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Molten Salt Reactor Fuel Qualification Considerations and Challenges

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Reactor and Nuclear Systems Division

MSR FUEL QUALIFICATION CONSIDERATIONS AND CHALLENGES

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ACRONYMS

AOO	anticipated operational occurrence
ARE	Aircraft Reactor Experiment
DOE	US Department of Energy
GDC	general design criteria
LOCA	loss-of-coolant accident
LWR	light water reactor
MOX	mixed oxide
MSBR	Molten Salt Breeder Reactor
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
NRC	US Nuclear Regulatory Commission
PIE	post-irradiation examination
QA	quality assurance
SAFDL	specified acceptable fuel design limit

1. PURPOSE

The purpose of this report is to describe the considerations for and challenges to fuel qualification for liquid-fueled molten salt reactors (MSRs).

Fuel qualification is a process which provides high confidence that physical and chemical behavior of fuel is sufficiently understood so that it can be adequately modeled for both normal and accident conditions, reflecting the role of the fuel design in the overall safety of the facility. Uncertainties are defined so that calculated fission product releases include the appropriate margins to ensure conservative calculation of radiological dose consequences [1].

2. USE IN THE REGULATORY PROCESS FOR LWR LICENSING (HISTORICAL)

Because of the crucial role that fuel plays in the safety of a reactor, the process of light water reactor (LWR) fuel qualification has historically been a major area of interest. However, there are no regulatory documents specifying explicit regulatory guidance for fuel qualification, so the process has been defined on the basis of experience and the availability of quality manufacturing and operating performance data. For example, 10 CFR 50.46 (b)(1)(2)(3) [2] establishes requirements for fuel's behavior during a loss-of-coolant accident (LOCA) event. This regulation defines the maximum cladding temperature, oxidation rate, release of combustible gases, and geometric stability. Lower level regulatory guidance is also available for normal and anticipated operational occurrence (AOO) conditions. Section 4.2 of NUREG-0800 [3] defines the expectations established for LWR fuel under these conditions. Regulatory Guide 1.206 Section C.1.4, "Reactor" [4], designates the content of information needed to address the expectations found in NUREG-0800 Section 4.2.

LWR fuel systems include not only the fuel material, but also the cladding, spacer grids, guide tubes, thimble, fuel rods, control rods, and channel boxes. The behavior of the fuel system is reviewed to ensure that the LWR design meets 10 CFR Part 50, Appendix A, General Design Criteria 10 (GDC 10) for normal and AOO conditions. In addition, the fuel failure mechanisms must be addressed during postulated accident conditions, including LOCA events. Mechanical effects, irradiation effects, and chemical effects such as oxidation, hydrating, and corrosion product buildup are all considered when addressing potential LWR fuel rod failures. This information is generated from both experimental data and fuel performance codes.

Along with the information on fuel performance within the reactor, the fuel qualification program must address the fabrication, transportation, and storage of fuel before and after it is used in the reactor.

To provide the data and input necessary to validate the fuel performance codes, an extensive experimental program is usually required. These experimental programs typically require separate-effects tests of the fuel, including unirradiated and irradiated testing, followed by extensive post-irradiation examination (PIE). This set of tests is generally followed by integral-effects testing or the use of Lead Test Assemblies in existing LWRs. The data quality assurance (QA) requirements are expected to be governed by an approved QA program.

The data obtained from these tests are used in code assessment and validation to develop and validate fuel performance models under both steady state and transient conditions. The current US Nuclear Regulatory Commission (NRC) fuel performance code is called FAST [5], which is a replacement for FRAPCON and FRAPTRAN. For steady state conditions, the assessment includes fission gas release, fuel centerline temperature, cladding strain, oxidation, and rod internal pressure—all of which make up the Specified Acceptable Fuel Design Limits (SAFDLs) for the fuel as required by 10 CFR Part 50, Appendix A, GDC 10. The integral-effects assessment associated with fuel performance modeling addresses the integral effect of all the models and correlations working together to analyze the thermal-mechanical behavior of the fuel rod under typical LWR conditions.

Several recent publications [6–10] have addressed the processes implemented or planned to collect the data needed to meet the NRC's expectations for fuel qualification of advanced LWR fuel types such as mixed-oxide (MOX) fuel, and more recently, accident-tolerant fuel, as well as fuel qualification for research and for non-LWRs. Work by Framatome and the NRC [6,7] address fuels that are similar to traditional LWR fuels except for the presence of MOX instead of uranium dioxide (UO_2). Work by Crawford [8], Daum et al. [9], and Idaho National Laboratory [10] may be more relevant to MSR fuel qualification in because it addresses fuels which have evolved from existing LWR fuels. These fuels include new fuel forms and new cladding materials or combinations of both, none of which have been used in LWRs.

The NRC plan for accident-tolerant fuel [7] addresses in-reactor performance—the traditional focus for fuel qualification—but the plan also emphasizes the importance of understanding fuel behavior as it relates to fabrication, transportation, and storage, all of which are within NRC's regulatory oversight.

3. UNIQUE MSR ASPECTS THAT IMPACT THE TRADITIONAL LWR FUEL QUALIFICATION APPROACH

3.1 IN-REACTOR ASPECTS

Liquid-fueled MSRs have certain unique aspects that NRC must consider in order to meet the advanced reactor fuel qualification process proposed in Section 1 above. For in-reactor performance, perhaps the most unique aspect of a liquid-fueled MSR design is that the fuel is also the coolant. Thus, while the fuel is not only the principal heat source, it also plays a major role in heat removal both for normal operations and off-normal events, as well as during shutdown (decay heat removal). Since fission products are produced in the fuel because of the fission process, the presence of these materials will change the chemical composition of the fuel salt, and in turn they will change the thermophysical properties of the fuel/coolant. Based on earlier studies of MSRs, it is well established that changing the composition of salt can affect properties such as melting point, boiling point, density, and viscosity. Understanding the impact of changes in the thermophysical properties of the fuel/coolant resulting from the addition of fission products and possible corrosion product impurities will be essential to the analyses of the ability to remove heat from the fuel during normal and off-normal conditions and after shutdown.

Another aspect of liquid-fueled reactors is that the fuel is molten and thus is a liquid during normal operations and during most off-normal events. Because the fuel is a liquid, the necessity to understand the mechanical behavior of the fuel/cladding/subassembly and core structures is restricted to the solid elements of the fuel system, as there are no permanent mechanical stressors in liquids. Apart from fission gas bubbles, MSR liquid fuels behave as single-phase Newtonian fluids. All MSR fuels being currently considered have low vapor pressures, so they will not require operation at high pressures to prevent the occurrence of two-phase conditions. The fuel normally operates at several hundred degrees Celsius below its boiling point. The fuel salt is mobile, so it circulates within the reactor vessel in some designs, and outside the reactor vessel in some designs. Other liquid-fuel MSR design variants restrict the fuel salt to natural circulation motion within fuel tubes that are externally cooled.

All fuel salts being considered can become highly corrosive. Some fuel salts under consideration contain hazardous materials such as beryllium, which may impact handling and maintenance operations, even for unirradiated fuel salt. Working with these types of salts requires protective clothing, respirators, or the use of glove boxes, as well as specialized training, policies, procedures, etc. Since some of the fission products, fissile materials, and other salt components are volatile, they may evolve from the liquid fuel at different rates and under different conditions, further changing the chemical composition of the fuel salt. Some of the fission and neutron transmutation products may decay into other elements as the fuel ages, further impacting the chemical composition of the fuel, especially during long-term storage. The chemical composition of fuel changes as a result of the fission process, refueling, chemistry control additions, or corrosion buildup. Except for deliberate changes, the inadvertent changes to fuel composition occur over long time intervals as compared to usual LWR accident time frames (years vs hours or minutes).

Since liquid-fueled MSRs have no cladding to contain the fission products, the liquid fuel should be chemically compatible with the materials that form the salt-wetted containment boundary, including heat exchangers, pump seals, valve components, flanges, gaskets, reactor vessel, storage tanks, and piping and any other materials it may contact, such as shielding or control assembly thimbles. Similarly, the cover gas contacting materials must be chemically compatible with the fission product gases, any spray or mists above the salt, as well as the fuel salt mixture in its vapor phase. Because the fuel chemical's composition will change slowly over time and will vary with temperature, the chemical compatibility with contacted materials may change over the lifetime of the reactor.

A significant part of MSR fuel may be contained outside the normal core, where prompt fission occurs, in places such as tanks, radioactive waste systems, cleanup systems, connecting piping, etc. This fission occurs as part of normal operation. Fuel found in these systems must be thermally managed since it will generate decay heat and has a freezing temperature of several hundred degrees Celsius. The amount of heat removal or supplemental heating needed will depend on the quantity, chemical composition, and age of the fuel found in these systems, all of which may vary from the characteristics of the fuel found in the core. Chemical compatibility with the surfaces of these containers must be ensured in a manner similar to that implemented for the vessel and other components mentioned earlier. The freezing point of MSR fuel is several hundred degrees Celsius above ambient temperature. Because it would be expected to freeze in some plant situations, its behavior as it cools below liquidus (freezing) needs to be well understood. Examples of such behavior are densification, stratification, plate-out, possible dimensional changes during freezing, and decomposition.

Fluoride-based MSR fuels have been shown to be radiation tolerant through the suite of experimental irradiations at power levels up to 8 GW/m3 when performed to support the Molten Salt Reactor Experiment (MSRE) [11,12] and the aircraft reactor program [13]. Radiolysis of liquid fluoride salts has no permanent consequences. Energies transferred to the salt constituents by irradiation are high enough to break the chemical bonding, but because the fuel salt is an ionic liquid, chemical re-bonding (recombination) occurs rapidly, and little if any of the fuel salt constituents remain in a disassociated state. Recombination is temperature dependent and has been shown to occur to a much lesser extent if the salt is deeply frozen (<200 °C) [14,15,16]. Chloride fuel salt behavior data during irradiation is not as readily available, but chloride fuel salt is expected to exhibit the same re-bonding in a liquid state as fluoride fuel salt since both are ionic liquids. The residual amount of radiolysis products within frozen salt depends on the dose, dose rate, radiation type, temperature, and specific salt composition.

3.2 UNIQUE MSR LIQUID FUEL ASPECTS THAT MAY IMPACT PREPARATION, STORAGE, AND TRANSPORTATION OF LIQUID MSR FUEL (NON-IN-REACTOR CONSIDERATIONS)

Plate-out of some fission products (mostly noble or semi-noble metals) will occur on surfaces that are in contact with irradiated liquid fuel. The deposition of insoluble materials on the surface of the container material will be dependent on temperature, time, and salt chemistry dependent, and in most cases, deposition will reduce the corrosion of the underlying material [17]. Deposition of tellurium is a notable exception to the protective effect. Over time at sufficiently high temperature, tellurium will embrittle the surface grain boundaries of nickel-based structural materials [18]. The deposits and overall structural material corrosion will impact the long-term storage and transportation of liquid irradiated fuel salts. However, once the salt has cooled sufficiently to form a solid crust on the container, the chemical interaction rate between the container wall and the salt will decrease substantially. Plate-out of highly radioactive materials will not only impact fuel storage, but it will also impact storage of removable components such as pumps, valves, heat exchangers, etc., as well as the shielding needed to protect plant personnel during routine maintenance and operations.

Additional fuel needed to make up for fissile material depletion during operation could be prepared on site. The first fuel loading could also be prepared on site. These options would more likely be implemented for a first-of-a-kind reactor. However, maintaining fuel salt synthesis capabilities at each reactor site would be more expensive than relying on centralized facilities beyond first reactor deployment. Without centralized fuel services, each site would only produce the limited quantities of fuel for its own use but would still need to be a fully qualified fuel synthesis facility. Also, loading a full core of fuel salt would require substantially more material than that required for ongoing operations. Therefore, the ability to produce fuel on site would increase efficiency when supplying multiple reactors from larger facilities. Separating the facilities also allows separation of chemical plant hazards from those of the nuclear reactor.

Some candidate MSR fuel salts have higher vapor pressure fissile forms that must be considered during the melting step of fuel preparation. UCl4 has a boiling point of 791 °C, and it consequently has an appreciable vapor pressure at fuel salt melting temperatures. Most chloride fuel salts include some UCl_4, along with the much less corrosive UCl_3, to minimize the amount of disproportion and potential plate-out of uranium metal [19].

Irradiated fluoride fuel salts stored as solids at less than 200 °C may release fluorine gas due to radiolysis. These salts may combine with U in the fuel to form UF_6, which will be gaseous at these temperatures. This formation of UF_6 may result in the transport of fuel to other locations in the reactor. This phenomenon was observed in the stored MSRE used fuel in 1994. Details of this phenomenon as it affected MSRE storage can be found in ORNL reports by Haubenreich [14], Williams [15], and Icenhour [16].

3.3 SAFETY IMPLICATIONS ASSOCIATED WITH INSUFFICIENT UNDERSTANDING OF LIQUID MSR FUEL SALTS DURING NORMAL AND ACCIDENT CONDITIONS

3.3.1 In-Reactor Safety Implications

Since fuel and coolant are one and the same, safety analysis codes used to analyze MSR behavior during AOOs and postulated accidents ideally reflect existing thermophysical properties of fuel salt (coolant) at the time an off-normal event is initiated. However, bounding accident analyses may be used to establish the fuel salt performance technical specifications. Safety analysis codes must also reflect any changes in the thermophysical properties that result from conditions occurring due to the off-normal event, such as those that might result from changes in temperature. A change in the melting point of the fuel salt would be of concern, as it could result in the fuel salt freezing in cold sections of the system, leading to a reduction in the ability to remove heat from the core following an event. The changing chemical properties of the fuel salt may impact the salt's fission product retention, fission product transport, and locations of fission product release points. In addition to fission product and corrosion product buildup and removal, the addition or removal of fuel (makeup) associated with normal operations would change fuel salt composition and thus would change the thermophysical properties of the fuel/coolant.

3.3.2 Plate-Out of Fission Products, Fuel, or Corrosion Products

Plate-out will cause increased heat deposition in certain locations and increased localized shielding requirements since most of the fission products (noble metals) are highly radioactive.

Plate-out of fission products will change the chemical composition of the fuel salt, which in turn impacts the thermophysical properties of the fuel/coolant. The safety implications of such changes are discussed in the preceding section.

Plate-out of fissile materials may have criticality safety implications. Fissile material solubility is a function of both salt composition and temperature. Increasing lanthanide fission product concentration decreases the solubility of fissile material in some fuel salts [20,21]. Plate-out of fissile materials could lead to unintended criticality if external materials such as additional or nearby fissile materials or moderators are in unfavorable geometry. Fuel salt technical specifications should require fuel salt composition to maintain adequate margin from fissile solubility limits under normal and accident conditions.

Corrosion buildup may impact the robustness of the fuel salt and/or salt vapor-wetted container material. Plate-out of noble metals on the salt-wetted barrier is generally thought to provide a protective effect [17]. However, deposition of tellurium can result in enhanced surface cracking by embrittling the surface grain boundaries of nickel-based structural materials [18]. Tellurium surface embrittlement has been addressed by development of alloys with substantially decreased sensitivity and by maintenance of the fuel salt in a sufficiently reducing chemical state to keep tellurium in an innocuous, dissolved form [22].

3.3.3 Fuel Preparation and Storage On-Site

For reactors using heterogeneous fuel, the fuel is fabricated in a separate facility and transported in finished form to the reactor, where it is inspected and typically stored dry or in the spent fuel pools until loaded in the reactor. The safety issues associated with wet or dry storage of fresh heterogeneous fuel are minimal. However, for MSR liquid fuel, the safety issues normally associated with the off-site fuel fabrication process must also be considered for the on-site MSR fuel preparation process.

Some fluoride fuel salts may contain beryllium, and the presence of this toxic material may impact fresh fuel preparation activities, as well as waste handling and storage of irradiated fuel. Beryllium is primarily an inhalation hazard, so all handling operations that could generate beryllium dust or vapors must be performed in sealed environments.

Volatile forms of fuel salt present during some aspects of fuel synthesis activities could result in relocation of fissile material, raising criticality safety concerns or complicating the material controls and accountability requirements needed to ensure safeguards. The fuel salt may solidify incongruently. If any fuel preparation steps involve slowly cooling fuel salt, stratification and fissile material concentration may result in criticality safety concerns not normally seen during solid fuel handling. These criticality safety concerns would be intensified if the fuel consists of high-assay low-enriched uranium or recycled fuel material containing Pu. The fissile content of freshly removed fuel salt will initially increase as the precursor isotopes bred in the reactor core (e.g. ^{239}Np or ^{233}Pa) decay into fissile elements. Small changes in fuel salt reactivity will continue to occur as shorter-lived minor actinides decay over time.

Improper storage of solid fluoride fuel salt under intense irradiation conditions (especially irradiated fuel salt containing U) could result in mobilization of fuel due to the presence of free fluorine gas combining with the U to form UF_6. If such a phenomenon is not addressed in the design and operations of a fluoride-fueled MSR, then criticality safety issues such as those seen at the MSRE could result. These issues included improper storage of spent fuel resulting in accumulation of fissile material in off-gas filters due to UF_6 mobilization [23]. In addition to criticality safety concerns, fissile material mobilization would complicate the material control and accountability relied upon for implementation of safeguards, and it would also create radiation hazards at additional locations.

3.4 DATA NEEDS AND POSSIBLE SOURCES OF INFORMATION TO SUPPORT ANALYSIS INCLUDING BENCHMARKS USED FOR CONFIRMATION (VERIFICATION OF MODELS) OR LICENSING APPLICATIONS

3.4.1 Separate-Effects Tests

A major portion of the fuel salt properties data needed to support an MSR fuel qualification program can be obtained from separate-effects tests. These include tests include measurements of physical properties of mixtures of fuel salts containing fission or corrosion products. The results from such measurements do not depend on geometry, and in most cases, the results do not depend on neutron irradiation behavior. Plate-out and corrosion data can be obtained from similar types of separate-effects tests. These separate-effects tests involve chemical property measurements that can be accomplished with nonradioactive forms of the elements, possibly eliminating the need for costly in-pile irradiation experiments, hot cells, or shielded PIE facilities. The lack of any geometry dependencies will allow the use of small sample tests, eliminate the need for lead test assemblies, and avoid the development of scaling rules. These rules are generally implemented for heterogeneous fuels in which geometry and mechanical damage (swelling, embrittlement) resulting from neutron irradiation are significant contributors to fuel behavior. As done for MSRE, most of the separate-effects tests can be conducted in parallel, each reflecting various compositions of fuel, salt, and nonradioactive chemical species of fission products or corrosion products. Use of separate-effects testing should reduce the cost and shorten the time needed for fuel qualification generally associated with heterogeneous reactor designs.

The MSRE and Aircraft Reactor Experiment (ARE) separate-effects tests are well documented in the open literature [24–26]. In addition to the nonradioactive separate-effects physical property measurements, several fluoride fuel samples also underwent irradiation testing. The results of these irradiation tests indicate that the irradiation damage to fluoride fuel salts was minimal. Details can be found in multiple reports [12,27,28]. Since the results from these early tests are generally applicable to any fluoride salt, they could serve as the basis for further fluoride fuel salt property measurements. However, the majority of tests were not conducted with the rigor of current nuclear QA methods as reflected in ASME NQA-1. Techniques must be developed to make use of this legacy information to support fuel qualification for advanced MSRs using fluoride-based fuel salts if supporting information is available. Rigorous ASTM standards [29] exist for measurement of molten salt properties; while these standards were developed for the chemical industry, they should be applicable to nuclear MSR fuel qualification salt property measurements.

3.4.2 Integral-Effects Testing and Flow Loops

For some of the MSR fuel qualification information needs identified in earlier sections of this report, geometry or flow rates may be influencing factors (e.g., densification, stratification, data needed for verification and validation tests to support safety analysis modeling and simulation, nuclear benchmarks, mechanistic source-term determination). Therefore, purely chemistry-based separate-effects testing may not be completely adequate. In such cases, integral-effects testing such flow loops may be necessary, along with appropriate scaling laws. In addition to the separate-effects physical property measurements made to support ARE and MSRE discussed in the preceding paragraph, several in-pile integral flow loop tests using fluoride salts were conducted to support the MSRE and ARE programs [30,31]. Multiple ex-pile natural and forced convection loops also supported the ARE and Molten Salt Breeder Reactor (MSBR) programs [32,33]. As in the MSRE separate-effects tests, lack of QA program rigor will prevent direct use of the results from the MSRE flow loop tests in an MSR fuel qualification program.

3.4.3 Chloride Fuel Salts Data Needs

For chloride-based fuel salts, some chemical property measurements exist, but the data are sparse compared to published data for fluoride salts [19,34–39]. As with fluoride salt information, the rigor of the QA program is less than that required in the current NRC expectations generally applied to fuel qualification. For chloride fuel salts few if any irradiation tests have been performed, so the data to support chloride salt irradiation behavior is minimal when compared to that available for the fluoride fuel salts. A similar situation exists for chloride fuel integral flow loop measurements. As with fluoride fuel salt irradiation testing, further chloride fuel salt irradiation testing results should be independent of geometry, so small sample irradiations are likely sufficient.

3.4.4 Delayed Neutron Drift

One phenomenon unique to MSR fuels is that for flowing liquid fuel, delayed neutrons will not be confined to the core. This results from the mobility of delayed neutron precursor fission products. Coupled multiphysics codes are being developed to enable proper modeling of this phenomenon [40]. Since no operational MSR exists today, it is difficult to benchmark this phenomenon. The MSRE data will likely need to be converted into a neutronic benchmark. This is being considered by the US Department of Energy (DOE)-sponsored MSR program office.

3.5 EXPECTATIONS FOR MSR APPLICATIONS REGARDING FUEL QUALIFICATION

As outlined in this report, the possible combinations of fission products and corrosion products that could exist in the fuel salt are so large as to make measurements of the thermophysical properties of all combinations cost prohibitive. However, an applicant might consider several approaches to make the process manageable. Currently, predictive models coupled with experimental results can provide a theoretical prediction of salt properties based on the combination of materials [41]. The DOE MSR campaign is also focusing on chemical property measurements and predictive models. Some of these predictive models must be validated using experimental data when they are available. Reliance on theoretical models alone would require extensive validation that is unlikely to be achieved with current MSR databases. However, these models have been used to determine that combinations of minor quantities of constituents generally result in minimal changes in overall mixture properties. Since most fission products are minor constituents in the MSR fuel salt, and since fission products build up and decay rather slowly, the fission product inventory is unlikely to significantly affect the thermophysical properties of the fuel salt in a manner that would impact the safety case over short time frames. However, the presence of corrosion products might present a larger challenge, as they are likely present in greater quantities and can build up much more quickly than fission products if chemistry controls are not implemented properly. Design basis accident analyses would therefore consider bounding values for these quantities.

Since reliance on predictive models is generally not possible for the reasons stated above, the applicants will likely need to institute a fuel salt sampling and property measurement plan over the operating life of the reactor to achieve the expectations for MSR fuel qualification addressed in the first paragraph of this report. Such a plan would be analogous to material surveillance programs in which extraction and measurement of coupons helps determine the irradiation behavior of LWR vessels. This approach would be analogous to routine chemistry and activity monitoring.

3.5.1 Examples of a Sampling and Measurement Plan

Below are some typical steps found in an MSR fuel qualification sampling and measurement plan.

1. Decide which fuel thermophysical properties impact the safety case. This can be accomplished by examining the input information used in the suite of design basis accident safety analysis codes.
2. Ensure that the properties used in the safety analysis are properly baselined. Each of these properties should be quantified by either experimental measurements or published literature for unirradiated fresh fuel salt as an initiating point for the safety analysis and a basis for establishing accident analysis bounding values.
3. Ensure that experimental results meet the NRC QA requirements established in 10 CFR Part 50, Appendix B (under which the NRC endorses certain editions of ASME NQA-1 via Regulatory Guide 1.28), if they are intended to be used to support the safety case.
4. Narrow the information requirements to focus on the most important information needs by (1) performing sensitivity studies to assess the importance of changes in thermophysical properties on the results of the safety analysis, and (2) using the results of these studies to inform selection and priority of thermophysical properties to be periodically measured.
5. Determine the inventory of contaminants in the fuel by using neutronic models to predict—as a function of time—the buildup and decay of the fission product inventory in the salt at fuel-salt–containing locations inside and outside the reactor core. These models should be able to account for inventory changes resulting from adding or removing fuel salt or salt treatment during normal operations.
6. Predict thermophysical changes occurring in the plant, using prediction models when possible to track changes in the most important thermophysical properties as a function of time based on the neutronic models and results from the sensitivity studies.
7. Measure thermophysical properties taken from samples, comparing predictive to actual measurements of physical properties taken from the fuel salt samples.
8. Develop a sampling plan that includes both frequency and types of sample measurements taken. The frequency and determination of these measurements will be based on the results of the neutronics models and the sensitivity studies.

3.5.2 Use of the Results of the Sampling and Measurement Plan

The design basis accident safety analyses results will incorporate uncertainty in a conservative manner, including the thermophysical property values. If the measurement results from the fuel salt samples approach or exceed the values assumed in the design basis accident safety analyses (including the uncertainty), then appropriate corrective actions must be taken to avoid violation of the plant's safety/licensing basis. Such actions would be included as part of the reactor's technical specifications, as discussed in 10 CFR 50.36(c), and they would be analogous to limiting conditions for operation analysis in Criterion 2. In addition to including requirements for thermophysical properties measurements, the plan might also include measures to ensure that fission products such as cesium or iodine are still being retained in the salt. This information would be relevant if it was used to determine the mechanistic source term. The sampling plan would also include measures of corrosion products, as well as fission products for requirements such as radiation protection and criticality safety. Evidence of fissile material plate-out should also be included in sample measurements. Possible corrective actions to be included are as follows:

1. Remove or reduce the quantity of fission products or corrosion products in the salt. (Changing the salt chemistry redox, which might increase plate-out on filters, may be one example of such a corrective action.)
2. Institute a chemical process to selectively remove certain contaminates.
3. Shut down the reactor and replace the fuel salt.

4. SUMMARY AND RECOMMENDATIONS

This report introduces a proposed technology neutral expectation for MSR fuel qualification. The report discusses the current approach to fuel qualification used for LWRs and how the unique aspects of liquid-fueled MSRs indicate that a new approach is needed to similarly document their fuel qualification. In addition, the report discusses why these unique aspects of MSRs require a broader view of fuel qualification than has been traditionally applied to LWRs, as qualification of LWRs has focused mainly on reactor-core fuel performance. This broader view must include examination of the safety aspects of fuel during fuel preparation, storage, transportation, and waste treatment in the core and in various parts of the reactor facility.

Since MSR experience is limited and a wide diversity of liquid fuels is being proposed by MSR designers, the report discusses possible approaches that might provide the data needed to support a comprehensive MSR fuel qualification program. The approaches focus on chemical property measurements and reduced reliance on irradiation data and mechanical measurements. The report emphasizes that the liquid-fueled systems' fuel qualification will have little if any need to replicate detailed geometry configurations compared to those developed for heterogeneous fuel systems.

Finally, a multistep process is proposed as an approach to MSR fuel qualification, along with some performance metrics to ascertain compliance of the reactor's safety design basis.

5. REFERENCES

1. US Nuclear Regulatory Commission. 2017. *Public Meeting on Improvements for Advanced Reactors, August 3, 2017.* Washington, DC, https:/www.nrc.gov/docs/ML1722/ML17220A315.pdf (accessed November 2018).
2. 10 CFR Part 50.46. "Acceptance Criterion for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." *Code of Federal Regulations,* Title 10, Energy, Part 50.46. Washington, DC.
3. US Nuclear Regulatory Commission. 2007. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*. NUREG-0800. NRC Accession Number ML092330826. Washington, DC.
4. US Nuclear Regulatory Commission. 2007. *Combined License Applications for Nuclear Power Plants.* Regulatory Guide 1.206. Washington, DC.
5. A. Prudil, B. J. Lewis, P. K. Chan, and J. J. Baschuk. 2015. "Development and Testing of the FAST Fuel Performance Code: Normal Operating Conditions, Part 1." *Nucl. Eng. Design* 282 (February).
6. Framatome. 2001. *MOX Fuel Qualification Plan.* FRA-ANP (US) No. 77-5005775-02, DCS-FQ- 1999-01, Rev. 2, NRC Accession Number ML011220513. Washington, DC.
7. US Nuclear Regulatory Commission. 2017. *Draft Project Plan to Prepare the US Nuclear Regulatory Commission to License and Regulate Accident Tolerant Fuel.* NRC Accession Number ML17325B772, December 14. Washington, DC.
8. Douglas C. Crawford et al. 2007. "An Approach to Fuel Development and Qualification." *J. Nucl. Mater.* 371: 232–242.
9. K. Daum et al. 2014. *Base Monolithic Fuel: Research, Development, and Qualification Plan.* INL/EXT-13-30238, Rev. 0, December. Idaho Falls, ID: Idaho National Laboratory.
10. Idaho National Laboratory. 2010. *NGNP Fuel Qualification White Paper.* INL/EXT-10-17686, Rev. 0, July. Idaho Falls, ID.
11. W. R. Grimes. 1963. *Radiation Chemistry of MSR System.* ORNL-TM-500, June. Oak Ridge, TN: Oak Ridge National Laboratory.
12. W. R. Grimes. 1967. *Chemical Research and Development for Molten-Salt Breeder Reactors*. ORNL-TM-1853. Oak Ridge, TN: Oak Ridge National Laboratory.
13. James A. Lane, H. G. McPherson, and Frank Maslan. 1958. Section 12-5, "Radiation Stability of Fluoride Mixtures," in *Fluid Fuel Reactors* Addison-Wesley Publishing Company, Reading, MA, USA for the US Atomic Energy Commission.
14. P. N. Haubenreich. 1970. *Fluorine Production and Recombination in Frozen MSR Salts after Reactor Operations.* ORNL/TM-3144, September. Oak Ridge, TN: Oak Ridge National Laboratory.
15. D. F. Williams et al. 1996. *A Descriptive Model of the Molten Salt Reactor Experiment After Shutdown: Review of FY 1995 Progress*. ORNL/TM-13142, January. Oak Ridge, TN: Oak Ridge National Laboratory.
16. A. S. Icenhour, D. F. Williams, L. D. Trowbridge, L. M. Toth, and G. D. Del Cul. 2001. "An Overview of Radiolysis Studies for The Molten Salt Reactor Remediation Project," Global 2001, Paris (France), September 9–13.
17. Mark T. Robinson, William A. Brooksbank, Samuel A. Reynolds, Henry A. Wright, and Thomas H. Handley. 1958. "The Behavior of Fission Products in Molten Fluoride Reactor Fuels." *Nucl. Sci. Eng.* 4(3): 288–296.
18. J. R. Keiser, 1977. *Status of Tellurium-Hastelloy N Studies in Molten Fluoride Salts*. ORNL-TM-6002, October. Oak Ridge, TN: Oak Ridge National Laboratory.
19. B. R. Harder, G. Long, and W. P. Stanaway. 1969. "Compatibility and Processing Problems in the Use of Molten Uranium-Chloride-Alkali-Chloride Mixtures as Reactor Fuels." *Nucl. Met.* 15: 405–432.
20. C. J. Barton. 1960. "Solubility of Plutonium-Trifluoride in Fused Alkali Fluoride Beryllium Fluoride Mixtures," *J. Phys. Chem.*, 64, 306.

21. Victor V. Ignatiev, Mikhail V. Kormilitsyn, Andrey A. Lizin, Alexander V. Zagnitko, Sergey A. Konakov, Alexander V. Merzlyakov, Sergey V. Tomilin, Alexander A. Khokhryakov, and Alexander G. Osipenko, "Key Experimental Results of the PYROSMANI Project," *Procedia Chem.*, 21, 2016, 417–424, https://doi.org/10.1016/j.proche.2016.10.058, accessed November 2018.
22. H. E. McCoy, Jr. 1978. "Status of Materials Development for Molten Salt Reactors," ORNL-TM-5920, January. Oak Ridge, TN: Oak Ridge National Laboratory.
23. F. J. Peretz, J. E. Rushton, R. L. Faulkner, K. L. Walker, and G. D. Del Cul. 1998. "Removal of uranium and salt from the Molten Salt Reactor Experiment," SPECTRUM `98: Nuclear and Hazardous Waste Management International Topical Meeting, Denver, CO (United States), ember 13–18.
24. S. Cantor. 1968. *Physical Properties of Molten-Salt Reactor Fuel, Coolant, and Flush Salts*. ORNL-TM-2316, August. Oak Ridge, TN: Oak Ridge National Laboratory.
25. C. F. Baes. 1974. "The Chemistry and Thermodynamics of Molten Salt Reactor Fuels." *J. Nucl. Mater.* 51: 149–162.
26. W. R. Grimes, 1970. "Molten-Salt Reactor Chemistry." *Nucl. Appl. Technol.* 8, 2: 137–155, published online 2017, dx.doi.org/10.13182/NT70-A28621, accessed November 2018.
27. G. W. Keilholtz, J. G. Morgan, and W. E. Browning. 1959. "Effect of Radiation on Corrosion of Structural Materials by Molten Fluorides." *Nucl. Sci. Eng.* 5: 5–20.
28. H. C. Savage, E. Compere, J. M. Baker, and E. G. Bohlmann. 1967. *Operation of Molten-Salt Convection Loops in the ORR.* ORNL-TM-1960, December. Oak Ridge, TN: Oak Ridge National Laboratory.
29. ASTM International. 2013. *Standard Test Method for Thermal Diffusivity by Flash Method.* Standard E1461-13.
30. D. B. Trauger and A. Conlin, Jr. 1961. "Circulating Fused-Salt Fuel Irradiation Test Loop." *Nucl. Sci. Eng.* 9: 346–356.
31. W. E. Browning, M. F. Osborne, H. E. Robertson, R. P. Shields, and D. E. Guss. 1956. "LITR Vertical In-Pile Loop." *Solid State Division Semiannual Progress Report for Period Ending August 30, 1956*. ORNL-2189. Oak Ridge, TN: Oak Ridge National Laboratory.
32. W. D. Manly et al. 1958. *Aircraft Reactor Experiment—Metallurgical Aspects.* ORNL-2349. Oak Ridge, TN: Oak Ridge National Laboratory.
33. R. B. Briggs. 1961. *Molten-Salt Reactor Program Progress Report for Period from March 1–August 31, 1961.* ORNL-3215. Oak Ridge, TN: Oak Ridge National Laboratory. https://www.osti.gov/scitech/servlets//purl/4804884, accessed November 2018.
34. C. J. Raseman, H. Susskind, G. Farber, W. E. McNulty, and F. J. Salzano. 1960. *Engineering Experience at Brookhaven National Laboratory in Handling Fused Chloride Salts.* BNL-627, June. Upton, NY: Brookhaven National Laboratory.
35. Alvan Glassner. 1957. *The Thermochemical Properties of the Oxides, Fluorides, and Chlorides to 2500 K.* ANL-5750. Lemont, IL: Argonne National Laboratory.
36. R. Benz and J. A. Leary. 1961. "Some Thermodynamic Properties of the System Plutonium Chloride–Sodium Chloride from Electromotive Force Data." *J. Phys. Chem.* 65 (June).
37. M. Furrer, M. 1977. "Chemical Behavior of Radiosulphur Obtained by $^{35}Cl(n,p)^{35}S$ during In-Core Irradiation of Alkali-Chloride Melts." *J. Inorg. Nucl. Chem.* 39(6): 1085–1086.
38. H. S. Friedman, 1963. *Preparation, Purification, and Properties of the Uranium Chlorides*. Literature survey. ORNL-CF-63-11-5. Oak Ridge, TN: Oak Ridge National Laboratory, DOI: 10.2172/1400261.
39. O. Benes, and R. J. M. Konings. 2008. "Thermodynamic Evaluation of the $NaCl$-$MgCl_2$-UCl_3-$PuCl_3$ System." *J. Nucl. Mater.* 375.2: 202–208.
40. B. R. Betzler et al. 2017. "Molten Salt Reactor Neutronics and Fuel Cycle Modeling and Simulation with SCALE." *Ann. Nucl. Energy* 101: 489–503.
41. O. Benes et al. 2008. "Thermodynamic Properties and Phase Diagrams of Fluoride Salts for Nuclear Applications." *J. Fluorine Chem.* 130: 22–29.