 provides a two-step licensing path that begins with an applicant applying for a CP and then an appropriate class OL as construction nears completion. A CP can be issued (10 CFR 50.50) by the NRC based on its review of a site environmental report per 10 CFR Part 51, "Environmental Protection Regulations," and a preliminary safety analysis report. A CP constitutes an authorization to the applicant to proceed with construction, but it does not constitute NRC approval of the safety of any design feature or specification.

Upon determining that an application for an OL meets the standards and requirements of AEA and NRC regulations and that any required notifications to other agencies or bodies have been duly made, the NRC will issue a license per 10 CFR 50.57. The issuance of an OL indicates that facility construction has been substantially completed, the facility will operate in conformity with the application as amended, and there is reasonable assurance that the activities authorized by the OL can be conducted without endangering the health and safety of the public. Opportunities for public comment and hearings are available in the CP and OL phase.

A.1.3 Licensing Requirements with Potential Significance for MSR Technologies Including Additional Fuel Cycle Activities

As noted in SECY-09-0082 [7], 10 CFR Part 50 provides the licensing framework for production and utilization facilities. This includes reprocessing facilities and certain frontend fuel cycle functions that may be associated with some MSR technologies because they are a type of production facility [7]. However, as noted in SECY-13-0093 [8], the staff concluded that the regulatory framework for licensing a reprocessing facility under 10 CFR Part 50 may not be efficient or effective. The licensing requirements located in 10 CFR Part 50 are principally based on the long NRC regulatory history with LWR technology. This will limit the ease with which 10 CFR Part 50 can be applied to other production facility designs and technologies [8].

A facility employing a broad-spectrum fuel cycle MSR technology must consider the intent of the regulations and make the case for an alternative method to meet a given requirement or seek a waiver. The following regulatory sections in 10 CFR Part 50 are specifically noted by the authors of this report for nuanced application to a broad-spectrum fuel cycle MSR facility applicant.

A.1.3.1 Standards for Licenses, Certifications, and Regulatory Approvals

The regulations in 10 CFR Parts 50.40, 50.42, and 50.43 provide for common standards and standards specific to the issuance of a Class 103 license.

This includes the requirement that an applicant will comply with the regulations found in 10 CFR Part 20, "Standards for Protection Against Radiation." The regulations in 10 CFR Part 20 address controlling "the receipt, possession, use, transfer, and disposal of licensed material by any licensee in such a manner that the total dose to an individual (including doses resulting from licensed and unlicensed radioactive material and from radiation sources other than background radiation) does not exceed the standards for protection against radiation prescribed in the regulations in this part".

Additionally, the requirements of 10 CFR 50.43(e) include additional conditions for reactor technologies that differ significantly from current LWR technologies or use simplified, inherent, passive, or other innovative means to accomplish their safety functions. Such technologies will only be approved under the following conditions.

- The performance of each design safety feature has been demonstrated through analysis, appropriate test programs, experience, or a combination thereof.
- Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof.
- Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

The regulation specifically states

> There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period.

It is unclear how an actual prototype plant would be licensed or how the use of a prototype would accelerate the licensing process. Therefore, MSR developers will likely consider pursuing an extensive test program that may include a test reactor. The review process for a test reactor application would be under the PB criteria found in NUREG-1537 [5]. A discussion of recommended changes to NUREG-1537 to facilitate nonpower MSR licensing is found in Section A 1.4.

A.1.3.2 10 CFR 50.34, "Content of Applications; Technical Information"

Each application for a production or utilization facility CP shall include a preliminary safety analysis report. The content of the report is described in this section, including the facility PDC. The requirement in 10 CFR 50.34(a)(ii)(E) states:

> (2) A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.
>
> (3) The preliminary design of the facility including:
>
> (i) The principal design criteria for the facility.
>
> (ii) The design bases and the relation of the design bases to the principal design criteria.
>
> (iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

10 CFR Part 52 contains the same language in 10 CFR 52.79 to include PDC in the facility safety analysis report. NEI incorporated consideration for PDC in some of its advanced reactor guidance documents. NEI issued technical report NEI 21-07 to inform the safety analysis report content for applicants using the methodology in NEI 18-04, endorsed by NRC in Regulatory Guide 1.233. NEI 21-07 provides a systematic approach for identifying PDC as an alternative to the traditional deterministic approach of identifying PDC used for LWRs. The methodology in NEI 18-04 and implementing guidance in NEI 21-07 will likely not lead to PDC that are identical to the GDC found in Appendix A of 10 CFR Part 50 or the ARDC found in RG 1.232 [19]. Neither resource provides regulatory requirements for non-LWR applicants. However, the GDC and the ARDC do provide guidance for developing the scope of proposed PDC to be developed by a non-LWR applicant under 10 CFR Parts 50 and 52.

NEI 21-07 notes three types of PDC.

- PDC–quality assurance (QA): Addresses the graded approach to special treatments for those SSCs performing safety significant functions, including design, fabrication, construction, and testing quality standards. PDC-QA should outline the design, fabrication, construction, and testing quality standards applicable to safety-significant SSCs. A sample PDC statement for QA is provided in Section 5.3.1 of NEI 21-07.
- PDC–required functional design criteria (RFDC): Establish the functional requirements of a plant that are required to meet the performance objectives of the FSFs and are satisfied by safety-related SSCs. The PDC-RFDC should outline the design criteria implemented to meet the required safety functions for a given technology. The required safety functions are determined through a process outlined in Figure 4-1 of NEI 18-04. The MSR PDC should ensure the successful completion of required safety functions for all identified DBAs. Sample-required safety functions include retention of radionuclides, reactor heat removal, the ability to shut down the reactor, and reactor shutdown diversity.
- PDC–complementary design criteria (CDC): Establish requirements for SSCs that are identified as non-safety–related with special treatment because they perform risk-significant functions or are identified as necessary for defense in depth. The PDC-CDC should outline the design criteria that are implemented to support safety-related SSCs or to provide defense in depth. These PDC relate to non-safety–related SSC in a bottom-up support role.

The draft content of 10 CFR Part 53 Framework A supports the approach proposed in NEI 21-07. The current proposed definition for a *DBA* in 10 CFR 53.020, “Definitions,” is a postulated event sequence used to set functional design criteria and performance objectives for the design of safety-related SSCs. DBAs are a type of licensing basis event and are based on the capabilities and reliabilities of safety-related SSCs needed to mitigate and prevent event sequences, respectively. *Functional design criteria* outline the requirements for the performance of safety-related SSCs and non-safety–related but safety-significant SSCs. *Special treatment* means the requirements (e.g., measures taken to satisfy functional design criteria, QA, and programmatic controls) that (1) assure that safety-related and non-safety–related but safety-significant SSCs provide defense in depth or perform risk-significant functions and (2) provide confidence that the SSCs will perform under the service conditions and with the reliability assumed in the safety analysis. The draft content of 10 CFR Part 53 Framework B supports the more traditional PDC approach found in Part 50 and Part 52.

The proposed Framework A requirements proposed in 10 CFR 53.210, “Safety Criteria for Design Basis Accidents,” provide the same radiation dose limitation as those found in the current regulations for LWRs. 10 CFR 53.210 also proposes that non-LWR applicants are required to provide PDC using the GDC or other generally accepted consensus codes and standards to inform the development of the provided PDC. 10 CFR 53.410, “Functional Design Criteria for Design Basis Accidents,” further proposes that functional design criteria be defined for each design feature that is relied upon to demonstrate compliance with the safety criteria defined in 10 CFR 53.210.

The requirements proposed in 10 CFR 53.220, “Safety Criteria for Licensing Basis Events Other than Design Basis Accidents,” compel the applicant to include design features and programmatic controls to ensure that plant SSCs, personnel, and programs provide the necessary capabilities and maintain the necessary reliability to address licensing basis events and provide measures for defense in depth. Associated dose requirements require that the overall cumulative plant risk from licensing basis events be maintained so that (1) the calculated risk to an average individual in the vicinity of the commercial nuclear plant of prompt fatalities remains below five in 10 million years and (2) the calculated risk to with the population in the area near a commercial nuclear plant of cancer fatalities effects remains below two in 1 million years. Subsequently, 10 CFR 53.420, “Functional Design Criteria for Licensing Basis Events Other than Design Basis Accidents,” proposes that functional design criteria be defined for each design feature relied upon to demonstrate compliance with the safety criteria in 10 CFR 53.220 as required by 10 CFR 53.400, “Design Features for Licensing Basis Events, and Evaluation Criteria in 10 CFR 53.450(e), Analysis Requirements.”

The proposed requirements for maximum public dose and protection of plant workers in 10 CFR Part 53 currently reference the requirements in 10 CFR Part 20.

The requirements in 10 CFR 53.440, “Design Requirements,” state that design features required to meet the safety criteria defined in 10 CFR 53.210 and 10 CFR 53.220 must be designed using generally accepted consensus codes and standards wherever applicable. Additionally, possible degradation mechanisms related to aging, fatigue, chemical interactions, operating temperatures, effects of irradiation, and other environmental factors that may affect the performance of safety-related and non-safety–related but safety-significant SSCs must be evaluated and used to inform the design.

If 10 CFR Part 53 is enacted as currently proposed, then a broad-spectrum fuel cycle MSR facility must consider the technology functional design criteria for safety-related SSCs and SSC special treatments for defense in depth or the performance of risk-significant functions. Non-LWR applicants would be required to provide PDC using the GDC or other generally accepted consensus codes and standards to inform PDC development.

A.1.3.3 10 CFR 50.36, "Technical Specifications"

Each applicant for a license authorizing operation of a production or utilization facility shall include proposed SSC technical specifications for the facility. Design certifications under 10 CFR Part 52 also require SSC technical specifications for the design. A broad-spectrum fuel cycle MSR technology must propose technical specifications for all aspects of the fuel cycle represented at the plant, including the reactor and any proposed frontend or backend facility. A technical specification-limiting condition for operating a nuclear reactor must be established for each item that meets one or more of the following criteria:

> (A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> (B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (C) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The criteria listed for establishing reactor technical specifications are based on LWR technology. For example, the reactor coolant pressure boundary has no meaning for an MSR. A closer MSR approximation for criterion 1 is to consider a degradation of the fuel system boundary. The fuel system boundary may comprise several different layers. Additionally, the fuel system boundary takes on the role of an FP barrier. Thus, the lines of distinction between the first three criteria are blurred for MSR technologies. Broad-spectrum fuel cycle MSR facilities must specify appropriate technical specifications that meet the reactor criteria and any other fuel cycle facilities associated with the MSR site.

The draft language for 10 CFR 53.710, "Maintaining capabilities and availability of structures, systems, and components (SSCs)," provides a broader basis for implementing technical specifications [9]: "(a) Technical specifications must be developed and implemented that define conditions or limitations on plant operations that are necessary to provide reasonable assurance that safety related SSCs fulfill the safety functions in 10 CFR 53.230, Safety Functions."

The safety functions defined in 10 CFR 53.230 are as follows [9]:

> (a) The primary safety function is limiting the release of radioactive materials from the facility and must be maintained during routine operation and for licensing basis events over the life of the plant.
>
> (b) Additional safety functions supporting the retention of radioactive materials during licensing basis events—such as controlling reactivity, heat generation, heat removal, and chemical interactions—must be defined.

If 10 CFR Part 53 is enacted as currently proposed, then 10 CFR 53.710 would provide more clarity for developing technical specifications for a broad-spectrum fuel cycle MSR facility. The NRC plans to release a draft guidance document for public comment in FY23 with proposed scoping criteria for advanced reactor technical specifications.

A.1.3.4 10 CFR 50.47, "Emergency Plans"

A production or utilization facility applicant must provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency in accordance with the regulations in 10 CFR 47 and Appendix E of 10 CFR 50, "Emergency Planning and Preparedness for Production and Utilization Facilities." A COL issued under 10 CFR Part 52 must also meet these requirements. In the case of LWRs, this has traditionally required that a plume exposure pathway EPZ for nuclear power plants comprise an area of ~10 mi (16 km) in radius and an ingestion pathway EPZ comprise an area of ~50 mi (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular production or utilization facility can be adjusted in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.

However, a rulemaking is in process for SMRs and ONT [10]. In SECY-11-0152 [11], the staff discussed potential changes to the EP and preparedness framework for SMRs. The staff indicated their expectation that the dose assessments could be a factor for the basis for the EPZ distances based on a spectrum of accidents using the plant design PRA, as well as including current insights on severe accident progression.

Subsequently, In SECY-15-0077 [12], the staff proposed a consequence-based approach to establishing requirements for off-site EP for SMRs and ONTs. In the related SRM [13], the NRC approved the staff's proposal to revise NRC regulations and guidance through rulemaking to "demonstrate how their proposed facilities achieve U.S. Environmental Protection Agency (EPA) Protective Action Guide (PAG) dose limits at specified EPZ distances, which may include the site boundary." The PAG manual was updated in 2017 [15].

Essentially, the proposed rule will provide [10]:

- a new alternative PB EP framework,
- a hazard analysis of any NRC-licensed or nonlicensed facility contiguous or nearby to an SMR or ONT that considers any hazard that would adversely affect the implementation of emergency plans,
- a scalable approach for determining the size of the plume exposure pathway EPZ, and
- a requirement to describe ingestion response planning in the EP, including the capabilities and resources available to prevent contaminated food and water from entering the ingestion pathway.

The proposed rule [10] offers a new 10 CFR 50.160 requirement for EP for SMRs and ONTs as an alternative to the current requirements in 10 CFR 47. The rulemaking also provides appropriate references to the new rule in 10 CFR 50.2, "Definitions," to provide definitions for non-LWRs, nonpower production or utilization facilities, and SMRs. Revisions are also included in 10 CFR 50.33, "Contents of Applications; General Information"; 10 CFR 50.34, "Contents of Applications; Technical Information"; 10 CFR 50.47; 10 CFR 50.54, "Conditions of Licenses," and Appendix E of 10 CFR Part 50 to include appropriate references and options for the new rule. Rule publication is targeted for July 2022.

Draft Guide (DG)-1350, *Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities* [16], provides PB guidance for the new rule. This EP approach requires that the plume exposure pathway EPZ be established as the area in which public dose is projected to exceed 10 mSv (1 rem) total effective dose equivalent over the first 96 h from the release of radioactive materials from a spectrum of credible accidents for the facility. As stated in DG-1350 [16], this is consistent with the PAGs published in a 2017 EPA PAG Manual, which notes that the duration of early-phase protective actions begins at the actual or projected start of a release and generally lasts up to 4 days (96 h).

DG-1350 [16] states that "an applicant or licensee that chooses to adopt the EP regulations in 10 CFR 50.160 must include in the emergency plan an analysis of any credible hazard from a contiguous or nearby facility that would adversely impact the implementation of emergency plans." This would include extended fuel cycle facilities associated with an MSR technology and the MSR off-gas system. Appendix A of DG-1350 [16] states that "analyses with design-basis-accident (DBA) source terms may simply present dose-distance curves conditional upon the occurrence of the source term without consideration of frequency. For BDBA, dose-distance results may be aggregated using frequency information developed to evaluate the likelihood of exceeding a total effective dose equivalent of one rem as a function of distance."

This EP discussion would be directly applicable to MSRs. The entirety of the broad-spectrum fuel cycle MSR facility would require EP to be included in a facility application.

The draft language in 10 CFR 53.1146, "Contents of Applications; Technical Information," supports the aforementioned EP discussion [9].

A.1.3.5 10 CFR 50.48, "Fire Protection"

Each holder of an OL issued under 10 CFR Part 50 or a COL issued under 10 CFR Part 52 must have a fire protection plan to meet design Criterion 3, "Fire Protection," in Appendix A of 10 CFR Part 50. The fire protection requirements for older LWRs are prescribed in Appendix R of 10 CFR Part 50. Appendix R requires that two separate water supplies be provided to furnish the necessary water volume and pressure to the fire main loop. A water-based fire suppression system is not the best technology choice for fire suppression at a broad-spectrum fuel cycle MSR facility because one advantage of MSR technologies is that they operate at low pressure with a functional containment that reflects the low risk for an accident-induced pressure spike spread of contamination. Introducing water into the high-heat MSR containment environment would provide an unnecessary potential for a steam explosion and subsequent spread of contamination.

The newer fire protection requirements in 10 CFR 50.48 are performance based and do not specify a water-based system. However, 10 CFR 50.48 endorses NFPA Standard 805 [17], which includes provisions for water-based design elements as part of a fire protection program, as well as other suppression technologies.

RG 1.205 [18] provides implementation guidance for the requirements in 10 CFR 50.48. The guidance clarifies that "the licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is adequate for the hazard" [18]. This applies to four specific sections of Chapter 3 of NFPA 805 [17] for which prior NRC review and approval would not be required:

- Fire alarm and detection systems (Section 3.8 of NFPA 805 [17])
- Automatic and manual water-based fire suppression systems (Section 3.9 of NFPA 805 [17])
- Gaseous fire suppression systems (Section 3.10 of NFPA 805 [17])
- Passive fire protection features (Section 3.11 of NFPA 805 [17])

Therefore, MSR technologies can meet the requirements of Appendix R of 10 CFR 50.48, NFPA 805 [17], and RG 1.205 [18]. However, MSR technologies may need to rely on combustible controls, controls for graphite applications outside the core (e.g., in an off-gas system), and chemical fire suppressants. The various aspects of the frontend or backend fuel cycle applications may require segmented cells for processing to limit the risk of fire or mitigate fire damage.

The draft language for 10 CFR 53.875, "Fire Protection," promotes PB requirements and does not specify a water-based fire suppression system [9].

A.1.3.6 10 CFR 50.54, "Conditions of Licenses"

The requirements specified in 10 CFR 50.54 provide a basis for licensing issues for any reactor design that intends to implement a high degree of autonomous control, as might be expected for a broad-spectrum fuel cycle MSR facility. These issues include plant staffing, manipulation of controls, licensed operators, EPs, technical specifications, cybersecurity, notifications, and the QA criteria in Appendix B of 10 CFR Part 50. The plant licensing basis may not be changed without NRC authorization.

Additionally, 10 CFR 50.54 states that a licensee shall prepare and maintain safeguards contingency plan procedures in accordance with Appendix C of 10 CFR Part 73. Physical protection is discussed further in Section 3.4 of this report. Additionally, each licensee subject to the requirements of 10 CFR Part 73 shall ensure that safeguards information is protected against unauthorized disclosure.

This discussion on conditions of licenses is directly applicable to MSRs. The entirety of the broad-spectrum fuel cycle MSR facility must consider the conditions of licenses in a facility application.

A.1.3.7 10 CFR 50.55a, "Codes and Standards"

This section lists codes and standards that are approved for incorporation by reference into an application. Because 10 CFR Part 50 is biased toward LWR technology, MSR designs must make the case for newer standards that support their technologies.

A.1.3.8 10 CFR 50.63, "Loss of All Alternating Current Power"

10 CFR 50.63 is a LWR-specific regulation that states that

> The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis.

This is based on Design Criterion (DC) 17 (Appendix A of 10 CFR Part 50), which requires that an on-site and an off-site electric power system be provided to permit the functioning of SSCs important to safety. "Sufficient capacity and capability should be provided to assure that (1) specified acceptable fuel design limits (SAFDLs) and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs and (2) the core is cooled, and containment integrity and other vital functions are maintained in the event of postulated accidents". SAFDLs are applicable to LWR and other heterogeneous fuel forms.

Although 10 CFR 50.63 is LWR specific, the underlying safety basis is reflected in the advanced reactor DC discussed in RG 1.232 [19. ARDC 17 shifts the safety emphasis from the fuel to the FP barriers and associated safety functions. "Electric power systems shall be provided when required to permit functioning of SSC. The safety function for each power system shall be to provide sufficient capacity and capability to ensure that (1) that the design limits for the FP barriers are not exceeded as a result of AOOs and (2) safety functions that rely on electric power are maintained in the event of postulated accidents".

An additional aspect of 10 CFR 50.63 is SBO coping time. Coping time for active safety systems is discussed in RG 1.155 [20] and is largely based on the operating status of emergency diesel generators. Because passive safety systems do not rely on the operating status of emergency diesel generators, a different approach to coping time is needed for advanced LWRs and non-LWRs. NUREG-0800 Chapter 8.4, "Station Blackout" [21], indicates that passive LWR technologies do not need to evaluate SBO coping duration if they can demonstrate that the technology being considered can perform safety-related functions for 72 h. The 72 h approach is consistent with the duration approved by the NRC for the AP 1000 design and has become a basic passive safety standard tenet. Thus, MSRs must include a discussion of their passive safety capabilities and consideration of SBO coping time within their technology license applications. Table 4 includes conditions for an MSR applicant to analyze.

Additionally, MSR technologies must demonstrate that the design electric power systems include the following for the entire facility [19]:

> …an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its safety function. If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided.

A.1.3.9 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

After an applicant is granted a Class 103 license (OL under that licensees monitor the performance or condition of SSC against licensee-established goals (i.e., technical specifications). Licensees shall provide reasonable assurance that these SSC can fulfill their intended functions. Industry-wide operating experience should be factored into SSC monitoring, evaluation, and maintenance.

SSC monitoring is not required if appropriate preventive maintenance is being effectively used to ensure the SSCs remain capable of performing their intended function. All maintenance activities must be evaluated to assess and manage the increase in risk that may result from the proposed maintenance activities. This includes safety-related SSCs and SSCs whose failure could negatively affect the performance of the safety-related SSC.

Because most MSR SSCs are always subjected to high-radiation environments, maintenance is more difficult and often must be performed remotely. Therefore, SSC monitoring, maintenance planning, and maintenance execution must be discussed in an MSR technology license application. Additionally, passive SSCs are used more extensively in advanced reactor applications, including MSRs. Passive SSCs more commonly exhibit performance degradation rather than outright failure. Such degradation may not constitute a system failure and may not compromise overall plant safety. MSR developers must provide a discussion of degraded passive SSC performance and work with NRC staff to develop advanced reactor guidance that would indicate that passive SSCs must remain capable of adequately fulfilling their intended functions so that overall plant operation continues to achieve the FSFs.

A.1.3.10 10 CFR 50.68, "Criticality Accident Requirements"

The requirement in 10 CFR 50.68 provides two options for preventing a criticality accident. A Class 103 license holder shall comply with either 10 CFR 70.24 or the requirements in paragraph (b) of 10 CFR 50.68.

Paragraph (b) applies to solid LWR fuel and is not directly applicable to MSR technologies. However, the underlying safety goal of 10 CFR 50.68(b) is to prevent a criticality condition away from the reactor core. To meet the paragraph (b) safety goal, MSR technologies must specify SSCs, design parameters, and technical specifications that are intended to prevent a criticality accident.

MSR technologies can also look to 10 CFR 70.24 for an alternate set of criticality accident requirements. The requirements in 10 CFR 70.24 are for detecting a criticality accident. Per 10 CFR 70.24, each licensee authorized to possess SNM in quantities necessary for reactor operations shall maintain a neutron and gamma radiation monitoring system. Two detectors shall provide coverage of all areas. This applies to the portions of an MSR fuel cycle at a plant beyond actual reactor operation. However, the underlying safety goal is still to prevent a criticality condition away from the reactor core. Therefore, it is still incumbent upon MSR designers to specify how they will meet this safety goal.

A.1.3.11 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

The requirements of 10 CFR 50.69 provide a voluntary approach for a risk-informed process to evaluate the safety significance of SSCs and establish the appropriate level of special treatment requirements for SSCs [22]. The grouping and integration of risk-informed requirements within this requirement provide the capability to apply a risk-informed approach more broadly within 10 CFR Parts 50 and 52. Under this approach, licensees and NRC staff can focus regulatory resources on SSCs with significant contributions to plant safety.

This voluntary requirement applies only to LWRs. However, because 10 CFR Part 50 applies to all production and utilization facilities, MSR SSCs associated with the frontend or backend of the fuel cycle production or utilization facility can make the case to treat their respective SSCs in accordance with 10 CFR 50.69. 10 CFR 50.69 states that "SSCs must be categorized as RISC–1, RISC–2, RISC–3, or RISC–4 SSCs using a categorization process that determines if an SSC performs one or more safety significant functions and identifies those functions".

The regulation provides the following category definitions:

- Risk-Informed Safety Class (RISC)–1 SSCs: Safety-related SSCs that perform safety-significant functions.
- RISC–2 SSCs: Non-safety–related SSCs that perform safety-significant functions.
- RISC–3 SSCs: Safety-related SSCs that perform low safety-significant functions.
- RISC–4 SSCs: Non-safety–related SSCs that perform low safety-significant functions.
- Safety-significant function: A function whose degradation or loss could result in a significant adverse effect on defense in depth, safety margin, or risk.

The initial draft language for 10 CFR Part 53 includes definitions in Part A that span the same breadth of information [9]. NEI issued technical report NEI 21-07 to inform the safety analysis report content for applicants using the methodology in NEI 18-04. NRC guidance for implementing NEI 18-04 is found in RG 1.233.

MSR technologies must specify safety-related SSC and important to safety SSC for the entire MSR facility to support technical specification development and accident analyses.

A.1.3.12 10 CFR 50.150, "Aircraft Impact Assessment"

The NRC promulgated the Aircraft Impact Rule [23] in June 2009. The rule requires that design and license applicants for new LWRs perform a rigorous assessment of their designs to identify design features and functional capabilities that could provide additional inherent protection to avoid or mitigate the effects of an aircraft impact. Applicants for new nuclear power reactors must perform a realistic design-specific assessment of their designs to identify design features and functional capabilities that could provide additional inherent protection to avoid or mitigate the effects of the impact of a large commercial aircraft.

Applicants are required to identify and incorporate into the design those design features and functional capabilities that avoid or mitigate—to the extent practical and with reduced reliance on operator actions—the effects of the aircraft impact on key safety functions [24]. The applicant must show that with reduced operator actions, the reactor core remains cooled or the containment remains intact and spent fuel pool cooling or spent fuel pool integrity is maintained. An MSR applicant will need to translate these functions to the attributes of their specific technology and safety systems for all fuel inventories at all site fuel cycle facilities.

In the statement of considerations for the aircraft impact assessment rule, the NRC stated that these four functions are applicable to LWRs, and each may not be applicable to non-LWR designs or may need to be supplemented by other key functions for non-LWR designs [25]. Additionally, extended fuel cycle facilities associated with an MSR technology may need to show similar analyses for key safety functions.

The draft language for 10 CFR 53.440 and 10 CFR 53.450 [9] proposes to extend the aircraft impact assessment rule to advanced reactor technologies.

The proposed 10 CFR 53.440(j) includes significant inventories of radioactive materials anywhere in the entire reactor facility:

> (j)(1) Design features must be provided, and related functional design criteria defined such that, with limited use of operator actions, one or more physical barriers are maintained to limit the release of radionuclides from reactor systems, waste stores, or other significant inventories of radioactive materials assuming the impact of a large, commercial aircraft.
>
> (2) The functional design criteria for those design features provided to address the requirements in paragraph (j)(1) of this section must be based on an assessment of the impact of a large, commercial aircraft used for long distance flights in the United States, with aviation fuel loading typically used in such flights, and an impact speed and angle of impact considering the ability of both experienced and inexperienced pilots to control large, commercial aircraft at low altitude representative of a nuclear power plant's low profile.

The proposed 10 CFR 53.450(g) and (g)(2) state that:

> (g) Analyses must be performed to assess:
>
> (g)(2) measures provided to protect against aircraft impacts as required by 10 CFR 53.440(j).

This is significant to a broad-spectrum fuel cycle MSR facility licensed under a new 10 CFR Part 53. If 10 CFR Part 53 is enacted as currently proposed, then 10 CFR 50.150 would be applicable to all facilities that comprise a broad-spectrum fuel cycle MSR facility.

A.1.3.13 10 CFR 50.155, "Mitigation of Beyond Design Basis Events"

In 10 CFR Parts 50 and 52, beyond design basis events (BDBE) are identified and assessed on an as-needed basis using a prescriptive approach in which uncertainties are addressed by exercising conservative assumptions. The requirement in 10 CFR 50.155 is heavily influenced by experience with LWRs, but a similar accident response discussion will be required for MSRs per 10 CFR 34. Under this regulation, licensees must (1) develop, implement, and maintain mitigation strategies for beyond-design basis external events and (2) develop extensive damage mitigation guidelines associated with the loss of large areas of the plant.

The rulemaking effort for 10 CFR Part 53 [9] has proposed a new Framework B that will provide technology-inclusive alternative requirements for commercial nuclear plants. Framework B is under development and will provide for more traditional accident evaluation.

The draft language for the Framework A portion of Part 53, 10 CFR 53.420 [9] incorporates the intent of the 10 CFR 50.155 mitigation of BDBEs rule to advanced reactor technologies. This proposed rule notes that

> functional design criteria must be defined for each design feature required by 10 CFR 53.400 and relied upon to demonstrate compliance with the safety criteria in 10 CFR 53.220 evaluation criteria in 10 CFR 53.450(e), or more restrictive alternative criteria adopted under 10 CFR 53.470. Corresponding human actions and programmatic controls and interfaces must be identified and implemented in accordance with this and other subparts to achieve and maintain the reliability and capability of SSCs relied upon to meet the safety criteria in 10 CFR 53.220 and evaluation criteria in 10 CFR 53.450(e), or more restrictive alternate criteria under § 53.470.

If 10 CFR Part 53 is enacted as currently proposed, then BDBEs must be identified and supporting SSCs must analyzed to confirm that the collective SSC provide an adequate response to mitigate the BDBE and maintain an adequate level of defense-in-depth supporting broad-spectrum fuel cycle MSR facility applications under 10 CFR Parts 50, 52, or 53.

A.1.3.14 Waste Management

The requirements in 10 CFR 50.36a state that an applicant shall

> identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable [ALARA], 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.

Additionally, the requirements in 10 CFR 50.34 indicate that the kinds and quantities of radioactive materials expected to be produced during operations, including AOOs, and the means to control and limit radioactive effluent releases and radiation exposures within the limits of 10 CFR Part 20 for members of the public must be included in the safety analysis report.

Appendix I of 10 CFR Part 50 and Chapter 11 of NUREG-0800 [26], [27], [28] provide LWR requirements and review standards for liquid waste management systems, gaseous waste management system, and solid waste management systems. As noted in NUREG-0800, the acceptance criteria for these systems are largely based on requirements found in 10 CFR Part 20 and 10 CFR Part 61, which are technology neutral.

Liquid Waste

According to Chapter 11.2 of NUREG-0800 [26], the application and review of an LWR liquid waste management system should encompass all tanks, piping, pumps, valves, filters, demineralizers, mobile equipment connected to permanently installed systems, and any additional equipment that may be needed to process and treat liquid wastes and route them to the point of discharge from the system. All SSCs included in a broad-spectrum fuel cycle MSR technology must also include an evaluation of any liquid waste streams. An MSR applicant must demonstrate compliance with regulatory limits on liquid effluent discharges and associated doses to members of the public to ensure that releases and doses are ALARA.

Gaseous Waste

According to Chapter 11.3 of NUREG-0800 [27], the application and review of an LWR gaseous waste management system should provide for the management of radioactive gases generated by a gaseous radwaste system or the off-gas system, which may include waste gas storage tanks, waste gas decay tanks, and charcoal delay beds, depending on the type of plant and design features. Apart from a gaseous waste handling system, all MSR technologies will employ a separate off-gas system to allow the removal of gaseous FPs in order to avoid pressurization of the reactor vessel. Therefore, all SSCs included in a broad-spectrum fuel cycle MSR technology must include an evaluation of gaseous waste streams and other gas handling systems while demonstrating compliance with regulatory limits on gaseous effluent discharges and associated doses to members of the public in ensuring that releases and doses are ALARA.

Solid Waste

According to Chapter 11.4 of NUREG-0800 [28], the application and review of an LWR solid waste management system should encompass design features that are necessary for collecting, handling, processing, and storing wastes in buildings that are part of the overall nuclear facility. Many MSR technologies plan for the periodic replacement of large, contaminated SSCs. An application for a broad-spectrum fuel cycle MSR technology must demonstrate that the collection, handling, storage, and off-site shipment of such contaminated SSCs have been considered within the plant life cycle. Compliance with limits on gaseous and liquid effluent discharges from the operation of the solid waste processing equipment and associated doses to members of the public must be evaluated to ensure that any releases and doses are ALARA.

Decommissioning

A decommission plan is not required by an applicant for a CP, OP, or COL. However, 10 CFR 50.33 specifies that "reasonable assurance will be provided that funds will be available to decommission the facility", as described in 10 CFR 50.75. The requirements in 10 CFR 52.77, "Contents of Applications; General Information," specify that an application for a COL must "contain all the information required by 10 CFR 50.33".

As noted on the NRC website for Financial Assurance for Decommissioning, "The total cost of decommissioning a reactor facility depends on many factors, including the timing and sequence of the various stages of the program, type of reactor or facility, location of the facility, radioactive waste burial costs, and plans for spent fuel storage" [29].

For LWR technologies, the NRC estimates the decommissioning costs for a reactor plant to range from $280 million to $612 million [29]. The requirements in 10 CFR 50.75 include a table that provides a decommissioning fund formula for LWRs based on the long history of that technology. Other advanced reactor technologies, including MSRs, must provide some background information on expected decommissioning tasks for that specific technology to satisfy regulatory staff that adequate decommissioning funds are being set aside. The draft language for 10 CFR 53.1020, "Cost Estimates for Required Decommissioning Funds" [9], indicates that the funding estimates for LWRs currently provided in 10 CFR 50.75(c) would not suffice for advanced reactor technologies. Instead,

> Cost estimates for decommissioning must be site-specific. Site-specific decommissioning cost estimates must account for the engineering, labor, equipment, transportation, disposal, and related charges needed to support termination of the license. They must include the costs for decontaminating structures, systems, and components and site environs; removal of contaminated components and materials from the plant and the site environs; disposal costs for removed components and materials in appropriate facilities; and any other associated costs supporting the release of the property and termination of the license.

The draft requirements in 10 CFR 53.1030 require that each licensee must "annually adjust the initial approved cost estimate for decommissioning to account for escalation in labor, energy, and waste burial costs" [9]. The draft supporting text for these proposed regulations suggests that decommissioning cost estimates be supported by guidance documents and "generic" analyses that are applicable to a subject design and site. All SSCs associated with a broad-spectrum fuel cycle MSR facility will require decommissioning analyses.

A.1.3.15 Siting of Fuel Reprocessing Plants and Related Waste Management Facilities

A broad-spectrum fuel cycle MSR technology may include fuel reprocessing on-site with the reactor. Appendix F of 10 CFR Part 50 states that 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. MSR plants, including the facilities for the temporary storage of HLW, may be located on privately owned property.

However, Appendix F also states that the inventory of HLW associated with the reprocessing process are 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. The dry solid shall be chemically, thermally, and radiolytically stable. 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.

The disposal of HLW material associated with fuel reprocessing is permitted only on land that is owned and controlled by the federal government. Thus, waste processing associated with fuel reprocessing must be considered by a broad-spectrum fuel cycle MSR technology that includes such a facility. In accordance with 10 CFR 50.33(f), decommissioning funds must also be projected.

If separated products from fuel reprocessing have a value or purpose for other aspects of the fuel cycle or other industrial uses, then an exception to Appendix F can be sought for those materials.

A.2 10 CFR PART 52, "LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS"

10 CFR Part 52 compliments the licensing requirements in 10 CFR Part 50. Specifically, 10 CFR Part 52 governs the issuance of early site permits, standard design certifications, COLs, standard design approvals, and manufacturing licenses for nuclear power facilities licensed under Section 103 of the AEA of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242).

10 CFR Part 50 provides a two-step licensing path. An initial CP for the construction of a production or utilization facility will be issued before issuing an OL if the application is otherwise acceptable and will be converted upon completion of the facility and NRC action into a license, as provided in 10 CFR 50.56. If a COL for a facility is issued under 10 CFR Part 52, then the CP and OL are combined—with certain provisions—in a COL.

A COL authorizes the construction and conditional operation of a nuclear power plant, once NRC authorizes fuel loading. The application for a COL must contain essentially the same information required in an application for an OL issued under 10 CFR Part 50. The application must also describe the inspections, tests, analyses, and acceptance criteria that are needed to ensure that the plant has been properly constructed and will operate safely. The permitting process for a COL is shown in Figure A.2.

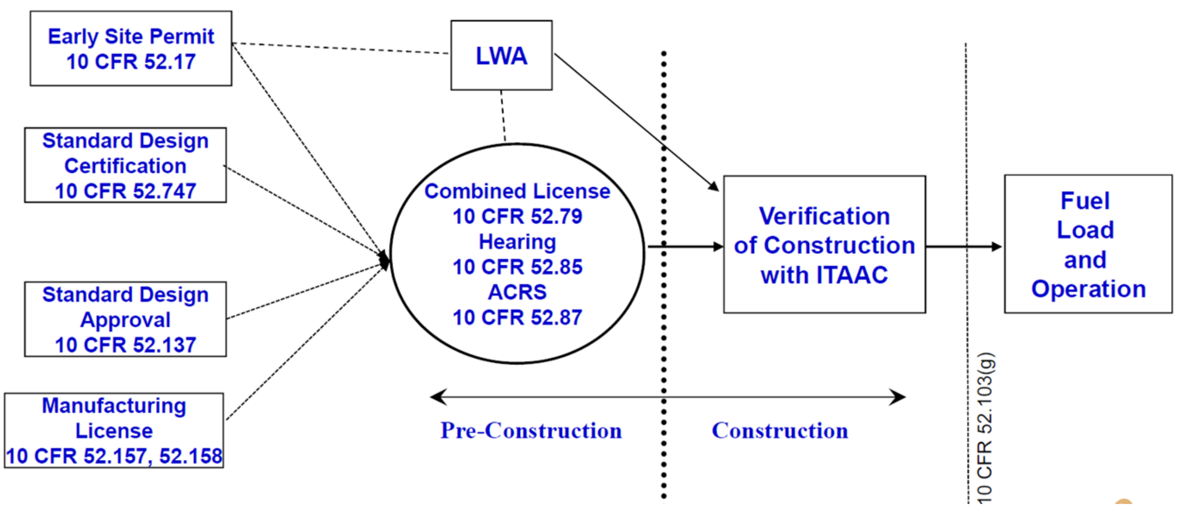

Figure A.2. 10 CFR Part 52 licensing process [30].

Most production and utilization facility requirements in 10 CFR Part 50 include a reference to the COL process in 10 CFR Part 52. COL applicants can reference an approved standard design and/or an early site permit. An MSR standard design or early site permit can include broad-spectrum fuel cycle MSR technology.

A.3 10 CFR PART 53, "RISK INFORMED, TECHNOLOGY-INCLUSIVE REGULATORY FRAMEWORK FOR ADVANCED REACTORS"

The NEIMA [31] directs the NRC to develop the regulatory infrastructure to support the development and commercialization of advanced nuclear reactors. In response to NEIMA, the NRC is preparing a rulemaking for a new 10 CFR Part 53, which is intended to establish a technology-inclusive regulatory framework for optional use by applicants for new commercial advanced nuclear reactors. The regulatory requirements developed in this rulemaking would use methods of evaluation, including risk-informed and

PB methods, that are flexible and practicable for application to a variety of advanced reactor technologies [32].

As shown in Figure 3, the NRC has proposed a general structure for Framework A and Framework B of the 10 CFR Part 53 rulemaking [33]. Draft language has been proposed for each subpart and is actively being debated among various advanced reactor technology stakeholders. Where appropriate, the 10 CFR Part 53 draft language is compared with existing regulations within this report to provide insight on what may be required for a broad-spectrum fuel cycle MSR technology license application. However, no 10 CFR Part 53 language has been finalized. Part 53 is scheduled to be finalized and available for use in 2025.

A.4 10 CFR PART 20, "STANDARDS FOR PROTECTION AGAINST RADIATION"

The requirements in 10 CFR Part 20 "establish standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the NRC". 10 CFR Part 20 controls the receipt, possession, use, transfer, and disposal of licensed material by any licensee to ensure that the total dose to an individual does not exceed the standards for protection against radiation.

The requirements in 10 CFR 20.1101 compel each licensee to "develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities". Such programs must provide procedures and engineering controls to "achieve occupational doses and doses to members of the public that are ALARA" . There is essentially no difference between an MSR facility and an LWR except that this will encompass all activities at a broad-spectrum fuel cycle MSR facility.

A.5 10 CFR PART 61, "LICENSING REQUIREMENTS FOR LAND DISPOSAL OF RADIOACTIVE WASTE"

10 CFR Part 61 regulates the procedures, criteria, and terms and conditions upon which the NRC issues licenses for the disposal of radioactive wastes containing byproduct, source, and SNM in a land disposal facility. Some important considerations for a broad-spectrum fuel cycle MSR facility include waste classification and waste characteristics [28].

A.5.1 10 CFR 61.55—Waste Classification

The Low-Level Radioactive Waste Policy Amendments Act of 1985 [34] and the Nuclear Waste Policy Act of 1982 [35] divide most radioactive waste into two categories: HLW and LLRW. HLW includes highly radioactive waste material associated with fuel reprocessing or other waste that requires permanent isolation. LLRW is material that is not HLW, spent fuel, or certain byproduct materials [36]. The 1985 Low-Level Radioactive Waste Policy Amendments Act provides for the classification of LLRW and gives responsibility for disposal to individual states, except for waste generated by DOE or the US Navy. The federal government is responsible for disposing of LLRW that is not defined in 10 CFR 61.55 or is classified as HLW.

According to 10 CFR 61.55, the classification of waste for near-surface disposal involves two considerations.

- Consideration must be given to the concentration of long-lived radionuclides and any shorter-lived precursors whose potential hazard will persist long after precautions such as institutional controls, improved waste form, and deep disposal cease to be effective.
- Consideration must be given to the concentration of shorter-lived radionuclides for which requirements on institutional controls, waste form, and disposal methods are effective.

The following waste classes are specified by 10 CFR 61.55.

- Class A waste is usually segregated from other waste classes at the disposal site.
- Class B waste must meet more rigorous requirements on waste form to ensure stability after disposal.
- Class C waste must meet more rigorous requirements on waste form to ensure stability but also requires additional measures at the disposal facility to protect against inadvertent intrusion.
- LLRW streams that contain radionuclide concentrations exceeding the limits for Class C waste are referred to as *GTCC waste*.

Classes A, B, and C waste can be disposed of using near-surface retention. Near-surface disposal can include surface disposal, trench disposal within 30 m of the surface, below ground vaults, earth mounds, and boreholes. Currently, GTCC waste is generally not acceptable for near-surface disposal because the form and disposal methods must be different—and, generally, more stringent—than those specified for Class C and lower waste. Under current regulations, GTCC waste must be held on-site to be disposed of in a geologic repository when available, except when allowed by the NRC on a case-by-case basis. GTCC waste typically includes neutron-activated components, in-core neutron detectors, activated metals, radioactive sources, and alpha emitting transuranics [28], [36].

Waste classification and disposal options are important to an MSR facility because of the potential for significant quantities of waste associated with associated fuel cycle activities and solid wastes associated with component replacements.

A.5.2 10 CFR 61.56, "Waste Characteristics"

Appropriate waste characteristics are intended to facilitate handling at the disposal site and protect the health and safety of personnel at the disposal site. The requirements in 10 CFR 61.56 state that:

> (a)(2) Liquid waste must be solidified or packaged in sufficient absorbent material to absorb twice the volume of the liquid.
>
> (a)(3) Solid waste containing liquid shall contain as little free standing and noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1% of the volume.

Broad-spectrum fuel cycle MSR facilities or technologies are expected to generate significant solid waste. MSR LLRW must meet the requirements in 10 CFR Part 61 for disposal. Additionally, MSR LLRW must be in a stable form to ensure that the waste does not structurally degrade and affect the overall stability of the site through slumping, collapse, or other failure of the disposal unit leading to water infiltration. The requirements in 10 CFR 61.56 further state that:

> (b)(1) A structurally stable waste form will generally maintain its physical dimensions and its form, under the expected disposal conditions such as weight of overburden and compaction equipment, the presence of moisture, and microbial activity, and internal factors such as radiation effects and chemical changes. Structural stability can be provided by the waste form itself, processing the waste to a stable form, or placing the waste in a disposal container or structure that provides stability after disposal.
>
> (b)(2) Liquid wastes, or wastes containing liquid, must be converted into a form that contains as little free standing and noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1% of the volume of the waste when the waste is in a disposal container designed to ensure stability, or 0.5% of the volume of the waste for waste processed to a stable form.

(b)(3) Void spaces within the waste and between the waste and its package must be reduced to the extent practicable.

Waste characteristics are important to an MSR facility because many byproduct materials can exist as a solid or a liquid, depending on material temperature and long-term heat removal paths.

A.5.3 Proposed Rulemaking with Implications for MSR waste

In SECY-20-0098 [37], the staff propose combining two NRC-directed activities that could result in amendments to 10 CFR Part 61.

- An LLRW disposal rulemaking to address the disposal of waste streams (e.g., depleted U) that were not envisioned to be disposed of in significant quantities when 10 CFR Part 61 was originally enacted in 1982.
- A GTCC waste rulemaking that would allow near-surface disposal of GTCC waste beyond the case-by-case approval currently authorized in 10 CFR Part 61. Additionally, the definition of *waste* in 10 CFR Part 61 is proposed for revision so that LLRW that is acceptable for disposal under 10 CFR Part 61 no longer excludes transuranic waste.

As part of SRM SECY-20-0098 [38], the commissioners directed the NRC staff to reexamine the technical basis for the performance objectives in 10 CFR Part 61 and ensure that the compliance period following the closure of a disposal facility is performance based using scientific data. Specifically, "rather than using the same compliance period for disposal sites containing significant amounts of depleted uranium, GTCC, or transuranic waste, the staff should consider a site-specific, graded approach based on when the peak dose is projected to occur or establish a longer compliance period for disposal sites containing significant quantities of mobile, long-lived radionuclides"

This rulemaking could have significant implications for MSRs with expanded fuel cycle facilities on-site. With this rulemaking, used MSR equipment that would currently be classified as GTCC waste requiring on-site storage until disposal in a geologic waste repository may gain a faster site removal path using near-surface disposal. This is discussed further in Section 4.1.1.6.

A.6 10 CFR PART 70, "DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL"

10 CFR Part 70 regulates the licensing required to receive title to, own, acquire, deliver, receive, possess, use, and transfer SNM. However, a separate 10 CFR Part 70 license is not required for possession or use of SNM for the operation of a nuclear reactor licensed under 10 CFR Parts 50 or 52 (10 CFR 70.22(b)).

As discussed in Section 1, a broad-spectrum fuel cycle MSR facility may require a license to possess and use SNM in an on-site fuel fabrication plant. SNM includes the fresh fuel for a reactor. Fuel for LWRs is in the form of fuel rods incorporated into fuel assemblies. Such heterogeneous fuel is relatively easy to inspect upon receipt, account for the individual assemblies, and provide for security on-site. New fuel for an MSR could be in solid or liquid from, could be fresh or recycled material, and could be ready to insert into the reactor core or require further batch processing. Fuel scrap recovery and conversion is also considered under 10 CFR Part 70. Applicants must make the case for whether any similar activities at a broad-spectrum fuel cycle MSR facility rise to the level of fuel fabrication.

10 CFR Part 50 requires preventative measures for criticality accidents. Broad-spectrum fuel cycle MSR facilities must meet the requirements found in 10 CFR 70.24. A discussion of the criticality accident requirements is provided in Section 2.1.2.9.

The primary regulations for fuel cycle facility safety are provided in 10 CFR Part 70.60 through 10 CFR 70.76. NUREG-1520 [39], the NRC's SRP for fuel cycle facilities license applications, describes acceptable methods for meeting the requirements of fuel cycle facilities. NUREG-1520 indicates that safety adequacy demonstration for fuel cycle facilities can be performed without high-fidelity performance and reliability data.

> Consequently, the licensing decision is ultimately based on information with a sufficient level of detail that permits reviewers to understand process system functions and, functionally, how items relied on for safety (IROFS) can perform as intended and be reliable. … For new facilities or new processes at existing facilities, there may not be complete detail or a final design available at the time of licensing. However, sufficient information must be available to permit the staff to understand the theory of operation and function of each IROFS, to have reasonable assurance that all credible accident sequences have been identified, that a sufficient set of IROFS has been defined, and that management measures will be sufficient to ensure IROFS will be available and reliable to perform their intended functions.

10 CFR 70.61 requires each applicant to evaluate its compliance with the performance requirements in 10 CFR 70.61(b), 10 CFR 70.61(c), and 10 CFR 70.61(d) via an ISA performed in accordance with 10 CFR 70.62, "Safety Program and Integrated Safety Analysis." NUREG-1513 provides general guidance on acceptable methods for performing an ISA and identifies the ISA's role in facility safety.

A.7 10 CFR PART 71, "PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIAL"

10 CFR Part 71 regulates the packaging, shipment preparation, and transportation of licensed material. Licensed material must be packaged in NRC-approved containers. 10 CFR Part 71 provides the requirements for obtaining a package license and does not impact licensing a broad-spectrum fuel cycle MSR facility except to note that such containers may not yet exist or be approved for certain MSR waste. MSR facilities are expected to experience more frequent SSC replacements than their LWR counterparts. Therefore, waste packaging, waste transportation, and waste storage will be an important MSR issue.

A.8 10 CFR PART 72, "LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL, HIGH-LEVEL RADIOACTIVE WASTE, AND REACTOR-RELATED GREATER THAN CLASS C WASTE"

10 CFR Part 72 regulates the licensing required to receive, transfer, and possess reactor spent fuel, reactor-related GTCC waste and other radioactive materials associated with spent fuel storage in an ISFSI. The regulations in 10 CFR Part 72 also establish the requirements, procedures, and criteria for issuing a spent fuel storage cask CoC.

MSR technologies must consider spent fuel handling and processing over the plant lifetime. If the ultimate disposition of the spent fuel will include an on-site ISFSI facility, then an applicant for a broad-spectrum fuel cycle MSR facility must describe "the receipt, handling, packaging, and storage of spent fuel, high-level radioactive waste, and/or reactor related GTCC waste as appropriate, including how the ISFSI will be operated". Dose evaluations and a discussion of SSCs must be included. A separate 10 CFR Part 72 license will be required for a collocated ISFSI. EP (10 CFR 72.32) must be coordinated with the emergency plan described in 10 CFR 50.47.

A.9 10 CFR PART 51, "ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS"

10 CFR Part 51 contains environmental protection regulations applicable to the NRC's domestic licensing and related regulatory functions. 10 CFR Part 51 implements the National Environmental Policy Act of 1969, as amended [40].

A.9.1 10 CFR 51.23, "Environmental Impacts of Continued Storage of Spent Nuclear Fuel Beyond the Licensed Life for Operation of a Reactor"

NUREG-2157 [41] provides the environmental impacts of continued storage of spent nuclear fuel (SNF) beyond the licensed life for reactor operation. NUREG-2157 is incorporated into the requirements specified in 10 CFR 51.23 to provide environmental impacts for

- a short-term time frame, which includes 60 years of continued spent fuel storage beyond the licensed life for operation of a reactor,
- an additional 100 year time frame (i.e., 60 years plus 100 years) to address the potential for delay in repository availability, and
- an indefinite time frame to address the possibility that a repository never becomes available.

High-temperature gas-cooled reactors and liquid metal fast reactors are not within the scope of NUREG-2157 because of the differences in the fuel form compared with LWRs. Although MSR fuel is not specifically mentioned, implicit in this assertion is that MSR fuel will also not be covered by NUREG-2157. Nonetheless, the environmental impact time frames stated still must be considered.

NUREG-2157 states that during the short-term storage time frame, LWR spent fuel in the spent fuel pool continues to generate decay heat from radioactive decay. The rate at which the decay heat is generated decreases the longer the reactor has been shut down. After approximately 5 years in a pool, LWR spent fuel can be transferred to a licensed dry storage system (DSS). NRC and licensee experience with ISFSIs and cask certification indicate that spent fuel can be safely and effectively stored using dry cask storage technology. No safety issues have been experienced with dry cask storage. LWR spent fuel becomes cooler and less radioactive over time, and the fuel cladding is relatively robust. Essentially, LWR spent fuel storage and handling becomes more stable with time. This provides an opportunity for LWRs to meet the indefinite storage provision in NUREG-2157 with dry storage and monitoring if a spent fuel repository never becomes available.

Unlike LWR fuel, MSR fuel salt tends to become less stable with time after irradiation. There is no cladding, and the radiolytic release of gas from the fuel salt will occur if fuel stabilization is not initiated. Some of the FP halide vapors can include radioactive materials, and ^{36}Cl is a long-lived beta emitter. Additionally, fluoride fuel salt can also release UF_6 or UO_2F_2 when cooled if the salt is not handled properly. MSR technologies must address the necessary controls for the indefinite storage of fuel on-site. These will likely be more complex than comparable LWR indefinite storage systems.

A.10 10 CFR PART 73, "PHYSICAL PROTECTION OF PLANTS AND MATERIALS"

10 CFR Part 73 regulates the establishment and maintenance of a physical protection system that will have capabilities for protecting SNM at production and utilization sites. The entire broad-spectrum fuel cycle MSR facility must be capable of defending against the design basis threat identified in 10 CFR Part 73, including appropriate safeguards systems to "protect against acts of radiological sabotage and to prevent the theft or diversion of special nuclear material".

The typical sabotage threats for MSR technologies may be different or on a different scale than the threats associated with LWRs. These threats must be noted when discussing MSR plant security. For example, MSRs will use more automation for plant maintenance and other activities because of the high radiation fields associated with plant operation. More plant automation can lead to greater cyber-security concerns. Additionally, the low-pressure MSR containment structure will be thinner, perhaps leading to different opportunities to penetrate containment.

Proposed rule changes for 10 CFR Part 73 for advanced reactors include a PB approach to physical security and a graded approach to cyber security and access authorization [36], [39], [40].

A.11 10 CFR PART 74, "MATERIAL CONTROL AND ACCOUNTING OF SPECIAL NUCLEAR MATERIAL"

10 CFR Part 74 regulates the control and accounting of SNM at nuclear facility sites. This is traditionally identified as NMAC. Requirements for the control and accounting of source material at enrichment facilities are also included. The general conditions and procedures for submitting a license application for the activities covered in 10 CFR Part 74 are detailed 10 CFR Part 70.

10 CFR Parts 50 and 70 and the proposed draft text for 10 CFR Part 53 all define SNM as:

- Plutonium, U^{233}, U enriched in the isotope U^{233} or in the isotope U^{235}, and any other material that the NRC pursuant to the provisions of Section 51 of the AEA, as amended, determines to be SNM but does not include source material or
- Any material artificially enriched by any of the foregoing but does not include source material.

10 CFR Part 53 proposes to define *fuel* as SNM, discrete elements that physically contain SNM, and homogeneous mixtures that contain SNM, intended to or used to create thermal power in a commercial nuclear plant.

10 CFR Part 74 requires that all nuclear facilities, including broad-spectrum fuel cycle MSR facilities, provide for NMAC for all SNM in the facility. Homogenous MSR fuel would be included under the traditional definition of SNM. However, under the proposed definition for fuel in 10 CFR Part 53, the use of homogenous fuel forms is specifically included.

The requirements of 10 CFR 74.17 specify that an inventory of SNM be taken annually and the results reported to the NRC by plant and total facility. NMAC techniques must be innovative for MSR liquid fuel forms.

A.12 10 CFR PART 110, "EXPORT AND IMPORT OF NUCLEAR EQUIPMENT AND MATERIAL"

10 CFR Part 110 regulates the export and import of nuclear equipment and material. The definitions included in 10 CFR 110.2 define HEU, LEU, and natural U.

- *HEU* means U enriched to 20% or greater in the isotope ^{235}U.
- *LEU* means U enriched below 20% in the isotope ^{235}U.
- *Natural U* means U as found in nature, containing about 0.711% of ^{235}U, 99.283% of ^{238}U, and a trace (0.006%) of ^{234}U.

Definitions for HEU and LEU are similar in 10 CFR Part 50. These definitions are problematic for MSR technologies contemplating the use of a Th-U fuel cycle.

A Th-U breeder reactor can remain critical based on the production of ^{233}U. Additionally, fuel salt containing ^{233}U could be separated and processed for use in other reactors. However, 10 CFR Parts 110 and 50 do not include ^{233}U in the current definition of LEU. Therefore, the existing definitions of LEU in 10 CFR 110.2 and 10 CFR 50.2 must be updated to enable future licensees to develop Th-U plants that meet regulatory expectations.

A.13 10 CFR PART 30, "RULES OF GENERAL APPLICABILITY TO DOMESTIC LICENSING OF BYPRODUCT MATERIAL"

10 CFR Part 50 defines byproduct material as "(1) Any radioactive material (except SNM) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or using SNM; or (2) Any discrete source of radium-226 that is produced, extracted, or converted after extraction, for use for a commercial, medical, or research activity".

10 CFR Part 30 regulates domestic licensing of byproduct material [1], [3] in that "no person shall manufacture, produce, transfer, receive, acquire, own, possess, or use byproduct material except as authorized in a specific or general license issued in accordance with the regulations in Part 30".

MSRs may strip actinides from spent fuel for waste stabilization. Although the actinides may be recycled or held for disposal, the remaining radioactive waste is technically a byproduct of the MSR operation. This could provide an MSR licensee with a byproduct material that is useful as a sealed industrial heat source. Any MSR facility contemplating the sale, distribution, or recycling of byproduct material resulting from expanded fuel cycle activities at its site must consider the licensing requirements found in 10 CFR Part 30.

A.14 INSIGHTS FROM NONPOWER REACTOR LICENSING

ORNL provided infrastructure support to the NRC staff for the regulatory review of nonpower MSRs [45]. The report proposed an MSR technology guidance alternative for NUREG-1537 Parts 1 and 2. The proposed text was based on the current version of NUREG-1537 as amended by the Aqueous Homogeneous Reactor interim staff guidance (ISG) and was endorsed by NRC staff. This section summarizes the proposed nonpower MSR guidance [45].

The introductory sections for Parts 1 and 2 of the NUREG-1537 nonpower MSR guidance defined terms often used when discussing an MSR. The generic MSR technology concepts and terms are used as illustrative examples and are not intended to identify, exclude, or limit any specific MSR technology under development. Such terminology may be useful for regulatory interactions regarding MSR technologies. However, the intention is not to set terminology for all possible MSR technologies. Vendors and applicants should introduce appropriate substitute terminology where necessary and provide definitions. The following terms are specific to MSR technologies or have a definition that is different from its LWR counterpart.

- **Active reactor core:** In an MSR, the active reactor core is the vessel region occupied by the fuel salt, which is where most prompt neutrons are generated and most fissions occur. In an MSR, the active reactor core geometry might change as a result of changes in density and voiding of the solution.
- **Coating or cladding:** Coating or cladding is the intervening protective layer of material between the fuel salt and the structural container alloy. Coating or cladding also includes surface modifications of the structural alloy to enhance its chemical or mechanical performance by altering its microstructure or composition (e.g., carbiding, phosphiding, or nitriding the surface).
- **Control elements:** Control elements are the objects employed to adjust reactivity. Control elements can act through fuel displacement, neutron absorption, neutron reflection, neutron spectral

adjustment, or a combination of these methods. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned.

- **Emergency cooling system:** The emergency cooling system provides decay heat removal from the reactor fuel following an accident (e.g., a direct reactor auxiliary cooling system or a reactor vessel auxiliary cooling system). Similar cooling systems for fuel drain tanks are also included, as well as systems such as in-floor heat pipes, to provide cooling to fuel located in outer containment layers in the event of fuel system breach–type accidents.
- **Fuel barrier:** The fuel barrier is the portion of the fuel system boundary in contact with the liquid fuel after being added to the fuel circuit and before being transferred to waste handling—principally, the vessel; chemical processing system boundary; drain tank, if used; heat exchanger; cooling thimbles; control element thimbles; instrumentation thimbles; piping; tanks; and valves.
- **Fuel system boundary:** The fuel system boundary is the material that mitigates the release of radionuclides from the reactor fuel, including volatile FPs (e.g., Kr, Xe, I). For an MSR, this includes the vessel; drain tank, if used; cooling thimbles; heat exchangers; chemical processing system boundary; waste-handling tank; pumps; valves; and piping. It essentially includes the radionuclide barrier and fuel barrier.
- **Gas management system:** The gas management system is the cover gas system provided to capture volatile FPs (e.g., Kr, Xe, I) until ultimate discharge and to provide venting of any pressure/density transients that could result in damage to the vessel or the fuel salt/primary cooling system salt heat exchanger, thus resulting in loss of the fuel system boundary.
- **Heat dissipation system:** The heat dissipation system is a set of components or systems that interface with the primary cooling system to provide the principal means of transferring the heat from the active reactor core to an ultimate heat sink. The heat dissipation system might use a variety of coolants (e.g., salt, liquid metal, gas, water) but does not contain fuel.
- **Neutron moderator:** In an MSR, the neutron moderator is the materials in or near the active reactor core that comprise light elements (e.g., H, Be, C). Moderators are generally solid in form.
- **Primary cooling system:** The primary cooling system directly interfaces with the fuel system boundary at the fuel salt/primary cooling system salt heat exchangers to provide the principal means of removing heat from the fuel salt during operation by transferring the heat to the heat dissipation system. The primary cooling system may employ a variety of coolants (e.g., nitrate salt, halide salt) but does not contain fuel.
- **Radionuclide barrier:** The radionuclide barrier is the portion of the fuel system boundary that serves as the innermost low-leakage barrier to the radionuclides within the gas management system.
- **Reactor fuel:** In an MSR, reactor fuel is the fuel salt that consists of fissionable and possibly fertile halide salts, FPs, and generally solvent halide salts.
- **Vessel:** For an MSR, the vessel is the structure that contains the active reactor core. In certain design configurations, other components such as heat exchangers might reside in the vessel but only outside the active reactor core.

The following subsections discuss some of the specific NUREG-1537 guidance changes that were suggested to accommodate nonpower MSR technologies. There are also implications for power MSR applications.

A.14.1 Chapter 4, "MSR Description"

LWR fuel elements that comprise rods, plates, or pins with fuel cladding acting as the initial FP barrier were eliminated. A liquid-fueled MSR uses homogenous fuel with no cladding. Therefore, except for soluble FPs, the initial MSR FP barrier is the fuel system boundary. The integrity of the MSR fuel system boundary replaces the function of LWR fuel cladding. Consequently, control of reactor fuel chemistry must be considered by an MSR applicant. The formation of gaseous, soluble, and insoluble FPs in the fuel salt will affect system chemistry. Chemistry control, mitigation, and redox tactics must be identified by the applicant for short- and long-term changes in the chemistry of the fuel salt. MSR fuel qualification is generally a thermophysical and thermochemical process rather than a mechanical process, as it is for heterogeneous LWR fuel [46].

Neutron-absorbing control rods are the typical means for controlling reactivity in an LWR, but there are many more ways to control reactivity in an MSR. MSR designs can control reactivity through fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned. An MSR applicant must address reactivity control.

Some delayed neutrons will be produced outside the active reactor core because of the nature of the flowing fuel solution and its movement out of the active reactor core. Any sudden reduction in flow, such as a transition from forced flow to natural circulation flow, will result in a subsequent reactivity addition. Therefore, applicants must consider the ability to control the core during normal operation and core flow changes.

Gaseous, soluble, and insoluble FPs are generated within the fuel salt as an MSR is operated. Gaseous FPs accumulate within the vessel or within a cover gas at a free surface boundary. An applicant must describe the design of the system for removing FP gases from the core and cover gas of the MSR. Any decay heat removal provided by the gas management system to the overall reactor cooling must be discussed.

A.14.2 Chapter 5, "MSR Cooling Systems"

Liquid homogeneous MSR fuel salt dissipates heat through a heat exchanger to cooling systems that do not include fuel. Therefore, the former heterogeneous fuel-based primary coolant system no longer exists with the same function as provided for LWR fuel. The applicant should address fuel salt heat transport within the fuel system boundary to the fuel salt/primary cooling heat exchangers. Subsequently, the applicant should describe how reactor heat is removed and transferred to the environment, typically through two or more cooling loops.

Applicants should describe any cleanup or salt polishing system included in the MSR design to handle the buildup of soluble and insoluble FPs. Additionally, applicants should discuss provisions for salt makeup in the various MSR salt loops. Applicants should also discuss the safe storage of the fuel salt in the event of a design-basis accident or for fuel system boundary maintenance. This storage often is provided by a drain tank.

A.14.3 Chapter 6, "Engineered Safety Features"

Multiple confinement or containment boundaries might be included for an MSR design, depending on the relative location of the gas management system and other cleanup systems to the fuel system boundary. An MSR technology may employ multiple barriers to satisfy confinement or containment requirements (i.e., functional containment). Therefore, although the traditional function of confinement and containment continues to be applicable to MSRs, the concepts may be altered slightly.

Loss-of-coolant accidents figure prominently in the discussion of the LWR engineered safety feature requirements because of the importance of cooling the heterogeneous fuel and preventing core melt, which can lead to uncoolable core geometry. However, the MSR loss-of-coolant accident concern is at the fuel system boundary, which must be protected from overheating because it can result in a boundary material failure, which could disrupt continued fuel salt cooling. Therefore, an MSR applicant should discuss how the homogenous fuel in an MSR maintains a decay heat removal path for continued boundary integrity. This is a crucial MSR safety issue and would constitute the technology-engineered safety features. Decay heat removal can be provided by multiple means, including a drain tank with a separate cooling system, a direct reactor auxiliary cooling system that passively rejects heat to air via a salt system, or a reactor vessel auxiliary cooling system that passively rejects heat to outside air via a water or air system.

A.14.4 Chapter 7, "Instrumentation and Control Systems"

Applicants should consider and include instrumentation and control parameters, such as temperature, flow, and level for the fuel system boundary, primary cooling system, gas management system, fuel salt cleanup system, fuel handling (i.e., addition) system, and any fuel cycle processing systems. As noted in Section A.14.1, reactivity controls should also be addressed. In an MSR, reactivity can be adjusted by changing flow, manipulating the control elements in their various forms, operating the fuel salt cleanup system, or adding fuel or fuel salt to the vessel.

A.14.5 Chapter 9, "Auxiliary Systems"

MSR auxiliary systems could include homogeneous MSR fuel handling and storage of SNM used for new and irradiated reactor fuel, including associated components (e.g., tanks, valves, pumps, instrumentation, controls), processes (e.g., chemical blending, purification, SNM transfers, waste storage, preparation for shipment), criticality monitoring, vaults, shielding, and contamination control. MSR applicants should discuss these topics and address the fuel form during storage and handling. For facilities designed to fabricate fuel on-site, the discussion should include a description of the form in which the fissile material is received, how and where it is stored before use, and how it is blended into a useable liquid fuel, including criticality control measures and monitoring.

Using liquid fuel adds numerous new issues that pertain to quantifying the quantity of byproduct, source, and SNM that differ from issues that pertain to using heterogeneous fuel elements. Applicants should define an effective means for limiting the SNM at the reactor site to establish an envelope used for safety and security analyses. Applicants should also consider the fact that fuel isotope quantities in the fuel salt will change during normal operation as U^{235} is depleted and that U^{238} converts to Pu^{239} or as Th^{232} is converted to U^{233}. Applicants should also describe how the quantity of byproduct materials to be created by routine operation will be limited.

A.14.6 Chapter 11, "Radiation Protection Program and Waste Management"

Waste management is significantly different for an MSR than for an LWR. In an MSR, soluble, nonsoluble, and gaseous FPs are released to the liquid salt fuel solution and contained by the fuel barrier. In an LWR, FPs are released to the fuel rod gap space and contained by the fuel clad. MSR gaseous FPs will migrate directly to the gas space in the top of the vessel or elsewhere and enter the gas management system to be processed within the radionuclide barrier. The gaseous FPs may require holdup for decay or further treatment before being recycled (i.e., into cover gas), released to the environment, or disposed of as waste. If applicable, residue from mechanical cleanup or polishing of soluble FPs will require treatment as radioactive waste. Criticality concerns could also exist in the waste treatment process for accident analysis and normal operation. Criticality concerns are addressed in 10 CFR 50.68 and 10 CFR Part 70. MSR applicants should address the unique aspects of MSR waste streams.

A.14.7 Chapter 13, "Accident Analyses"

Nonpower reactors use a maximum hazard analysis approach to accident analyses. MSRs that use liquid fuel produce both liquid and gaseous FPs that must be contained within the facility barriers rather than within heterogeneous fuel cladding. This impacts consideration of the maximum hazard analysis. Limiting phenomena for MSRs include fuel salt precipitation, fission-product precipitation and plate out on fuel system boundary surfaces, fuel salt chemistry/physical properties, delayed neutron production, core voiding, and tritium production.

A previous ORNL report [45] provides a list of postulated events that are more applicable to MSRs and includes typical parameters to be tracked during the postulated events. Applicants must identify their own technology-specific scenarios as part of their application.

A.14.8 Chapter 14, "Technical Specifications"

A previous ORNL report [45] adapted the discussion of technical specifications for nonpower reactors to nonpower MSRs. This included terminology updates to address specific aspects of MSR technology. Elements of the nonpower MSR technical specifications may be applicable to a power MSR application.

A.15 INTERNATIONAL COOPERATION

International cooperation on specific MSR technologies is beyond the scope of this report. However, international cooperation on basic MSR research is an acceptable activity. Conversely, international cooperation on specific technology applications is likely prohibited by the AEA [3] under Section 123(a)(7), which states:

> No cooperation with any nation, group of nations or regional defense organization shall be undertaken until
>
> a. the proposed agreement for cooperation has been submitted to the President, which proposed agreement shall include the terms, conditions, duration, nature, and scope of the cooperation; and shall include the following requirements:
>
> (7) …a guaranty by the cooperating party that no material transferred pursuant to the agreement for cooperation and no material used in or produced through the use of any material, production facility, or utilization facility transferred pursuant to the agreement for cooperation will be reprocessed, enriched or (in the case of Pu, U^{233}, or uranium enriched to greater than twenty percent in the isotope 235, or other nuclear materials which have been irradiated) otherwise altered in form or content without the prior approval of the United States.

Any MSR would contain nuclear material that has been irradiated following startup. Therefore, necessary safety-related activities—such as maintaining the redox state of the fuel salt or operational activities, including mechanical filtering of solid FPs from the fuel salt—would require specific authorization. Even online refueling by adding LEU to the active fuel salt circuit (i.e., altering the content of irradiated nuclear material) could trigger a requirement for specific authorization. The specific wording of the regulations implies that the fuel salt would not be allowed to freeze or even drain into a critically safe tank during an accident because both would alter the form (i.e., phase in the former case and shape in the latter case) of the fuel salt.

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APPENDIX B. REGULATORY PRECEDENCE

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Current regulations are tailored to efficiently evaluate the safety issues raised by the once-through fuel cycle implemented with solid fuel rods at large LWRs. The presumptions embedded into the regulatory process about how fuel will be fabricated, used, and stored do not match the characteristics of MSRs, which will incorporate additional elements of the fuel cycle beyond those of large LWRs. This appendix provides some examples of alternative approaches to fuel cycle licensing.

B.1 SHINE MEDICAL TECHNOLOGIES

B.1.1 SHINE Description

SHINE is a nonreactor-based, subcritical fission process, medical isotope production facility owned and operated by SHINE Medical Technologies Inc. in Janesville, Wisconsin. The purpose of the facility is to produce ^{99}Mo. Molybdenum-99 is the precursor of ^{99m}Tc, a diagnostic imaging isotope used in over 30 different diagnostic imaging procedures and in 80% of nuclear diagnostic procedures worldwide [1].

Previous techniques for producing ^{99}Mo involve the use of HEU targets. The SHINE irradiation unit (IU) components include a neutron driver, a subcritical operating assembly, a light-water pool, and surrounding biological shielding. The irradiation facility comprises eight IUs and their supporting systems. Each IU is licensed as a separate utilization facility. The SHINE facility also includes the radioisotope production facility (RPF) in which the irradiated material is processed to separate medical isotopes. The RPF includes material packaging for shipment to medical customers. The separate RPF is licensed as a production facility [1].

In the IU, a deuterium source is accelerated into a tritium gas to spall neutrons. The neutrons fission in an aqueous LEU uranyl sulfate target solution. The fission process creates the desired ^{99}Mo as a byproduct. The irradiated target solution is then processed in the RPF to extract and purify the ^{99}Mo. Radioactive waste materials are processed and/or converted to solid wastes for shipment to off-site disposal facilities. The SHINE facility is designed to be a zero radioactive liquid effluent discharge facility [1].

Like a typical MSR facility, the entire SHINE irradiation facility and RPF constitute a radiologically controlled area because radioactive materials are present throughout the SHINE facility buildings [1]:

- Irradiation facility
  - Eight IU cells
  - Off-gas shielded cells
  - Tritium production areas
- RPF
  - Target solution preparation and storage areas
  - Uranium extraction hot cell
  - Extraction, purification, and packaging hot cells
  - Pump transfer cell
  - Waste processing hot cells
  - Noble gas storage cell
  - Radiologically controlled area ventilation equipment areas
- RPF (other areas)
  - Receiving area
  - Shipping area
- Waste staging and shipping building

SHINE submitted a CP application in February 2013 and received a CP in February 2016. The CP safety evaluation is documented in NUREG-2189, *Safety Evaluation Report Related to SHINE Medical Technologies, Inc. Construction Permit Application for a Medical Radioisotope Production Facility* [2]. Construction of the 91,000 ft^2 five-building facility began in August 2017 and is ongoing. The SHINE production facility building includes the irradiation facility and the RPF. Although these are functionally separate, they share the same building. At or near the conclusion of construction, SHINE must apply for an OL to permit isotope production.

B.1.2 Licensing Approach

Although the SHINE facility is a production facility, the preliminary safety evaluation report (PSAR) associated with the CP application was submitted and reviewed by generally following content and organization of NUREG-1537 Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content" [3], as augmented by the *Final Interim Staff Guidance (ISG) Augmenting NUREG-1537, Part 1, Guidelines for Preparing and Reviewing Applications for Licensing Non-Power Reactors: Format and Content for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors* [4]. SHINE compared its technology with other nonpower reactor facilities and noted that each IU will be operated so that the effective neutron multiplication factor is much less than 1.0. After consultation with NRC staff, the application and review process provided by NUREG-1537 was deemed to be the most appropriate application and review [5], [6] approach [1]. This conclusion required a rule change for the definition of a utilization facility to include accelerator-driven subcritical operating assemblies.

Potential SHINE technology DBAs were identified by applying hazard analysis methodologies to evaluate the preliminary design of the facility and processes for potential hazards, initiating events (IEs), scenarios, and associated controls. These methodologies were applied to the irradiation facility and the RPF using the following hazard analysis methodologies [1]:

- Hazards and operability study
- Preliminary design hazard analysis

Based on the analyses, an ISA was prepared in accordance with NUREG-1513. The list of accident categories and IEs identified in NUREG-1537 Part 1 [3] and the associated *Final ISG Augmenting NUREG-1537 Part 1* [4] were the basis for identifying potential DBAs. According to the SHINE PSAR, the following accident categories and IEs are addressed for the SHINE facility. Some listed items are applicable to the irradiation facility, some are applicable to the RPF, and some are applicable to both [1].

- Maximum hypothetical accident
- Insertion of excess reactivity/inadvertent criticality
- Reduction in cooling
- Mishandling or malfunction of target solution
- Loss of off-site power
- External events
- Mishandling or malfunction of equipment that affect the primary system boundary
- Large undamped power oscillations
- Detonation and deflagration in the primary system boundary
- Unintended exothermic chemical reactions other than detonation
- Primary system boundary system interaction events
- Facility-specific events
    - Inadvertent exposure to neutrons from the neutron driver

    - IF fires
    - Tritium purification system malfunction
- Critical equipment malfunctions
- Inadvertent nuclear criticality in the RPF
- RPF fire
- Accidents with hazardous chemicals

As a result of transportation and on-site operations involving radioactive materials, members of the public and the staff are at risk to receive some exposure. Analyses associated with the licensing process indicate that doses to workers and the public during normal operation are within the limits of 10 CFR 20.1201 and 20.1301, respectively. Likewise, potential exposures to the public from postulated accidents are also within the limits of 10 CFR 20.1201 and 20.1301, respectively [1].

B.1.3 Summary

The SHINE facility and generic MSR operations have many similarities. There may be an argument for some MSR technologies to follow a licensing path similar to that taken for SHINE. Although NUREG-1537 [3] is directly applicable to nonpower reactors and is generally applied solely to research and test reactors, the use of this application and review path for the SHINE nonreactor production technology may set some precedent for this approach.

NUREG-2189 Appendix A, "Post Construction Permit Activities – Construction Permit Conditions and Final Safety Analysis Report Commitments" [2], provides conditions on the CP. Numerous conditions are placed on issues such as a criticality accident alarm system, criticality accident analyses, solution precipitates, ventilation, and seismic evaluations. All noted SHINE license conditions are relevant to an application for a broad-spectrum fuel cycle MSR facility.

B.2 MIXED OXIDE FUEL LICENSING

The United States and the former Soviet Union (Russian Federation) began dismantling thousands of nuclear weapons when the Cold War ended in the late 1980s. The dismantlement resulted in large quantities of surplus weapons-grade HEU and Pu. Subsequently, in September 2000, the United States and the Russian Federation signed an agreement committing each country to dispose of 34 MT (approximately 75,000 lb) of surplus Pu. DOE evaluated different strategies to dispose of this material and ultimately developed the Surplus Plutonium Disposition Program. Under this program, DOE planned to convert the surplus weapons-grade Pu into mixed oxide (MOX) fuel to be irradiated in commercial nuclear power reactors [7].

B.2.1 Program Description

DOE planned to design, construct, and operate the Mixed Oxide Fuel Fabrication Facility and eventually supply commercial fuel to an affiliated nuclear utility to be irradiated in its reactor. The facility would be owned by the US government and would be used to dispose of surplus Pu and some waste from DOE's nuclear processes (e.g., alternate feedstock) [7]. The NRC was granted regulatory and licensing authority over the facility to use existing fuel cycle facility regulations to license the facility. The NRC developed NUREG-1718, *Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility* [8], to facilitate the review. On February 28, 2001, the applicant selected by DOE to design and build the facility submitted a construction authorization request to the NRC seeking authority to construct the fuel fabrication facility on the DOE's Savannah River Site in South Carolina [9].

B.2.2 Licensing Approach

Under the applicable requirements of 10 CFR Part 70, before a license to possess and use licensed material at the Mixed Oxide Fuel Fabrication Facility may be issued, the NRC must first authorize facility construction. The requirements found in 10 CFR 70.23(b) state that the NRC will approve construction of a Pu processing and fuel fabrication facility if it finds that the design bases of the principal SSCs and the QA program provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. Other 10 CFR Part 70 requirements applicable to authorizing construction of the facility include the following baseline design criteria listed in 10 CFR 70.64(a) [7]:

- Quality standards and records
- Natural phenomena hazards
- Fire protection
- Internal environmental conditions and dynamic effects
- Chemical protection
- Emergency capability
- Utility services
- Inspection, testing, and maintenance
- Criticality control
- Instrumentation and controls

Acceptable risk control must be demonstrated as part of the license review by the NRC staff to grant a license to possess and use SNM. The starting point for demonstrating acceptable risk control is the safety assessment of the design bases: the ISA [8]. The Mixed Oxide Fuel Fabrication Facility applicant submitted an ISA summary for the facility as set forth in 10 CFR 70.62(c). Each control identified by the applicant to meet facility performance requirements must be designated as an item relied on for safety (IROFS). Each IROFS must be available and reliable to perform its intended function when needed [7].

B.2.3 Integrated Safety Analysis

The staff evaluated the applicant's license application to ensure that it contained appropriate commitments, including commitments to [7]

- perform and maintain an ISA,
- compile and maintain process safety information,
- engage personnel with appropriate training to conduct the facility ISA,
- use appropriate methods to conduct the ISA, and
- ensure that the ISA stays accurate and up to date.

Subsequently, the staff evaluated [7] the ISA performed by the applicant to identify and evaluate the hazards and potential accidents associated with the facility and to establish engineered and administrative controls to ensure that facility operation will be within the bounds of the 10 CFR 70.61 performance requirements. The staff confirmed that the ISA summary

- identified the hazards at the facility,
- analyzed for accident sequences through the use of process hazards analysis,
- evaluated and assigned consequences to the accident sequences, and
- evaluated the likelihood of each accident consistent with the guidance in NUREG-1718 [8].

The staff evaluated the facility license application in the following areas.

B.2.3.1 Nuclear Criticality Safety

The NRC staff evaluated the nuclear criticality safety program [7] detailed in the license application. Important elements of the nuclear criticality safety program included (1) staff qualified to develop, implement, and maintain the program in accordance with the description of facility organization, administration, and management measures in the license application; (2) planned operation based on technical practices sufficient to ensure that licensed material will be possessed, stored, and used safely according to the requirements of 10 CFR Part 70; (3) development, implementation, and maintenance of a criticality accident alarm system in accordance with the requirements of 10 CFR 70.24; and (4) established safety limits and controls sufficient to ensure subcriticality, including an appropriate margin of subcriticality for safety, and the baseline design criteria of 10 CFR 70.64.

Pursuant to 10 CFR 70.66(a), the NRC staff found [7] that there is reasonable assurance that the MOX facility applicant will establish controls on all credible accident sequences leading to criticality sufficient to ensure that (1) credible accident sequences will be highly unlikely, (2) all processes will be subcritical under normal and credible abnormal conditions, and (3) all processes will adhere to the double contingency principle. The double contingency principle means that process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

B.2.3.2 Fire Protection

The NRC staff evaluated the applicant's proposed fire equipment, facilities, and procedures to ensure that they provide a reasonable level of assurance that adequate fire protection will be provided and maintained for those IROFS to meet the safety performance requirements and the baseline design criteria of 10 CFR 70.61 [7].

B.2.3.3 Chemical Safety

The NRC staff evaluated the applicant's facility and system design and facility layout pertaining to chemical safety to ensure that they are based upon defense-in-depth practices. Pursuant to 10 CFR Part 70, the facility design and items relied on for safety must provide reasonable assurance of chemical safety at the facility for routine operations, off-normal conditions, and potential accidents. The staff found that the facility license application and ISA summary identified chemical process hazards and potential accidents and established safety controls to ensure safe facility operation [7].

B.2.3.4 Radiation Safety

An effective radiation protection program is essential for meeting the requirements of 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations"; 10 CFR Part 20; and 10 CFR Part 70 for a license to possess and use SNM. The NRC staff reviewed elements of the applicant's radiation protection program, including the following:

- The program to ensure that occupational radiological exposures are ALARA
- Radiation protection procedures
- Radiation safety training
- Air sampling program
- Radiological contamination controls
- Respiratory protection program
- Personnel dose monitoring program

B.2.3.5 Plant Systems

The applicant provided design basis information for the MOX process and identified IROFS for the overall facility. Plant system controls identified by the applicant as IROFS must be available and reliable to perform their intended function when needed. A summary of important Mixed Oxide Fuel Fabrication Facility systems that include IROFs are discussed as follows.

- **Aqueous polishing process and chemistry:** The applicant provided design basis information for chemical process safety identified as IROFS for the facility. The staff evaluated the design bases of the IROFS pursuant to 10 CFR 70.23(b) to ensure that they will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.
- **Ventilation and confinement systems:** The applicant provided information for the ventilation and confinement systems that are identified as IROFS for the facility. The staff evaluated the system design and operation pursuant to 10 CFR Part 70 to ensure that the systems are adequately available and reliable to perform their intended functions when needed.
- **Electrical systems:** The applicant provided information regarding the design of the electrical power systems, including concepts for defense in depth. Pursuant to 10 CFR 70.64, the staff evaluated the electrical systems design and operation and whether it will fulfill the functional requirements of providing reliable power to enable the facility IROFS to reliably perform their required safety actions.
- **Instrumentation and control systems:** The applicant provided information regarding the design of the instrumentation and control systems, including concepts for defense in depth. The staff evaluated the instrumentation and control system design and operation pursuant to 10 CFR 70.64 to ensure that the various systems will be available and reliable to enable the facility IROFS to perform their required safety actions when needed.
- **Material handling systems:** The applicant provided information regarding the design of proposed material handling equipment, facilities, and procedures. Pursuant to 10 CFR 70.64, the staff evaluated the ability of the material handling systems to provide a reasonable level of assurance that load-handling events that cause a release of radioactive material or radiation exposures in excess of the performance requirements of 10 CFR 70.61 are highly unlikely, given the use of the designated IROFS, codes and standards, and management measures, as well as the facility QA program.
- **Fluid transport systems:** The applicant provided information regarding the design of proposed fluid transport equipment, controls, and procedures. Pursuant to 10 CFR 70.64, the staff evaluated the ability of the fluid transport systems to provide a reasonable level of assurance that events that cause a release of radioactive material or radiation exposures in excess of the performance requirements of 10 CFR 70.61 are highly unlikely, given the use of the designated IROFS, codes and standards, and management measures, as well as the facility QA program.

B.2.3.6 Material Control and Accounting

The license application must discuss the nuclear material control and accounting (MC&A) program that tracks and verifies the SNM that is on-site pursuant to 10 CFR Part 74. The material control system must be designed to protect against, detect, and respond to the loss or diversion of nuclear materials. The material accounting system must be designed to determine the quantities of nuclear materials in a licensee's possession, maintain knowledge of such materials, verify the presence of such materials, and detect the loss or diversion of such materials [9]. Requirements for an MC&A program are discussed in Section 2.10 of this report.

The NRC states [10] that MC&A regulations ensure that the information collected by the licensee about SNM is accurate, authentic, and sufficiently detailed to enable a licensee to (1) maintain current knowledge of its SNM and (2) manage its program for securing and protecting SNM. The MC&A program, together with physical protection of facilities and information security requirements, comprise the primary elements of the NRC's SNM safeguards program. The MC&A component of the larger safeguards program helps ensure that SNM within a licensed facility is not stolen or otherwise diverted from the facility.

B.2.3.7 Physical Protection

Applicants must outline and discuss their physical protection plan and show that it meets the requirements in 10 CFR 73.20, "General Performance Objectives and Requirements"; 10 CFR 73.45, "Performance Capabilities for Fixed Site Physical Protection Systems"; 10 CFR 73.46, "Fixed Site Physical Protection Systems, Subsystems, Components, and Procedures"; and Appendices B, C, G, and H to 10 CFR 73.46 [9]. Requirements for a physical protection program are discussed in Section 2.9 of this report.

The NRC states [10] that together with the MC&A program, a facility physical protection program and information security requirements comprise the primary elements of the NRC's SNM safeguards program.

B.2.3.8 Emergency Management

Applicants must consider all appropriate accident initiators and ensure that the baseline design criteria for emergency capability are met. Consideration for hazards from internal, external, and natural phenomena must be included. Any accident initiators that are not used as part of the facility design basis must be adequately explained [9].

B.2.4 Summary

The Mixed Oxide Fuel Fabrication Facility licensing experience discussion is included in this report because it is directly applicable to frontend fuel cycle activities that may parallel frontend activities at an MSR plant site. Although the MOX facility was never completed, considerable NRC staff review efforts were made that is recent and relevant to any MSR application. Many plant aspects that must be discussed and considered by an MSR applicant are included in the MOX facility licensing experience and documentation.

B.3 BLENDED LEU FUEL

In 1996, DOE began implementing plans to support the disposition of HEU. The Tennessee Valley Authority (TVA) and DOE entered into an agreement with Nuclear Fuel Services (NFS) in Erwin, Tennessee to support an HEU downblend process to manufacture commercial nuclear power plant fuel. The project was called the Blended Low Enriched Uranium (BLEU) project, and it used off-specification HEU that included high concentrations of unusual U isotopes. In 1998, TVA inserted four BLEU project pressurized water reactor (PWR) lead test assemblies (LTAs) into Sequoyah Unit 2 for testing. TVA subsequently used BLEU boiling water reactor fuel in Browns Ferry Units 2 and 3 starting in 2005. BLEU PWR fuel was later introduced at Sequoyah in 2008. Finally, in 2012, TVA also began using BLEU boiling water reactor fuel in Browns Ferry Unit 1.

B.3.1 BLEU Project Description

HEU production has primarily been produced to support military purposes. A limited amount of HEU is also used in research reactors. HEU is classified as U that has been enriched to a concentration of 20 wt % or more ^{235}U, which is the only naturally occurring fissile isotope of U. As the Cold War ended, the need for HEU to support military weapons programs abated. Proliferation concerns also provided incentives for many research reactors to shut down or transition to LEU fuel. In September 1993, President Clinton issued the *Nonproliferation and Export Control Policy*, which established a framework to prevent the proliferation of weapons of mass destruction.

Subsequently, DOE and the US Department of Defense undertook an in-depth review of current HEU inventory and the amount of HEU supporting national security needs. In December 1994, approximately 174 MT of HEU were determined to be surplus for weapons needs. In March 1995, President Clinton declared an additional 200 MT of HEU and Pu as excess to national security needs. The DOE Office of Fissile Materials Disposition was charged with making the HEU incapable of supporting its use in nuclear weapons [11].

The excess HEU came from a variety of sources, existed in several forms, and included varying material assays. Approximately 51 MT of the excess HEU included off-specification U isotopes. Section 12 of the *USEC Privatization Act* authorized DOE to transfer off-specification HEU to the TVA for downblending for use as LEU fuel [12]. Initially, 33 MT of HEU were made available by DOE to downblend to LEU fuel.

The off-specification HEU provided for downblend contained high concentrations of the U isotopes ^{234}U and ^{236}U. The ^{234}U isotope concentration in the excess HEU is typically in excess of 1 wt %, whereas the ^{236}U isotope is typically in excess of 20 wt % [13]. After downblending the off-specification HEU to an isotope concentration of less than 5 wt % ^{235}U, the off-specification blended LEU fuel still did not meet the chemical and isotopic specifications of the American Society for Testing and Materials (ASTM) for enriched commercial-grade reactor fuel. Otherwise, enriched commercial-grade LEU and downblended LEU are similar. Each comprises at least 94 wt % ^{238}U, 3–5 wt % ^{235}U, and typically less than 1 wt % of other elements. It is in the remaining 1 wt % where the ^{234}U and ^{236}U isotopes do not meet the ASTM specifications for enriched commercial-grade LEU.[2] The ASTM limit for ^{234}U is 0.2000 wt % in downblended LEU, and the ASTM limit for ^{236}U is 0.025 wt % in enriched commercial-grade LEU or downblended LEU. However, ASTM Standard C996-15 notes that a "buyer may consider acceptance of a lot above the [^{236}U isotope] limit on the basis of the total significance of all the measured levels of radionuclides to determine the suitability for intended use in fuel fabrication and irradiation" [14], [15]. Therefore, a utility can opt to use off-specification LEU fuel based on a review of the effects on plant operation.

The ^{234}U contributes to higher radiation exposure from the fresh LEU fuel, and the ^{236}U acts as a core poison because it has a high neutron absorption cross section, which tends to inhibit the nuclear chain reaction in the core. Additional quantities of ^{235}U can be included in the off-specification LEU fuel mix[3] to account for the poison effect of ^{236}U, and additional fresh fuel handling precautions can be implemented to protect against the increased radiation dose associated with ^{234}U [16]. Some additional fresh fuel assembles can also be introduced at each refueling to compensate for the natural increase in ^{236}U as the BLEU fuel is irradiated in each fuel cycle. However, this approach increases fuel receipts, handling, and storage.

B.3.2 Licensing Approach

TVA and DOE entered an agreement with NFS to initiate the BLEU oxide production process. NFS subsequently submitted three license amendment requests to the NRC to permit alterations to its processing operations to allow the downblending of the off-specification HEU material. This process was

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[2] The ASTM specifications reported for ^{234}U and ^{236}U refers to UF_6 enriched to less than 5 wt % ^{235}U used to manufacture LEU fuel (ASTM C996-15). There are separate ASTM specifications for unenriched UF_6. The unenriched isotope limits are 0.0480 wt % for ^{234}U and 0.8400 wt % for ^{236}U (ASTM C787-15).

[3] Total enrichment remains less than 5 wt % ^{235}U.

completed in October 2003 [17], [18], [19], and the NRC granted the NFS licensing amendments through July 2004. NFS subsequently began processing off-specification HEU into BLEU oxide. The BLEU oxide was shipped to the AREVA fuel fabrication facility in Richland, Washington where the BLEU fuel assemblies were manufactured.

Although NFS obtained the license authority to produce BLEU oxide, TVA proceeded to obtain the permissions required to insert four PWR BLEU LTAs into Sequoyah Unit 2. TVA also adopted the 1996 DOE EIS for the HEU Program. TVA submitted a LAR to the NRC in August 1998 to change the Sequoyah Unit 2 Technical Specifications to allow a limited number of LTAs to be inserted into the core based on a Topical Report (BAW-2328) prepared by Framatome Cogema Fuels. The LTAs were intended to demonstrate the feasibility of using BLEU fuel in a commercial nuclear power plant and to verify the analytical tools for predicting BLEU fuel behavior. The BLEU fuel was noted to increase radiation exposure rates by as much as 20% over standard LEU fuel assemblies. This necessitates heightened operator awareness of ALARA techniques during receipt inspection of fresh fuel. The NRC granted the Technical Specification Amendment to insert up to 4 BLEU PWR LTAs into Sequoyah Unit 2 for one fuel cycle in April 1999 [20].

B.3.3 Summary

This licensing experience discussion is included in this report because there are similarities between the increased radiation exposure rates experienced in the implementation of the BLEU program and the increased and the radiation exposure rates expected for fuel preparation associated with MSR fuel cycle processes. The TVA facilities had to modify fuel receipt procedures and consider impacts on fuel receipt, handling, and storage overall. MSRs must consider similar procedural controls up to and including broader use of remote handling operations. This was a successful demonstration of a collaborative effort between DOE and a utility to use an unusual fuel blend coupled with a successful regulatory response to allow the use of the BLEU fuel in reactors.

B.4 INTEGRATED SPENT FUEL STORAGE INSTALLATIONS

B.4.1 Background

Every LWR in the United States has at least one spent fuel pool on-site to store and cool used fuel that is no longer viable for reactor operation. In the United States, the government owns the spent fuel and is responsible for its disposition after removal from the core. Therefore, fuel residence in each reactor plant spent fuel pool has always been seen as a temporary measure until the spent fuel could be removed by the government for reprocessing. Because reprocessing was eventually banned in the United States out of concern for the proliferation of nuclear material, an alternative storage solution has become necessary. The government continues to own the spent fuel and ultimately pivoted to a new plan to store spent fuel in a repository. However, the plan to store the spent fuel in a central repository has been hindered by numerous factors over time. Meanwhile, spent fuel at reactors has continued to accumulate, and storage capacity at most spent fuel pools has long been exceeded. This has led to the need for on-site interim storage of the spent fuel in dry casks or cannisters [21].

B.4.2 Licensing Approach

NUREG-2215 [22] provides the consolidated SRP for a CoC for a DSS for use at a general license facility or a specific license for a dry storage facility (DSF) that is either an ISFSI or a monitored retrievable storage installation (MRS). 10 CFR Part 72 requires a license to receive, transfer, and possess reactor spent fuel, reactor-related GTCC waste, and other radioactive materials associated with spent fuel storage in an ISFSI. An ISFSI license can either be a specific or a general license. An MRS license is an example of a specific license. A general license authorizes the storage of SNF in an ISFSI at a power reactor site using NRC-approved DSSs. The regulations in 10 CFR Part 72 also establish the requirements, procedures, and criteria for issuing a spent fuel storage cask CoC. NUREG-2215 provides consolidated SRP guidance for the various DSS and DSF license forms by incorporating previous review guidance contained in earlier NUREGs, such as NUREG-1536 [23] and NUREG-1567 [24].

All the review guidance is focused on heterogeneous fuel, and the predominant industry experience is with LWR fuel. Therefore, MSR facilities can only use the NUREG-2215 SRP as a general reference to the types of evaluations needed for spent fuel storage that does not require a pool. This section summarizes those evaluations.

B.4.2.1 Principle Design Criteria

An MSR applicant must provide the PDC and bases related to various SSC to ensure that the PDCs comply with the requirements found in 10 CFR Part 72. This will require a discussion of the rationale for systems classified as SSCs important to safety, a subset of the SSCs important to safety that are characterized as safety protection systems, and other SSCs. Factors that must be considered in the classification of SSC include structural, thermal, shielding, confinement, radiation protection, criticality, and material properties. These are discussed further in the following sections [22].

The specific discussion of stored SNF parameters and characteristics may need to include thermal and radiological characteristics, history, burnup, initial enrichment, post-processing of the material, and any other characteristics unique to molten salt fuel. Bounding conditions under which the DSS or DSF is expected to operate and perform its design functions should be discussed, including normal conditions, off-normal conditions, and accident conditions.

B.4.2.2 Structural and Materials Evaluation

The storage facility structural review ensures that the integrity of the associated SSCs are appropriately maintained under all credible loads and their combinations for normal, off-normal, and accident conditions and natural phenomena effects. The evaluation should demonstrate a reasonable assurance that storage systems and associated facilities will maintain their intended function to provide safe storage of SNF.

SSC materials must meet applicable codes, standards, and specifications and support the intended SSC functions under all credible loads and environments for normal, off-normal, and accident conditions. This includes an evaluation of adequate materials performance.

B.4.2.3 Thermal Evaluation

The thermal review ensures that the heat transfer and flow characteristics of facility storage containers and other fuel materials will remain within the allowable limits for normal, off-normal, and accident conditions. The review will confirm that the temperatures and fuel boundary conditions will be maintained throughout the storage period. This includes an evaluation of SSCs associated with decay heat removal. Thermal evaluations of the storage facility will also confirm that acceptable analytical and testing methods were employed to design and operate the facility.

B.4.2.4 Shielding Evaluation

The storage facility shielding review ensures that the design features relied on for shielding provide adequate protection against direct radiation. The shielding features should limit the direct radiation dose to the operating staff and members of the public so that the total dose (i.e., dose due to direct radiation and any effluents or releases) remains within regulatory requirements during design-basis normal operating, off-normal, and accident conditions. The occupational doses from operating the storage facility should align with ALARA principles.

B.4.2.5 Criticality Evaluation

The nuclear criticality safety review ensures that any SNF in the storage facility remains subcritical under normal, off-normal, and accident conditions that involve handling, packaging, transferring, and storing SNF. This analysis extends to the storage of any HLW or GTCC waste at the facility.

B.4.2.6 Containment or Confinement Evaluation

The storage facility containment or confinement review of the storage facility ensures that radiological releases to the environment will be within the limits established by the regulations.

B.4.2.7 Conduct of Operations

The storage facility conduct of operations review ensures that the applicant has (1) described an appropriate infrastructure to manage, test, operate, and maintain the facility—including provisions for effective training, EP, and physical security programs—and (2) developed appropriate acceptance tests and maintenance programs to ensure that storage facility SSCs are fabricated and maintained in accordance with the design described in the safety analysis report.

B.4.2.8 Accident Analysis

The accident analysis review provides for a systematic evaluation of the identification and analysis of hazards for off-normal and accident conditions that involve storage facility SSCs. The accident analysis review ensures the following:

- All relevant off-normal conditions for the storage facility are identified
- All credible accidents for the storage facility are identified
- The envelope or bounding set of off-normal conditions and accident conditions that are relevant to the storage facility design and operations and for which the facility is analyzed to ensure performance of its design functions are identified

B.4.2.9 Technical Specifications

The storage facility technical specifications are reviewed to ensure that the design and operations of the facility will meet the requirements of 10 CFR Part 72.

B.4.3 Summary

This licensing experience discussion is included in this report because MSRs must also plan for on-site storage of spent fuel. This discussion is based on the extensive licensing experience gained from establishing ISFSIs at most US nuclear power plants. These facilities handle cladded heterogeneous LWR fuel. NUREG-2215 [22] was used to govern the aforementioned licensing approach outline and discussion. In the discussion, specific references to fuel clad and dry storage facilities were removed. The included topical areas must be considered for an MSR spent fuel storage facility collocated with the reactor. Any facility that links MSR operation to the MSR spent fuel storage facility must also be considered. This would include any facility used for postirradiation processing of the MSR fuel salt to facilitate long-term storage.

B.5 OKLO AURORA REACTOR

B.5.1 Aurora Description

The Aurora reactor is a compact fast reactor that uses metal U Zr (U-10Zr) [25] fuel in 316L stainless-steel clad to generate heat. Instead of flowing coolant, the OKLO technology uses heat pipes to transport reactor heat to the power conversion system. The Aurora reactor builds on the experience gained through the operation of the Experimental Breeder Reactor-II and will use high-assay LEU enriched up to 19.75%. It has a rated thermal output of 4 MW and a core power density of 3.91 W/cm^3. The fast spectrum, small size, and lack of flowing coolant in the core enable a simple design with few moving parts. The lower power of the Aurora also leads to low decay heat production [26]. Some operational parameters and design characteristics are detailed in Table B-1.

Table B-1. OKLO Aurora reactor parameters.

Parameter	Value
Reactor type	Fast reactor [25]
Coolant/moderator	Liquid metal, no moderator [25]
Thermal/electrical capacity	4 MW(t)/1.5 MW(e) [25]
Primary circulation	Heat pipes [27]
Nuclear steam supply system operating pressure (primary/secondary)	Not pressurized [25]
Fuel type	Metal fuel [25]
Fuel enrichment	19.75% [26]
Core discharge burnup	20–60 GWd/t [26]
Refueling cycle	Up to 20 years [25]
Reactivity control mechanism	Rotating reflector drums and gravity aided shutdown rods
Approach to safety systems	Passive [27]
Design life	20 years per deployment [25]
Seismic design (safe shutdown earthquake)	0.50 g [27]
Distinguishing features	Microreactor with no refueling
Design status	COL application accepted by the NRC [25]

Source: [25], [26], [27]

B.5.2 Licensing Approach

Substantial guidance exists to support licensing and regulation of existing LWR technologies. But from the OKLO licensing perspective, this guidance is generally unsuitable for use with advanced reactor technologies, which typically take far greater advantage of inherent design features. Therefore, OKLO provided two topical reports for consideration by the NRC staff that offer an alternative licensing approach to the approaches now being considered and developed for 10 CFR Part 53 for advanced non-LWRs. The current 10 CFR Part 53 development effort has recognized that regulations cannot remain prescriptive based on large LWR experience (e.g., historical plant types, fuels, coolants, energy spectra, and sizes). Instead, the advanced reactor regulation work is evolving toward specific PB goals related to safety, security, and the environment. This is similar to the approach taken in NUREG-1537 [28]. Applicants must subsequently prove how their design meets these PB goals instead of attempting to either comply with or exempt from specific and possibly outdated or inapplicable requirements [29], [30].

However, OKLO has indicated that the 10 CFR Part 53 development efforts and the associated guidance do not fully incorporate a means of assessing the inherent features of new and advanced reactor technologies. Although regulatory guidance is a tool for demonstrating previous implementations of the requirements, the requirements themselves have assured adequate protection of the public health and safety over the decades. Through its topical report submittals, OKLO is proposing an alternate licensing application approach that focuses on addressing the regulations directly while accounting for important functions and features[4] of advanced reactor technologies [31], [32].

The *MCA Methodology Topical Report* [29] provides a technology-neutral guide for managing event analyses with the goal of identifying a bounding MCA in a clear systematic way. The MCA then becomes the licensing basis event for the reactor design. Although the intent of the *MCA Methodology Topical Report* [29] is to provide an MCA methodology that is applicable to many reactor technologies, OKLO notes that each technology type may need to implement this methodology in different ways. Applicants must justify and document their MCA implementation in the NRC license application and during the review process [31].

The *MCA Methodology Topical report* [29] may be used as a stand-alone approach for licensing basis event selection with application of defense-in-depth considerations. The MCA method may also be integrated with the second OKLO topical report *Performance-Based (PB) Licensing Methodology* [30], which describes a flexible PB approach for assessing the functions and features of a design so that the technology features are accounted for in the evaluation of the MCA and in the design bases. The *Performance-Based (PB) Licensing Methodology* [30] report provides a methodology to systematically identify those functions and features of the design to designate for regulatory controls and to track and uphold those functions and features from design to construction to operation. OKLO contends that both methodologies are appropriate for use in the licensing of novel advanced reactor technologies that not only use active functions but also those technologies that may rely on passive functions and features to assure the health and safety of the public and environment.

Regarding regulatory controls, OKLO notes that “regulatory controls include design bases, design commitments, and programmatic controls (pre-operational testing, startup testing, license conditions, technical specifications, and quality assurance). Collectively, the regulatory controls are developed with defense-in-depth in mind and assure the safety of the plant throughout the lifecycle of the facility” [32].

The *Performance-Based (PB) Licensing Methodology* [30] report provides guidance on where regulatory controls must be applied, but the guidance does not specify how any regulatory controls must be applied because that is typically driven by the individual advanced reactor technology [32].

B.5.3 Summary

The proposed MCA methodology is one way of systematically selecting a bounding event for a reactor technology, which becomes the licensing basis event. The safety analysis supporting the MCA includes a thorough review of all possible licensing basis events, screens those events for applicability and credibility, allows for the grouping of events, performs bounding analyses, and finally selects a

[4] Within the OKLO topical reports [29], [30], functions are usually passive or active system responses (e.g., valve actuation, shutdown rod insertion), and features are typically inherent or intrinsic system characteristics (e.g., reactivity feedback, heat transfer properties, structural configurations).

technology MCA. The MCA is evaluated using a defense-in-depth approach. Under this approach, the most limiting single failure for the event is assumed to occur simultaneously with the MCA event. The safety analysis result is compared directly against the regulatory dose acceptance criteria. After the MCA is selected, external hazards are analyzed against the design to assure that the MCA continues to be upheld [29]. This could be one accident analysis approach for an expanded MSR facility.

At this point, the PB licensing methodology may be applied. The methodology is designed to clearly identify the functions and features of the technology that are relevant to safety. It is also designed to create the proper regulatory controls, including QA measures, to uphold those functions and features. Furthermore, this methodology is designed to verify that the assumptions and conclusions of the MCA analysis are valid. In this manner, the license application is scoped to include the appropriate level of information to evaluate the adequacy of the identified regulatory controls while limiting or excluding information about design aspects that do not have importance to safety. The confirmation of the identified regulatory controls occurs during construction, pre-operational testing, and startup and is likely performed via regulatory audits and inspections. Ultimately, the OKLO PB licensing methodology proposes an approach that prohibits any new design from achieving important operational milestones without satisfying all its regulatory controls [30].

OKLO's intent for the proposed MCA and PB evaluation of reactor functions and features is to provide technology-inclusive, risk-informed, PB licensing methodologies that are available for use by any advanced reactor technology, thus enabling broader deployment as directed by NEIMA.[5] One goal of NEIMA is to make reviews of advanced non-LWR more effective and efficient than the piecemeal LWR application reviews using prescriptive LWR requirements. OKLO has indicated that it hopes its proposed methodologies can substantially reduce the regulatory risk that US companies assume when designing new technologies and can streamline the effort of the regulator by explicitly identifying the regulatory controls used to demonstrate the safety of as-built systems from design to construction to operation of advanced reactor technologies. This should help ensure that review resources are efficiently focused on the relevant functions and features of the proposed design [30].

The NRC has recently denied [33] the combined OL application submitted by OKLO in March 2020. The denial letter stated:

> OKLO has repeatedly failed to provide substantive information in response to NRC staff requests for additional information (RAIs) on the maximum credible accident (MCA) for the Aurora design, the safety classification of structures, systems, and components (SSCs), and other issues needed for the NRC staff to establish a schedule and complete its technical review. A custom COL application must contain all design information as well as site-specific information needed for licensing.

As a result, the NRC will not evaluate the MCA and the PB technical reports because the staff was looking to these reports to close some of the application technical gaps. The OKLO application was denied without prejudice. Therefore, OKLO can resubmit a revised application and supporting technical reports that address the information gaps.

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