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Molten Salt Reactor Technical and Safety Considerations Outside of Guidance Documents

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Nuclear Energy and Fuel Cycle Division

MOLTEN SALT REACTOR TECHNICAL AND SAFETY CONSIDERATIONS OUTSIDE OF GUIDANCE DOCUMENTS

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CONTENTS

ACRONYMS

AEA	Atomic Energy Act
AMSE	American Society of Mechanical Engineers
AOO	anticipated operational occurrences
BPVC	Boiler and Pressure Vessel Code
CFR	*Code of Federal Regulations*
DOE	US Department of Energy
GTCC	greater than class C
HLW	high-level waste
GIF	Generation IV International Forum
IAEA	International Atomic Energy Agency
LEU	low enrichment uranium
LWR	light-water reactor
MCA	maximum credible accident
MSR	molten salt reactor
NPP	nuclear power plant
NRC	US Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act
PRA	probabilistic risk assessment
RIM	reliability and integrity management (programs)
SARRDLs	Specified Acceptable Radionuclide Release Design Limits
SNM	special nuclear material
SSC	structures, systems, and components

ABSTRACT

This document provides information on distinctive characteristics of liquid salt–fueled molten salt reactors (MSRs) for US Nuclear Regulatory Commission (NRC) consideration as it seeks to achieve effective and efficient advanced reactor mission readiness. The NRC has requested that Oak Ridge National Laboratory provide advice regarding technical and safety considerations that might be advantageous to address via policy, rulemaking, or license conditions (i.e., beyond the scope of guidance documents) for accommodating MSRs. It is recognized that MSRs could be regulated based on existing rules, with exceptions, to reflect their distinctive characteristics and technologies. However, directly applying the existing regulatory processes that evolved with the light water reactor fleet to MSRs could be unduly burdensome to the point that their application would inhibit deployment of MSRs in the United States.

The additional technical and safety considerations that the NRC might need to address outside of guidance documents are broad and diverse. Some of the issues identified are fully within the scope of the NRC's authority over nuclear power plant safety. Others have shared jurisdiction with other US government agencies (e.g., environmental protection, export control, and proliferation resistance). The issues arise from the physics and chemistry of liquid fuel salts, anticipated MSR configurations, the larger fraction of the fuel cycle potentially included at the reactor site, changes in national waste policies, and advancement in technologies since applicable regulations were formulated—in particular, changes in which fuel cycle technologies are now readily available outside the United States.

Developing performance-based regulations would be particularly useful for MSRs given their design diversity and divergence from existing reactors. Performance-based regulations are substantially more technology independent and resilient against obsolescence as they embody the regulatory objective instead of prescriptive technology limits to accomplish the objective.

1. INTRODUCTION

The US Nuclear Regulatory Commission (NRC) vision and strategy for achieving effective and efficient non-light-water reactor (LWR) mission readiness (NRC 2016) and the *Report to Congress, Advanced Reactor Licensing* (NRC 2012) both call for identifying technical and policy considerations that are distinct to advanced reactors. This document supports the objectives of the higher-level vision- and strategy-related documents by describing technical and safety considerations that might be advantageous for the NRC to address such as policy, rulemaking, or license conditions that are outside of guidance documents to accommodate liquid salt–fueled molten salt reactors (MSRs). Some of the issues identified are fully within the scope of the NRC's authority over nuclear power plant safety. Others have shared jurisdiction with other US government agencies (e.g., environmental protection, export control, and proliferation resistance). The NRC could regulate MSRs through its existing processes. However, MSRs introduce a significant number of technical issues that would then need to be addressed by exception or through additional requirements, resulting in an unnecessarily lengthy and expensive regulatory process. An efficient regulatory process would reduce the regulatory burden on applicants and NRC staff and support the introduction of technologies with substantial potential benefits to the US economy and environment.

MSRs have several technical issues that are substantially different from those of solid-fueled reactors. MSR technical issues affecting fuel cycle technology and high-level waste generation, transport, and storage have implications for policies that derive directly from compliance with federal law (i.e., the Atomic Energy Act [AEA] and the Nuclear Waste Policy Act [NWPA]). Relying on the safety issues of solid-fuel reactors when evaluating MSRs could introduce excessive conservatism in some areas, while overlooking potential risks in others.

The regulatory implications of MSRs are relevant for the NRC to consider as it modernizes its rules and policies. The NRC's modernization of its capabilities and regulations centers on making them more performance based, risk informed, and technology independent. Regulatory structures that evolved with the current LWR fleet using solid, cladded fuel are generally prescriptive and focused on the risk characteristics of these reactors. Prescriptive technology regulations inevitably reflect the maturity of the technology at the time the regulations were formulated. Diverse MSR technologies continue to be developed however. Performance-based regulations are substantially more resilient against obsolescence as they embody the regulatory objective instead of prescriptive technology limits to accomplish the objective.

This document is limited to issues that pertain predominately to liquid-fueled MSRs. Additional topics that would be appropriately addressed at the policy or rulemaking level but that pertain broadly to all advanced reactors—such as harmonization of the safety evaluation process and reporting structures with the International Atomic Energy Agency (IAEA) to reflect the increasing internationalization of the nuclear industry—are not addressed.

The additional technical and safety considerations the NRC might need to address outside of guidance documents are technically diverse:

- Inspections within biological shielding
- Expanding risk-significance beyond the core
- Different first-barrier radionuclide release risk probability
- Releasable stored energy as an accident bounding concept
- Alternate risk representation
- Integrated reactor and fuel-cycle facility regulation

- Performance-based proliferation resistance requirements
- Performance-based export control rules
- Structural material qualification
- Performance-based waste rules
- An inclusive definition of low enrichment uranium (LEU)

The issues arise from the physics and chemistry of liquid fuel salts, anticipated MSR configurations, the larger fraction of the fuel cycle potentially included at the reactor site, changes in national waste policies, and advancement in technologies since applicable regulations were formulated—in particular, changes in which fuel cycle technologies are now readily available outside the United States. This document sequentially discusses the technical basis for each identified issue. Although the topics include all the issues identified to date, additional issues could arise with further evaluation as the first MSRs proceed through the licensing process.

2. INSPECTIONS WITHIN BIOLOGICAL SHIELDING

Inspections verify the continued functionality and anticipated remaining useful life of structures, systems, and components (SSCs). Inspections are widely used at LWRs as part of providing reasonable assurance that their safety-related SSCs are capable of adequate performance if called on. The passive nature of MSR safety responses significantly reduces or eliminates the need for active safety systems and, thus, their inspection (e.g., emergency diesel generator oil condition monitoring has no logical equivalent at MSRs). The safety elements of an MSR's off-gas system would be part of the overall plant passive safety. Both decay heat rejection and radioactive material containment would be passively provided for the off-gas systems under accident conditions. MSRs are dependent on the proper functioning of passive structures and systems to provide adequate safety. Rulemaking could provide a blanket exemption for the elements of currently required inspections that do not pertain to MSRs, as described in the following paragraphs. This section provides an overview of the technical basis for the policy changes and related guidance needed to enable appropriate inspections within the biological shielding of MSRs as part of an efficient and effective regulatory process. A regulatory recommendation summary is provided at the end of the section.

MSRs would benefit from the development of performance-based, risk-informed, technology-neutral inspection guidance. MSRs will require development of tailored inspection validation procedures and will require exemptions from LWR technology-specific requirements. Solid fuel centric language is embedded throughout the NRC inspections documents (e.g., the *NRC Inspection Manual* [NRC 2022b]). Developing a technology-neutral equivalent set of inspection processes will require updates to a substantial fraction of guidance documents. For example, the technical basis for fire protection in the NRC Inspections Manual employs an increase in core damage frequency to determine acceptance criteria. The objectives of MSR inspections are the same as those for any reactor. Consequently, inspection validation processes that have direct analogy with LWR processes would not require rulemaking or policy changes but would benefit from development of focused guidance documents. An example of focused guidance would be to explicitly consider performance degradation within the monitoring requirements of 10 CFR 50.65(a)(1), which currently requires SSCs to remain *capable of fulfilling their intended function* (NRC 10 CFR 50.65). Passive SSCs more commonly exhibit performance degradation (as opposed to failure), which might not compromise overall plant safety. Guidance would shift the focus from individual SSC performance to overall plant safety by indicating that SSCs must remain capable of *adequately* fulfilling their intended functions such that the overall plant continues to achieve the FSFs. For example, startup of a passive decay cooling loop might be somewhat slower than intended because of fouling within its tubing resulting in a higher than intended temperature and undesirable container deformation. However, provided the container deformation does not prevent the plant from achieving the FSFs (likely demonstrated by remaining within its technical specification for deformation), the slower passive decay loop startup would remain acceptable.

The NRC has existing procedures for verifying that licensees perform required inspections and that the results of the inspections are properly acted on. MSRs will require different inspections than LWRs because of their technology differences. For example, an NRC inspector at an MSR might validate that natural circulation-based decay heat removal systems are (1) performing within specification for systems that are continuously operating or (2) that they initiate and perform within specification during periodic testing for systems that initiate in response to accidents by observing heat transfer (temperature and flow) instrument readings.

The high dose rates within biological shielding at operating MSRs effectively and permanently preclude any direct observations by inspectors. Licensees will need fully automated procedures for performing any inspections and maintenance using remote or fully automated means for SSCs within the biological shielding. Much as with any nuclear power plant (NPP), NRC inspectors will need evidence that required inspections and maintenance activities have been adequately performed.

Developing the technical specifications for inspection and validation at MSRs is based on understanding (1) which SSCs are required to provide adequate safety under both normal operations (including anticipated operational occurrences [AOOs]) and under accident conditions, (2) the critical performance parameters of those SSCs, and (3) how those parameters degrade during use. Consequently, a detailed inspection plan cannot be formulated until a plant has reached the conceptual design phase. An associated issue is what measurements can be made to assess critical performance parameters, how frequently the measurements need to be made, and what should be the acceptance criteria for the measurements. As MSR SSCs have finite lifetimes, an alternative approach to in-service inspection is prequalification of specific components, acceptance testing before entering service, and replacement before the first required inspection.

The containment barriers at MSRs, other than the normally salt-wetted barrier, are not normally under stress. Proper functioning of the normally unstressed barriers can be validated by observing their performance under conditions representative of the conditions under which they are required to perform their safety function. Because the unstressed containment barriers are not identical to any LWR feature, uncertainty remains as to which regulations govern their inspection interval and acceptance criteria. 10 CFR 50.65(a)(3) indicates that *performance and condition monitoring activities and associated goals and preventive maintenance activities shall be evaluated at least every refueling cycle provided the interval between evaluations does not exceed 24 months*. MSR containment layer testing could alternately be considered to be more conceptually similar to periodic containment leakage testing at LWRs. However, the primary containment stressor at MSRs is elevated temperature, whereas the primary barrier stressor at LWRs under accident conditions is pressure. Elevating containment temperature to mimic accident conditions is undesirable as the temperature would substantially stress instruments and components other than those whose performance is being assessed. The specific testing requirements for containment leakage testing at LWRs are provided in Appendix J to Title 10, *Code of Federal Regulations* (CFR), Part 50 (10 CFR 50 2021). MSRs will need a functionally equivalent containment leakage testing rule, as well as guidance to implement the containment leakage testing process (similar to ANSI/ANS-56.8, 2020 Edition, *Containment System Leakage Testing Requirements* [ANS 2020]) that is oriented towards locally heating the potentially vulnerable portions of the barrier layers (i.e., seals and penetrations). More fundamentally, without a postulated degradation mechanism, stressing barrier layers to validate their continued performance increases the risk of containment failure without a commensurate benefit. Policy level guidance is needed on acceptable methods and intervals for normally unstressed containment layer performance degradation monitoring.

Inspection requirements for the normally salt-wetted barrier layer will depend on its safety function. Verification and validation that the normally salt-wetted components have been properly manufactured and installed can be performed using existing quality assurance and manufacturing guidelines. MSRs will use different materials and thinner walled components than LWRs because of the differences in service requirements. Applicants who do not elect to credit the normally salt-wetted barrier to achieve a safety function avoid the need to assess the remaining useful life of the component, although instrumentation will be required to assess whether the barrier is currently performing its function.

Applicants who do elect to credit the normally salt-wetted barrier to achieve a safety function need to verify that it can perform its functions under normal operations, including AOOs, and that it can continue to adequately perform its function under accident conditions. Applicants who elect to credit their normally salt-wetted boundary to perform a safety function will need inspection rules that are functionally equivalent to those for other reactor classes. Note that gross failure of an MSR container layer would typically have lower consequences than gross failure of an LWR vessel or primary piping because of the low system pressure and low chemical reactivity of the fuel salt (no potential for hydrogen deflagration). Development of a mechanistic source term for a particular MSR under its complete set of design basis event accident conditions would, however, be required to establish acceptable performance requirements for credited barrier layers. 10 CFR 50.65(b)(1) specifies that the *integrity of reactor coolant pressure boundary* is to be within the scope of the monitoring program. The MSR fuel salt boundary has some of the safety characteristics of the reactor coolant pressure boundary. However, as the fuel salt boundary is not a pressure boundary, MSRs will require exemption from the specific language.

The material performance requirements for LWR reactor vessels are documented in 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements" (10 CFR 50 2021). Additionally, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI, provides rules for in-service inspection for LWR NPP components (ASME 2021). However, neither the 10 CFR 50 appendixes nor ASME BPVC, Section XI, includes rules that incorporate the distinctive characteristics of MSRs. No consensus standard currently exists on what would constitute a critical flaw in an MSR vessel, means to detect such a flaw, or the rate at which a flaw could develop or progress in service. MSR applicants who elect to credit their fuel salt boundary to perform a safety function will require equivalent material performance rules and guidance documents reflective of the different safety consequences of failure, the extended distribution of the radionuclides, and the different material degradation processes.

One acceptable means of confirming the predicted behavior of normally salt-wetted structural material performance adequacy would be via in situ material coupons. The primary in-service material stressors will be a combination of exposure to fuel salt, temperature, neutron fluence, and mechanical stress. Procedures for implementing coupons that provide an equivalent stress environment remain under development. Applicants will need to provide sufficient evidence that their combination of materials and stressors will result in adequate margin from component damage during normal operations, including AOOs, and so much damage under accident conditions that the container fails to perform its function.

An alternative means of providing reasonable assurance of container adequacy would be to employ components with additional mechanical margin fabricated from materials known to have (1) adequate temperature-strength profiles over time, (2) acceptable neutron embrittlement characteristics, (3) acceptable salt compatibility, and (4) a salt chemistry monitoring and control program. For example, the 316H grade of stainless steel has both well-known temperature strength characteristics over time and well-known radiation damage characteristics. Furthermore, 316H stainless steel does not corrode rapidly in fluoride salts under reducing conditions, and its extent of generalized corrosion can be monitored by observing changes in the chromium fluoride content of the fuel salt. Prior demonstration of acceptable fuel salt compatibility is a necessary element of relying on in-use salt chemistry monitoring to assess the extent of corrosion. Other forms of corrosion are possible with some fuel salt and structural material combinations that would not be as readily detectable using chromium content monitoring. For example, some fuel salts have strongly temperature-dependent corrosion product solubility. Single-point corrosion component concentration monitoring would not capture the transfer of material from a hot to a cold zone.

Inspection of the thin heat exchanger tube walls would be an element of containment inspection if the heat exchanger containment performs a credited safety function. No technical means is currently available to assess the remaining useful life of thin, single-tube walls at MSRs. Consequently, it is anticipated that the functional containment strategy for the heat exchanger leak path will include the coolant salt and its boundary. The coolant salt piping would extend beyond the biological shielding.

Monitoring of other salt contacting SSCs (e.g., control rod guide tubes, flow control elements, or graphite structures) will be necessary for designs in which proper SSC functioning is necessary to achieve (or their failure could impede the achievement of) an FSF. Specific monitoring requirements will depend on the details of the design and accident sequences. For example, failure of control rod guide tubes might result in a containment layer failure and loss of reactivity control. The failure might be accounted for in the reactor design (e.g., by not crediting the salt-wetted containment layer and relying on inherent salt negative reactivity feedback to provide adequate reactivity control) or through a combination of quality control in design, manufacturing and installation, and health monitoring during operation. The existing language in 10 CFR 50.65 (a)(1) indicating that specific monitoring requirements for SSCs relied on to accomplish FSFs will rely on *licensee-established goals* remains applicable to MSRs.

The permanent lack of accessibility within biological shielding also shifts the material control and accountability inspections paradigm. Traditionally, fuel rods and fuel assemblies are accounted for as they leave and enter the vessel, as well as while they are in the spent fuel pool. MSRs lack discrete fuel units to inspect. Additional policy guidance will be required on acceptable alternative (indirect observation) fissile material accountability methods.

The *Maintenance Rule* (10 CFR 50.65) is central to monitoring requirements for licensed NPPs. It is recommended to modify the rule to focus on the overall plant achievement of the FSFs rather than the proper functioning of specific SSCs. Focus on plant-level safety accommodates the performance degradation that is more common for passive SSCs than the failure-on-demand model typical of active systems. It is also recommended to clarify which regulations on inspection intervals and methods (10 CFR 50.65 or Appendix J) would be appropriate to follow for both salt-wetted and normally unstressed container layers and indeed whether normally unstressed containment layers require periodic testing. It is also recommended to develop inspection procedures for both operational issues (normally addressed in the NRC Inspection Manual) and fissile material accountability (further discussed in Section 8) that are compatible with remote or indirect observation.

3. EXPANDING RISK SIGNIFICANCE BEYOND CORE

The primary risk arising from NPPs is the potential for the release of radionuclides. In solid-fueled reactors the radionuclides are largely in the fuel in the core and in the used fuel pool. Risks are, thus, centered on phenomena that have the potential to adversely impact either the core or the used fuel pool. In contrast, radionuclides are distributed over much more of the nuclear island at MSRs. Moreover, significant quantities of the radionuclides can be substantially more mobile under normal conditions than those trapped in solid fuel forms. The shift in risk characteristics at MSRs provides a technical basis for shifting the regulatory basis away from preserving the integrity of the fuel towards maintaining containment of the radionuclides. Although each MSR could request exemption from core and used fuel integrity-focused regulations, providing a general exemption from fuel mechanical integrity requirements would increase regulatory efficiency.

Recent guidance-level documents, such as NRC Regulatory Guide 1.232 (NRC 2018b), have focused on preserving the safety intent of each element of the existing solid fuel–focused safety rules but have not broadened the hazards evaluation to reflect the additional risks resulting from increased radionuclide mobility and broader distribution. The American Nuclear Society's working group seeking to develop an MSR design safety standard (working group 20.2) has partially addressed the issue by introducing additional candidate MSR design criteria focused on distinctive MSR characteristics. Guidance-level documents, however, are restricted to adapting existing policies and rules. Although the guidance adaption process is efficient for applying existing policies to closely related situations, substantial technology and risk changes would be more efficiently addressed by customized rules. The NRC's safety regulations for fuel cycle facilities that contain extended distributions of substantial quantities of mobile radionuclides are provided in 10 CFR 70, "Domestic Licensing of Special Nuclear Material (10 CFR 70 2020). Policy-level changes would be required to reflect the radionuclide release risks arising from both the reactor and chemical-processing facility aspects of MSRs into a single safety adequacy evaluation process.

An MSR's cover gas, in particular, contains a significant fraction of the labile radionuclides. Hence, activities with the potential to adversely affect the headspace above the fuel salt have substantially greater risk significance than those for other reactor classes. Even if the liquid fuel salt is drained, the gas space will contain significant quantities of radionuclides. Thus, offline maintenance activities that could introduce containment bypass mechanisms have greater risk significance at MSRs than at solid-fueled reactors. Bypass of otherwise intact containment layers has a substantially greater risk impact for systems with higher fractions of mobile radionuclides. For example, a fuel salt heat exchanger (either power cycle or decay heat removal) tube rupture could provide a pathway to the bypass of normal containment layers, increasing the risk of releasing significant quantities of radionuclides. In addition, any fuel treatment systems at MSRs, such as mechanical filters, will accumulate large quantities of radionuclides, increasing the risk significance of maintenance and waste-handling activities. Verifying that containment has been appropriately reestablished following maintenance activities also has a higher risk significance for MSRs because of the potential for loss of substantial fractions of radionuclides from small leaks.

4. DIFFERENT FIRST BARRIER RADIONUCLIDE RELEASE PROBABILITY

Much of LWR safety accident progression analysis is typically based on the ductile response of container materials combined with the segmented nature of solid-fuel containment (fuel pins or particles). The configuration and physics of LWRs result in progressively increasing radionuclide releases with increasing stressors. A single fuel rod crack tends to retain the fuel pellets, and small break loss-of-coolant accidents at an LWR with otherwise functional safety-related SSCs would not lead to core damage.

Much of an MSR's radionuclide content would be releasable though a crack in its fuel salt boundary. Without a mitigating action, all of the gaseous materials and the liquid down to the hydraulic level of the crack could flow out. In many MSR designs, nearly all of the radionuclides are confined within a common first barrier layer—the primary system boundary. Structural alloys can become embrittled following neutron irradiation. Loss of tensile strength is especially problematic in high-nickel alloys irradiated at high temperature. The physics and chemistry of MSRs tend to promote a bimodal radionuclide release probability from the first barrier. Either no (or small) amounts of radionuclides are released, or large quantities leak into the next containment layer. On the other hand, releasing large amounts of radionuclides from the first barrier layer might not significantly increase the risk of release from the plant, substantially reducing the failure consequences in comparison with those for a large containment layer failure accident at LWRs.

Progressive failure consequence distribution presumptions are embedded within current regulations. Concepts such as “limited local yielding” and “leak before break” do not align with the failure mechanics of salt-wetted containment layers, especially those exposed to high neutron fluence. Corrosion and neutron embrittlement, along with temperature, are the primary first container layer material stressors at MSRs. The ASME BPVC requirements, mandated for LWRs in 10 CFR 50.55a, focus on ensuring adequate material strength over time and do not address the most significant materials stressors for MSRs. MSRs would either require alternate containment material performance rules reflecting their difference in failure mechanics and consequences of failures or could rely on the performance-based requirements embodied in functional containment.

Maintaining adequate margin from design limits for safety-related SSCs is required during normal operations. MSRs will require inspection and/or surveillance methods, along with validated degradation models, to provide reasonable assurance that their inner container layer continues to have adequate margin from its design limits if the container is safety related. Heat exchanger tube walls are anticipated to be especially problematic to validate as they need to be thin for effective heat transfer. Consequently, small amounts of tube-wall thinning result in a greater fractional loss in strength. Moreover, no inspection technology is currently available to evaluate the remaining material thickness in the high-radiation environment.

MSRs are required to adequately retain radionuclides with a conservative bias to uncertainty under design-basis accident conditions. Although it might be possible to develop sufficient evidence that failure of a comparatively thick reactor vessel is adequately unlikely, heat exchanger tube leakage will be sufficiently probable for all known technologies as to require sufficiently mitigating release consequences. Developing a performance-based surrogate damage metric, conceptually equivalent to a core damage frequency employed in solid-fueled reactors, would be useful to avoid needing to completely model and validate fuel salt leakage accident progression to regulated limits (e.g., dose to the public). Although the Specified Acceptable Radionuclide Release Design Limits (SARRDLs) employed in guidance documents captures the intent of developing the surrogate metric, policy-level guidance will be required to define and quantify SARRDLs.

5. RELEASABLE STORED ENERGY AS AN ACCIDENT BOUNDING CONCEPT

The safety evaluation objective for NPPs is to provide reasonable assurance of adequate protection of the public and the environment. A logical approach to accomplishing the objective is postulating the worst accident that could credibly occur and evaluating whether the public and the environment would be adequately protected. Indeed, maximum credible accident (MCA) evaluation was central to the original reactor safety adequacy evaluation methods and continues to be an acceptable method for demonstrating adequate safety.

Although MCA evaluation remains central to current nonpower reactor safety adequacy evaluations, MCA evaluation ceased to be part of commercial NPP safety evaluation with the development of progressively larger LWRs for which some postulated accidents could not be adequately contained. In addition, attempting to develop a containment capable of containing all credible accidents was a key part of the large cost escalation that contributed to the cancellation of the Clinch River Breeder Reactor project.

A primary objection to relying on containing the MCA for safety adequacy evaluation is establishing that any particular accident is actually the maximum credible accident. MSRs, however, can employ total stored energy release to provide a maximum potential accident. An accident at an MSR cannot be larger than complete release of the stored chemical and physical energy. MSR cores are at their maximum reactivity and therefore lack the potential for large reactivity (or hypothetical core disruptive) accidents. Molten salt fuel has low chemical potential energy and therefore cannot chemically react vigorously with any material. Decay heat is a continuing energy release phenomenon. Stored energy release evaluation is commonly employed in the chemical industry to evaluate required containment barriers.

MCA containment adequacy shifts accident progression evaluation towards mechanistic evaluation of the energy transfer processes that generate stress on the containment layers and mechanisms that reject decay heat to the environment. Defense-in-depth adequacy is assessed by requiring at least one minimally stressed barrier layer at the conclusion of the accident sequence. Most of the safety adequacy evaluation would be deterministic, with the principal remaining risk being in adequately representing the degradation of passive SSCs (e.g., decay heat transfer loops might have become partially blocked by corrosion products, thereby taking longer to initiate and transferring less heat than anticipated).

A substantial body of work has been devoted to devising the regulatory structures and methods to demonstrate via probabilistic component performance models that the risks of NPPs are adequately low (e.g., NUREG 1860 and NEI 18-04 [NRC 2007; NEI 2019]). Less design and operating experience with MSRs than with other advanced reactor classes results in substantial additional conservatism to employ high-fidelity probabilistic models focused on component failure. Although NRC Regulatory Guide 1.233 (NRC 2020a) indicates that alternate (potentially nonprobabilistic) formulations of risk might be acceptable, no guidance is currently available on how to develop and apply a graded approach to integrating probabilistic and nonprobabilistic risk representation methods. MSRs are in danger of being orphaned as either not having sufficient data to efficiently employ existing probabilistic methods or not having the proven regulatory structures in place to employ an alternate safety adequacy evaluation process that aligns with their characteristics.

Unnecessarily complex, conservative, and costly safety evaluation methods directly impact the viability of new technology. Being able to attract sufficient investment in the technology directly depends on the cost and time of obtaining regulatory approval. Not providing a safety adequacy evaluation methodology matching MSR reactor class characteristics results in an undesirable bias against a reactor class that has potentially desirable safety and performance characteristics.

6. ALTERNATE RISK REPRESENTATION

Multiple policy and guidance-level activities are currently under way to incorporate risk into NPP regulation. The leading risk assessment method proposed for NPPs is based on probabilistic risk assessment (PRA) as described in Regulatory Guide 1.233. PRA models NPP accident progression as a series of component failure probabilities through the use of quantified, binary fault trees. Each step along an accident progression sequence is represented as a probability of a component to perform its function on demand. Fault tree accident sequence representation matches the processes involved with active systems driven by external energy sources such as the probability of a pump to activate or a generator to start. Much of the value in PRA comes from its capability to illuminate the combined risk arising from multiple, individually less severe, component failures.

PRA is structured to assess whether, for a reactor design of interest, the frequency of accidents with a particular dose consequence is below an acceptable limit. The characteristics of MSR accidents, however, do not align well with progressively increased accident consequences for increasing numbers of interconnected component failures. MSRs rely on the performance of passive barrier layers to prevent the release of radionuclides and on the performance of passive heat rejection systems to provide adequate decay heat rejection. MSR safety-related SSC failure modes tend to be partial and time dependent, which are conceptually different from the binary, on-demand failures characteristic of active safety systems. For example, slower-than-anticipated initiation of natural circulation decay heat rejection could result in fuel salt reaching a higher than anticipated temperature, which in turn places a higher stress on its container layer. However, natural circulation heat rejection increases substantially at higher temperature. Although the higher fuel salt temperature would result in increased mechanical creep of its container (potentially reducing its remaining useful lifetime), much higher temperatures (hundreds of degrees) for an extended period would be required to result in sufficient mechanical damage to cause containment failure.

Passive system response instead tends to be initiated by smaller forces that progressively establish conditions such as the start-up of buoyancy-driven natural convection flow. MSRs lack rapidly progressing accidents and so do not depend on rapidly acting safety mechanisms. Passive SSC failures tend to be incomplete and time dependent, such as slow or partial activation, which is not well represented in binary fault trees. The risks of MSRs are likely to be dominated by either external events (e.g., flooding of decay heat rejection heat exchangers) or progressive degradation (e.g., corrosion product buildup that prevents adequate decay heat removal system initiation). Although MSRs do incorporate components, and some accidents at MSRs can be represented by component failure on demand-focused accident progression modeling, the lack of PRA alignment with the physics of failure at MSRs renders it unable to provide transparent representation of the dominant contributors to plant risk and highlights the need for an alternative MSR-tailored risk representation.

Several possible alternative risk representation schemes are possible for MSRs. Barrier failure analysis (also known as lines of defense and layers of protection analysis) is an example of a potential alternative quantitative screening tool the physics of which align with accident progression at MSRs. Alternative accident progression schemes have been previously employed at similar facilities. For example, MCA analysis was employed in establishing the safety basis for the Molten Salt Reactor Experiment (Beale 1964). Process hazard assessment, which can incorporate layers of protection analysis, remains central to evaluation of fuel processing facility safety adequacy (10 CFR 70, Subpart H). Other risk representation schemes, such as the IAEA integrated decision-making process and the Generation IV International Forum’s (GIF) integrated safety assessment methodology, have yet to be fully considered for MSRs.

Unnecessarily complex and costly safety evaluation methods directly impact the viability of new technology. Being able to attract sufficient investment in the technology directly depends on the cost and time of obtaining regulatory approval. Providing a risk representation methodology matching MSR reactor class characteristics removes an implicit, undesirable bias against a reactor class with potentially desirable safety and performance characteristics.

The NRC is currently developing 10 CFR Part 53. The current draft (NRC 2022a) of 10 CFR Part 53 includes three different risk representation pathways with differing amounts of emphasis on PRA. The issues identified in this section are anticipated to be addressed through the development of the decreased PRA emphasis pathways. It is also recommended to develop the technical basis to evaluate the potential application of the IAEA integrated decision-making process and the GIF integrated safety assessment methodology to MSR risk representation.

7. INTEGRATED REACTOR AND FUEL CYCLE FACILITY REGULATION

The expanded fuel cycle elements of MSRs result in substantial regulatory uncertainty as to which process to apply. 10 CFR 50 is entitled “Domestic Licensing of Production and Utilization Facilities.” 10

CFR 50.2 includes in its definition of production facility "Any facility designed or used for the processing of irradiated materials containing special nuclear material." Some fuel salt processing (i.e., gaseous and solid fission product separation) is integral to any MSR. Other MSRs anticipate including additional elements of the fuel cycle. The scope of 10 CFR 70, "Domestic Licensing of Special Nuclear Materials," covers licensing requirements for the *use* of special nuclear material (SNM).

SECY-13-0093, "Reprocessing Regulatory Framework - Status and Next Steps," states that the NRC did not envision that 10 CFR 70 would encompass a reprocessing facility (NRC 2013). By itself, 10 CFR 70 lacks features that would be an element of MSR licensing because of its concurrent status as a reactor. The NRC Glossary indicates that for a fuel cycle facility to be considered a reprocessing facility it needs to separate unused fissionable material from waste material (NRC 2021). Because the fuel salt in an MSR active circuit is not waste material, fuel salt processing would not be considered reprocessing according to NRC Glossary terminology.

NUREG 1520, revision 2 (NRC 2015), *Standard Review Plan for Fuel Cycle Facilities License Applications*, was issued after SECY-13-0093 and indicates that its purpose includes providing the means to demonstrate compliance with 10 CFR parts 20 and 70 and providing guidance on "safety and environmental impact reviews of applications to construct or modify and operate nuclear fuel cycle facilities." Although it appears that NUREG 1520 provides an approved, appropriate means to evaluate the safety of the additional fuel cycle elements integrated into MSRs, merging the requirements for reactor safety evaluation along with fuel cycle facility regulation does not result in a clear regulatory process reflective of the potential safety issues of the complete facility. Policy guidance is needed to provide the regulatory clarity necessary for applicants to reasonably evaluate the regulatory requirements and risks of including fuel cycle features into their MSR designs.

All liquid salt–fueled MSRs will require some form of used fuel salt stabilization or processing to meet the requirements of the waste confidence rule to have a waste form capable of indefinite storage at the reactor site following the reactor's licensed period of operation (NRC 2014b). The US Department of Energy's (DOE's) Advanced Research Projects Agency-Energy Program has initiated programs (DOE 2021a; DOE. 2021b) that will, in partnership with industry stakeholders, develop advanced technologies for processing and reuse of spent LWR and used MSR fuel. Developing a performance-based used fuel regulatory infrastructure would, thus, be especially timely given the increasing level of stakeholder interest, including US Government support, multinational corporations with presence in both the United States and Canada (Moltex 2021), and Canadian Government support (The Energyst 2021).

8. PERFORMANCE-BASED PROLIFERATION RESISTANCE REGULATIONS

MSRs have the potential to include several distinctive SNM production and use features that would require appropriate regulation to enable compliance with the AEA as amended. For example, MSRs have distinctive capabilities to efficiently combine the thorium/uranium fuel cycle along with integral fuel salt processing.

MSRs that incorporate the thorium/uranium fuel cycle along with integral fuel salt processing could be presented to the NRC for licensing under Section 103b of the AEA, which indicates that the Commission *shall* issue licenses on a nonexclusive basis for utilization or production facilities that meet requirements. Thorium/uranium fuel cycle reactors could meet the public benefit, safety, and information disclosure tests called out in Section 103b as necessary for a license.

The existing material control and accountancy rules focus on the issues involved with detecting diversion of fissile materials from solid-fueled reactors and reprocessing facilities. 10 CFR 74, "Material Control and Accounting of Special Nuclear Material," (10 CFR 74 2021) provides regulations with regard to accounting for SNM, and 10 CFR 73, "Physical Protection of Plants and Materials," (10 CFR 73 2021) provides security provisions for physical protection of SNM. 10 CFR 75, "Safeguards on Nuclear Material - Implementation of Safeguards Agreements Between the United States and the International Atomic Energy Agency," (10 CFR 75 2021) also provides accounting and inspection requirements for compliance with the additional protocol of the nonproliferation treaty. Fissile material tracking at solid-fueled reactors is based on discrete item counting (i.e., accounting for fuel rods or pebbles), which is the focus of 10 CFR 74. The safeguards requirements provided in 10 CFR 75 also provide material accountancy rules appropriate to bulk handling facilities, such as fuel reprocessing facilities. However, unlike reprocessing facilities, MSRs both create and consume fissile material. Consequently, accountancy regulations based on the characteristics of bulk reprocessing facilities can neglect potential MSR diversion scenarios.

Regulations governing the assurance of adequate protection against proliferation during fuel salt processing at MSRs would derive from the NRC's general authority. Section 161 of the AEA grants the NRC general authority to establish regulations governing the possession and use of SNM as needed to promote the common defense and security or to protect health or to minimize danger to life or property.

Because Section 103 of the AEA indicates that the NRC shall issue commercial licenses on a nonexclusive basis, a regulatory framework is needed to describe the plant and organizational features the NRC deems necessary to provide adequate security with regard to the possession and use of SNM. Without any mention of additional requirements or a supplementary regulatory process, liquid-fueled MSR applicants that include substantial fuel processing could reasonably conclude that no additional requirements beyond the material accountancy required under 10 CFR Parts 74 and 75 need to be fulfilled to allow extensive fuel processing in their plants.

Development of a regulatory framework to provide reasonable assurance that US-based commercial MSRs would not represent an unacceptable proliferation risk has not been necessary until recently, as no US-based MSR development program existed. Moreover, fuel salt processing technology was sufficiently technologically difficult that developing working systems required national government-sponsored efforts, which still do not exist. The general advancement of multiple integrated technologies ranging from industrial use of halide salts to electrochemistry, advanced modeling and simulation, remote operations technologies, and high-temperature materials has substantially reduced the degree of difficulty to undertake MSR fuel cycle technology development. Over the past decade, multiple US-based commercial entities have begun to develop MSR designs, some of which include elements of the fuel cycle with the technical potential for SNM diversion. Developing a performance-based regulatory framework that provides clear rules for US industry about the necessary features to demonstrate that their technology does not represent an unacceptable risk to the common defense and security is now timely.

The lack of a clear regulatory framework for approval of thorium/uranium MSRs could inhibit US leadership in this subclass of MSRs, which would be disadvantageous to the United States economically as well as potentially to national security. The thorium/uranium fuel cycle at MSRs can be implemented with differing levels of proliferation resistance. Substantial first-mover advantages could accrue to whichever MSR technologies are first implemented. It is in the US national interest for highly proliferation-resistant technologies to be an available alternative for developers, the use of which would not incur a significant economic penalty.

The United States is not the sole supplier of nuclear technology. Multiple nation states are currently pursuing development of the thorium/uranium fuel cycle (some using MSRs). Other nations to date have not placed nearly the amount of emphasis on proliferation resistance as has been US policy. If the United States does not make available technology for a highly proliferation-resistant version of the thorium/uranium fuel cycle for MSRs, less proliferation-resistant versions of the technology of non-US origin are more likely to be deployed.

9. PERFORMANCE-BASED EXPORT CONTROL RULES

A principal purpose for nuclear technology export control rules is to prevent the spread of technologies that could facilitate nuclear weapons development to unfriendly nations while not unduly impeding commerce with friendly nations. The responsibility for export control regulations is subdivided among multiple US government agencies and derives from multiple federal laws. However, in general, the NRC is responsible for elements of export control that relate to the safe operation of nuclear reactors, as well as to the export of nuclear equipment and materials. Several operations at liquid-fueled reactors have substantial potential impact on the protection of people and the environment that would not ordinarily be the case at solid-fueled reactors. For example, online chemical adjustment of the content of fuel salt, the inherent separation of fission gases, and chemical stabilization of used fuel salt all involve technologies that, under existing prescriptive rules focused on solid-fueled technologies, are considered to require specific authorization. Hence, the NRC would be anticipated to have increased involvement with the development and implementation of revisions to export control rules for MSRs.

The NRC has primary responsibility for the nuclear material export control rules provided in 10 CFR 110, "Export and Import of Nuclear Equipment and Material," (10 CFR 110 2021) and a supporting role in the development of other nuclear-related export control regulations. The current version of 10 CFR 810.6(a), in "Assistance to Foreign Atomic Energy Activities," (10 CFR 810 2022) indicates that cooperation between the United States and foreign entities, from authorized countries, on SNM production is a generally authorized activity provided no sensitive nuclear technology is involved. However, because fission gases and insoluble materials inherently separate from liquid salt fuel during operation, MSRs could be considered nuclear fuel reprocessing facilities and thus involve sensitive nuclear technology. 10 CFR 810.3 does not include a definition of fuel processing or reprocessing. However, conventional definitions of processing are based on the separation of fission products from the actinides, which occurs inherently to some extent in all MSRs. The NRC Glossary indicates that for a fuel cycle facility to be considered a reprocessing facility it needs to separate unused fissionable material from waste material. As the fuel salt in an MSR active circuit is not waste material, fuel salt processing would not be considered reprocessing under this NRC description. Also, Appendix I to 10 CFR 110 indicates that "reprocessing irradiated nuclear fuel separates plutonium and uranium from intensely radioactive fission products and other transuranic elements." Under the 10 CFR 110 description, separations that keep the uranium or plutonium with the other transuranic elements would not be considered reprocessing. In contrast, 10 CFR 810.7(c)(6) indicates that specific authorization is required for "reprocessing of irradiated nuclear fuel." This could be interpreted to mean that chemical processing of liquid salt fuel would be considered reprocessing and require specific authorization. Definitions of fuel processing and reprocessing would provide clear guidance on what activities would require specific authorization. Because multiple elements of normal MSR operations—such as redox adjustment through chemical addition—involve processing of irradiated nuclear fuel, the regulations need to be updated to rectify their internal conflicts and align their requirements with the governing statutes (primarily the AEA and the Nuclear Non-Proliferation Act of 1978).

The export control language of the AEA is prescriptive and includes specific restrictions that align with proliferation-sensitive technical issues of solid-fueled reactors. These prescriptive restrictions could inhibit safe operations at liquid-fueled MSRs without providing a corresponding proliferation vulnerability reduction. For example, AEA Section 123 a (7) indicates that no material containing plutonium, uranium-233, or other nuclear materials that have been irradiated (except for low-enrichment uranium) can be "altered in form or content" without prior approval from the United States. As any MSR would contain "nuclear material that has been irradiated" following start-up, necessary safety-related activities such as maintaining the redox state of the fuel salt, or operational activities including mechanical filtering of solid fission products from the fuel salt, would require specific authorization. Even online refueling by adding low-enrichment uranium to the active fuel salt circuit (altering the content of irradiated nuclear material) could trigger a requirement for specific authorization. The ability to avoid excess reactivity (instead of withdrawing control rods over time as is conventionally done at LWRs) has substantial potential reactor safety benefits. Moreover, the specific wording of the regulations implies that the fuel salt would not be allowed to freeze or even be drained into a critically safe tank during an accident as both would alter the form (phase in one case and shape in the other) of the fuel salt. Enabling equivalent international commerce in liquid salt–fueled reactors and solid-fueled reactors will, therefore, require developing a performance-based legal framework that incorporates the purpose of the rule without unnecessarily restricting unrelated activities.

Prescriptive technology regulations inevitably embody the state of the art at the time they were written. Chemical processing technology has undergone revolutionary improvements since the time the AEA and the Nuclear Non-Proliferation Act were written. A major purpose of export control restrictions is to prevent unauthorized diffusion of US technology. Decades ago, fuel salt processing technology was effectively an exclusive US technology. However, the key processing technologies for removing actinides from fuel salt are now well known, having been published decades ago by non-US authors in the open scientific literature. The effective result of the current prescriptive regulations is that the proprietary engineering and development work for implementing MSR fuel cycle technology (which is restricted purely for commercial purposes) is being performed outside of the United States, substantially reducing US influence. More generally, the lack of regulatory clarity on what aspects of MSR technology involve only *production of SNM,* and thus be generally authorized, versus what elements involve *sensitive technology,* and thus require specific authorization, is contributing to MSR developers deciding to perform their engineering development and manufacturing outside of the United States, resulting in adverse US economic impact, as well as less US influence in the control of sensitive technologies.

Prescriptive technology regulations also bias against innovation in favor of technologies that were available at the time the regulations were formulated. The development of technology-dependent, export-control regulatory infrastructures focused on the issues relevant to solid-fueled reactors inevitably reflect the technical issues of separating fissile materials from solid fuels. The dominant technical challenge in performing chemical processing operations on used reactor fuel is withstanding the high-radiation-dose environment. Solid fuel cannot be readily processed integrally to reactor power operations (although processing could be integral to the plant). An advantageous feature of liquid-fueled reactors is the ability to directly couple fuel processing to the operating reactor. Used fuel, the radionuclides of which have decayed for years following removal from the core or that has been only briefly irradiated, has multiple orders of magnitude less radiation dose than used fuel salt that has been removed from the active fuel salt circuit for only hours. Equipment reconfiguration at an MSR that has operated would require substantial, technically difficult work in the much higher radiation environment. Performance-based export control regulations that have the objective of limiting the spread of fissile material separation technologies would reflect the technical difficulty of performing such separations for any reactor type.

Use of prescriptive technology regulations in one area can also impede other beneficial outcomes. For example, restrictions on the ability to strip uranium from used fuel salt, which can be accomplished by well-known techniques such as contact with oxide- or carbide-forming chemicals, could prevent formation of a more stable waste form. The same conversion of uranium in used fuel salt into an oxide or carbide form could be defined as either waste stabilization or fuel processing. Waste stabilization would be generally authorized, while fuel processing would require specification authorization. Thus, clarification on requirements and definitions is needed.

A key recommended regulatory activity is to develop a generic specific authorization under Section 131 of the AEA that is equivalent to generally authorized activities described in 10 CFR 810 to cover MSR operational and safety activities. It is also recommended to develop the technical basis for a technology-neutral, performance-based fissile material processing proliferation vulnerability evaluation process. The technical basis would enable subsequent development of a regulatory guide that provides a clear pathway for review and approval of proposed fissile material processing related technology exports.

10. STRUCTURAL MATERIAL QUALIFICATION

LWRs comply with ASME Section III of the BPVC as a requirement of 10 CFR 50.55(a). The NRC is continuing to evaluate whether to mandate or endorse the high-temperature section of the BPVC (Section III, Division 5) for high-temperature reactors, including MSRs. Compliance with the BPVC has been mandated for LWRs because of the high consequences of failure of the reactor coolant pressure boundary. Although similar logic might be applicable to other advanced solid-fueled reactors, failure of the salt-wetted containment layer would not necessarily result in radionuclide release to the environment at liquid-fueled MSRs.

Mandating MSR compliance with the high-temperature section of the BPVC (Section III, Division 5) would indicate that the NRC views compliance as necessary for an MSR to achieve adequate safety. However, with the approval of functional containment for MSRs (SECY-18-0096 [NRC 2018a]), no single layer of containment would necessarily be more important to safety than any other. MSR licensees could maintain adequate plant-level radionuclide containment following failure of their salt-wetted containment layer.

Endorsing the high-temperature portion of the nuclear section of the BPVC would indicate that the NRC accepts that compliance with the high-temperature portion of the BPVC is an acceptable means of demonstrating that items fabricated according to its requirements would adequately perform their safety-related functions. However, Section III, Division 5, of the ASME BPVC does not focus on the primary stressors/failure mechanisms of the normally salt-contacting containment layer and so would not, by itself, provide reasonable assurance of achieving their safety functions. The focus of Section III, Division 5, of the ASME BPVC is on maintaining adequate material strength over time, which is key for high-pressure containments. However, it does not address either corrosion or radiation embrittlement, which are primary material stressors for MSRs. For example, simultaneous use of different alloys approved by ASME BPVC Section III, Division 5, in a single MSR loop could result in rapid containment failure due to dissimilar material corrosion. Also, issues such as corrosive thinning of heat exchanger tube walls, which could have significant safety implications for MSRs, are not addressed. ASME BPVC Section III, Division 5, also does not address the containment of tritium, which could diffuse through an intact barrier at high temperature.

Other portions of the ASME BPVC are also candidates for endorsement by the NRC. The current version of ASME BPVC, Section XI, “Rules for Inspection of Nuclear Power Plant Components,” includes Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants.” Division 2 is intended to be technology neutral; thus, it would be applicable to MSRs (Roberts 2020). However, RIM requires monitoring and nondestructive examination for each plant SSC based on a PRA. Components that employ run-to-failure or preventative replacement strategies would not fit the monitoring and nondestructive examination paradigm. Moreover, the risk significance of particular components can be established via non-PRA–based accident consequence evaluation methods. ASME BPVC, Section XI, Division 2, does not provide guidance on appropriate methods or technologies for performing inspections or monitoring or provide acceptance criteria and so forth for MSRs.

It is recommended to only endorse structural material performance standards for MSRs that address all relevant stressors. In addition, any standard endorsement is recommended to clearly indicate its range of applicability (e.g., limited to helium-cooled, high-temperature reactors). Further, any endorsement of an SSC reliability standard needs to indicate that no specific accident consequence evaluation model is required and that any SSC failure consequence evaluation model that provides reasonable assurance that the plant continues to achieve the FSFs would be acceptable.

11. PERFORMANCE-BASED WASTE RULES

The NRC's current high-level nuclear waste rules require additional policy statements or rulemaking to clarify their application to liquid salt fuel. The NRC description of high-level waste stipulates that it must be both highly radioactive and the byproduct of the reactions that occur inside of nuclear reactors (NRC 2020b). The definition indicates that high-level waste takes one of two forms: either (1) spent (used) nuclear fuel when it is accepted for disposal or (2) waste materials remaining after spent fuel is reprocessed. As liquid salt fuel can be processed as part of normal reactor operations; disposal of its actinides can be indefinitely deferred. Further, liquid fuel salt might not become spent since adding additional fissile material (as would normally be done to compensate for burnup) would enable it to continue to be used to generate energy. The NRC definition also does not indicate how to address highly radioactive materials that are the byproduct of reactions occurring inside nuclear reactors that are neither used fuel nor waste materials from fuel reprocessing. For example, tritium is both highly radioactive and a byproduct of reactions that occur inside of nuclear reactors but largely separates from the remainder of the liquid fuel salt. The definition also does not clearly apply to other highly radioactive materials that inherently separate from the fuel salt. For example, the ^{137}Cs that results from the decay of ^{137}Xe that inherently separates from the bulk of fuel salt would be part of neither the spent fuel nor the waste material resulting from reprocessing.

The AEA and NWPA define high-level waste as

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

Current NRC policy does not provide specific guidance as to what "highly radioactive" or "requires permanent isolation" means. DOE has interpreted (Federal Register 2019) the AEA and NWPA statutory language to mean that material would be considered non-high-level radioactive waste (HLW) if the waste meets either of the following two criteria:

1. Does not exceed concentration limits for Class C low-level radioactive waste as set out in 10 CFR 61.55 and meets the performance objectives of a disposal facility.
2. Does not require disposal in a deep geologic repository and meets the performance objectives of a disposal facility as demonstrated through a performance assessment conducted in accordance with applicable requirements.

Because of the relative ease of chemical (or electrochemical) separations from liquid salt fuel, fuel salt processing facilities are more likely than solid-fueled reactors to separate the short-lived radionuclides from the longer-lived ones to produce

- a waste stream that will become less radioactive than natural uranium in a few centuries (and so would not require permanent isolation),
- an actinide fuel salt stream,
- a depleted uranium stream, and
- a long-lived radionuclide stream.

As the long-lived radionuclides constitute such a small fraction of the used fuel salt, they would likely be incorporated into the actinide fuel salt stream for transmutation. Consequently, no fuel salt wastes would necessarily be considered HLW by statute. However, NRC policy statements would be needed to provide the necessary regulatory clarity for industry to undertake the risks necessary to develop used fuel salt processing facilities.

The NRC's rules for continued storage of spent nuclear fuel (NRC 2014a) (formerly known as the waste confidence rule) require the capability to indefinitely store used nuclear fuel onsite following plant closure. The radiochemical stability of liquid salt fuel is based on recombination (promoted by ionic mobility) dominating radiolysis. Once the salt has become deeply frozen, its ions will no longer be sufficiently mobile to overcome radiolysis, and the fuel salt composition can chemically evolve. Halide gases are likely to be produced and released from the bulk fuel salt. Consequently, chemically stabilizing used fuel salt will be a necessary element of post use treatment. However, some of the most effective chemical stabilization techniques separate fission products from actinides and could thus be interpreted to be reprocessing. Effective waste stabilization processes should not be inhibited by the lack of regulatory clarity. Clear policy statements from the NRC about the safety, material accountancy, and security requirements for fuel salt processing, as well as an efficient process for obtaining any additional approvals (e.g., specific authorization for working with international partners), are needed to enable indefinite continued storage of used fuel salt.

MSR design variants are currently being developed with a primary purpose to consume the long-lived radioactive materials (e.g., the actinide wastes) from other reactors. Synthesizing fuel salt is substantially technically easier than fabricating solid fuel from previously used reactor fuel. However, development of MSR waste burning in the United States is currently being inhibited by lack of regulatory clarity.

The NRC is currently engaged in rulemaking to promulgate requirements for the near-surface land disposal of Greater-Than-Class-C (GTCC) and transuranic waste (SRM-SECY-20-0098 [NRC 2022c]). Used fuel salt-wetted SSCs are likely to be classified as GTCC waste due to fission products deposited onto their surfaces and, in the case of graphite, penetration of radionuclides into pores and cracks. Also, MSR graphite and structural materials have the potential to become activated to GTCC levels during use. Consequently, waste disposal pathways for MSRs have the potential to be substantially impacted by the ongoing rulemaking.

Waste stability remains a cornerstone to minimize migration of radionuclides, avoid the need for long-term active maintenance, and minimize potential exposures to intruders. Encapsulating MSR wastes in grout has the potential to create physically and chemically stable and robust GTCC waste forms. DOE has recognized this potential in its reconsideration of the disposal pathway for MSRE components and fuel salt residues. The experience gained in grouting waste tanks at the Hanford and Savannah River sites (Harbour 2005) that include highly radioactive residues shows the potential for used MSR SSCs to be encapsulated into stable, robust waste forms.

No specific regulatory language has yet been released for the 10 CFR Part 61 update. The primary recommendation for the revision to ensure its applicability to MSRs is to establish technology-independent, performance-based GTCC waste stability criteria that if met would enable near-surface disposition. Updated criteria would enable NRC staff and prospective license applicants to efficiently and effectively evaluate the potential for near-surface disposal of MSR GTCC waste forms.

Providing clear guidance that (1) returning the actinides and other long-lived radionuclides from used solid reactor fuel to fuel salt along with (2) creating radioactive material waste streams that either have concentrations less than Type C or are sufficiently short lived to decay to below naturally occurring radioactive materials in a few centuries would remove the used fuel from being considered HLW. Clear guidance would provide the regulatory clarity necessary to enable adequate private investment in waste remediation technology development. Substantially reducing the radiotoxicity of used solid fuel waste and reducing the quantities of existing HLW support the NRC mission of protecting the environment. Lack of regulatory clarity should not inhibit achieving the overall agency mission.

12. CREATE INCLUSIVE DEFINITION OF LOW-ENRICHMENT URANIUM

The NRC currently provides a definition of low-enrichment uranium in both 10 CFR 110.2 and 10 CFR 50.2 that refers only to the ^{235}U content. MSR designers might elect to use a mixture of ^{233}U, ^{235}U, and ^{238}U (or only ^{233}U and ^{238}U) in their fuel salt as an alternate means to provide adequate proliferation resistance. Section 103 b of the AEA indicates that the NRC shall issue commercial licenses to persons whose proposed activities *will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized*. The fissile isotopes of uranium have different technical characteristics that are relevant to enrichment limits. The NRC needs to update its definition of LEU to include uranium compositions that include ^{233}U. An updated definition would facilitate development of plant and fuel cycle designs that meet the expectation to both serve a useful purpose and promote the common defense and security.

13. SUMMARY AND CONCLUSIONS

The highest-level regulatory objectives are the same for any reactor—protecting people and the environment while promoting national security without unduly burdening commercial activities. Regulatory authority to ensure achievement of the highest-level objectives is divided among multiple agencies. NRC is the lead agency for ensuring nuclear reactor safety. The NRC has shared jurisdiction for protecting the environment, export control, and preventing proliferation.

The fundamental safety functions are the same for any reactor—contain the radionuclides, provide adequate cooling, and control the reactivity. The high-level objective of preventing diversion of fissile materials from peaceful purposes by preventing the spread of technologies that separate fissile materials (export controls) and ensuring that diversion is both technically difficult (proliferation resistant) and readily detectable (safeguards) is applicable to any fuel cycle.

MSRs could be regulated based on existing rules with exceptions to reflect their distinctive characteristics and technologies. However, directly applying the existing regulatory processes could be unduly burdensome to the point that their application would inhibit US deployment. Moreover, applying a solid-fuel-centric process to MSRs might not provide adequate emphasis on the risks that are more significant for liquid-fueled systems. MSRs have multiple technical and policy considerations that would be more efficiently addressed through a tailored regulatory process. Developing customized regulatory processes for each reactor, however, would require substantial resources. Much of the regulatory inefficiency arises from applying prescriptive rules that were formulated based on the characteristics of solid-fueled reactors and the suite of technologies available at the time the rules were formulated.

Performance-based regulations can increase regulatory efficiency and transparency by focusing on the achievement of objectives that are common to any nuclear reactor rather than on the means of achieving the objectives. The NRC is currently engaged in modernizing its regulatory framework and has indicated that performance-based regulations should be considered when possible (NRC 2018c). Performance-based regulations provide applicants with flexibility on how to demonstrate achievement of the objectives, which is key to minimizing unnecessary regulatory requirements.

MSRs can have highly desirable performance and safety characteristics and, thus, the potential to substantially contribute to reliable, economic, low-carbon energy production. Multiple, diverse MSR designs could be presented to the NRC for approval over the next few years. An appropriate MSR regulatory process would facilitate achievement of the NRC mission in an efficient and effective manner.

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