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Assessment of Adequacy and Completeness of Standards Endorsed by Regulatory Guides to Molten Salt Reactors

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June 2023

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Nuclear Energy and Fuel Cycle Division

ASSESSMENT OF ADEQUACY AND COMPLETENESS OF STANDARDS ENDORSED BY REGULATORY GUIDES TO MOLTEN SALT REACTORS

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ABBREVIATIONS

ac	alternating current
ACI	American Concrete Institute
ACS	auxiliary cooling system
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	(ASME) Boiler and Pressure Vessel Code
BSR	Board of Standards Review
BTP	branch technical position
°C	degrees Celsius
CFR	Code of Federal Regulations
DBA	design-basis accident
dc	direct current
DG	diesel generator
DOE	US Department of Energy
DRACS	direct reactor auxiliary cooling system
EM	electromagnetic
EPRI	Electric Power Research Institute
GDC	general design criteria
GE	General Electric
HAA	head access area
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
IAP	implementation action plan
ID	identification number
IEEE	Institute of Electrical and Electronic Engineers
IHTS	intermediate heat transport system
IHX	intermediate heat exchanger
IMC	inspection manual chapter
IP	inspection procedure
ISA	Instrumentation Society of America
ISI	in-service inspection
ISG	interim staff guidance
LWR	light-water reactor
mHTGR	modular high-temperature gas reactor
MSR	molten salt reactor
MWt	megawatt thermal
N/A	not applicable
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	US Nuclear Regulatory Commission
NUREG	US Nuclear Regulatory Commission Regulation
OMB	Office of Management and Budget
PCS	plant control system
PRISM	Power Reactor Inherently Safe Module
PSID	preliminary system information document
PSPS	primary sodium processing subsystem

PWR	pressurized water reactor
RCPB	reactor coolant pressure boundary
RG	regulatory guide
RHR	residual heat removal
RO	reactor operator
RPS	reactor protection system
RS	reactor system
RVACS	reactor vessel auxiliary cooling system
SDO	standards development organization
SER	safety evaluation report
SNM	special nuclear material
SRO	senior reactor operator
MSR	sodium fast reactor
MSR-DC	sodium fast reactor–design certification
SRP	standard review plan
SSC	structures, systems, and components
UFSAR	updated final safety analysis report
V	volt

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EXECUTIVE SUMMARY

The benefits of using voluntary consensus standards and industry guidance documents and reports are evident in the design and licensing of the current generation of nuclear power plants. The use of voluntary consensus standards and industry guidance and reports facilitates the design and licensing of advanced reactors to improve the effectiveness and efficiency of the licensing and regulation of non-light-water reactor (non-LWR) technologies. Most of the regulations, guidance, and standards applicable to nuclear power plants were developed for water-cooled plants, so they may not adequately address factors such as the coolants, materials, temperatures, operations, testing, and maintenance proposed for advanced reactors.

Consistent with Office of Management and Budget (OMB) Circular A119, it is the US Nuclear Regulatory Commission's (NRC's) policy to use standards developed by voluntary consensus standards bodies if available and appropriate. The NRC incorporates by reference consensus standards to provide the certainty and predictability desired by stakeholders. This approach also minimizes the expenditure of NRC resources that would otherwise be necessary to develop new regulations at a level of detail comparable to that provided by existing consensus standards. To review and regulate a new generation of non-LWRs, the NRC's near-term strategies include the following:[1]

- Work with stakeholders to determine the currently available codes and standards applicable to non-LWRs and to identify the technical areas where gaps exist;
- Participate with the standards development organizations (SDOs) that are actively involved in developing codes and standards for non-LWRs; and
- Review codes and standards for endorsement.

The NRC's mid/long-term action plan recognizes that it typically takes years to develop consensus codes and standards or to promulgate a new or revised regulation.

The NRC's regulatory framework is specific to LWRs. Similarly, the guidance for meeting regulatory requirements is primarily applicable to water-cooled nuclear power plants. Not surprisingly, many industry guidance documents and reports cited or referenced in regulatory documents such as the NRC Standard Review Plan (LWR edition) (NRC Regulation [NUREG]-0800), regulatory guides (RGs), the Code of Federal Regulations (CFR), NRC bulletins, information notices, circulars, generic letters, and policy statements are specific to LWRs.

To understand the size and scope of work required to expand the regulatory framework to address non-LWRs, a program was initiated to provide the US Department of Energy (DOE) with the following items:

1. An estimate of the number of standards that need revision;
2. An estimate of the levels of effort required to revise those standards;
3. A description of the process for revising or creating a new standard; and
4. A description of the NRC's process for endorsing a standard.

This review focused on the adequacy and completeness of standards to a molten salt reactor (MSR). The standards selected for the focused review were evaluated at a high level with respect to their relative adequacy and completeness to less developed technologies. It is outside the scope of this review to

[1] *NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness, December 2016.* (ML16356A670)

prioritize the NRC or SDO standard development activities or to relate their development to the NRC mission.

The first step in estimating the size and scope of the effort to ensure that the standards support the industry's activities for advanced reactors was to obtain a list of all standards cited in RGs. This step identified more than 865 standards cited in RGs.

The second step was to narrow down the number of standards for an in-depth review to assess their potential application to MSRs. The objective of the down-selection process was to limit the review to standards endorsed, partially endorsed, or endorsed with exceptions by RGs in Division 1 (Power Reactors), Division 3 (Fuels and Materials Facilities) and Division 5 (Plant Protection) that are active (i.e., the RG has not been withdrawn). This step identified 182 standards. The next step was to identify those standards approved for use in NRC Regulation (NUREG)-0800 (the Standard Review Plan). This step identified 9 unique standards in addition to those endorsed by RGs. Finally, a review was performed to identify those standards required by the CFR. This step identified an additional 12 standards for review. In total, 197 standards were selected for review.

The third step was to review the 197 standards for their adequacy and completeness to MSRs and to identify the need for new standards unique to MSRs. This step identified 14 potential new standards.

Of the 197 reviewed standards endorsed, approved for use, or required, about 40% will not require any changes—18 standards require no changes, and 5 standards are not applicable. Of the other ~60% of standards reviewed, 16 will require minor changes, 19 will require significant changes, and there was insufficient information available to assess the adequacy and completeness of 2 standards. Significant changes are needed because of the higher energy spectrum, higher temperatures, and corrosive coolants. Material properties for metals, concrete, and protective coatings must be addressed.

The 12 new standards needed are likely to be developed by an SDO such as the American Nuclear Society (7) or the American Society of Mechanical Engineers (5). These new standards are the result of the increased use of passive systems and the molten salt environment.

This review then evaluated the basic process to develop a new standard or the process to revise an existing standard. The amount of time required to develop or modify a standard is related to the complexity of changes needed, up to and including the development of a new standard. The development and approval of a new standard by an SDO is likely to take 5–8 years. Because of the long time to develop or even make small changes to a standard through an SDO, other means to provide the necessary guidance—such as DOE or NRC funding or industry participation—should be explored.

After a standard is available for use, it must be endorsed by NRC in regulations (i.e., codified) or as guidance (e.g., in a RG, NUREG, or the SRP) for that standard to be used in the regulatory process. The review and NRC endorsement of codes and standards (with possible clarifications and exceptions) can only follow the development and issuance of a standard by the SDOs. This endorsement process could add years to the adoption of a standard in the regulatory process, with additional time needed if the approval is made through a regulation as opposed to guidance.

With respect to licensing an MSR, the 12 new standards should be prioritized, and standards requiring significant changes should be a close second. Delays in addressing these changes will directly affect the licensing timeline and commercial deployment. Furthermore, a staggered submittal of requests will be necessary to prevent overwhelming an SDO.

The use of codes and standards will be integral to the NRC's strategy to improve readiness to regulate non-LWR technologies. If a consensus standard is not available, then NRC can create its own guidance. If a standard is available, then NRC must justify why it is not being used. There is a great advantage to industry if they participate with the SDOs to create the standards rather than have a standard or guidance imposed on them.

Designs can proceed without approved standards, but approved standards can help with multiple licensees. Advanced reactor technology licensing and deployment will likely be delayed significantly if applicable and endorsed standards are not available for use by both technology developers and the NRC. Delays in providing the NRC with the knowledge base and tools for reviewing non-LWR applications will increase the effort needed to review an application, thereby delaying its approval.

ABSTRACT

The benefits of using voluntary consensus standards and industry guidance documents and reports are evident in the designing and licensing of the current generation of nuclear power plants. The use of voluntary consensus standards and industry guidance documents and reports the design and licensing of advanced reactors to improve the effectiveness and efficiency of the licensing and regulation of non-light-water reactor (non-LWR) technologies. However, most of the regulations, guidance, and standards applicable to nuclear power plants were developed for water-cooled plants and may not adequately address factor such as the coolants, materials, temperatures, operations, testing, and maintenance proposed for advanced reactors.

The first step taken to understand the size and scope of work required to expand the regulatory framework to address non-LWRs, was to obtain a list of all standards cited in RGs that are frequently used to endorse standards for providing an acceptable method for satisfying NRC's regulations. The second step was to narrow down the number of standards for an in-depth review to assess their potential application to a molten salt reactor (MSR). The objective of the down-selection process was to limit the review to standards endorsed, partially endorsed, or endorsed with exceptions by RGs in Division 1 (power reactors), Division 3 (Fuels and Materials Facilities), and Division 5 (Materials and Plant Protection) that are active (i.e., the standard is not inactive or withdrawn). Standards can also be approved for use via NUREG-0800 (the Standard Review Plan) or required via the CFR. The last step was to review the 197 standards selected for detailed review and identify gaps that would need to be addressed by the development of new standards. This process added 14 potential new standards to the list.

Just as standards with no or limited changes would not be prioritized for updates, not all standards that would require significant updates would need to be addressed immediately. Therefore, the final step was to categorize those standards as high, medium, or low priority in terms of needing updates.

1. INTRODUCTION

The benefits of using voluntary consensus standards and industry guidance documents and reports are evident in designing and licensing of the current generation of nuclear power plants (NPPs). The use of voluntary consensus standards and industry guidance documents and reports facilitates the licensing of advanced reactors to improve the effectiveness and efficiency of the licensing and regulation of non-light-water reactor (non-LWR) technologies. However, most of the regulations, guidance, and standards applicable to NPPs were developed for light water-cooled plants and may not adequately address the coolants, materials, temperatures, operations, testing, maintenance, etc., proposed for advanced reactors.

Consistent with Office of Management and Budget (OMB) Circular A119 [1], it is the US Nuclear Regulatory Commission's (NRC's) policy to use standards developed by voluntary consensus standards bodies if available and appropriate [2, 3]. The NRC incorporates by reference consensus standards to provide the regulatory certainty and predictability desired by stakeholders, minimizing the expenditure of NRC resources that would otherwise be necessary to develop regulations with a level of detail comparable to that provided by consensus standards [4]. To review and regulate a new generation of non-LWRs, the NRC's near-term strategies include the following [5]:

- Work with stakeholders to determine the currently available codes and standards applicable to non-LWRs and to identify the technical areas where gaps exist,
- Participate with the standards development organizations (SDOs) that are actively involved in developing codes and standards for non-LWRs, and
- Review codes and standards for endorsement.

The NRC's mid/long-term action plan recognizes that it has typically taken years to develop consensus codes and standards and promulgate a new or revised regulation [6]. The number of standards involved and the level of effort (LOE) needed to revise or develop new standards applicable to non-LWRs remain to be determined.

The NRC's regulatory framework is specific to LWRs, so the guidance for meeting these regulatory requirements was developed for water-cooled nuclear power plants. Not surprisingly, many industry guidance documents and reports cited or referenced in regulatory documents such as the *NRC Standard Review Plan (LWR edition)* (NRC Regulation [NUREG]-0800), regulatory guides (RGs), the Code of Federal Regulations (CFR), NRC bulletins, information notices, circulars, generic letters, and policy statements are also LWR-specific.

It is not known how many existing standards would apply in designing and licensing a non-LWR, what changes would be needed so that the scope of the standard address issues related to non-LWRs, and whether new standards would be required to address new technological issues introduced by non-LWR technology. The time required to revise, develop, approve, and endorse a new or revised standard for adequacy and completeness to an advanced reactor must also be estimated.

To understand the size and scope of work required to expand the regulatory framework to address design and licensing issues for a molten salt reactor (MSR), a program was initiated to provide DOE with the following items:

1. An estimate of the number of standards that need revision,
2. An estimate of the LOE required to revise those standards,
3. A description of the process for revising or creating a new standard, and
4. A description of the NRC's process for endorsing a standard.

In a pilot program to evaluate the use of standards endorsed by Division 1 RGs, the scope was limited to sodium fast reactors (SFRs) and focused on RGs in Division 1 (Power Reactors) because they are among several guidance documents that describe an acceptable method for applicants and licensees to meet specific provisions of the NRC's regulations, techniques used by the staff members to evaluate specific problems or postulated accidents, or data needed by staff members to review applications for permits and licenses [7].

Similar to the pilot program for SFRs, the first step in evaluating standards applicable to MSRs was to obtain a list of all standards cited in Division 1 (Power Reactors) of the RGs. For an approved standard to be used in the regulatory process, it must be endorsed by NRC in regulations (i.e., codified) or as guidance (e.g., in an RG, NUREG, or the SRP). This list of standards to be considered for review was expanded from the pilot program to include RGs endorsed from not only Division 1 but also from Division 3 (Fuels and Materials Facilities) and Division 5 (Materials and Plant Protection). The list of standards for review was then further increased to evaluate those approved for use in NUREG-0800 [8] (the SRP) or required through the CFR. From this list, a down-selection process was used. This process focused on standards being endorsed by an active RG (i.e., not withdrawn), approved for use in the SRP, or required by the CFR. This focus served three purposes:

1. To narrow down the number of standards for review to endorsed, approved, or required standards to discern how many would require review,
2. To categorize the LOE[2] required to develop or revise each standard for adequacy and completeness to an MSR to determine the LOE needed, and
3. To prioritize the standards needing modifications for adequacy and completeness to MSRs.

The first step is estimating the size and scope of standards to be revised or created to aid in designing and licensing an MSR. The next task is to provide detailed assessments and inputs in support of (1) revision of existing consensus standards and (2) development of new standards to be used to justify the need for a new or revised standard to the SDO. Assessments and inputs will also help to prioritize the 33 standards for revision or development (19 with significant changes and 14 new). The ranking would be based on adequacy and completeness to other reactor types, whether data exist or research is needed to collect data, and the data's effect on an SDO. Efforts should be made to gauge the volume of requests and stagger the submittal of requests to prevent overwhelming an SDO. This review focused on MSRs and could be expanded to other reactor technologies (e.g., gas reactors). Engagement with NRC and the SDOs is essential during these efforts.

This review then evaluated the basic development process for a new standard or the process to revise an existing standard. The amount of time to develop or modify a standard is related to the complexity of changes needed, up to and including the development of a new standard. The development and approval of a new standard by an SDO is likely to take 5–8 years. The review and NRC endorsement of codes and standards (with possible clarifications and exceptions) can only follow the development and issuance of a standard by the SDOs. This NRC endorsement process could add years to the adoption of a standard in the regulatory process, with additional time needed if the approval is made through a regulation as opposed to guidance.

This project began by identifying standards endorsed by RGs in Division 1 (Power Reactors), Division 3 (Fuels and Materials Facilities), and Division 5 (Materials and Plant Protection). However, standards endorsed by RGs are not the only standards that provide acceptable methods for meeting NRC

[2] "Level of effort" represents the number of changes that might be required and not the amount of resources and time to make those changes.

regulations. For example, the acceptance criteria in the *Standard Review Plan* (SRP) occasionally cite standards that provide an approved method but are not endorsed by an RG. Furthermore, the CFR may require certain standards to be applied to the design and operation of an NPP.

Section 2 of this report provides a short description of an MSR.

Section 3 provides an overview of the selection and ranking process.

Section 4 provides the results of the reviews and identifies the level of effort to revise a standard (if needed), the prioritization of revising those standards, and new standards that should be considered.

Section 5 describes the process for creating or revising a consensus standard and the NRC endorsement process for a standard.

Section 6 presents an overview of the results of the review.

Appendix A presents the detailed review results for the 197 consensuses endorsed, approved for use, or required.

Appendix B lists the SDO standards endorsed, approved for use, or required by the CFR that are withdrawn or inactive.

Appendix C lists the non-SDO documents endorsed, approved for use, or required by the CFR.

Appendix D provides a review of the sources that endorse, approve for use, or require a standard.

2. MOLTEN SALT REACTOR DESIGN DESCRIPTION

2.1 OVERALL REACTOR DESCRIPTION

The Molten Salt Reactor Experiment (MSRE), which operated from 1965 to 1969, and the proposed Molten Salt Breeder Reactor (MSBR) program, which ran from 1970 to 1976, are described to illustrate design features typical of MSRs. The MSRE was an 8 MWt technology demonstration reactor without a power conversion system (PCS). The proposed MSBR was a 2,250 MWt (1,000 MWe) MSR with a superheated steam PCS.

MSRs generally use molten salts to transfer heat away from the reactor core, although liquid metals have also been proposed to transfer the heat from liquid salt fuel. The salt component can function solely as the coolant, or it can function as both the fuel and the coolant. The heat can then be used to produce electricity or to support an industrial process. As shown in Figure 1, MSRs fall into two classes: salt-cooled reactors, in which the core contains a solid fuel and liquid salt coolant; and salt-fueled reactors, in which the fuel forms part of the liquid halide salt mixture. Salt-fueled MSR facilities are significantly different from previously licensed nuclear fuel and reactor facilities. Salt-fueled reactors are further subdivided into thermal-spectrum and fast-spectrum designs [9]. The types of radiological systems and locations of fissile material vary significantly by MSR concept. This MSR standard review focuses on salt-fueled MSRs because salt-cooled MSRs (i.e., those with solid fuel) are like other advanced reactor technologies, such as sodium-cooled reactors, in terms of systemic operation. Kairos Power is the sole US-based salt-cooled reactor developer; its Hermes demonstration reactor is planned for operation in Oak Ridge, Tennessee by 2026 [10].

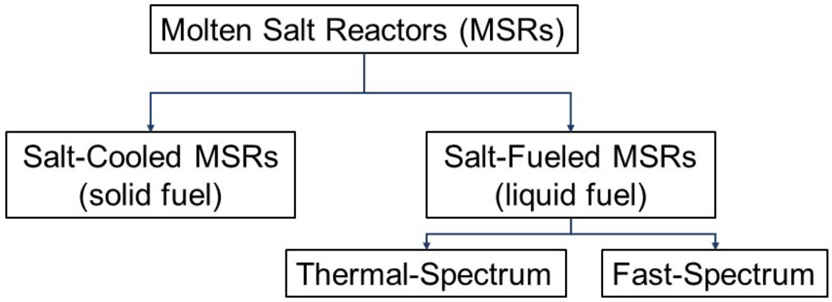

Figure 1. Categorization of common MSR designs.

The use of liquid salt fuel in MSRs provides substantial design and operational flexibility. MSR fuel salts consist of mixtures of fissile and generally fertile materials, halides, and carrier salts. For example, FLiBe fuel salt consists of uranium, fluoride, lithium, and beryllium. All salt-fueled MSRs use molten fluoride, chloride, or mixed halide salts as the carrier salt at low pressure. Fast-spectrum MSRs require a substantially higher concentration of fissile materials in their fuel salts than thermal-spectrum MSRs to maintain criticality. Optimized fuel salts will have a high boiling point, high radiolytic stability, large volumetric heat capacity, acceptable thermal conductivity, low parasitic neutron capture, and strong retention of fission products (FPs). Different designs are currently under development to achieve different performance objectives [11]. For example, some designs focus on consuming the actinides in spent LWR fuel to reduce long-term radiotoxicity, whereas others seek to implement a modernized version of the MSRE technology to minimize their development risks [9].

The fuel salt is highly radioactive after operation because of the presence of FPs and actinides. Therefore, all systems containing fuel salt will be in some type of functional containment with very restricted access even when the reactor is shut down. Most maintenance is likely to require remote handling [12].

Thermal-spectrum technologies, such as the MSRE demonstrated at Oak Ridge National Laboratory (ORNL), typically rely on frequent removal of FPs from the molten fuel salt to maintain high neutron flux in the reactor. However, one advantage of using a fast-spectrum design is that online fuel processing is much less intensive because FP compounds have lower neutron capture cross sections at higher neutron energies. Some level of FP removal is inherent in any liquid fuel because gaseous materials bubble out and solids deposit onto surfaces.

Both thermal- and fast-spectrum MSR technologies must account for gaseous FPs by incorporating an off-gas system in the design. Otherwise, accumulating gases would pressurize the fuel–salt boundary and potentially affect reactivity by building up in the core area. Fast-spectrum MSRs will generate larger quantities of fission gases per unit fuel volume because of their higher power densities. A typical salt-fueled MSR functional diagram is shown in Figure 2.

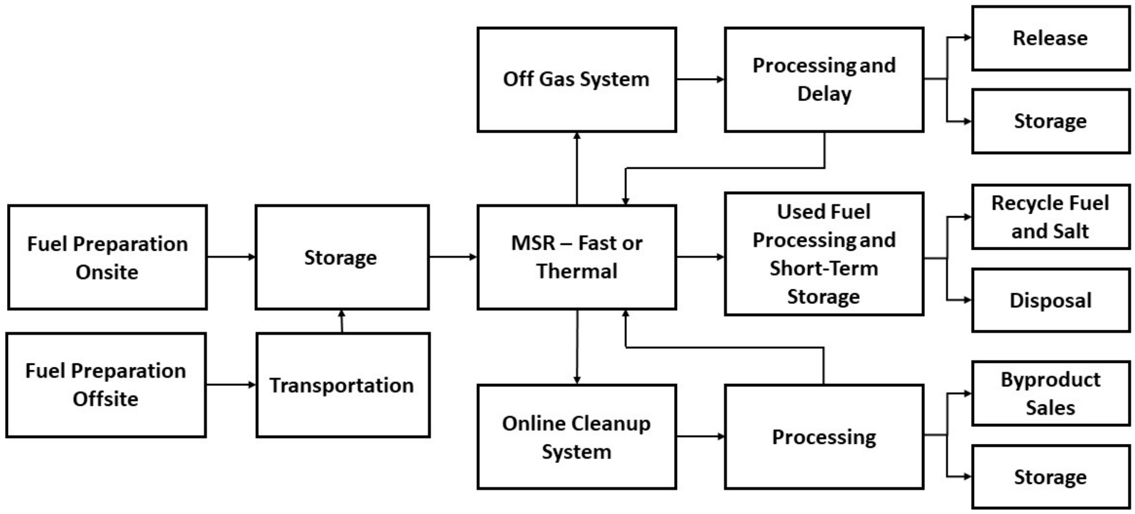

Figure 2. Generic salt-fueled MSR functional block diagram.

Most MSR technologies involve fuel salt flowing in and out of the core region. Because the delayed neutron precursors are mobile, reactivity will be affected by the core fuel salt flow rate. In addition, most MSRs have a large negative temperature coefficient of reactivity. If an accident or transient causes an increase in core temperature, then the fuel expands out of the core region, adding negative reactivity. Such passive safety in advanced reactor design is consistent with the NRC advanced reactor policy statement [13] regarding such features.

Most of the core in a thermal-spectrum MSR will be filled with neutron moderator material and flow channels for the fuel salt. This configuration will necessitate moderator support structures for the thermal-spectrum core. The thermal-spectrum core will be surrounded by additional moderator/reflector material to improve the neutron economy and a subsequent outer layer of neutron absorber material to minimize the neutron flux on the reactor vessel. The core of most fast-spectrum MSRs will contain no materials other than the fuel salt and possibly sparging gas to aid in the removal of FP gases. Alternatively, the fast-spectrum fuel salt may be confined to fuel channels cooled by an external high-atomic-mass coolant similar to an MSR. As with a thermal-spectrum MSR, fast-spectrum MSR cores will be surrounded by neutron reflector/shielding materials to improve neutron economy and reduce radiation damage to the reactor vessel. [12]

Both thermal- and fast-spectrum MSRs can be configured as integral units in which the heat exchange from fuel salt to primary coolant salt is included within the reactor vessel, or they can be configured as loops in which the heat exchanger is in a separate shell connected to the reactor vessel by piping [12]. Basic diagrams of an integral-type MSR and a loop-type MSR are shown in Figure 3 [14]. The core area is a simple representation of the core material. Most designers have opted for an integral design or a loop design with extremely short connecting pipes because those designs require less fuel salt [12].

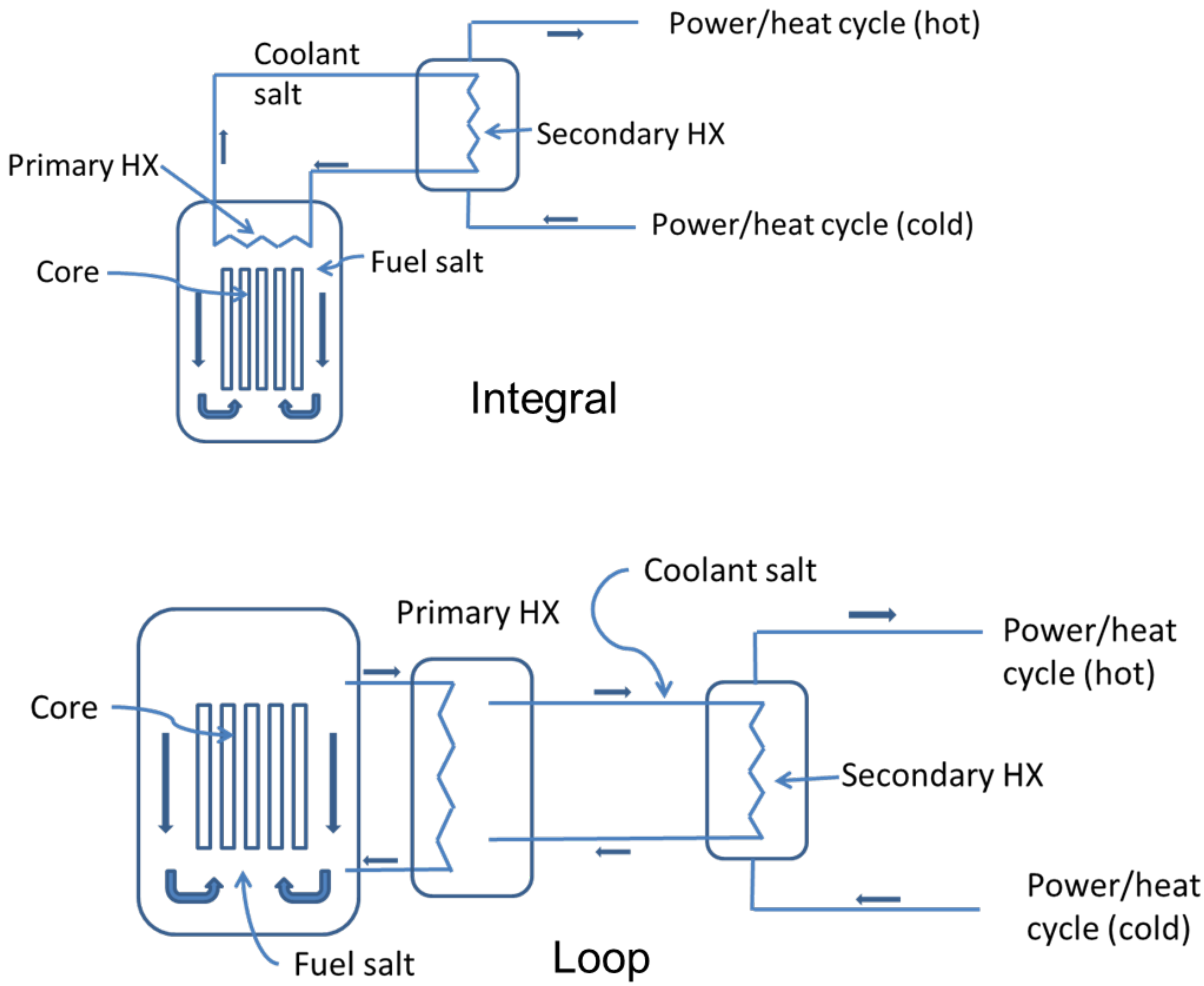

Figure 3. Schematic examples of an integral MSR and a loop MSR. [14].

2.1.1 Fuel Preparation

Fuel can be prepared at an offsite facility and then transported to the site for use by the MSR, or fuel can be prepared at the MSR site or at a centralized site among several MSRs where the owner/operator controls the land. A fuel storage facility at the MSR will be necessary.

2.1.1.1 Fuel Preparation Off-site and Transportation

Off-site fuel preparation would occur at a facility separate from the MSR site. Transportation between the fuel preparation site and the MSR site would be governed by 10 CFR 71, Packaging and Transportation of Radioactive Material. The regulations found in 10 CFR 71 note that licensed material is subject to NRC jurisdiction as well as the regulations associated with other agencies such as the US Department of Transportation [15].

2.1.1.2 Fuel Preparation On-site

Some MSR technologies plan to recycle used fuel at the same site as the MSR. This process could involve either using fuel from a sister reactor or processing spent fuel from a large LWR. A collocated MSR fuel processing facility would prepare and transfer MSR fuel for storage at an individual MSR unit.

2.1.1.3 Storage

One advantage of MSRs is that the fuel salt's lifetime is not determined by radiation damage. As such, MSRs are not limited by the life of a heterogeneous fuel element and associated control system into which excess reactivity is designed to account for fuel burnup. MSRs can be operated with minimal excess reactivity by adding fuel continuously or in batch mode during operation. This process requires a fuel storage system to support reactor operations.

The fuel storage system operates to support storage, fuel addition, and fuel removal to maintain the reactivity within operational bounds by adding or removing fuel salt. The frequency of fuel salt addition/removal is determined by the amount of reactivity adjustment available via the control system. Daily additions/removals are anticipated to be typical [12].

The amount of fuel storage required is determined by a combination of the reactor power rating, the recent MSR power generation history, and the observed reactivity. Fuel will be added (MSR in burner configuration) or removed (MSR in breeder configuration) to maintain the desired power level at the specified operating temperature [12].

2.1.2 MSR Heat Removal

A fast or thermal MSR primary system consists of the core, heat exchangers that transfer heat from the fuel salt to the coolant salt, and pumps that circulate the fuel–salt mixture. In most designs, all the equipment is contained within the reactor vessel in an integral scheme [16]. Some designs may employ a loop scheme (both are shown in Figure 3). The MSR primary system is not coolant solid. A surface layer for the fuel salt is below a cover gas layer and an off-gas system or cover gas handling system. The functional block diagram in Figure 2 reflects this primary system interface with the off-gas system and the possible interface with a cleanup system. Some MSR technologies may also employ a fuel salt drain-tank system. Fuel salt or FPs are present in all these interfacing systems, and each system will require a cooling system.

2.1.2.1 Reactor Decay Heat Removal

The inherent characteristics of MSRs (e.g., quantity of heat being produced, thermal storage potential within containment, practical temperature limits of engineering materials) dictate how decay heat can be passively transferred from the FPs to the environment [17]. Decay heat can be removed indirectly from an MSR by passively transferring heat by radiation and convection from the exterior surface of the reactor vessel to a natural draft-driven cooling system, which is typically referred to as a *reactor vessel auxiliary cooling system* (RVACS). Alternatively, decay heat can be removed directly from an MSR by inserting a natural circulation-driven decay heat removal loop into the fuel salt. This system is typically referred to as a *direct reactor auxiliary cooling system* (DRACS) [12]. The MSR fuel salt circuit can also be immersed into a large pool of coolant salt to remove decay heat. This type of cooling configuration is referred to as a *pool reactor vessel auxiliary cooling system* (PRACS).

As with any fission reactor, immediately following shutdown after extended operation, the FPs produce ~7% of full power. For the MSRs that rapidly and proactively remove and isolate fission gases, roughly

40% of the decay heat could be generated in the cover-gas handling and storage systems. This process reduces the cooling needed in the reactor core at shutdown but adds locations that require cooling to ensure safe plant operations [17].

Practical heat rejection to air in an RVACS or DRACS generally employs a structural chimney to drive natural air circulation past an air draft heat exchanger. Even a PRACS system eventually must reject decay heat to the environment, although the pool for a PRACS would provide additional coping time owing to the large thermal mass. Whereas a substantial portion of an MSR structure may be located below grade, the cooling chimneys must extend above grade to provide adequate heat removal. Severe natural phenomena such as tornadoes, floods, earthquakes, and airborne missiles can damage above-ground structures, systems, and components (SSCs). Therefore, MSR cooling chimneys must be mechanically robust to withstand such phenomena. Additionally, large civilian aircraft impacts, especially engine impacts, could damage or destroy even a reasonably robust chimney or cooling tower [17].

2.1.2.2 Off-Gas or Cover-Gas Decay Heat Removal

An MSR off-gas system or cover-gas system will couple to the headspace above the fuel salt. The headspace will contain volatile FPs, activation products, particulates, and mists generated from the movement of the salt through the primary heat transport system. As was done for the MSRE, the fuel salt may also be sparged with an inert gas, such as helium, to liberate entrained FPs or poisons from the fuel salt and transport them to the headspace region. The gases will be highly radioactive and will constitute a substantial heat load (a few tens of megawatts for a large reactor) [18].

Roughly 40% of FPs have a gaseous phase in their decay chains. Noble gases are not highly soluble in fuel salts, so much of the FP burden will accumulate in an MSR's cover gas. This cover gas can be kept mostly within the reactor vessel, allowing only the long-lived fission gases to escape, or the cover gas can be removed into an off-gas system to prevent daughter FPs from accumulating in the fuel salt. Activated carbon beds or hydroxide-based scrubbers have been proposed to prevent the fission gases from entering the off-gas system [12].

The off-gas total decay heat load for the MSBR design (a large MSR) was estimated to be 21.3 MWt, including entrained aerosols and mists. Almost 80% of the gaseous heat load is generated in the first hour following release from the fuel salt and significantly less than 1% of decay heat remains in the gas stream after 2 days [19]. As a result, the upstream portions of the cover-gas system could have a significant thermal load during normal operations. Therefore, an active cooling system will need to provide cooling during normal operations, and a passive decay heat removal system will be needed in the event of a loss of forced cooling [17]. Some MSR designs elect to keep the fission gases within the reactor vessel for the first couple of days to avoid needing a separate, safety-related cooling system.

2.1.2.3 Cleanup System Decay Heat Removal

The fuel salt cleanup system or polishing system consists largely of a high surface area mechanical filter. This filter promotes deposition of suspended, undissolved fission and/or corrosion products. The filter is likely to be made from porous metal such as Hastelloy and will likely be placed in a recirculating side stream from the reactor core [11]. The accumulated FPs and other materials will generate decay heat and will require a passive heat removal system. Floatation-based solid separation systems have also been proposed. In this case, the solids would accumulate in a scum layer that would be decanted from the bulk of the fuel salt in a side stream.

2.1.2.4 Drain Tank Decay Heat Removal

As was done for the MSRE, some designs include a fuel salt drain tank as a safety and/or a maintenance system. The fuel salt drain tank can be connected to the bottom of the reactor vessel via a drain line with a freeze plug to serve as a safe storage volume for the fuel salt when it is drained from the circulation loop. Alternatively, MSRs can employ a lower goose-neck type connection and a gas accumulator to blow the salt from the core. A goose-neck and accumulator type fuel volume control system has previously been employed for aqueous fuel reactors. Passive cooling is required to remove decay heat from the fuel salt, and heater equipment is available for longer term storage to maintain the fuel salt above its liquidus temperature.

2.1.3 Reactivity Control

Neutron-absorbing control rods are the typical means for controlling reactivity in an LWR, but there are many more ways to control reactivity in an MSR. MSR designs can control reactivity by using fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. As a result, MSRs typically use the term *control elements* in place of the term *control rods*. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned. References to *fully inserted* or *fully withdrawn* elements must be carefully considered regarding what condition relates to the most or the least reactivity state [20].

In designs that incorporate a fuel salt drain tank, such as the MSRE, the fuel salt drain tank is hydraulically connected to the reactor core and serves as a safe storage volume for the fuel salt when it is removed from the circulation loop. A critical mass cannot exist in the storage tank owing to geometry (increased leakage) and insufficient neutron moderation.

Freeze plugs act as the isolation mechanism and would thaw in the event of a major loss of electric power or failure of the plug cooling system because electrical power is required to maintain the freeze. The drain system can also be designed to provide leak protection for the fuel salt circulating loop by providing a storage location for fuel salt leakage into any guard system [16].

The design of the drain tank system will allow the system to be initiated by the reactor protection system in response to a transient or accident condition. However, system drainage is not normally used as an emergency procedure because numerous other reactor control and safety mechanisms can quickly take the reactor subcritical while fuel-salt circulation continues to remove FP decay heat via the primary heat exchangers or other dedicated decay heat removal heat exchangers [16]. Nevertheless, a fuel salt removal system can provide shutdown defense-in-depth.

2.1.4 Radionuclide Retention

The functional block diagram in Figure 2 shows the numerous potential primary system interfaces. Because of the radioactive content of all these interfacing systems, they will all be enclosed in a functional (layered) containment structure located behind bio-shields. Many areas will be difficult to access directly by site personnel after the reactor has operated for any amount of time. This section discusses some of the means available for radionuclide retention in MSRs.

2.1.4.1 Fuel Salt

In MSRs, gaseous, soluble, and insoluble FPs accumulate in the fuel salt as the MSR is operated. Gaseous FPs, such as xenon and krypton, bubble off continuously and are typically removed from the cover-gas space by a gas management system without significantly disrupting reactor operation. Soluble and insoluble FPs remain in the fuel salt. Therefore, the fuel salt in the MSR becomes highly radioactive. Insoluble FPs tend to plate out on reactor surfaces. If necessary to support continued reactor operation, soluble FPs can be removed from the fuel salt by chemical processing, polishing, or filtration, either by batch operations or continuously in a side stream. In some MSR designs, the filtering system or chemical processing loop can also be used to add additional fuel to the fuel salt. Because soluble FPs remain in the fuel salt, the fuel salt acts as a limited physical barrier. The purity of the fuel salt is monitored and maintained as high as reasonably possible to limit the chemical corrosion of the fuel system boundary [20].

2.1.4.2 Containment

The fuel salt's distributed nature supports segmenting the containment into various functional cells containing the equipment needed for operations within each functional area, similar to the segmentation found in hot-cell facilities. This configuration lowers the risk of releasing radionuclides from a failure in one containment cell to another containment cell [17].

Although the fuel salt provides a limited physical barrier, the fuel system boundary (e.g., vessel, cover-gas boundary, and associated piping) forms the first nominal leak-tight containment layer (essentially equivalent to the fuel clad in an LWR). The second containment layer, more typically referred to as *reactor containment*, is the first containment layer not directly contacting or wetted by the fuel salt or cover gas (outside the fuel system boundary). This layer may include guard vessels, guard pipes, and individual reactor cells [17].

The second containment layer is assumed to be contained within a third barrier layer, typically thought of as a *reactor building*. As noted, multiple system cells within the reactor building may limit the risk of widespread radionuclide release. The requirement to withstand large civilian aircraft impact (10 CFR 150) may provide an economic incentive to locate low-pressure containments below grade. Below-grade system cells will be protected beneath significant bio-shields to protect workers, sensitive equipment, and the environment. The shielding mass can also serve to absorb and/or deflect impacts from above [17]. Moreover, underground structures exhibit excellent earthquake performance because they are constrained and supported by the surrounding medium and generally do not move independently of the soil or rock medium nor are they subjected to vibration amplification. The interface between the reactor containment layer and the reactor building layer determines the functional requirements for containment during accident scenarios [17]. The reactor building layer is more likely to contain typical building services, thereby contributing the most complex system interactions.

Molten salts are readily contaminated by air exposure, but they do not have any energetic chemical reactions with air or water. Consequently, the reactor building environment, or the environment in the individual functional cells, may be inerted to minimize the potential for contamination during any maintenance activities in which the salt boundary is broken. The containment floor will almost certainly be made of stainless steel and a thermal barrier to prevent hot salt from directly interacting with concrete in the event of a severe accident [11].

2.1.4.3 Off-Gas System

Following fission of the different fuel types, a variety of radioactive gaseous FPs or nonradioactive gases/vapors mixed with radionuclides could be released, including particulates, aerosols, reactive gases, hydrogen (e.g., tritium), water, nitrogen, oxygen, and noble gases (e.g., xenon, krypton). These radionuclides will be generated continuously during reactor operation, and releases must be maintained below regulatory limits [21].

The mass load of fuel salt constituents in the off-gas system will be determined by the thermochemical and thermophysical properties of the salt and the transport of fluids through the reactor. The long-lived noble gas FPs must be removed from the fuel salt to avoid pressurization of the fuel–salt boundary. Shorter-lived gaseous FPs may also be removed from the fuel salt as a matter of convenience [12]. Operational experience with MSRs is limited to the MSRE, which provided substantial information to support the conceptual design for the larger MSBR. In the MSBR design, the off-gas system was to rely on charcoal beds, in which radionuclides can decay for 90 days, leaving ^{85}Kr as the most important contributor to the source term [18].

Because some FPs tend to plate out on system surfaces, there is also a potential for plug formation in the off-gas system. Cover-gas piping can accumulate pressure after plug formation from deposited material, progressively increasing pressure within the reactor vessel. This issue affected the MSRE. The potential for material accumulation in the cover gas lines suggests that some capability must exist to detect the build-up and correct it as part of normal operations, perhaps chemically or mechanically [17].

2.1.4.4 Online Cleanup System

The need for an online cleanup system or polishing system will vary with MSR system design. Most thermal MSR designs will require removal of FP poisons and periodic fuel additions to maintain criticality. A single-fluid, thermal-spectrum MSR implementing a thorium-uranium breed and burn fuel cycle will require substantial fuel salt processing to achieve breeding gain. Fast-spectrum breed and burn MSR designs operating a uranium-plutonium fuel cycle may require little (if any) fuel salt processing beyond timely redox adjustment to compensate for fission being an oxidative process. FP decay back toward stability is likely to be a reductive process, so redox control will be time dependent [17]. Polishing systems can be run in either continuous or batch-style processing routes in which FPs can be removed during operation or at the reactor's end of life (or end of batch) [21].

As with other major systems, any required fuel salt processing will likely be performed in containment cells separated from the reactor core to minimize the potential for disruptions in one system to affect another [17]. If needed, the fuel salt polishing system may consist of a high surface area mechanical filter, likely a nickel mesh, to promote deposition of suspended, undissolved fission and/or corrosion products. Alternatively, a floatation-based suspended-particle separation system may be implemented to remove insoluble materials from the fuel salt without requiring a mechanical filter [12]. The filter will require periodic changing, which will involve pulling the filter out of the salt, allowing any remaining salt inside the filter to drain, installing a new filter, and then packaging the old filter for disposal. Like the filters that may be used in the off-gas system, routine maintenance and exchange of particle filters while the reactor is operating may be particularly problematic. The dose rate on the old filter will be very high because of the insoluble FPs and any residual fuel salt remaining on the filter [11]. If necessary, the fuel addition system will likely consist of a fuel salt melting pot connected to the polishing system piping via a valve [12].

2.1.4.5 Leaks and Leak Detection

MSRs operate at pressures just above ambient because the vapor pressures of the salts are very low at normal operating conditions. Therefore, a small breach in the salt-wetted boundary will likely allow the fuel salt to ooze through the crack and ultimately solidify rather than disperse into a pressurized spray of aerosolized particles and gases such as in an LWR severe accident [11]. If the MSR primary system is breached, then the consequences depend on the location of the breach, the size of the breach, mitigation measures, and how much of the fuel salt or fission gases leak into a confined space [22].

Many MSR designs intend to use a guard vessel and/or guard pipes to limit the likelihood of fuel salt leakage into the reactor building. Employing a system to measure the fuel salt level will indicate significant fuel salt out-leakage or coolant salt in-leakage. Detection of salt within the guard vessel would indicate that the salt-wetted boundary has leaked. Gaseous radionuclides outside the fuel system boundary also indicate a fuel system leak. Detecting the presence of radionuclides where they are not intended—beyond salt boundaries—is a major safety goal for MSRs [12]. It is possible for leakage into the guard system to be directed to a drain tank where drain tank level can be monitored.

In one type of salt spill that could occur from a breach in the primary system boundary in a reactor vessel vapor region, liquid fuel salt is not spilled from the primary system. Instead, radioactive material evolves from the salt and continuously flows in a gas/vapor phase out of the primary system into a guard vessel or reactor cell. The continuous nature of the release at this location causes reasonable potential for a release outside of containment [22].

A second type of salt spill could occur from a breach in the primary system (or off-gas system as an extension of the primary system boundary) into a reactor cell. In this scenario, a low-energy spray could form at the exit point from the primary system, or splash droplets could form as the salt contacts the cell floor. Potential interaction with the floor includes a distortion of the steel lining the reactor cell or a chemical interaction with the steel. The horizontal spread of the leaked salt pool will depend on the leak geometry, the cell geometry, and the physical properties of the salt, such as viscosity, thermal conductivity, and freezing point. Radioactive material within the salt pool could vaporize from the pool surface into the cell atmosphere and eventually leak to a subsequent containment layer, depending on the form of the radioactive material and the driving forces for transporting the material. Subsequent freezing of the salt at the upper surface of the leak pool would effectively prevent further release of radioactive material to the cell atmosphere [22].

Water near the primary system is avoided in MSR designs because an event in which high-temperature molten salt contacts a pool of water creates the potential for an energetic steam explosion. An explosion would create the conditions for an energetic release of radioactive material, conceding the MSR advantage of low-pressure operation.

2.1.4.6 Maintenance

After reactor operation, radiation levels near any system containing fuel salt will be much too high for staff access, even after shutting down the reactor and draining the fuel salt. Access inside primary containment will not be possible, and access to certain areas within the reactor building may be extremely limited, depending on the amount of shielding available and any shutdown operations in progress. Therefore, many maintenance activities will require remote handling using cameras and long-handled tools or electronics that are highly tolerant of radiation with local shielding [12]. Some designs require significant core component reconfigurations/replacements to establish an economically viable plant life. These designs will also require consideration for remote component replacements and storage of activated components.

2.1.5 Considerations for the Back End of the MSR Fuel Cycle

The back end of the fuel cycle for an LWR includes the following:

1. Spent fuel storage,
2. Streams for operating wastes, and
3. Streams from decommissioning and decontamination (D&D).

By comparison, the typical MSR backend waste materials will be more complex. Typical generic MSR waste streams include the following [21]:

1. Used fuel,
2. Unseparated salt-based waste streams,
3. Separated salt-based waste streams,
4. Carbon-based waste streams,
5. Metal-based waste streams,
6. Streams for operating wastes, and
7. Streams from D&D.

2.1.5.1 Used Fuel

The concepts of used fuel and spent fuel are different for a salt fueled MSR. By design, solid fuel accumulates FPs and radiation damage. It also ages with time, eventually requiring replacement. Therefore, solid fuel is considered spent after a period of operation. Liquid fuel does not accumulate radiation damage.

To the extent that its chemical composition can be maintained, liquid fuel will not age or need to be replaced.[3] MSR fuel salts inherently contain the FPs that do not escape during operations. They also contain accumulated amounts of corrosion and environmental contamination products. For MSR designs that do not implement FP removal, the fuel salt will eventually reach an equilibrium level of FPs as FPs burn out, transmuting into a gaseous or insoluble form, and build in. For designs that provide for reconditioning the fuel salt and those with equilibrium FP contents that result in acceptable thermophysical properties, the fuel salt would be usable in future generations of MSRs [12]. In this case, MSR fuel can be referred to as used but not spent. The liquid fuel would not become waste until MSRs cease to be an operating reactor class.

Therefore, MSR fuel salt can be reused indefinitely provided it has adequate fissile content. Consequently, MSRs do not necessarily have an equivalent to semi-permanent spent fuel storage [17]. Nevertheless, MSR fuel salt must be stored between uses. Processing and storage of the used fuel may be performed nearby the reactor site at a facility designed for this purpose. Likewise, processing may be performed off-site, requiring temporary storage of the fuel salt to allow for radionuclide decay before transportation to the off-site processing facility. Additionally, the radionuclides that separate from the liquid fuel salt (gases, vapors, and insoluble materials) require containment, and used fuel salt will require cooling while in storage.

2.1.5.2 Salt-Based Waste Streams

If the used fuel is no longer required for reuse in another MSR, then constituents in the fuel salt may be separated and recycled. An important reason to remove the halide from the salt is to impart radiolytic and chemical stability to the waste form [11]. Because of its solubility in water, the MSR salt is not a good medium for long-term immobilization. Therefore, conversion of unseparated salt streams to another form is necessary. Depending on the salt type, a variety of options for immobilizing unseparated salt streams are available. Options include glass, ceramic, glass-ceramic, glass-bonded ceramic, and ceramic-metallic forms [3, 11].

Another option is to separate the salt-based waste stream because a market may exist for many of the constituents. For example, other industries may be interested in the recovery of FP metals from fluoride

[3] Additional fuel may be added over time as the MSR is operated to account for fuel burnup.

salts. Salt separation processes can also be used to remove the volatile FPs and recover/recycle expensive carrier salts. Actinides can also be recovered either for reuse in a reactor or for segregated disposal [3]. Any remaining materials can be immobilized in glass, metal, or ceramic form [21].

2.1.5.3 Carbon-Based Waste Streams

Graphite will be a major waste form in the case of thermal-spectrum MSRs, and it may be the limiting factor of the reactor lifetime [11]. Graphite damage is introduced by neutron irradiation in the structure of graphite core components. Dimensional changes tend to occur over time with irradiation. The rate of change depends on the carbon grade and the manufacturing process [21].

The graphite will be contaminated with FPs captured by recoil and fluoride from the salt itself. The graphite may also have a significant loading of tritium from activation of lithium in the core. Recovery of graphite by removing the tritium has been proposed and would greatly reduce the volume of the remaining waste [11]. Another option is to decrease the waste volume by hot-pressing the carbon-based waste for disposal [21].

2.1.5.4 Metal-Based Waste Streams

MSR infrastructure metals can be an intermittent waste as components are replaced or an end-of-life waste as the reactor is decommissioned. Typical metals suggested for MSR construction include Hastelloy-N, Alloy 800H, MONICR, Inconel 600, and 316L stainless steel [21]. Depending on the economics, one option is decontamination and recycling. The more likely metal disposal option includes decontamination along with melting or compaction into a metal waste form for disposed as low-level waste (LLW) or greater than Class C (GTCC) waste.

2.1.5.5 Operations-Based Waste Streams

Operations-based MSR waste streams are similar to LWR operations-based waste streams but may be somewhat more radioactive. Wastes include failed equipment, materials cans, job-control wastes (e.g., personal protective equipment), facility filters (e.g., HEPA), water cleanup wastes (e.g., ion-exchange resins), glove-box gloves and manipulator boots, and laboratory samples. These wastes can likely be reduced in size, decontaminated, packaged, and disposed of as LLW or GTCC wastes [21].

2.1.5.6 Decommissioning and Decontamination

The proposed Yucca Mountain repository application for construction authorization[4] restricted waste to nonhazardous waste in accordance with the Resource Conservation and Recovery Act (RCRA). Untreated salt waste and salt-contaminated wastes from MSRs are likely to be characterized as GTCC waste (hazardous) for toxicity (e.g., beryllium) and/or reactivity (e.g., lithium-containing salts). For reference, MSRE fuel salts have been declared hazardous mixed waste under RCRA [21]. Salt and salt-contaminated wastes may be treated to remove the toxicity and reactivity hazards associated with the salt, rendering the waste nonhazardous. Regulations require that GTCC waste must be held on-site to be disposed of in a geologic repository when available, except when allowed by the NRC on a case-by-case basis. [23, 24]

[4] DOE/RW-0573, Update No. 1, NRC Docket No. 63–001, 2008

2.1.6 Support Systems

The current US vendors have closely held significant design features for their respective designs. Therefore, detailed information on support systems is not available. However, considerable information for the proposed MSBR is available, as is general information on proposed support systems for advanced passive reactors. The following discussion is based on this information, and it will vary by design.

2.1.6.1 Intermediate Loop

An intermediate circulating coolant salt is typically used to transport the heat generated in the primary system to the steam-power system rather directly coupling the primary fuel salt loop to the PCS [16]. Other fluids, such as liquid metal or gas, can also be used in the intermediate loop, but this configuration is not typical. A fluid with characteristics like the salt used in the primary is desirable in the event of a primary heat exchanger leak.

The intermediate loop serves the following purposes [16]:

- Provides an additional barrier for containing the FPs in the fuel salt in the event of a primary heat exchanger tube failure,
- Reduces the possibility of freezing the fuel salt in the primary heat exchanger because of low feedwater temperatures in the PCS,
- Isolates the high-pressure steam in the PCS from the primary (fuel salt) system, and
- Reduces the potential for water entering the primary system, which could cause oxidation and precipitation of uranium and thorium (MSBR design).

The MSBR coolant-salt circulation system consisted of four independent loops, each containing a salt circulation pump, steam generators, steam reheaters, coolant-salt piping, and the shell side of one primary heat exchanger. Multiple loops were desirable to improve the coolant flow's reliability [16].

Steam Generator

The four MSBR U-tube steam generators were to be operated in parallel with respect to both the coolant-salt and steam flows. The feedwater supplied to the steam generators was to be preheated to 700°F and at a pressure of about 37S0 psia in the inlet region of the unit [16].

Superheater

In the MSBR, the steam reheater (superheater) associated with each steam generator was planned as a horizontal, counterflow, single-pass shell-and-tube exchanger with disk and doughnut baffles, which transfer heat from the coolant salt in the shell side to steam in the tubes [25].

Rupture Disks

Each of the four MSBR salt coolant loops (intermediate loops) were to be provided with rupture disks to prevent system overpressurization should a steam generator tube leak occur [16]. The rupture disks would help protect the primary heat exchangers from damage caused by overpressure.

Coolant Salt Drain System

A series of four tanks was proposed for the MSBR design to allow maintenance on the salt coolant loop (intermediate loop). A series arrangement was adopted to facilitate heat removal if the salt coolant became contaminated with fuel salt. The tanks are in a cell directly beneath the steam generator cells. This cell is heated to about 800°F by electric resistance heaters to maintain the salt above its liquidus temperature [16].

2.1.6.2 Power Conversion (MSBR Superheated Steam)

The thermal energy generated by the MSBR was to be converted to electric power in a steam cycle employing once-through steam-generator superheaters, a turbine generator, and a regenerative feedwater heating system [16]. Terrestrial Energy, Exodys Energy, and ThorCon International also plan to use steam generators providing superheated steam to a turbine generator [10]. Flibe Energy has proposed using a supercritical CO_2 gas turbine [16] to provide electricity. Other power conversion systems are possible, such as storing the reactor heat in molten salt storage tanks to provide steam to on-demand turbine generators (Moltex Energy in Canada has proposed this option [16]). Designers may also opt to provide process heat directly to nearby industry.

2.1.6.3 Ultimate Heat Sink

After reactor shutdown, under normal operating conditions, decay heat is removed from the fuel salt via the steam system. The steam can be dumped to a condenser and removed as normal until steam can no longer be generated. At that point, the diminishing decay heat is dumped to the atmosphere or to a controlled-volume surface pond to control overcooling the fuel salt via the RVACS, DRACS, or PRACS, as discussed in Section 2.1.2.1. During a loss of forced circulation, decay heat can immediately be removed via the RVACS, DRACS, or PRACS. In this case, there will be a higher heat load on these heat removal systems and the surrounding/supporting SSC.

2.1.6.4 Diesel Generator (Not Safety-Related)

Most passive reactor designs continue to include a support diesel generator (DG). The support DG is not required to start and load in a short period of time to support safety-related SSC. Because the support DG does not power safety-related SSC, it does not require weekly testing that often leads to maintenance and related safety-related SSC outages. However, the support DG provides defense-in-depth by supplying backup power to such systems as lighting, instrumentation, and batteries.

2.1.6.5 Battery System

In the event of a loss of off-site power and a failure of the standby DG(s), battery systems will be used to supply backup power to such systems as lighting, instrumentation, and batteries. In the MSBR, batteries were also planned to supply power to pony motors on the salt cooling pumps (intermediate loop) to provide for forced circulation for decay heat removal to continue for an unspecified interval of time [25]. This system was intended to ease the heat load on the MSBR RVACS. It is unknown whether current MSR vendors are planning to use pony motors for decay heat removal defense-in-depth.

2.1.6.6 HVAC

The MSBR reactor building was a high bay over the various below-grade system cells. The high bay was serviced by an HVAC system and filtration [25]. Likewise, an HVAC system is required for the reactor control room to protect the operators against radioactive materials, fire, and toxic gas. MSRE also had a separate remote maintenance control room, and this space would also require an HVAC system. These systems would require periodic filter medium monitoring and replacement as well as damper testing.

2.1.6.7 Pipe Heat

If the fuel salt or coolant salt is overcooled, then a phase change will occur in the salt. The salt will become solid. Electric pipe and vessel heating systems may be necessary to maintain each salt system above its liquidus temperature to avoid possible system damage caused by volume change.

2.1.7 Typical Plant Footprint

The typical layout for most proposed MSR designs is a single high-bay reactor building over below-grade cells containing the reactor, intermediate loops, off-gas system, fuel salt conditioning, and other systems. This configuration is typified in Figure 4 for the proposed MSBR. An additional structure for the PCS would be necessary. Other separate or combined structures may include support DG(s), breaker rooms, waste handling, used fuel handling and processing, gas stack, control room, administrative, training, security, and a warehouse. The overall footprint would directly correlate with plant thermal energy.

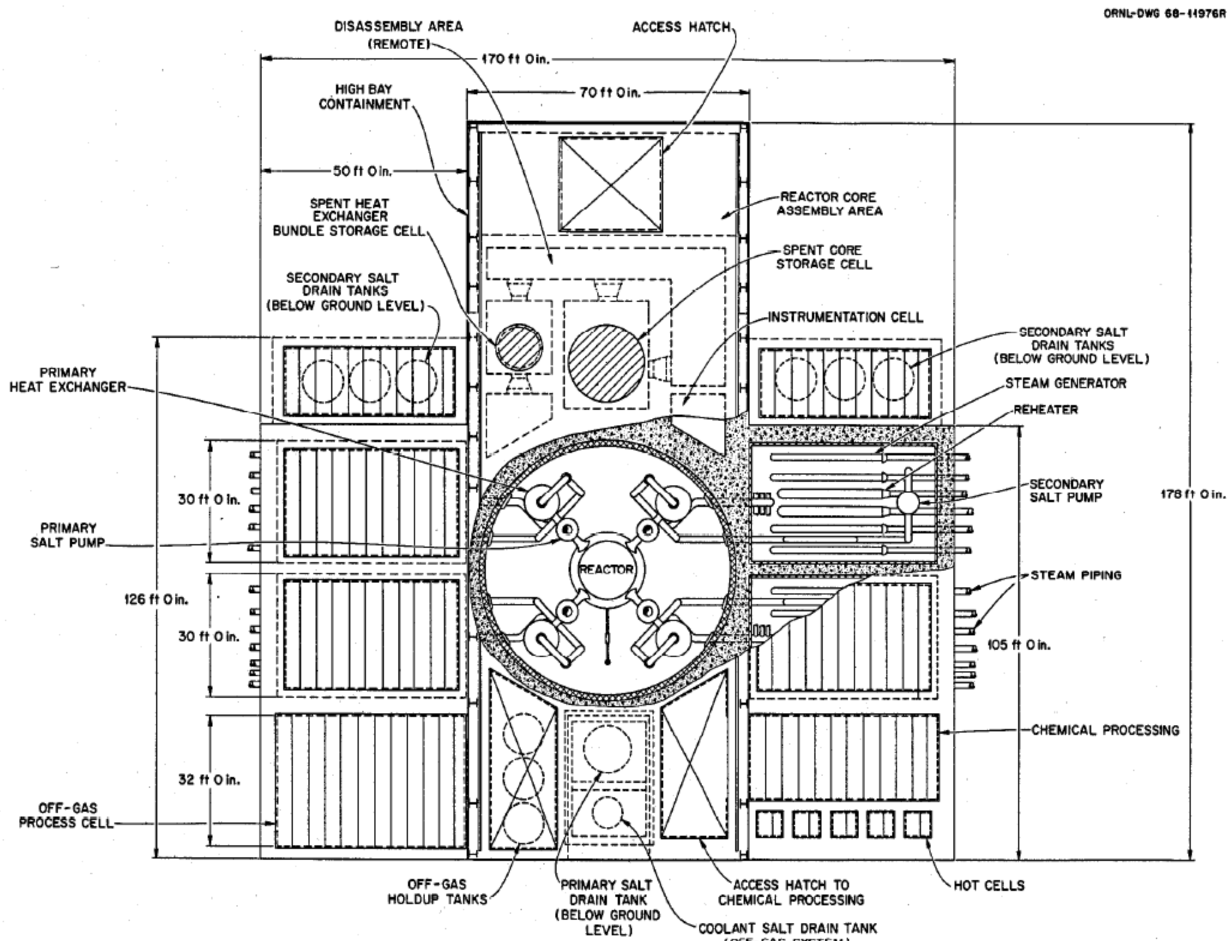

Figure 4. MSBR reactor building. [16]

2.2 MAJOR DIFFERENCES BETWEEN MSRs AND LWRs

Molten salts are an attractive medium for heat transfer as compared to other heat transfer mediums, such as water, liquid metal, or gas [9]. Fluoride and chloride salts have been demonstrated to be very stable thermodynamically with no radiolytic decomposition in the liquid phase. In addition, molten salts are chemically inert—there are no chemical reactions with air or water. The typical salts are compatible with nickel-based structural alloys and graphite.

Most importantly, with respect to using water as a coolant, molten salts have a very large heat capacity and very high boiling points with a low vapor pressure at operating temperatures. This translates to low operating pressures with less driving force for radionuclides in the event of an accident, reducing the source term compared to a similar sized LWR. In addition, many fission products are soluble in the fuel salt and would be retained in the event of an accident, further reducing the potential source term compared to an LWR.

In summary, there are a number of key aspects that differentiate an MSR design from traditional LWR designs [9]:

1. High MSR operating temperatures provide the opportunity for more efficient electricity production. The higher temperatures also better support the high temperatures requirements associated with many industrial uses, predominantly with chemical processes.
2. Low MSR operating pressures lead to less stringent containment requirements. Functional containments will be employed for MSRs versus leak-tight containments for LWRs.
3. The fuel salt materials that contact the MSR boundary materials are subject to different stressors, including fluence, corrosion, and temperature (instead of pressure for LWRs). For MSRs, salt chemistry control, structural alloy cladding, internal shielding, and material replacement are employed to mitigate impact of stressors.
4. MSRs operate with small amounts of excess reactivity, whereas LWRs must include significant amounts of excess reactivity for adequate fuel cycle lengths. MSRs can add fuel online to maintain criticality as needed or in the case of breeder reactors, they can generate fuel. Therefore, the MSR fuel cycle length is determined by material conditions and not by the expenditure of available excess reactivity.
5. Safeguards and proliferation resistance will be conceptually different. Fuel assemblies in LWRs are relatively easy to track as distinct items as opposed to actinides in solution in fuel salts.
6. MSR operations and maintenance need to accommodate a much more severe radiation environment. Actinides and fission products are dispersed throughout all systems containing fuel salt. These materials are contained within the fuel assemblies associated with LWRs. The development of remote maintenance techniques will likely be necessary.
7. Molten salts provide a strong negative reactivity coefficient of reactivity. In the case of an accident that causes system temperatures to rise, the MSR will tend to shut down.
8. Typically analyzed LWR accidents are different for an MSR. Large LWRs analyze loss-of-coolant-accidents (LOCAs) as the dominate, bounding accident sequence. Much of the safety equipment in an LWR is for protection against LOCAs. LOCAs will not be a dominate accident sequence for an MSR and will not require extensive safety systems. Passive heat removal systems will be important for MSRs.

3. SELECTION AND REVIEW OF STANDARDS

A structured review process was developed to guide the reviews from an MSR perspective.

3.1 STANDARDS SELECTED FOR REVIEW

At the start of this study, it was not known how many existing standards would be applicable in designing and licensing an MSR, what changes would be needed so that the scope of the standard address those issues related to MSRs, and whether new standards would be required to address new technological issues introduced by MSR technology. The number and significance of any changes that may be required to revise, develop, approve, and endorse a new or revised standard for adequacy and completeness to an advanced reactor must also be estimated. Added to the time to develop or revise a standard is the amount of time for the NRC to endorse a new or revised standard.

This review focused on the adequacy and completeness of the standards to an MSR. The identification of standards for review consisted of RGs endorsed from not only Division 1 but also from Division 3 (Fuels and Materials Facilities) and Division 5 (Materials and Plant Protection). The list of standards for review was then further increased to evaluate those approved for use in NUREG-0800 [8] (the SRP) and required by the CFR. From this list, a down-selection process was used. This process focused on standards being endorsed by an active RG (i.e., not withdrawn), approved for use in the SRP, or required by the CFR. This focus served three purposes:

1. To narrow down the number of standards for review to endorsed, approved, or required standards to discern how many would require review;
2. To categorize the LOE[5] required to develop or revise each standard for adequacy and completeness to an MSR to determine the LOE needed; and
3. To prioritize the standards needing modifications for adequacy and completeness to MSRs.

The first step in estimating the size and scope of the effort was to obtain a list of all standards cited in RGs. This list was obtained by performing a query on the NRC's internal standards database.[6] The database has a total of 865 standard-to-RG cross reference citations in 486 RGs for Division 1. The revision to existing RGs and the publication of new Division 1 RGs added 19 endorsed standards to the review, for a total of 113 standards. The expansion to Division 3 and Division 5 added 41 and 28 endorsed standards to the review, respectively. The number of citations exceeds the number of RGs and standards because RGs may endorse numerous standards, and several RGs may endorse the same standard.

From all sources, 197 unique standards were designated for review. Many standards had several sources. For example, ASME NQA-1-2008 is endorsed by RGs 1.28-5, 1.8-3, 1.97, 3.48-1, and 3.75-0, approved for use by SRP Chapters SRP Ch 4.5.1, SRP BTP 7-14, and SRP 17.5, and required by 10 CFR 50.55a. This standard however, accounts for only one unique standard. However, this process is not a review of the RGs; it identifies the standards that are endorsed by an RG.

The second step was to narrow down the number of standards from the list of more than 865 citations to focus on those to be considered for in-depth review. The down-select process limited the review to standards endorsed, partially endorsed, or endorsed with exceptions by active RGs (i.e., the RG has not been withdrawn) in Division 1 (Power Reactors), Division 3 (Fuels and Materials Facilities), and Division 5 (Plant Protection). This step identified 182 standards (Figure 5).

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[5] "Level of effort" represents the number of changes that might be required and not the amount of resources and time to make those changes.
[6] Database distributed by NRC at the Nuclear Energy Standards Coordinating Collaborative (NESCC) circa 2012 (unpublished).

The next step was to identify the number of standard citations in the NUREG-0800 (the Standard Review Plan). The SRP has 3,364 citations to standards, reports, and articles. Narrowing the selection to standards from SDOs that are approved for use identified eight unique standards in addition to those endorsed by RGs.

Finally, a review was performed to identify those standards required by the CFR. This step identified an additional seven standards for review.

In total, 197 standards were selected for review (Appendix A).

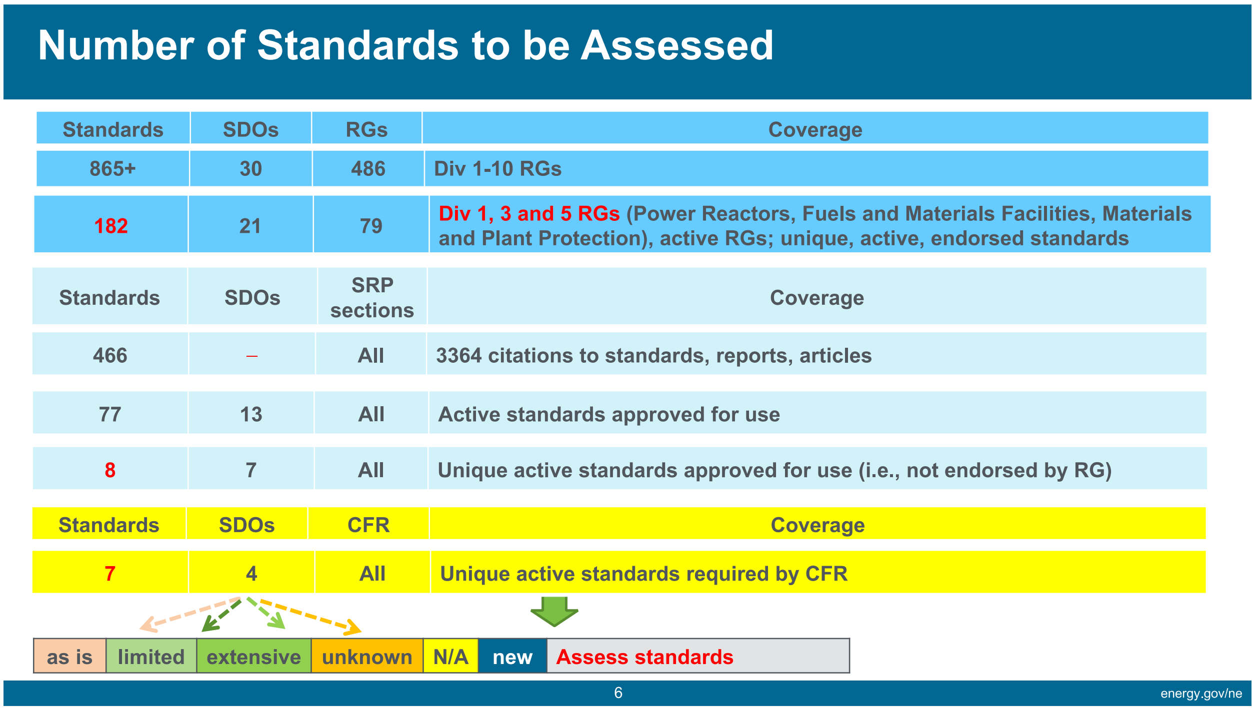

Figure 5. Number of standards to be assessed.

3.2 SDOs OF STANDARDS SELECTED FOR REVIEW

Table 1 and Figure 6 show the number of consensus standards or industry guidance documents and reports endorsed by an RG, approved for use via the SRP, or required by the CFR by SDO or industry group. Appendix A provides a list of standards reviewed, Appendix B provides a list of withdrawn or inactive standards, Table 1 and Figure 7 show the number of industry guidance documents and reports, and Appendix C provides a list of documents from non-SDOs.

Table 1. Number of standards/documents endorsed by RGs by SDO/industry group.

SDO or industry group		Number of standards/documents
American Concrete Institute	ACI	3
American Global Standards	AGS	1
American Institute of Steel Construction	AISC	1
American Nuclear Society	ANS	33
American National Standards Institute	ANSI	8
Department of Defense (Army)	AR	1

Table 1. Number of standards/documents endorsed by RGs by SDO/industry group (continued).

SDO or industry group		Number of standards/documents
American Society of Heating, Refrigerating and Air-Conditioning Engineers	ASHRAE	1
American Society of Mechanical Engineers	ASME	21
American Society for Testing and Materials	ASTM	38
American Welding Society	AWS	1
Builders Hardware Manufacturers Association, Inc.	ANSI/BHMA	3
Deutsches Institut fur Normung E.V.	DIN	1
International Electrotechnical Commission	IEC	3
Institute of Electrical and Electronic Engineers	IEEE	43
Instrumentation Society of America	ISA	2
International Organization for Standardization	ISO	11
Department of Defense (MIL standards)	MIL	4
National Bureau of Standards	NBS	1
National Fire Protection Association	NFPA	7
National Institute of Justice	NILECJ	1
National Institute of Standards and Technology	NIST	7
The Society for Protective Coatings	SSPC	1
Underwriters Laboratories	UL	5
Total SDOs		197
Atomic Energy of Canada, Ltd.	AECL	1
Argonne National Laboratory	ANL	1
Department of Energy	DOE	1
Environmental Protection Agency	EPA	2
Electric Power Research Institute	EPRI	27
General Electric	GE	1
U.S. General Services Administration	GSA	3
Institute of Nuclear Power Operations	INPO	1
Japan Atomic Energy Research Institute	JAERI	1
Lawrence Livermore National Laboratories	LLNL	1
Nuclear Energy Institute	NEI	37
Nuclear Regulatory Commission	NRC	22
Nuclear Management and Resources Council	NUMARC	3
Journal articles, books, etc.	papers	10
US Army Corps of Engineers	USAEC	3
Westinghouse	Westinghouse	3
Total industry group		117
Overall total		314

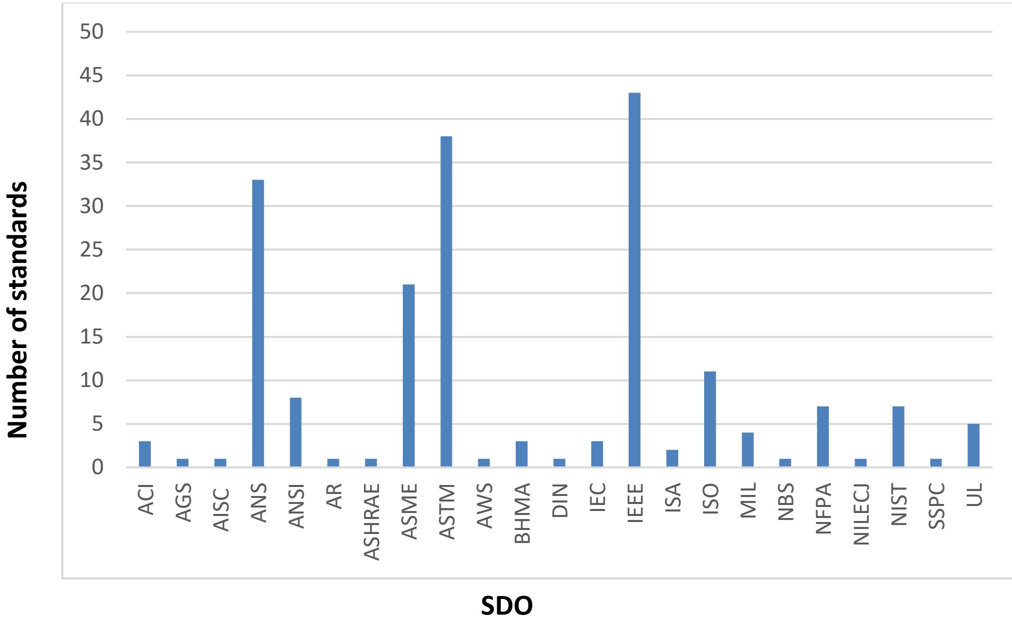

Figure 6. Number of standards by SDO.

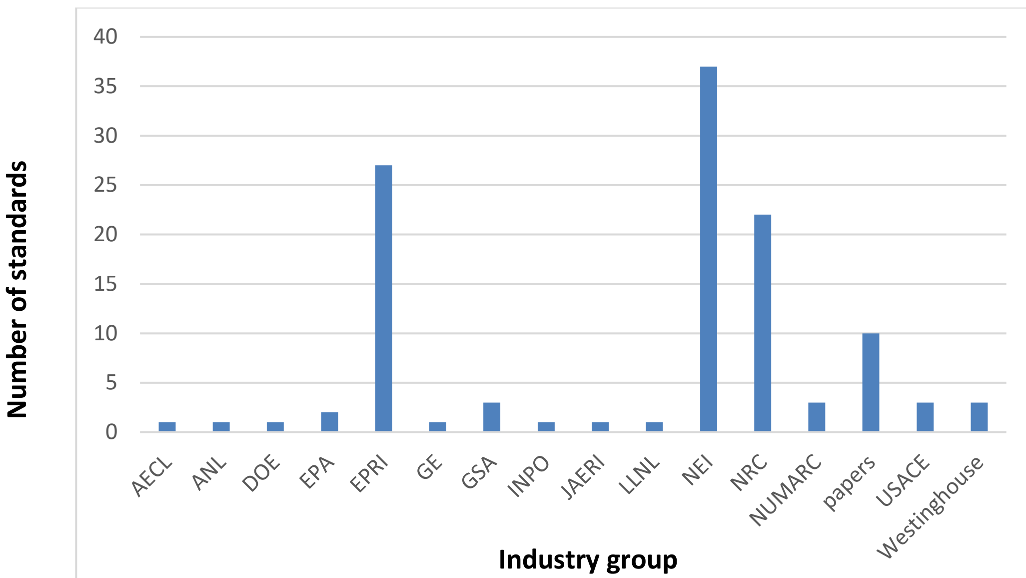

Figure 7. Number of documents by industry group.

ANS, ASME, ASTM, and IEEE were the most cited SDOs overall. A breakdown by source (e.g., RG division, SRP, or CFR) is provided in Appendix A. EPRI, NEI and NRC were the most cited non-SDOs; a breakdown by source is provided in Appendix C.

3.3 LEVEL OF EFFORT

For consistency, a spreadsheet was developed for reviewers to follow, with criteria shown in Table 2. The table addresses objective information such as section numbers and titles as well as subjective information such as summaries of recommended changes, key technical issues, and basis for changes. It also includes qualitative information such as the ease or difficulty in implementing each change and whether a new method or new approach is presented. Table 2 also provides guidance for performing the review. This process was performed on those standards selected for review.

Table 2. Guidance for performing reviews of standards.

Criterion	Notes for content of each column
ID	The identification number (ID) is used for sorting purposes and typically identifies the RG and the number of citations for the RG. Example: 1.05202, which represents RG 1.052, second revision.
Standard	Identify the standard(s) endorsed by the RG, approved for use by the SRP, or required by the CFR. Only cite one standard per ID. Example, RG 1.52 endorses five standards, each of which is provided its own ID: ASME AG-1-2009 ASME N509-2002 ASME N510-2007 ASME N511-2007 ASTM D3803-1991
Standard title	Provide the title of the standard. Example, the title of ASME AG-1-2009 is "Code on Nuclear Air and Gas Treatment."
RG endorsing standard	The RG number and its revision number. Example RG 1.52-4, which signifies the 4^{th} revision of RG 1.52.
RG (or CFR) cited in SRP	An endorsed standard is not always identified in NUREG-0800 (the SRP) but rather is identified through the citation of the RG or the CFR section. For example, SRP 3.8.3 cites RGs 1.57, 1.69, 1.136, 1.142, 1.143, 1.160, 1.199, and 1.221 as acceptable for guidance regarding design, construction, quality control, tests, and inspections that are acceptable. However, the SRP only cites ACI 349, ANSI/AISC N690-1994, and ASME Section III Divisions 1 and 2 as acceptable. Using RG 1.69 as an example, RG 1.69 endorses ACI 349-2013, ACI 349-1R-07, ANSI/ANS 6.3.1-1987, and ANSI/ANS 6.4-2006.
Standard accepted in SRP	A standard may be approved for use in one or more subsections in NUREG-0800 (the SRP). The standard will be "approved for use," "are acceptable," or "in accordance with" or similar wording. Example: The design, materials, fabrication, erection, inspection, testing, and in-service surveillance, if any, of containment internal structures are covered by codes, standards, and guides that are applicable either in their entirety or in part. The following codes and guides are acceptable: American Concrete Institute (ACI) 349 (supplemented with additional guidance by RGs 1.142 and 1.199).
Standard required by CFR	A CFR section may require or approve a standard for use. Example: 10 CFR 34.20 states that "Each radiographic exposure device, source assembly or sealed source, and all associated equipment must meet the requirements specified in American National Standards Institute, N432-1980."
SDO	Provide the name of the SDO. Example, the SDO for ASME AG-1-2009 is ASME.
LOE	Include the number as it applies to each column Change code: 1 = no changes needed 2 = limited changes needed (e.g., only change terminology) 3 = substantive changes needed 4 = insufficient design info to know how extensive the changes might be 5 = not applicable to the design reviewed 6 = new design-specific requirement to add

Table 2. Guidance for performing reviews of standards (continued)

Criterion	Notes for content of each column
Priority	High: affects design or licensing Medium: reduces component fabrication or plant construction time and operations and maintenance costs Low: other effect not cited in High or Medium or LOE 1 or 2
Key technical issues	Summarize what the key technical issues of the standard. Provide the purpose of the standard and what the standard addresses.
Comments, notes	Specify whether a complete or partial review; include additional notes that might aid in rewriting the section.

This review focused on the adequacy and completeness of the standards to an MSR. It is outside the scope of this review to prioritize the endorsement activities of a standard by NRC, to prioritize the development activities of an SDO, or to relate the development of a standard to NRC's mission.

The 197 standards selected for review for adequacy and completeness to MSRs were categorized in one of five LOE categories:

1. No changes needed (i.e., use standard as-is),
2. Limited changes for adequacy and completeness to MSRs,
3. Substantive changes needed for adequacy and completeness to MSRs,
4. Insufficient design information available, and
5. Not applicable to MSRs.

The LOE estimates the significance of the changes to revise a standard for adequacy and completeness to an MSR and not necessarily the hours needed or the availability of data to revise the standard.

A sixth LOE was added to track any standards recommended for new standards that would be beneficial in the design or licensing of an MSR.

This section provides examples of the LOE categorizations. Appendix A provides reviews of all 197 standards endorsed, approved for use, or required by guidance or regulations along with the 14 proposed new standards. Information on key technical issues and the comments provided in the following tables and in the Appendixes are largely quoted from the referenced standard.

3.3.1 No Changes (LOE 1)

There were 141 standards with no changes necessary for adequacy and completeness to an MSR. An excerpt of the changes is shown in Table 3. These standards are technology neutral and would be applicable to any reactor design type.

Table 3. MSR review: examples of no changes to standard for adequacy and completeness (LOE 1)

Standard	Change summary	Key technical issues
ASME NQA-1-2008	—	NQA-1 is a multipart Standard that provides/includes requirements and nonmandatory guidance to establish and implement a quality assurance (QA) program for any nuclear facility application. Part I contains QA program requirements for the siting, design, construction, operation, and decommissioning of nuclear facilities. Part II contains QA requirements for the planning and conducting of the fabrication, construction, modification, repair, maintenance, and testing of systems, components, or activities for nuclear facilities. Part III contains nonmandatory guidance. Part IV contains NQA position papers and other quality program information.
ASTM D3843-16	—	QA, as covered in ASTM D3843, comprises all those planned and systematic actions necessary to provide adequate confidence that safety-related coating work in nuclear facilities as defined in ASTM D5144, will perform satisfactorily in service. Safety-related coating work shall be governed by programmatic and procedural quality provisions that ensure the requirements of 10 CFR 50, Appendix B as defined are satisfied.

3.3.2 Limited Changes (LOE 2)

There were 23 standards with limited changes necessary for adequacy and completeness to an MSR. An excerpt of the changes is shown in Table 4. Standards cited as LOE 2 are typically technology neutral; removing the LWR-based terminology (e.g., LWR or design-basis accident [DBA]) makes that standard applicable to all reactor technologies. To avoid overwhelming an SDO and to prevent scope creep, these standards are not recommended for revision.

Table 4. MSR review: examples of limited review comments (LOE 2)

Standard	Change summary	Key technical issues
ANSI/ANS 3.1-2014	Requirements for experience at a comparable facility and equivalent position will need to be addressed for senior reactor operator (SRO) and reactor operator (RO). Other managerial and staff requirements seem applicable.	The purpose of this standard is to provide guidance for functional levels and job positions as they exist in the operating organization. Qualification requirements include education, experience, and training. This standard provides qualification guidance to meet the particular organizational needs that are derived from the requirements contained in this standard.
ASTM D7167-05	Coating Service Level III lining systems subject to this guide are generally those applied to metal substrates comprising raw water, condensate-quality water, or fuel oil wetted (that is, full or intermittent immersion) surfaces. The establishing procedures to monitor the performance applies to MSRs and the scope should be expanded to include MSRs.	This guide covers procedures for establishing a program to monitor the performance of Coating Service Level III lining (and coating) systems in operating nuclear power plants. Monitoring is an ongoing process of evaluating the condition of the in-service lining systems.

3.3.3 Substantive Changes (LOE 3)

In 19 cases, reviewers stated that more substantive changes are needed. An excerpt of these comments is shown in Table 5. Standards were cited as LOE 3 because of the higher energy spectrum, higher temperatures, and corrosive coolants. Material properties for metals, concrete, and protective coatings must be addressed.

Table 5. MSR review: examples of substantive review comments (LOE 3)

Standard	Change summary	Key technical issues
ASME AG-1-2009	Materials of construction for all components and accessories shall conform to the ASME or ASTM material specifications listed in Table AA-3100. Because of the presence of sodium, the list of allowable materials listed in Table AA-3100 may need to be updated for MSRs. The Process Gas section is incomplete and needs to be completed. The entire section needs to address the use of a cover gas such as helium.	This Code provides requirements for the performance, design, fabrication, installation, inspection, acceptance testing, and quality assurance of equipment used in air and gas treatment systems in nuclear facilities. The code is divided into the following divisions: Division I: General Requirements Division II: Ventilation Air Cleaning and Ventilation Division III: Process Gas Treatment Division IV: Testing Procedures.
ASME Boiler and Pressure Vessel Code Division 1 and 2, Subsection NCA	The containment barrier is "…essentially leak-tight…" rather than an "…effective barrier…" to describe a flexible containment function for concepts that may rely on acceptable design condition leak rates.	The rules of Subsection NCA constitute requirements for the design, construction, stamping, and overpressure protection of items used in nuclear power plants and other nuclear facilities. This Section consists of the three divisions: (a) Division 1. Metallic vessels, heat exchangers, storage tanks, piping systems, pumps, valves, core support structures, supports, and similar items. (b) Division 2. Concrete containment vessels. (c) Division 3. Metallic containment systems for storage or transportation of spent nuclear fuel and high-level radioactive materials and waste.

Table 5. MSR review: examples of substantive review comments (LOE 3) (continued)

Standard	Change summary	Key technical issues
ANSI/ISA-67.02.01-2014	Pressure and level measurements may use different technologies or apply existing technology in a different manner. Pressure measurements may use impulse lines, bubblers, or use direct measurement sensors. Level measurements may use guided-wave microwave, guided-wave ultrasonic, or heated lance. Temperature alone will require changes to the methodology for pressure and level measurements. Sodium presents problems with visibility and does not boil which will eliminate some measurement techniques. In an MSR, the reactor coolant pressure boundary is the primary coolant boundary.	Routing of instrument sensing lines in the standard are concerned with water level indication during and after rapid depressurization involving flashing, degassing, or non-condensable gas events has been identified in industry as a concern, specifically in the pressurizer reference legs of PWRs and reactor vessel water level instrumentation of BWRs and shall be considered. Sensing lines and level measurements will have different fluids and possibly types of sensors. MSRs may also use optical sensors.

3.3.4 Unknown (LOE 4)

Only six standards were cited as LOE 4. Because of the unknowns of the postulated accidents, the adequacy and completeness of the standards were unknown. In addition, it is unknown whether an MSR will use/require an emergency DG. An excerpt of the changes is shown in Table 6.

Table 6. MSR review: examples of reference information not applicable to the MSR design (LOE 4)

Standard	Change summary	Key technical issues
ANSI/ANS 59.51-1997	The purpose of this standard is to define those features of fuel oil systems required to ensure an adequate fuel supply to safety-related emergency diesel generators, and to provide performance and design criteria to ensure sufficient fuel is available for supply to the emergency diesel generators under all plant conditions. Although the criteria may be useful, it is unknown whether MSRs will use Class 1E emergency DGs.	The fuel oil system shall be capable of supplying an adequate supply of suitable fuel oil to the emergency diesel generators under all Plant Conditions that are defined ANSI/ANS-51.1-1983 (for PWRs) and ANSI/ANS-52.1-1983 (for BWRs). Both ANS 51.1 and ANS 52.1 have been withdrawn so replacement with an MSR-specific set of plant conditions would not be necessary.
ASTM D4082-10	Based on the assessed lifetime radiation of coating and radiation during a DBA, the irradiation dose rate, irradiation accumulated dose, and radiation source will need to be revised. For an MSR, DBA should be Postulated Accident.	This test method covers a standard procedure for evaluating the lifetime radiation tolerance of coatings to be used in nuclear power plants. This test method is designed to provide a uniform test to assess the suitability of coatings, used in nuclear power facilities, under radiation exposure for the life of the facilities, including radiation during a DBA.

Table 6. MSR review: examples of reference information not applicable to the MSR design (LOE 4) (continued)

Standard	Change summary	Key technical issues
ASTM D3803-1991	Guidance for testing new and used carbons using conditions different from the test method in ASTM D3803 is offered in Annex A1 of the standard. The appropriateness of the test method will need to be evaluated when a more detailed design is available.	The test method in ASTM D3803 is a very stringent procedure for establishing the capability of new and used activated carbon to remove radio-labeled methyl iodide from air and gas streams. The conditions employed in the standard were selected to approximate operating or accident conditions of a nuclear reactor which would severely reduce the performance of activated carbons.

3.3.5 N/A (LOE 5)

Eight standards were classified as LOE 5. An excerpt of the changes is shown in Table 7.

Table 7. MSR review: examples of reference information not applicable to the MSR design (LOE 5).

Standard	Change summary	Key technical issues
ANSI/ANS 8.17-2004 (R2014)	—	Standard is specific to handling, storage, and transport of LWR fuel (including individual fuel rods) outside of the reactor core. Consideration of fuel salts is necessary.
DIN 25463-1	—	Not applicable to MSRs — German National Standard that will not be updated for MSRs.

3.3.6 New Standards Needed (LOE 6)

The process for identifying new standards involved reviewing the following documents for insights:

- The Evaluation Findings in ORNL/TM-2020/1478, "Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application" [20];
- The high priority needs in ORNL/TM-2021/2176, "Molten Salt Reactor Fundamental Safety Function PIRT" [22];
- The Advanced Reactor Design Criteria in RG 1.232, Rev. 0, "Guidance for Developing Principal Design Criteria for Non-light-water Reactors" [26, 27, 28]; and
- The MSR design criteria in The American National Standards Institute (ANSI)/ANS-20.2-202x (DRAFT), Performance Requirements for Liquid-Fuel Molten Salt Reactor Nuclear Power Plants.

Based on these reviews and a review of various proposed MSR designs, 14 new standards were identified for review (Section 4.3).

3.4 PRIORITY

Similar to the review by the Nuclear Energy Institute (NEI) of ORNL's assessment of standards requiring significant changes for applicability to SFRs [7, 29], the priority of changes or updates to standards is based on high, medium, or low impact. However, if the LOE for the standard was LOE 1 (no changes), LOE 2 (limited changes), or LOE 5 (N/A), then the standard was identified as low impact. The priority levels are defined as follows:

- High — affects design or licensing,
- Medium — reduces component fabrication or plant construction time and operations and maintenance costs, and
- Low — other effect not cited in High or Medium, no changes (LOE 1), limited changes (LOE 2) needed, or N/A (LOE 5).

If not already commenced, the highest priority activities should begin as soon as resources can be provided. Work on the lower priority items can be done in later phases, depending on available resources. It is clear from the number of codes and standards identified in this report that a great deal of work remains to be completed to facilitate the efficient design, licensing, construction, and operation of advanced reactors.

Details of the priority assessment of the standards identified as requiring significant changes (LOE 3) or new standard required (LPE 6) are provided in Subsection 4.2.

4. RESULTS

The results are measured in terms of LOE to make the standard applicable in whole to an MSR and the priority of revising the standard. Included in this analysis are the list of new proposed standards that will likely be needed to license an MSR.

Of the 197 standards reviewed, 149 will not require any changes (LOE 1 and 5).

A total of 35 inactive or withdrawn standards were identified (Appendix B). These documents were not evaluated further. Most of the inactive standards (10) were ANSI standards that addressed information technology, device calibration, and test procedures. Three additional inactive standards were ASTM standards that addressed plant performance monitoring of various special nuclear material (SNM) monitors. The inactive documents were not investigated.

Materials and plant protection requirements and associated standards are not unique to any one reactor technology. Therefore, this outcome is anticipated. However, many of the standards referenced by the Division 5 RGs do not reference the latest revision or reaffirmation of some standards. This issue will likely be addressed by NRC staff as individual RGs come up for review. For this review however, the latest approved standard was used.

Of the 51 fuels and material facilities RGs reviewed, 13 provide guidance on reprocessing, 9 provide guidance on fuel storage, 8 provide guidance on fuel fabrication, 4 provide guidance on fuel cycle facilities, 2 provide guidance on facility decommissioning, 1 provides guidance on criticality safety, and the remaining 14 are not applicable to MSRs. These RGs cover the breadth of topics that may be applicable to an MSR facility that includes a larger portion of the fuel cycle on-site than a comparable LWR. Furthermore, the individual RGs and some of the related standards may require modifications to include guidance unique to MSRs. However, the need to create new standards was not identified. Individual MSR designers may subsequently identify the need for a standard that is specific to their distinctive design, but information at that level of detail is not currently available.

Materials and plant protection requirements and associated standards are not unique to any one reactor technology. Therefore, no new standards were identified for this subject area.

4.1 LEVEL OF EFFORT

Of the 197 standards reviewed, about 75% will not require any changes (i.e., no changes [141] or not applicable [8]). About 12% of the standards reviewed will require minor changes (23) and 10% will require significant changes (19). Insufficient information available to assess the adequacy and completeness of about 3% of the standards (6) (Table 8 and Figure 8).

Table 8. LOE for standards identified by source

LOE	Division 1	Division 3	Division 5	SRP	CFR	Total
1	72	31	28	5	5	141
2	21	1			1	23
3	16	2			1	19
4	4	3				6
5		4		3		8
Total	113	41	28	8	7	197

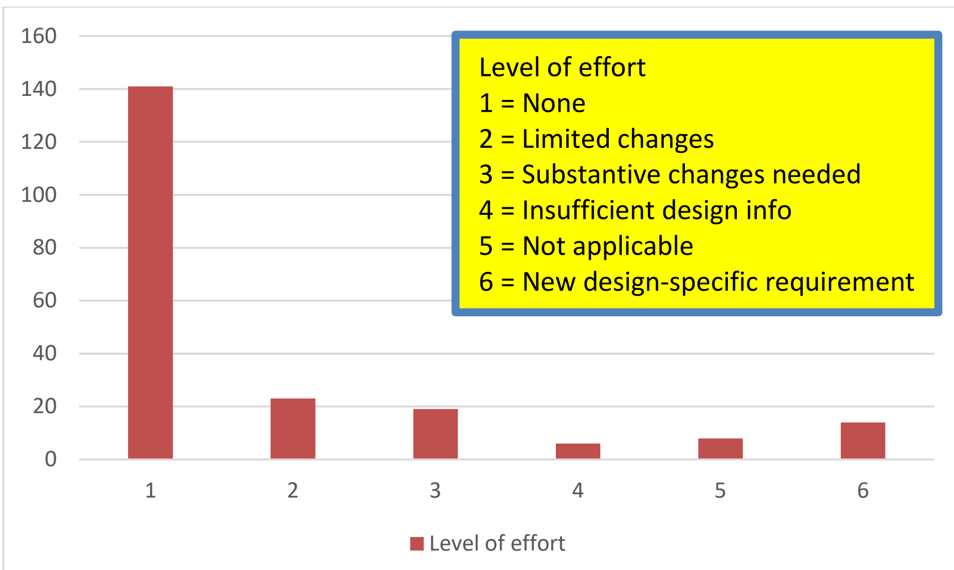

Figure 8. LOE.

Of the SDOs with standards endorsed by an RG, approved for use by the SRP, or required by the CFR, the SDOs must be involved in revising or developing the standard for adequacy and completeness to an MSR (Figure 11).

As shown in Table 9, four SDOs with active standards endorsed by a fuels and material facilities RG must be involved in revising or developing at least one standard for adequacy and completeness to an MSR. In addition, NEI 14-03 Rev 2, *Format, Content, and Implementation Guidance for Dry Cask Storage Operations Based Aging Management*, will need to be revised if MSR fuel is to be stored in dry cask facilities.

Because MSR fuel and the related fuel cycle is so unique, any SDO standard changes would not be technology neutral. In that respect, SDOs may opt to generate new standards on similar fuel cycle topics that are directed solely at MSR technologies.

Industry and NRC have shown interest in leveraging standards to accelerate licensing of advanced non-LWRs [30]. This strategy depends on forming a group of volunteers to serve as a "coalition of the willing" composed of end-user organizations, SDO representatives, NRC representatives, and other stakeholders. One lesson learned in the development of standards is the problem of not having a true champion to help plan a roadmap for the development of a standard (i.e., technical basis, intended standard, RG, schedule).

Table 9. LOE by SDO

SDO	Number Of standards	Level of effort					
		1	2	3	4	5	6
ACI	3	1	2				
AGS	1	1					
AISC	1	1					
ANS	33	20	5	5	1	2	13
ANSI	8	6				2	
AR	1	1					
ASHRAE	1	1					
ASME	21	11	4	7			1
ASTM	38	20	10	4	3	1	
AWS	1	1					
ANSI/BHMA	3	3					
DIN	1					1	
IEC	3	3					
IEEE	43	39	1		1	2	
ISA	2	1		1			
ISO	11	10		1			
MIL	4	3			1		
NBS	1	1					
NFPA	7	4	1	2			
NILECJ	1	1					
NIST	7	7					
SSPC	1	1					
UL	5	5					
Total SDOs	197	141	23	19	6	8	14

4.2 PRIORITY OF REVISIONS

A standard identified as requiring a significant number of changes for adequacy and completeness to an MSR may not present a high need for the MSR community. For example, although in-service inspections (ISIs) are a critical part of operations and the ISI program for an MSR will require extensive changes, AMSE Boiler and Pressure Vessel Code (BPVC) Section XI on ISI was rated low in terms of priority of standards to be revised because it does not directly affect the design.

A total of 19 standards have been evaluated as high priority with the potential to provide the greatest benefit for near-term development. Table 10 lists the 19 standards, and details of the review are provided in Appendix A. The high-priority codes and standards are listed in no particular order. To enable progress on development of these codes and standards in the near term, support from the federal government through DOE and NRC is needed in addition to industry involvement.

Table 10. Standards with LOE 3

Standard	Title	LOE	Priority
ASME BPVC Section III Division 1 Subsection NE Subsection NF Subsection NG	Rules for Construction of Nuclear Power Plant Components	3	High
ASME BPVC Section III Division 2	Code for concrete containments	3	High
ASME AG-1-2009	Code on Nuclear Air and Gas Treatment	3	High
ASME N509-2002 (SRP accepts N509-1989)	Nuclear Power Plant Air-Cleaning Units and Components	3	High
ASTM D3803-1991	Standard Test Methods for Nuclear-Grade Activated Carbon	3	High
ASTM D7491-08	Standard Guide for Management of Non-Conforming Coatings in Coating Service Level I Areas of Nuclear Power Plants	3	High
ASME QME-1-2017	Qualification of Active Mechanical Equipment Used in Nuclear Power Plants	3	High
ASME BPVC Section II, Parts A, B, and C	ASME BPVC Section II, "Materials," Parts A, B, C, and D	3	High
NFPA 251	Standard Methods of Tests of Fire Resistance of Building Construction and Materials	3	High
NFPA 805	Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants	3	High
ANS 5	Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors	3	High
ASTM D3911-16	Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design-Basis Accident (DBA) Conditions	3	Low
ANSI/ANS 6.4-2006	Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants	3	Low
ANSI/ANS 56.2-1984 (ANSI N271-1976)	Containment Isolation Provisions for Fluid Systems	3	Low
ASME BPVC Section XI	Rules for Inservice Inspection of Nuclear Power Plant Components	3	Low
ANSI/ANS 3.5-2009	Nuclear Power Plant Simulators for Use in Operator Training and Examination	3	Low
ANSI/ISA 67.02.01-2014	Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants	3	Medium
ISO 10645:1992	Nuclear Energy — Light Water Reactors — Calculation of The Decay Heat Power in Nuclear Fuels	3	Medium
ANSI/ANS 5.1-1979 ANSI/ANS 5.1-2014	Decay Heat Power in Light Water Reactors	3	Medium

4.3 NEW STANDARDS

After identifying the number of standards to review and the LOE to revise those standards for adequacy and completeness to an MSR, the next step was to review the standards selected for detailed review to identify the need for new standards unique to MSRs. This step identified 14 potential new standards, listed in Table 11.

The process for identifying new standards consisted of reviewing the following documents for insights:

- The Evaluation Findings in ORNL/TM-2020/1478, “Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application” [20];
- The high priority needs in ORNL/TM-2021/2176, “Molten Salt Reactor Fundamental Safety Function PIRT” [22];
- The Advanced Reactor Design Criteria in RG 1.232, Rev. 0, “Guidance for Developing Principal Design Criteria for Non-light-water Reactors” [26, 27, 28]; and
- The MSR design criteria in ANSI/ANS-20.2-202x (DRAFT), Performance Requirements for Liquid-Fuel Molten Salt Reactor Nuclear Power Plants.

Table 11. MSR review: examples of proposed new standards applicable to the MSR design (LOE 6)

Priority	Key Technical Issues	Comments, Notes	SDO
Medium	A requirement of the qualification of passive equipment is needed to address the nuclear analysis and design of passive heat removal systems, such as the concrete for passive heat removal. This new standard would be similar to ASME QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants that provides the requirements and guidelines for the qualification of active mechanical equipment whose function is required to ensure the safe operation or safe shutdown of a nuclear facility. NEW standard based on review of ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.” A standard should be developed for the qualification of passive equipment. This standard would describe the requirements and guideline for qualifying passive mechanical equipment, such as valves not requiring external motive force, used in many advanced reactors. The requirements and guidelines would include the principles, procedures, and methods of qualification.	Nonmandatory Appendix QR-A, “Seismic Qualification of Active Mechanical Equipment,” to ASME QME-1-2017 also includes the use of experience data as a method for the seismic qualification of active mechanical equipment. A similar standard should be developed for the qualification of passive equipment whose function is required to ensure the safe operation or safe shutdown.	ASME

Table 11. MSR review: examples of proposed new standards applicable to the MSR design (LOE 6) (continued).

Priority	Key Technical Issues	Comments, Notes	SDO
High	Higher energy neutrons and photons may affect the characteristics of the concrete. That is, the radiation and thermal environment of MSRs may be different from concrete used for LWR applications and result in different shielding and thermal properties. In addition, changes in the structural characteristics of concrete resulting from the radiation and thermal environment may affect the ability of concrete to meet its structural requirements. NEW standard to address the higher energies imparted on concrete being used for passive heat removal.	A new standard would be similar to ANSI/ANS 6.4-2006, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants. Because the types of steel, concrete, and source term and function of the concrete may differ significantly from those addressed in ANSI/ANS 6.4-2006, a new standard is recommended rather than revising the existing standard.	ANS
High	The need for new standards to address fluoride MSRs was highlighted in the 1960s through the Molten Salt Reactor Experiment at Oak Ridge National Laboratory (ORNL). This experiment confirmed that tritium is produced within the heat exchanger tubes, which impacts corrosion and potential for release, and that this could be addressed through material standards. Additionally, methods to decommission and fully deconstruct MSRs are still being developed. These lessons can be applied to the molten salt reactor technology and the development of standards in future.	Quality of the intermediate coolant boundary. Components that are part of the intermediate coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. ANS-20.1-201x, Nuclear Safety Criteria and Design Criteria for Fluoride Salt-cooled High-temperature Reactors may fill this need once completed.	ANS
Low	Some MSR designs use dissolved fuel rather than the dispersed fuel approach. This represents a novel approach to reactor fueling (and Fluoride34) that is not directly addressed by existing experience and could lead to new standard requirements. However, the same concepts used to prove the safety case of existing reactors are assumed to hold and can be used by vendors to address this new technology application.		ANS
Low	To support the Evaluation Findings in Section 4.2.2 in ORNL/TM-2020/1478, because the control elements in a MSR may be significantly different than those of other reactor types, sufficient information should be provided to show that the functional and safety-related design bases can be achieved by the control elements designs, that the control elements conform to the design bases and can control and shut down the reactor safely from any operating condition, reasonable assurance exists that the reactor trip features designed will perform as necessary,	The control elements in an MSR are designed to change reactivity by changing the amount of neutron absorber (or fuel) or reflection in or near the active reactor core. Control elements can be designated by their material, phase, and their intended function in the reactor. To	ANS

Table 11. MSR review: examples of proposed new standards applicable to the MSR design (LOE 6) (continued).

Priority	Key Technical Issues	Comments, Notes	SDO
	acceptable shutdown margin exists, maximum scram times and maximum rates of insertion of positive reactivity cause my malfunctions are acceptable. Other systems may also act to reduce reactivity on a reactor trip such as adjusting reactor fuel flow, dumping fuel salt to a drain tank, securing the addition of fresh fuel, and securing operation of the fuel salt cleanup system. ARDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The status of understanding of reactivity control (FOM3) is either adequately known or would be addressed by research performed to improve the understanding of potential for release of fission products and/or barrier integrity failure (FOM1) or potential for unbalanced heat removal (FOM2) (ORNL/TM-2021/2176), making this a low priority issue.	trip the reactor, the negative reactivity of the control elements is usually added passively and quickly. Because the control elements can serve a dual function (control and safety), control and safety systems for non-power reactors are usually not completely separable. In non-power reactors, a reactor trip does not challenge the safety of the reactor or cause any undue strain on any systems or components associated with the reactor. Other systems may also act to reduce reactivity on a reactor trip such as adjusting reactor fuel flow, dumping fuel salt to a drain tank, securing the addition of fresh fuel, and securing operation of the fuel salt cleanup system.	
Low	To support the evaluation findings in ORNL/TM-2020/1478, each nuclear reactor should contain a neutron startup source that ensures the presence of neutrons during all changes in reactivity. This is especially important when starting the reactor from a shutdown condition. Areas of review should include the following: -- Type of nuclear reaction, Energy spectra of neutrons, Source strength, Source material phase (e.g., solid material stored in a holder or liquid dissolved in the fuel salt), Interaction of the source and holder (if applicable), while in use, with the chemical, thermal, and radiation environment, Design features that ensure the function, integrity, and availability of the source, TS. Acceptance criteria for information on the neutron startup source include the following: -- The source and source holder should be constructed of materials that will withstand the environment in the active reactor core and during storage, if applicable, with no significant degradation. — The type of neutron-emitting reaction in the source should be comparable to that at other licensed reactors, or test data should be presented in this section of the SAR to justify use of the source. — The natural radioactive decay rate of the source should be slow enough to prevent significant decay over a 24-hour period or between reactor operations. — The design should allow easy replacement of the source and its holder and a source check or calibration. — Neutron and gamma radiation from the	Early data from the Molten Salt Reactor Experiment (MSRE) at Oak Ridge National Laboratory (ORNL) showed acceptable compatibility of highly pure fluoride salts with Hastelloy-N, but data acquired under other conditions are scattered and not well controlled. Data on materials in chloride salts are especially limited. It is recommended that standards be developed for experimentation in molten salts to produce a consistent data set so alloy behavior can be understood in different salt conditions. Thermodynamic data on alloy and salt combinations are also needed to better understand interfacial phenomena in MSRs. Further, there is a need for rate modeling to predict lifetimes of salt-facing materials. Thus, there	ANS

Table 11. MSR review: examples of proposed new standards applicable to the MSR design (LOE 6) (continued).

Priority	Key Technical Issues	Comments, Notes	SDO
	reactor during normal operation should not cause heating, fissioning, or radiation damage to the source materials or the holder. — If the source is regenerated by reactor operation, the design and analyses should demonstrate its capability to function as a reliable neutron startup source in the reactor environment. — TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements, and should be justified in this section of the SAR.	is a gap in knowledge as to how the more common chloride salt compositions will perform which limit their adequacy and completeness	
High	ANS 20.2, Criterion 71 recognizes that systems shall be provided as necessary to maintain the composition of the fuel salt within specified limits. These limits shall be based on the ability of the fuel salt to perform its safety functions. This new criteria needs to be reflective of the role of the fuel salt in the overall facility safety. Fuel salt is an essential element of providing adequate containment, heat removal, and reactivity control. Fuel salt properties are determined by its composition, which must be maintained within acceptable limits.	There are various activities being undertaken and on-going to supplement Division 5. Activities such as extending the qualified lifetimes of Class A materials to support a 60-year design life, qualification of additional materials, development of analysis methods to simplify the Division 5 design analyses, development of design rules for integrally cladded components with weld overlay on Class A materials to support molten salt reactor applications, incorporation of graphite irradiation data to support graphite design rules, and incorporation of ceramic composite design rules. ANS 20.2 added a series of design criteria to the modified GDC list provided in 10 CFR 50, Appendix A. This is classified as Series VII: salt systems and control and adds 5 new design criteria.	ANS
High	To support the Evaluation Findings in Section 4.2.3 in ORNL/TM-2020/1478), if the moderator (if applicable) and reflector are integral constituents of an active reactor core, the designs should take into account interactions between the moderator (if applicable) or reflector and the reactor environment. Reasonable assurance should be provided such that degradation rates of the moderator (if applicable) or reflector will not affect safe reactor operation, prevent safe reactor shutdown, or cause uncontrolled release of radioactive material to the		ANS

Table 11. MSR review: examples of proposed new standards applicable to the MSR design (LOE 6) (continued).

Priority	Key Technical Issues	Comments, Notes	SDO
	unrestricted environment. (Fuel salt infiltration could cause changes in neutron scattering and absorption, thereby changing active reactor core reactivity.		
High	To support the Evaluation Findings in ORNL/TM-2020/1478, the design features of the fuel system boundary and components must give reasonable assurance of boundary integrity under all possible reactor conditions, including potential accident scenarios. The fuel system boundary should be designed to remove sufficient fission heat from the fuel salt to allow all licensed operations without exceeding the established LSSSs that are included in the TS. In addition, the design and location of fuel system boundary components needs to have been specifically selected to avoid fuel salt loss that could lead to fuel system boundary failure, an uncontrolled release of excessive radioactivity, or damage to safety systems or experiments. Similarly, ANS 20.2, Criterion 70, states that any reactor coolant system whose moving fluids may become activated shall be designed with sufficient margin to assure that its containment function is adequately maintained. Rationale is that MSRs are vulnerable to coolant activation due to its proximity to the neutron field. This DC addresses the need to maintain containment of the activated coolant.	ANS 20.2 added a series of design criteria to the modified GDC list provided in 10 CFR 50, Appendix A. This is classified as Series VII: salt systems and control and adds 5 new design criteria.	ANS
High	ANS 20.2, Criterion 72 — salt temperature control systems—Heating systems shall be provided as necessary for systems and components important to safety, which contain or could be required to contain salt. These heating systems and their controls shall be appropriately designed to ensure that the temperature distribution and rate of change of temperature in systems and components containing salt are maintained within design limits assuming a single failure. This new criteria needs to be reflective of the importance of preventing freezing of salt and thermally damaging fuel salt contacting containment layers. Examples of salt systems could include salt sampling lines and or decay heat removal systems that are important to safety.	ANS 20.2 added a series of design criteria to the modified GDC list provided in 10 CFR 50, Appendix A. This is classified as Series VII: salt systems and control and adds 5 new design criteria.	ANS
High	ANS 20.2, Criterion 74 — fuel salt system interfaces—Where the fuel salt boundary interfaces with a structure, system, or component containing fluid that if allowed to freely interact with the fuel salt would cause the loss of a safety function, the interface location shall be designed to ensure that the fuel salt is separated from the fluid by two redundant, passive barriers. This new criterion is derived from SFR-DC 78, which describes the safety function of interfaces.	ANS 20.2 added a series of design criteria to the modified GDC list provided in 10 CFR 50, Appendix A. This is classified as Series VII: salt systems and control and adds 5 new design criteria.	ANS

Table 11. MSR review: examples of proposed new standards applicable to the MSR design (LOE 6) (continued).

Priority	Key Technical Issues	Comments, Notes	SDO
High	To support the Evaluation Findings in ORNL/TM-2020/1478, the information on the reactor fuel should include a description of the required characteristics. Further, RG 1.232 states that "An MSR designer may need to develop new PDC [principal design criteria] for liquid fuel and systems to support this design." The PIRT [ORNL/TM-2021/2176] indicates that understanding the phenomena of the mass/volume and energy of the molten salt (fueled salt) pool is of high importance and the knowledge base is insufficient making this a high priority for further research. The NRC reviews the fuel system description and design drawings with emphasis on product specifications rather than process specifications (SRP 4.2). The closest standard approved for use (SRP 4.2) is ASTM C776-89, Part 45, Standard Specification for Sintered Uranium Dioxide Pellets, which specifies the chemical, nuclear, and physical characteristics of UO2 pellets. A standard similar to ASTM C776-89 should be developed for molten salt.		ASTM
High	To support the Evaluation Findings in ORNL/TM-2020/1478, the design of the gas management system helps the MSR operate with a gas-tight vessel. The gas management system is designed to prevent the uncontrolled release of radioactive material and interference with safe reactor operation or shutdown. Design considerations of the gas management system include its integrity, showing that a credible failure would not lead to loss of fuel system boundary integrity, that adequate heat removal mechanisms are available to the gas management system. Similarly, ANS 20.2, Criterion 73 notes that if plugging of any cover gas line due to condensation, solidification of salt, plate out of salt aerosol, reaction product, or vapor could prevent accomplishing a safety function, then the line shall include corrective measures with adequate reliability. This new criteria needs to reflect the distinctive characteristics of molten salt cover gas. Salt deposits may not be removable by melting. A new standard should address the overall design of the gas management system to ensure that it is designed to ensure that the required type of gas, the acceptable concentrations of constituents (including processing, storing, and recombining of reactive gases, as applicable), and the design-basis pressure are maintained.	ANS 20.2 added a series of design criteria to the modified GDC list provided in 10 CFR 50, Appendix A. This is classified as Series VII: salt systems and control and adds 5 new design criteria.	ANS
High	There are two types of molten salt reactors: chloride salt and fluoride salt. Historically, most MSR designs have been fluoride based, however, chloride-based MSR designs have been proposed. This proposed standard may only be applicable to fluoride-based salts. The Molten Salt Reactor Experiment at Oak Ridge National Laboratory (ORNL) confirmed that tritium is produced		ANS

Table 11. MSR review: examples of proposed new standards applicable to the MSR design (LOE 6) (continued).

Priority	Key Technical Issues	Comments, Notes	SDO
	within the heat exchanger tubes, which impacts corrosion, and that this could be addressed through material standards. Thus, a new standard should be developed that addresses materials being exposed to tritium and the susceptibility to corrosion.		

The NRC staff is actively evaluating security requirements for advanced reactors (light-water small modular reactors and non-LWRs). The requirements found in 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," allow for alternative measures to be considered. These measures may be less prescriptive than current guidance for meeting the CFR requirements by meeting performance objectives if they can be shown to be equivalent to the measures provided by the specific requirement for which it would substitute. This may lead to the development of additional standards and industry guidance documents in the future. However, consideration of a risk-based approach for meeting performance objectives is beyond the scope of this MSR standards review.

4.4 SUMMARY OF RESULTS

With respect to those standards requiring significant changes, Figure 9 shows that SDOs, plant design, and start of operations could experience significant disruptions. Although none of the standards with significant changes needed for adequacy and completeness had the American Concrete Institute (ACI) as the lead author, its participation is critical for the concrete and concrete-coating-related standards. These cross-cutting issues will probably be applicable to multiple reactor designs and will require the collaboration of multiple SDOs and industry.

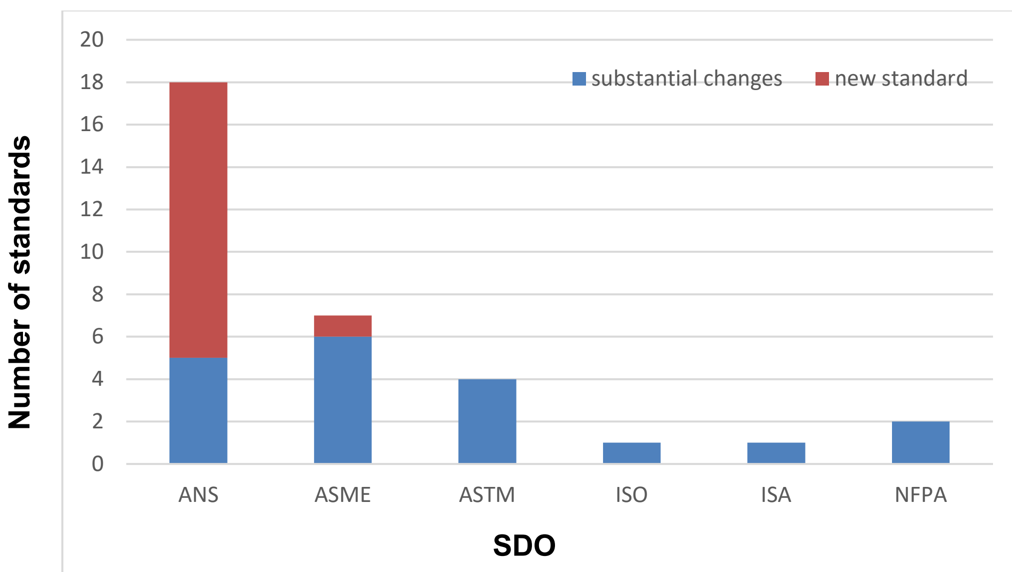

Figure 9. SDOs most affected.

Of the standards reviewed, 19 will likely require substantive changes. These 19 standards address the following topic areas:

- Protective coatings and test methods for protective coatings may differ;
- Temperatures in MSRs may exceed concrete and steel limits in standards;
- Types of steel, concrete, and source terms may differ greatly for MSRs compared with LWRs;
- Those components required to function during a DBA (postulated accident) will be different for MSRs and will require modification to some standards (e.g., seismic, dynamic qualifications);
- Containments will be different from current plants;
- Fire issues (e.g., fire-induced failures, testing) will differ; and
- Presence of sodium affects factors such as environmental qualification, habitability, fire.

Fourteen new consensus standards will be required for the following areas:

- Different operating environment,
- Passive cooling, and
- Passive equipment.

The IEEE standards are technology neutral. However, because of the differences in the function of containment, insufficient design information is available to evaluate the adequacy and completeness of IEEE 317-1983, *IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations*. Moreover, insufficient design information is available to evaluate the adequacy and completeness of IEEE 387-1995, *Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations*, to the MSR.

Of the instrumentation and control (I&C) standards not published by IEEE, ANSI/ISA-67.02.01-2014, *Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants*, will require substantial changes. Routing of instrument-sensing lines in the Instrumentation Society of America (ISA) standard are included to address water level indication during and after rapid depressurization involving flashing, degassing, or noncondensable gas events. Depressurization has been identified in industry as a concern—specifically in the pressurizer reference legs of PWRs and reactor vessel water level instrumentation of BWRs—and shall be considered. Sensing lines and level measurements will include different fluids and possibly different types of sensors. MSRs may also use optical sensors.

Changes necessary for adequacy and completeness to MSRs for ISA-67.02.01-2014 include the following:

- Pressure and level measurements may use different technologies, or they may apply existing technology in a different manner. Pressure measurements may use impulse lines, bubblers, or direct measurement sensors. Level measurements may use guided-wave microwave, guided-wave ultrasonic, or heated lance.
- Temperature alone will require changes to the methodology for pressure and level measurements. Sodium presents problems with visibility and does not boil, which will eliminate some measurement techniques.
- In an MSR, the reactor coolant pressure boundary (RCPB) is the primary coolant boundary.

Of the voluntary consensus standards endorsed by RGs in Division 3, six will likely need substantive changes. These six standards address the following topic areas:

- Decay heat considerations (level and location) may differ for MSRs compared to LWRs with some MSR FPs removed in an off-gas system or a polishing system.
- Temperatures in MSRs may exceed concrete and steel limits in standards.

- Types of steel, concrete, and source terms may differ greatly for MSRs compared with LWRs.
- Criticality safety considerations for handling and storage may be different for MSR liquid or solid fuel compared to LWR solid fuel.
- The use of water for fire suppression must be considered carefully for MSR facilities.
- Dry cask storage and aging management may be different for MSR liquid, or solid fuel compared to LWR solid fuel.

Of the voluntary consensus standards endorsed by RGs in Division 5, none should need substantive changes because the materials and plant protection subject areas are technology neutral.

5. SDO AND INDUSTRY INVOLVEMENT

Some of the standards will require extensive changes, and addressing the needs of new standards will require the development of a new standard. The number of standards requiring extensive changes is 19.

5.1 SDO APPROVAL

The basic development process for a new standard or the process of revising an existing standard, although it differs for each SDO, includes the following steps:

1. Submit a need (and justification) for a new (revised) standard
   - Include background information
   - Explain the significance of the revisions
2. Prepare a draft standard or revision of an existing document for internal review and comment
3. Revise standard based on internal reviews and issue first draft report
   - Committee approves document for public comment
4. Issue draft for public comment
5. Revise standard based on public comments and issue revised draft
   - Submission of committee approved responses to public comments
6. Submission of revised standard to standards Board
7. Approval of standard for use by Board

Based on a review of the standards development process [31, 32] and a survey of the seven SDOs with standards endorsed by an RG (including ANSI), a range of time periods for development or modification is estimated as follows:

1. Time for minor changes to a standard to be approved: 0.5–2 years;
2. Time for significant changes to a standard to be approved: 1–3 years; and
3. Time for the development and approval of a new standard: 2–8 years.

However, ASTM states that, depending on a committee's commitment to timely development and approval, "standards can take as little as nine months to become full consensus standards" [33].

Many variables that can affect the time to develop or modify a standard, such as the following [34]:

- Which committee(s) are involved in the approval process,
- The technical complexity of the standard,
- Whether any research is needed to support a revision,
- Whether there is a strong champion for the revision or new standard, and
- How many other significant revisions are being considered by the committee(s) in the time frame.

Any modification or creation of a new standard would benefit from interactions with NRC staff members throughout the development/modification process [16, 35].

After the standard is certified as a full consensus standard, it may be forwarded to ANSI for review. Standard development is a rigorous consensus process that for many SDOs has been approved by ANSI. ANSI is a private, nonprofit organization that administers and coordinates the US voluntary standards and conformity assessment system. *ANSI Standard* is technically a misnomer because ANSI does not develop the standards. ANSI Standards are actually developed by one of more than 230 ANSI-accredited SDOs, and then they are approved by ANSI's Board of Standards Review (BSR) as meeting certain criteria for openness, balance, due process, and consensus in standards development [36].

ANSI verification does not include any evaluation or review of the standard. ANSI audits the SDO to ensure that complete records are retained and that the records fully substantiate the decision to certify that due process was achieved. Proposals for new ANSI standards and proposals to revise, reaffirm, or withdraw approval of existing standards undergo a public comment period of 30–60 days. A revised standard based on public comments should be submitted to ANSI within 1 year from the close of the comment period.

5.2 NRC ENDORSEMENT

As the NRC prepares to review and regulate a new generation of non-LWRs, a vision and strategy has been developed to ensure NRC readiness to efficiently and effectively conduct its mission for these technologies [37, 38]. One of NRC's near-term strategies (i.e., 0–5 years) is to "Facilitate industry codes and standards needed to support the non-LWR life cycle (including fuels and materials)" [35].

Contributing activities for the near term include the following [37, 38]:

- Work with stakeholders to determine the currently available codes and standards applicable to non-LWRs and their associated fuels and waste and to identify the technical areas,
- Participate with the SDOs in developing codes and standards for non-LWRs, and
- Review codes and standards for endorsement.

The NRC's mid- and long-term action plans developed as part of the NRC non-LWR implementation action plans (IAPs) recognize that it has typically taken years to develop consensus codes and standards and to promulgate a new or revised regulation [39]. Contributing activities for mid- and long-term activities include the following:

- Continue efforts to ***facilitate development*** of industry codes and standards and
- Develop RGs and conduct rulemaking, as needed, to ***endorse*** industry codes and standards.

These activities will yield available consensus codes and standards endorsed by the NRC to improve the effectiveness and efficiency of the designing and licensing and regulation of non-LWR technologies. However, the NRC's endorsement of codes and standards—in either regulations or guidance—can only follow the development and issuance of the codes and standards by SDOs.

The NRC's endorsement process is described in Management Directive (MD) 6.5 [2, 3], which states that NRC's participation in the development and use of consensus standards consists of three steps:

1. Identifying and prioritizing needed new and revised technical standards,
2. Participation in codes and standards development, and
3. Endorsement of codes and standards.

In the case of a new consensus standard, MD 6.5 states that "it is preferable to determine how the consensus standard is to be used before the consensus standard is written."

NRC endorses consensus standards through incorporation by reference in regulations and through reference in such documents as RGs, NUREG reports, and the SRP. MD 6.6 describes the method used to endorse standards and the process for issuing an RG [40]. The process and timing of a standard being incorporated into a regulation or being approved as guidance is not addressed in this review.

6. CONCLUSIONS

This project represents a first look at the magnitude of work required to use standards in the design and licensing of MSRs. This project shows that 19 of consensus standards endorsed by RGs, approved by the SRP, or required by the CFR would require significant revisions. An additional 14 standards should also be created.

A prioritization must be developed for ranking the standards. In terms of advanced reactor design and licensing, the 14 new standards should be a priority, and the 19 standards requiring significant changes should be a close second. Furthermore, requests must be submitted using a staggered schedule to prevent overwhelming an SDO. Related to this is consideration of standards that require coordination between SDOs or between an SDO and industry. Prioritization should consider those standards that address cross-cutting topics (i.e., adequacy and completeness to several advance reactor designs). Other options could include providing volunteer SDOs with technical guidance or directly providing resources (perhaps including funding) for new standards development. Delays in addressing these changes will directly affect the licensing timeline and commercial deployment.

To aid in ranking the 14 new standards and 19 standards requiring significant changes, the top 4 in each category were identified to prioritize efforts. These topics cover the broadest spectrum of proposed MSR designs.

The top four new standards should address:

1. Where the fuel salt boundary interfaces with a structure, system, or component containing fluid that if allowed to freely interact with the fuel salt would cause the loss of a safety function, the interface location shall be designed to ensure that the fuel salt is separated from the fluid by two redundant, passive barriers. This is derived from SFR-DC 78 and ANS 20.2, Criterion 74 and describes the safety function of interfaces.

2. The information on the reactor fuel should include a description of the required characteristics and its fuel development program under which all fuel characteristics and parameters that are important to the safety operation of the reactor were investigated. The closest standard approved for use (SRP 4.2) is ASTM C776-89, Part 45, Standard Specification for Sintered Uranium Dioxide Pellets, which specifies the chemical, nuclear, and physical characteristics of UO2 pellets. A standard similar to ASTM C776-89 is needed to address fuel characteristics for MSRs.

3. Systems shall be provided as necessary to maintain the composition of the fuel salt within specified limits. These limits shall be based on the ability of the fuel salt to perform its safety functions. This new criteria should be reflective of the role of the fuel salt in the overall facility safety. Fuel salt is an essential element of providing adequate containment, heat removal, and reactivity control. Fuel salt properties are determined by its composition, which must be maintained within acceptable limits. This is derived from ANS 20.2, Criterion 71.

4. A new standard would be similar to ANSI/ANS 6.4-2006, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants. Because the types of steel, concrete, and source term and function of the concrete may differ significantly from those addressed in ANSI/ANS 6.4-2006, a new standard is recommended rather than revising the existing standard.

The top four standards that require extensive revisions for applicability to MSRs are:

1. ASME QME-1 provides the requirements and guidelines for the qualification of active mechanical equipment whose function is required to ensure the safe operation or safe shutdown of a nuclear facility. In addition to requirements and guidelines put forth in this Standard, the active mechanical equipment shall comply with the requirements of the applicable design and construction codes and standards. As MSRs will rely on passive mechanical equipment, this standard should be updated to provide guidance.

2. The four goals of NFPA 805, and thus NEI 04-02, are: the nuclear safety goal, the radioactive release goal, the life safety goal, and the plant damage/business interruption goal. Many fire issues addressed in NFPA 805 are specific/involve BWR and PWR specific designs. Changes require addressing MSR-specific fire issues.

3. ASME N509-2002 covers requirements for the design, construction, and qualification and acceptance testing of the air-cleaning units and components that make up Engineered Safety Feature (ESF) and other High efficiency air and gas treatment systems used in nuclear power plants. Because ASME AG-1 supplements ASME N509-2002, it is this relationship that should be reviewed more closely.

4. Molten salt aerosols and byproducts may affect the applicability of this standard. ASME AG-1-2009 provides requirements for the performance, design, fabrication, installation, inspection, acceptance testing, and quality assurance of equipment used in air and gas treatment systems in nuclear facilities. Materials of construction for all components and accessories shall conform to the ASME or ASTM material specifications listed in Table AA-3100. Because of the presence of molten salt, the list of allowable materials listed in Table AA-3100 may need to be updated for MSRs. The Process Gas section is incomplete and needs to be completed. The entire section needs to address the use of a cover gas such as helium.

Given the large number of codes and standards identified as needing to be developed or revised to accommodate advanced non-LWRs, it is recommended that industry and federal government resources be focused on those that are the highest priority to near-term deployment of the advanced reactor designs.

The use of codes and standards is expected to be an integral part of the NRC's strategy to improve its readiness to regulate non-LWR technologies. If a consensus standard is not available, then NRC can create its own guidance. If a standard is available, then NRC must justify why it is not being used. There is a great advantage to industry if they participate in creating standards rather than have a standard or guidance imposed. Regardless, advanced reactor technology licensing and deployment will likely be significantly delayed if applicable and endorsed standards are not available for use by technology developers and the NRC. Delays in providing the NRC with the knowledge base and tools for reviewing non-LWR applications will increase the effort needed to review an application, thus delaying its approval.

Designs can proceed without approved standards, but the benefits of approved standards include their ability to aid in obtaining multiple licensees. The use of standards is an integral part of the NRC's strategy to improve its readiness to regulate non-LWR technologies (IAP).

Based on the results of this study, it is recommended that future efforts focus on the following:

- Work with DOE, NRC, and industry to develop guidance applicable to the MSR designs.
- Begin to set up a database of component failure rates with adjustments to account for the environment encountered in an MSR. This moves toward the licensing and application phase for MSRs.
- Review the documents endorsed, approved for use, or required from non-SDOs for applicability to MSRs.
- Review the inactive/withdrawn standards to determine if they should be updated and approved by the SDO.

7. REFERENCES

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APPENDIX A. REVIEW OF SDO-BASED STANDARDS

ID	Standard	Standard title	RG endorsing standard	RG (or CFR) cited in SRP	Standard accepted in SRP	Standard required by CFR	SDO	Level of Effort 1=none 2=limited changes 3=substantive changes needed 4=insufficient design info 5=not applicable 6=new design-specific requirement	Priority High — impacts design or licensing Medium — reduce component fab or plant construction time, O&M costs Low — other impact not cited in High or Medium or LOE 1, 2, or 5	Key Technical Issues	Comments, Notes
1.06901	ACI 349-2013 ACI 349-06	Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary	1.69-1 1.142-3 1.199-1	SRP 3.8.3 SRP 3.8.4 SRP 12.3-12.4 SRP 3.8.4 SRP 3.8.3 SRP 3.8.4	SRP 3.8.3 SRP 3.8.4 SRP 3.8.5	—	ACI	2	High	This Code provides minimum requirements for design and construction of nuclear safety-related concrete structures and structural members for nuclear facilities. Safety-related structures and structural members subject to this Code are those concrete structures that support, house, or protect nuclear safety class systems or component parts of nuclear safety class systems. Specifically excluded from this Code are those structures covered by “Code for Concrete Containments,” ASME Boiler and Pressure Vessel Code Section III, Division 2, and pertinent General Requirements (ACI 359). Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	Elevated temperatures, even slightly, must be investigated. MSRs offer potential for reaction product generation different from those associated with clad metal-water interactions. Therefore, hydrogen generation may need to be changed.
1.06901	ACI 349.1R-07	Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures	1.69-1	SRP 3.8.3 SRP 3.8.4 SRP 12.3-12.4	—	—	ACI	2	High	ACI 349.1R presents a design-oriented approach for considering thermal effects on reinforced concrete structures. The standard should be reviewed to ensure that the ΔT’s at non-LWRs are still applicable. In addition, the term “Loss of coolant accidents (LOCAs)” should be replaced with “postulated accidents.” This Code covers the design and construction of concrete structures inside and outside the containment system. Thermal effects cause expansion or contraction of the components in a structural system. If the components are restrained, stresses are induced. There are three types of thermal effects: Bulk temperature change. In this case, the entire structural component (or segments of the component) is subject to a uniform temperature change; Thermal gradient. A temperature cross-fall or thermal gradient is caused by different thermal conditions on two faces of a structure, such as two sides of a wall or the top and bottom of a beam; and Local thermal exposure. Elevated temperature at a local surface caused by an external source such as operating equipment or piping or an abnormal event such as a fire. The High temperature and constant exposure for passive heat removal systems needs to be reviewed.	Thermal effects can arise from many sources including, but not limited to, process fluid transport; proximity to hot gases, steam, or water passage (for example, reactor vessel or steam piping from reactor building to turbine); fire; or gradients formed when opposing faces of a structure are exposed to differing temperatures (for example, spent fuel pool) or cyclic gradients from plant startup and shutdown.
1.19901	ACI 355.2-07	Qualification of Post-Installed Mechanical Anchors in Concrete and Commentary	1.199-1	SRP 3.8.3 SRP 3.8.4	—	—	ACI	1	Low	ACI 355.2 prescribes the testing programs required to qualify post-installed mechanical anchors for use with the design method of ACI 318-19 Chapter 17, where it is assumed that anchors have been tested either for use in uncracked concrete or for use in cracked and uncracked concrete. This testing is performed in concrete specimens controlled by the testing laboratory as a means of simulating concrete, both cracked and uncracked, that might occur in actual structures. ACI CODE-355.2 is intended to develop the data required by ACI 318-19 Chapter 17 to confirm an anchor’s reliability and place it in the appropriate anchor category.	
3.01201	AGS-G001	Guideline for Gloveboxes	3.12-1	—	—	—	AGS	1	Low	Use of this RG assumes MSR fuel fabrication may be implemented at a facility on or adjacent to the reactor site.	Technology Neutral Standard
—	ANS 5	Decay Energy Release Rates Following Shutdown of	—	SRP 4.2 SRP 4.4 SRP BTP 4-1 SRP 6.1.1 SRP 6.2.1	—	10 CFR 50, Appendix K	ANS	3	High	This standard covers the heat generation rates from radioactive decay of fission products. Research is needed for heat generation rates in MSRs.	Heating from the decay of radioactive nuclides in shutdown reactors plays an important role in the safety evaluation of nuclear power plants. Although there are many other important uses for this

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		Uranium-Fueled Thermal Reactors		SRP 6.2.1.3 SRP 6.2.1.5 SRP 6.2.1.1.A SRP 6.2.4 SRP 6.3 SRP BTP 6-2 SRP BTP 6-4 SRP 15.0.2 SRP 15.6.5							information, the need for more accurate data for the analysis of hypothetical reactor accident scenarios has been the main impetus for recent research activity that has led to a major revision of the Draft American Nuclear Society 5.1 Standard, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Reactors" (published in 1971). The 1978 revised standard, titled "Decay Heat Power in Light Water Reactors," is based on new experiments and summation calculations. Very accurate determination of the decay heat is now possible for light water reactors, especially within the first 10^4 s after shutdown, where the influence of neutron capture in fission products may be treated as a small correction to the idealized zero capture case. The new standard accounts for differences among fuel nuclides. It covers cooling times to 10^9 s, but provides only an "upper bound" on the capture correction in the interval from 10^4 to 10^9 s. The current version is ANSI/ANS-5.1-2014.
—	ANSI MH5.1 (1971)	Basic Requirements for Cargo Containers	—	—	—	10 CFR 73.26	ANSI	1	Low	10 CFR 73.26 requires that *Shipment by sea* shall be made only on container-ships. The strategic special nuclear material container(s) shall be loaded into exclusive use cargo containers conforming to American National Standards Institute ANSI) Standard MH5.1—"Basic Requirements for Cargo Containers" (1971) or International Standards Organization (ISO) 1496, "General Cargo Containers" (1978). Locks and seals shall be inspected by the escorts whenever access is possible.	
3.02100	ANSI N101.4-72	Quality Assurance for Protective Coatings Applied to Nuclear Facilities	3.21-0	—	—	—	ANSI	5	Low	ASTM D3843-16 (R2021) is most recent and broadly applicable standard to cover coatings. Although the standard covers requirements applicable to quality assurance-level protective coatings for safety-related nuclear facilities, the RG is cited for fuel reprocessing and to plutonium processing and fuel fabrication plants. Quality assurance-level protective coatings are those coatings whose failure could adversely affect the operation of those structures, systems, and components of safety-related nuclear facilities that are essential: (1) to prevent postulated accidents that could affect the public health and safety, or (2) to mitigate the consequences of these accidents. The practice includes a discussion of general quality assurance requirements, control of selection and qualification of coating materials, control of coating manufacturing, control of coating application and subsurface preparation of substrates, control of coating inspection, and records.	Use of this RG assumes MSR fuel fabrication may be implemented at a facility on or adjacent to the reactor site.
5.02902	ANSI N15.8-2009	Methods of Nuclear Material Control—Material Control Systems—Special Nuclear Material Control and Accounting	5.29-2	—	—	—	ANSI	1	Low	Physical Protection Standard is applicable to MSRs	Reaffirmed in 2015

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		Systems for Nuclear Power Plants									
3.04001	ANSI N170-76	Determining Design Basis Flooding at Power Reactor Sites	3.40-1	—	—	—	ANSI	5	Low	ANSI N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites," presents standards to establish design basis flooding for safety-related features at power reactor sites. ANSI N170-1976 also contains, among other things, methodology for estimating probable maximum surges and seiches at estuaries and coastal areas on oceans and large lakes. Although flooding is of concern at NPPs, the RG that endorses this is for plutonium processing and fuel fabrication plants. ANSI/ANS 2.8 is more recent.	First published as ANSI N170/ANS-2.8-1976. Now ANSI/ANS 2.8-2019
—	ANSI N432-1980 published as NBS Handbook 136	Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography	—	—	—	10 CFR 34.20	ANSI	1	Low	This standard applies to the design and construction of apparatus used for industrial gamma radiography which employs radioactive material as the energy source. It establishes the criteria to be used in the proper design and construction of the various components to ensure a high degree of radiation safety at all times. This includes the classification and labeling criteria for the exposure device; and factors which should be considered in the design and construction of exposure devices, controls, and source assemblies. The testing procedures and equipment for the various classifications of the exposure devices and source assemblies are detailed.	
—	ANSI N45.2.1-1973	Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants	—	—	SRP 10.2 SRP 10.3.6	—	ANSI	1	Low	This Standard covers the management of cleaning and cleanness control of fluid systems and associated components for nuclear power plants during manufacturing, construction, repairs, and modifications. This Standard is not a procedure for cleaning and cleanness control but provides a basis for development of such procedures. This Standard requires close attention to cleanness control so that only flushing or rinsing may be required to render the item ready for service. When more than a flush or rinse is needed to produce the specified cleanness, additional cleaning, in accordance with this Standard, may be necessary.	
1.08800	ANSI N45.2.9-1974	Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants	1.88 RG 1.88 was withdrawn in 1991; however, the standard is approved for use in SRP 17.1	SRP 17.1	SRP 17.1	—	ANSI	1	Low	NQA-1 is a multipart Standard that provides includes requirements and nonmandatory guidance to establish and implement a QA program for any nuclear facility application. Part I contains QA program requirements for the siting, design, construction, operation, and decommissioning of nuclear facilities. Part II contains QA requirements for the planning and conducting of the fabrication, construction, modification, repair, maintenance, and testing of systems, components, or activities for nuclear facilities. Part III contains nonmandatory guidance. Part IV contains NQA position papers and other quality program information.	The NRC staff performed a review and identified that differences exist between the previously endorsed guidance (NQA-1-2008 and NQA-1a-2009 addenda) and the most recently issued guidance (NQA-1b-2011, NQA-1-2012 and NQA-1-2015). Additional time and resources are required to understand the impact of these changes. The NRC staff continues to endorse the previous guidance and is not aware of any issues that would preclude its use.
—	ANSI S3.6-1969	Specifications for Audiometers	—	—	—	10 CFR 73, Appendix B	ANSI	1	Low	Hearing: (a) Individuals shall have no hearing loss in the better ear greater than 30 decibels average at 500 Hz, 1,000 Hz, and 2,000 Hz with no level greater that 40 decibels at any one frequency by ISO 389 ANSI S3.6-1969.	
1.24300	ANSI/AISC N690-18	Specification for Steel-Related Steel Structures for Nuclear Facilities	1.243-0	—	SRP 3.5.3 SRP 3.8.3 SRP 3.8.4 SRP 3.8.5	—	AISC	1	Low	Structures and structural elements subject to the Nuclear Specification are those steel structures and structural elements that are part of a safety-related system or that support, house or protect safety-related systems or components, the failure of which could credibly result in the loss of capability of the structure, system or	

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										component to perform its safety functions. Concrete that is part of steel-plate composite (SC) walls is also subject to the Nuclear Specification. Safety categorization for nuclear facility steel structures and structural elements shall be the responsibility of the owner and shall be identified in the contract documents.	
1.16601	ANSI/ANS 2.10-2017	Criteria for Retrieval, Processing, Handling, and Storage of Records from Nuclear Facility Seismic Instrumentation	1.166-1	SRP 3.7.4	—	—	ANS	1	Low	The primary purpose of this standard is to provide criteria for retrieval, processing, handling, and storage of records from nuclear facility seismic instrumentation and to specify related activities such that a high-quality standard is achieved in obtaining pertinent information from the seismic instrumentation to adequately support the decision making at power and non-power nuclear facilities following an earthquake event	This standard describes methods for qualifying static battery chargers and inverters for Class 1E installations outside containment in nuclear power generating stations.
1.01203	ANSI/ANS 2.2-2016	Earthquake Instrumentation Criteria for Nuclear Power Plants	1.12-3	SRP 3.7.4	—	—	ANS	2	Low	The purpose of ANSI/ANS 2.2-2016 is to specify the minimum requirements for earthquake instrumentation. This standard defines the minimum requirements for an earthquake instrumentation system to be installed at nuclear power plants. These instruments are intended to provide timely (within 4 hours) information on the earthquake ground motion at the site and the corresponding response vibratory motion of Seismic Category I structures, when subjected to earthquake ground motion. By comparing this information with the vibratory motions used in the facility's seismic design,2) an evaluation can be made as to whether or not the design basis vibratory motions have been exceeded.	The references to light-water cooled plants is not of concern.
1.16601	ANSI/ANS 2.23-2016	Nuclear Power Plant Response to an Earthquake	1.166-1	SRP 3.7.4	—	—	ANS	1	Low	ANS 2.23 describes actions that the owner of a nuclear power plant shall take to prepare for and respond to a felt earthquake at the plant(s), including the need for plant shutdown; actions to determine the readiness of the plant to resume operation; and those evaluations necessary to verify the long-term integrity of safety-related (SR) and important structures, systems, and components (SSCs).	
3.04801	ANSI/ANS 2.8-1981	Probabilistic Evaluation of External Flood Hazards for Nuclear Facilities	3.48-1	—	SRP 2.4.2 SRP 2.4.4 SRP 2.4.5 SRP 2.4.6	—	ANS	1	Low	The purpose of this standard is to present requirements for performing a probabilistic flood hazard assessment (PFHA) for a nuclear facility. This standard also can be used to update an existing PFHA. This standard also provides guidance for conducting efficient, site-specific screening analyses to screen out noncredible external flood hazard sources and mechanisms. A PFHA considers sources of flooding and flooding mechanisms that pose a credible hazard to a nuclear facility. Results from the PFHA can be used in conjunction with applicable authority or regulatory body requirements to evaluate the design-basis flood (DBF) for a nuclear facility or for individual structures, systems, and components (SSCs) important to safety. PFHA results also may be used as input to external flood probabilistic risk assessments (PRAs), for development of emergency response plans, for assessment of safety-related SSCs and site vulnerability, and to design flood protection or mitigation systems.	Most recent version is 2019. RG 3.48-1 refers to 1981 version. 1981 version provides for a deterministic evaluation of flood hazards at reactors. 2019 version provides for probabilistic evaluation of flood hazards at all nuclear facilities. Technology Neutral Standard Endorsed in RG 3.40-1 (superseded version) and RG 3.48-1
1.00803	ANSI/ANS 3.1-2014 (SRP says RG 1.8 endorsed -1993 version)	Selection, Qualification, and Training of Personnel for Nuclear Power Plants	1.8-3 5.73-0	SRP 12.1 SRP 12.3-12.4 SRP 12.5 SRP 13.1.1 SRP 13.1.2-13.1.3 SRP 13.2.1 SRP 13.2.2	SRP 13.2.1 SRP 13.2.2 SRP 13.5.1 SRP 13.5.1.2 SRP 17.1	—	ANS	2	Low	The purpose of this standard is to provide guidance for functional levels and job positions as they exist in the operating organization. Qualification requirements include education, experience, and training. This standard provides qualification guidance to meet the particular organizational needs that are derived from the requirements contained in this standard.	Based on its periodic review, the NRC plans to revise this RG following the formal review of ANS 3.1-2014. (The RG 1.8 endorses ANS 3.1-1993.) The experience requirements will be NPP experience because non-LWRs are not in operation in the U.S.

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				SRP 17.1 SRP 17.5						Requirements for experience at a comparable facility and equivalent position will need to be addressed for SRO and RO. Other managerial and staff requirements seem applicable.	
1.03303	ANSI/ANS 3.2-2012	Managerial, Administrative, and Quality Assurance Controls for Operational Phase of Nuclear Power Plants	1.33-3 1.189-4	SRP BTP 5-4 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP BTP 11-6 SRP 12.1 SRP 12.5 SRP 13.1.1 SRP 13.1.3 SRP 13.5.1.1 SRP 13.5.1.2 SRP 13.5.1 SRP 13.5.2.1 SRP 13.5.2.2 SRP 17.2 SRP 17.3 SRP 17.5 SRP 3.2.1 SRP 7.4 SRP Appendix 7.1-A SRP Table 7-1 SRP 9.5.1.1 SRP 9.5.1.2 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP BTP 11-3 SRP 13.1.2 SRP 13.2.1 SRP 17.5	SRP 13.5.1.1 SRP 13.1.2-13.1.3 SRP 13.5.1 SRP 13.5.2.1 SRP 13.5.2.2 SRP 17.5	—	ANS	2	Low	This standard provides requirements for implementing managerial and administrative controls consistent with requirements of 10 CFR 50, Appendix B. Appendix A of the standard provides typical procedures for PWRs and BWRs, however this appendix is not part of the standard. The managerial and administrative controls provided in the standard are applicable to non-LWRs. However, the operating organization should have knowledge of molten salts; standard needs to be updated to reflect this. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	Minor changes to endorsed (normative) part of standard. An appendix similar to Appendix A (informative) of the standard, which provides typical procedures for PWRs and BWRs, should be developed to provide typical procedures for non-LWRs.
1.13403	ANSI/ANS 3.4-1996	Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants	1.134-3	—	—	—	ANS	2	Low	This standard defines the physical and mental requirements in order to be licensed as a nuclear reactor operator. It also addresses the content, extent, and methods of examination. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	Remove acronyms for BWR and PWR and references to them. Training requirement requires training on a comparable facility. Because a comparable facility does not exist this requirement will need to be modified.
1.14904	ANSI/ANS 3.5-2009	Nuclear Power Plant Simulators for Use in Operator Training and Examination	1.149-4	SRP 13.2.1 SRP 13.2.2	SRP 13.2.1 SRP 18.0 SRP Appendix 8-A	—	ANS	3	Low	This standard establishes the functional requirements for full-scope nuclear power plant control room simulators for use in operator training and examination. The standard also establishes criteria for the scope of simulation, performance, and functional capabilities of simulators. This standard establishes the functional requirements for full-scope nuclear power plant control room simulators for use in operator training and examination. The standard also establishes criteria for the scope of simulation, performance, and functional capabilities of simulators. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs. The standard is LWR specific. It needs to be updated for	Although the general concept of the standard is applicable to non-LWRs, the standard is LWR specific. Although Appendix B (informative) provides guidelines for the conduct of simulator operability testing, these tests consist of running the transient events identified in Sec. B.2 for BWRs and B.3 for PWRs. A section should be added for MSRs.

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										an MSR. For example, loss of coolant accidents (LOCAs) should be replaced with postulated accidents. Emergency electric power may not be applicable.	
3.06000	ANSI/ANS 5.1-1979 ANSI/ANS 5.1-2014	Decay Heat Power in Light Water Reactors	3.60-0 3.54-3	—	SRP 9.2.5	—	ANS	3	Medium	LWR-specific. Require calculation of decay heat in molten salts.	Updated 2014 and Reaffirmed 2019. RG 54 references 2014 version. RG 60 references 1979 version. Need decay heat calculations for the constituents of molten fuel salts, including recycled LWR fuels and fuels containing U-233 and Thorium. Endorsed in RG 3.54-3 and RG 3.60-0
—	ANSI/ANS 5.4	Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel	—	—	SRP 4.2	—	ANS	5	Low	This standard provides an analytical method for calculating the release of volatile fission products from uranium dioxide ~UO 2! Fuel pellets during normal reactor operation. When used with nuclide yields, this method will give the release-to-birth ratio R0B or the so-called “gap release,” which is the inventory of volatile radioactive fission products that could be available for release from the fuel rod if the cladding were breached.	
1.14101	ANSI/ANS 56.2-1984 (ANSI N271-1976)	Containment Isolation Provisions for Fluid Systems	1.141-1	SRP 6.2.1 SRP 6.2.4	—	—	ANS	3	Low	The purpose of ANSI ANS 56.2 (ANSI N271-1976) is to specify minimum design requirements for fluid systems that penetrate the primary containment boundary of LWRs to provide for isolation of the containment after a LOCA. Some containment penetrations in an MSR may not provide a release path to the atmosphere. The appendices illustrate methods of application of the standard for BWRs and PWRs but are not mandatory or part of the standard. These would have to be replaced for an MSR but would still be a nonmandatory part of the standard.	Not all penetrations will provide a release path to the atmosphere. In fact, containment isolation valves may not be required. The requirement will be to have the redundancy, reliability, and performance capabilities necessary to perform the containment safety function and which reflect the importance to safety of preventing radioactivity releases from containment through these piping systems.
1.13702	ANSI/ANS 59.51-1997	Fuel Oil Systems for Safety-Related Emergency Diesel Generators	1.137-2	SRP 9.5.4	SRP 9.5.4	—	ANS	1	Low	The purpose of this standard is to define those features of fuel oil systems required to ensure an adequate fuel supply to safety-related emergency diesel generators, and to provide performance and design criteria to ensure sufficient fuel is available for supply to the emergency diesel generators under all plant conditions. Although the criteria may be useful, it is unknown if MSRs will use Class 1E emergency DGs. The fuel oil system shall be capable of supplying an adequate supply of suitable fuel oil to the emergency diesel generators under all Plant Conditions that are defined ANSI/ANS-51.1-1983 (for PWRs) and ANSI/ANS-52.1-1983 (for BWRs). Both ANS 51.1 and ANS 52.1 have been withdrawn so replacement with an MSR-specific set of plant conditions would be necessary.	RG 1.137 R1 endorsed ANSI N195-1976. This standard was revised as ANSI/ANS 59.51-1977. RG 1.137 R2 endorses ANSI/ANS 59.51-1977.
1.06901	ANSI/ANS 6.3.1-1987 (R2007)	Program for Testing Radiation Shields in Light Water Reactors (LWR)	1.69-1	SRP 3.8.3 SRP 3.8.4 SRP 12.3-12.4	—	—	ANS	2	Low	This standard describes a test program to be used in evaluating biological radiation shielding in nuclear reactor facilities under normal operating conditions including anticipated operational occurrences. The program encompasses examining and testing to be performed before startup, during startup, and testing subsequent to the startup phase. Post startup tests are required for the shielded components which do not contain sufficient radioactivity during the startup phase to allow valid testing. Post startup shield tests are also required whenever radioactive or potentially radioactive equipment which could affect the adequacy of the installed shielding is introduced into the plant or relocated within the plant, or when previously tested	Nuclear heating shall be considered during the determination of the operating temperature and water content of a concrete primary reactor shield and of any other concrete shields that are exposed to an incident energy flux greater than 1010 MeV0cm2 s and that will operate at a temperature of 65°C or greater. concrete shielding designed to protect plant personnel should be tested in accordance with American National Standard “Program for Testing Radiation Shields in Light

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										shielding has been modified. A simple change is to remove LWR-specific words. Loss of coolant accidents (LOCAs) should be replaced with postulated accidents. The collection of concrete shielding data presented in this standard is applicable to most practical nuclear power plant shielding designs; however, the data should be reviewed for adequacy and completeness to non-LWR designs with fast spectrums.	Water Reactors (LWR),” ANSI-ANS 6.3.1-1987 (R1998).
1.06901	ANSI/ANS 6.4-2006	Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants	1.69-1	SRP 3.8.3 SRP 3.8.4 SRP 12.3-12.4	—	—	ANS	3	Low	This standard contains methods and data needed to calculate the concrete thickness required for radiation shielding in nuclear power plants. Appendix C, which is not a part of the standard, provides Gamma-ray dose rates from Schedule 80 and Schedule 160 steel pipe, containing airborne or waterborne radioactivity.	Types of steel, concrete, and source terms may differ greatly for non-LWRs.
3.07103	ANSI/ANS 8.1-2014	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors	3.71-3	—	—	—	ANS	1	Low	Use of RG assumes MSR fuel fabrication may be implemented at a facility on or adjacent to the reactor site. Materials for onsite fuel preparation system not defined.	Current version is 2018, Reference lists 2014 version. Technology Neutral Standard
3.07103	ANSI/ANS 8.10-2015	Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement	3.71-3	—	—	—	ANS	1	Low	Applicable to operation outside of an MSR in which shielding and confinement are provided for U-233, U-235, Pu-239, and other fissionable materials.	2015 Version reaffirmed in 2020. Technology Neutral Standard
3.07103	ANSI/ANS 8.12-1987 (R2016)	Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors	3.71-3	—	—	—	ANS	1	Low	Applicable to operation outside of an MSR with fissionable materials.	1987 Version reaffirmed in 2021. Technology Neutral Standard
3.07103	ANSI/ANS 8.14-2004 (R2016)	Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors	3.71-3	—	—	—	ANS	1	Low	Applicable to operation outside of an MSR with fissionable materials.	2004 Version reaffirmed in 2019. Technology Neutral Standard
3.07103	ANSI/ANS 8.15-2014	Nuclear Criticality Safety Control of Selected Actinide Nuclides	3.71-3	—	—	—	ANS	1	Low	Applicable to operation outside of an MSR involving nuclides other than U-233, U-235, and Pu-239.	2014 Version reaffirmed in 2019. Technology Neutral Standard
3.07103	ANSI/ANS 8.17-2004 (R2014)	Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors	3.71-3	—	—	—	ANS	5	Low	Standard is specific to handling, storage, and transport of LWR fuel (including individual fuel rods) outside of the reactor core. Consideration of fuel salts is necessary.	Reaffirmed 2019
3.07103	ANSI/ANS 8.19-2014	Administrative Practices for Nuclear Criticality Safety	3.71-3	—	—	—	ANS	1	Low	Standard provides criteria for the administration of a nuclear criticality safety program for operations with fissile materials outside of nuclear reactors. Addresses the responsibilities of management, supervision, and nuclear criticality safety staff.	2014 Version reaffirmed in 2019. Technology Neutral Standard
3.07103	ANSI/ANS 8.20-1991 (R2015)	Nuclear Criticality Safety Training	3.71-3	—	—	—	ANS	1	Low	Standard provides criteria for nuclear criticality safety training for personnel associated with operations outside reactors where a potential exists for criticality accidents.	1991 Version reaffirmed in 2020. Technology Neutral Standard
3.07103	ANSI/ANS 8.21-1995 (R2011)	Use of Fixed Neutron Absorbers in Nuclear	3.71-3	—	—	—	ANS	1	Low	Standard provides guidance for the use of fixed neutron absorbers as an integral part of process equipment outside reactors, where such absorbers provide criticality safety control.	1995 Version reaffirmed in 2019. Technology Neutral Standard

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		Facilities Outside Reactors									
3.07103	ANSI/ANS 8.22-1997 (R2016)	Nuclear Criticality Safety Based on Limiting and Controlling Moderators	3.71-3	—	—	—	ANS	1	Low	Standard applies to limiting and controlling moderators to achieve criticality safety in operations with fissile materials in a moderator control area.	1997 Version reaffirmed in 2021. Technology Neutral Standard
3.07103	ANSI/ANS 8.23-2007 (R2012)	Nuclear Criticality Accident Emergency Planning and Response	3.71-3	—	—	—	ANS	1	Low	Standard provides criteria for minimizing risks to personnel during emergency response to a nuclear criticality accident outside reactors. The criteria address management and technical staff responsibilities, planning, equipment, evacuation, rescue, reentry, stabilization, classroom training, drills, and exercises.	Current version is 2019, Reference lists 2007 version. Technology Neutral Standard
3.07103	ANSI/ANS 8.24-2017	Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations	3.71-3	—	—	—	ANS	1	Low	Standard provides requirements and recommendations for validation, including establishing adequacy and completeness, of neutron transport calculational methods used in determining critical or subcritical conditions for nuclear criticality safety analyses.	May require some additional consideration for Liquid fuel salt in an MSR.
3.07103	ANSI/ANS 8.26-2007 (R2016)	Criticality Safety Engineer Training and Qualification Program	3.71-3	—	—	—	ANS	1	Low	Standard presents the fundamental content elements of a training and qualification program for individuals with responsibilities for performing the various technical aspects of criticality safety engineering.	2007 Version reaffirmed in 2022. Technology Neutral Standard
3.07103	ANSI/ANS 8.27-2015	Burnup Credit for LWR Fuel	3.71-3	—	—	—	ANS	4		Standard provides criteria for accounting for reactivity effects of fuel irradiation and radioactive decay in criticality safety control of storage, transportation, and disposal of commercial LWR UO2 fuel assemblies.	Reaffirmed 2020
3.07103	ANSI/ANS 8.3-1997 (R2017)	Criticality Accident Alarm System	3.71-3	—	—	—	ANS	1	Low	Applicable to MSRs	Current version is 2022, Reference lists 1997 version. Technology Neutral Standard
3.07103	ANSI/ANS 8.5-1996 (R2017)	Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material	3.71-3	—	—	—	ANS	1	Low	Use of RG assumes MSR fuel fabrication may be implemented at a facility on or adjacent to the reactor site. Materials for onsite fuel preparation system not defined.	—
3.07103	ANSI/ANS 8.6-1983 (R2017)	Safety in Conducting Subcritical Neutron-Multiplication Measurements in Sit	3.71-3	—	—	—	ANS	1	Low	Use of RG assumes MSR fuel fabrication may be implemented at a facility on or adjacent to the reactor site. Materials for onsite fuel preparation system not defined.	1983 Version reaffirmed in 2022. Technology Neutral Standard
3.07103	ANSI/ANS 8.7-1998 (R2017)	Nuclear Criticality Safety in the Storage of Fissile Materials	3.71-3	—	—	—	ANS	1	Low	Standard is applicable to the storage of fissile materials. Biased toward LWR fuel assemblies.	Current version is 2022, Reference lists 1998 version.
1.18904	ANSI/ASME B31.1	Power Piping	1.189-4	SRP 3.2.1 SRP Appendix 7.1-A SRP 7.4 SRP 9.5.1 SRP 9.5.1.1 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP BTP 11-3 SRP 11.4 SRP 11.5 SRP 13.1.2-13.1.3 SRP 17.5	SRP BTP 3-1 SRP BTP 3-2 SRP BTP 3-3 SRP BTP 3-4 SRP 3.3	—	ASME	1	Low	The ASME B31 Code for Pressure Piping consists of a number of individually published Sections, each an American National Standard, under the direction of ASME Committee B31, Code for Pressure Piping. This is the B31.1 Power Piping Code Section. The general philosophy underlying this Power Piping Code is to parallel those provisions of Section I, Power Boilers, of the ASME Boiler and Pressure Vessel Code, as they can be applied to power piping systems. The allowable stress values for power piping are generally consistent with those assigned for power boilers. This Code is more conservative than some other piping codes, reflecting the need for long service life and maximum reliability in power plant installations.	

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										The endorsement of ASME B31.1 is for fire protection system piping and headers.	
5.05301	ANSI/ASTM E178-80	Practice for Dealing with Outlying Observations	5.53-1	—	—	—	ASTM	1	Low	Physical Protection Standard is applicable to MSRs	Most recent version is 2020. RG refers to 1987 version.
5.01201	ANSI/BHMA A156.2 -2003	American National Standard for Bored and Preassembled Locks & Latches	5.12-1	—	—	—	BHMA	1	Low	Physical Protection Standard is applicable to MSRs	Current revision is 2017
5.01201	ANSI/BHMA A156.25-2013	Electrified Locking Devices	5.12-1	—	—	—	BHMA	1	Low	Physical Protection Standard is applicable to MSRs	
5.01201	ANSI/BHMA A156.5-2014	Cylinders and Input Devices for Locks	5.12-1	—	—	—	BHMA	1	Low	Physical Protection Standard is applicable to MSRs	
1.15102	ANSI/ISA 67.02.01-2014	Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants	1.151-2	SRP 3.2.1 SRP 3.2.2 SRP Appendix 7.1-A SRO Appendix 7.1-B SRP Appendix 7.1-C SRP 7.5 SRP 7.7 SRP 11.5	SRP Appendix 7.1-A	—	ISA	3	Medium	Routing of instrument sensing lines in the standard are concerned with water level indication during and after rapid depressurization involving flashing, degassing, or non- condensable gas events has been identified in industry as a concern, specifically in the pressurizer reference legs of PWRs and reactor vessel water level instrumentation of BWRs and shall be considered. Sensing lines and level measurements will have different fluids and possibly types of sensors. Non-LWRs may also use optical sensors.	RG 1.151 endorses ANSI/ISA 67.02.01-1999. The 1999 revision of this standard does not have the correct information for air or gas sensing lines. Because NRC plans to revise RG 1.151 to address ISA 67.02-2014, ISO 2186-2007, and issues identified from operating experience, this review is based on NSI/ISA- 67.02.01-2014. Pressure and level measurements may use different technologies or apply existing technology in a different manner. Pressure measurements may use impulse lines, bubblers, or use direct measurement sensors. Level measurements may use guided-wave microwave, guided-wave ultrasonic, or heated lance. Temperature alone will require changes to the methodology for pressure and level measurements. Molten salt presents problems with visibility and does not boil which will eliminate some measurement techniques.
1.10504	ANSI/ISA S67.04.01-2018 (current version) ISA-S67.04-1994 (Part I) is withdrawn and replaced by ISA-S67.04.01-2006.	Setpoints for Nuclear Safety-Related Instrumentation	1.105-4	SRP 7.2 SRP 7.3 SRP 7.5 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-C SRP Table 7-1 SRP BTP 7-12 SRP BTP 7-21 SRP 15.1.1-15.1.4 SRP 15.2.1-15.2.5 SRP 15.2.6 SRP 15.2.7 SRP 15.3.1-15.3.2 SRP 15.4.4-15.4.5 SRP 15.5.1-15.5.2 SRP 15.6.1	SRP Appendix 7.1-A SRP BTP 7-12 SRP BTP 7-21	—	ISA	1	Low	Setpoints of nuclear safety-related instruments are selected such that resultant actions will correct the monitored condition or mitigate the consequences of the monitored condition. The uncertainties and combining those uncertainties should be the same for an MSR.	Part II of ISA-S67.04-1994, "Methodologies for the Determination of Setpoints for the Nuclear Safety-Related Instrumentation," is not endorsed by RG 1.105 R3.
5.02000	AR 40-501	US Army Standards of Physical Fitness	5.20-0	—	—	—	AR	1	Low	Physical Protection Standard is applicable to MSRs	Although a standard from the US Army is not a standard from an SDO, it is cited as standards and included in this review.

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3.03200	ASHRAE 52-68	Method of Testing Air Cleaning Devices Used in General Ventilation for Removing Particulate Matter	3.32-0	—	—	—	ASHRAE	1	Low	This standard (ASHRAE 58.1-2019 that superseded ASHRAE 58-68) establishes test procedures for evaluating the performance of air-cleaning devices for removing particulate matter, to establish specifications for the equipment required to conduct the tests, to define methods of calculation from the test data, and to establish formats for reporting the results obtained.	ASHRAE 52.68 (1971) has been superseded by ASHRAE 58.1-1992.
1.05204	ASME AG-1-2009	Code on Nuclear Air and Gas Treatment	1.52-4 1.140-3 3.12-1	SRP 6.1.1 SRP 6.2.3 SRP 6.3 SRP 6.4 SRP 6.5.1 SRP 6.5.3 SRP 6.5.5 SRP 9.1.2 SRP 9.1.3 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5 SRP 11.3 SRP 11.5 SRP 12.3-12.4 SRP 15.6.5 SRP 6.1.1 SRP 6.5.1 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5 SRP 11.1 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 12.3-12.4 SRP 14.2	SRP 3.9.2 SRP 6.1.1 SRP 6.4 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4	—	ASME	3	High	This Code provides requirements for the performance, design, fabrication, installation, inspection, acceptance testing, and quality assurance of equipment used in air and gas treatment systems in nuclear facilities. The code is divided into the following divisions: Division I: General Requirements, Division II: Ventilation Air Cleaning and Ventilation Division III: Process Gas Treatment, Division IV: Testing Procedures.	Materials of construction for all components and accessories shall conform to the ASME or ASTM material specifications listed in Table AA-3100. Because of the presence of molten salt, the list of allowable materials listed in Table AA-3100 may need to be updated for MSRs. The Process Gas section is incomplete and needs to be completed. The entire section needs to address the use of a cover gas such as helium.
1.09103	ASME B31.8S	Managing System Integrity of Gas Pipelines	1.91-3	SRP 2.2.1-2.2.2 SRP 3.5.1.5 SRP 3.8.1 SRP 3.8.4	—	—	ASME	1	Low	This Code applies to onshore pipeline systems that are constructed with ferrous materials and transport gas. The principles and processes embodied in integrity management are applicable to all pipeline systems. This Code is specifically designed to provide the operator with the information necessary to develop and implement an effective integrity management program using proven industry practices and processes. The processes and approaches described within this Code are applicable to the entire pipeline. RG 1.91 is concerned about the thermal effects or radiative heat flux exposure on power plant structures, the information in NUREG/CR-3330, 49 CFR Part 192.903, and ASME B31.8S is acceptable for determining the potential impact of radiative heat flux on power plant structures exposed to explosions and fires.	
1.14302	ASME BPVC Section II, Parts A, B, and C	ASME BPVC Section II, "Materials," Parts A, B, C, and D	1.143-2	SRP 3.8.3 SRP 3.8.4 SRP 12.3-12.4	SRP 3.13 SRP 4.5.1 SRP 4.5.2 SRP 10.3.6	10 CFR 50.55a	ASME	3	High	The approved tables 1 and 2 are specific to LWRS only. There may be some of the cases that might be applicable to an MSR, but they are only currently approved for LWR use. It there are existing code cases that a MSR applicant wants to use for their design they would have to get ASME to review them for that purpose. Code cases are specifically focused on an aspect of the design.	Provisions of the ASME BPV Code have been used since 1971 as one part of the framework to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

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1.02004	ASME BPVC Section III Division 1 Subsection NE Subsection NF Subsection NG	Rules for Construction of Nuclear Power Plant Components	1.20-4 1.84-36 1.100-4 1.107-2 1.136-3 1.142-3 1.147-20 1.189-4 1.207-1 3.27-1 3.60-0	SRP 14.2 SRP 3.1.2 SRP 3.1.3 SRP 3.2.1 SRP 3.2.2 SRP 3.8.1 SRP 3.8.2 SRP 4.5.2 SRP 5.2.1.2 SRP 5.2.2 SRP 5.2.3 SRP 5.4.2.1 SRP 6.1.1 SRP 10.2 SRP 10.3.6 SRP 3.9.3 SRP 3.10 SRP 5.4.12 SRP BTP 7-10 SRP 3.8.1 SRP 3.8.1 SRP 3.8.3 SRP 3.8.4 SRP 3.5.3 SRP 3.8.3 SRP 3.8.4 SRP 3.8.5 SRP 3.8.1 SRP 3.8.2 SRP 5.2.1.2 SRP 54.2.4 SRP 3.2.1 SRP 7.4 SRP Appendix 7.1-A SRP 9.5.1.1 SRP 9.5.1.2 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP BTP 11-3 SRP 13.1.2-13.1.3 SRP 13.2.1 SRP 17.5 SRP 3.12 SRP 5.4.2.1	SRP 3 SRP 4 SRP 5 SRP 6 SRP 9 SRP 10 SRP 14 SRP 15 SRP 17	10 CFR 50.55a 10 CFR 50, Appendix G 10 CFR 61	ASME	3	High	Provisions of the ASME BPV Code have been used since 1971 as one part of the framework to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. Section III consists of Division 1 (N), Division 2 (C), Division 3 (W), and Division 5 (H). Each Subsection is published separately, with some exceptions for Divisions 2, 3, and 5. Division 1 has the following subsections: NB — Class 1 Components, NC — Class 2 Components, ND — Class 3 Components, NE — Class MC Components, NF — Supports, NG — Core Support Structures, and NH — Class 1 Components in Elevated Temperature Service. Division 2 is the Code for Concrete Containments with Subsection CC — Concrete Containments.G6 Changes are necessary to address the varied advanced reactor designs as well as functional containment concepts.	ASME BPVC Division 1 and 2, Subsection NCA, Containment Barrier - Changes are necessary to reflect functional containment concept. For example, the containment barrier is "…essentially leak-tight…" rather than an "…effective barrier…" to describe the containment function for concepts that may rely on acceptable design condition leak rates. The rules of Subsection NCA constitute requirements for the design, construction, stamping and overpressure protection of items used in nuclear power plants and other nuclear facilities.
1.02004	ASME BPVC Section III Division 2	Code for concrete containments	1.20-4 1.84-36 1.100-4 1.107-2 1.136-3 1.142-3 1.147-20	SRP 14.2 SRP 3.1.2 SRP 3.1.3 SRP 3.2.1 SRP 3.2.2 SRP 3.8.1	SRP 3 SRP 4 SRP 5 SRP 6 SRP 9 SRP 10 SRP 14	10 CFR 50.34 10 CFR 50.55a	ASME	3	High	Subsection CC establishes rules for material, design, fabrication, construction, examination, testing, marking, stamping, and preparation of reports for prestressed and reinforced concrete containments. The containments covered by this Subsection shall include the following: (a) structural concrete pressure resisting shells and shell components (b) shell metallic liners (c) penetration liners	ASME BPVC Section III, Division 2 is also known as ACI Standard 359-01 The CFR requires that containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and

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			1.189-4 1.207-1	SRP 3.8.2 SRP 4.5.2 SRP 5.2.1.2 SRP 5.2.2 SRP 5.2.3 SRP 5.4.2.1 SRP 6.1.1 SRP 10.2 SRP 10.3.6 SRP 3.9.3 SRP 3.10 SRP 5.4.12 SRP BTP 7-10 SRP 3.8.1 SRP 3.8.1 SRP 3.8.3 SRP 3.8.4 SRP 3.5.3 SRP 3.8.3 SRP 3.8.4 SRP 3.8.5 SRP 3.8.1 SRP 3.8.2 SRP 5.2.1.2 SRP 5.4.2.4 SRP 3.2.1 SRP 7.4 SRP Appendix 7.1-A SRP 9.5.1.1 SRP 9.5.1.2 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP BTP 11-3 SRP 13.1.2-13.1.3 SRP 13.2.1 SRP 17.5 SRP 3.12 SRP 5.4.2.1	SRP 15 SRP 17					extending the containment liner through the surrounding shell concrete. Containments having a Design Pressure greater than 5 psi (35 kPa) that are classified as Subsection CC containments shall be constructed in accordance with the rules of Subsection CC. However, an MSR will have a significantly different containment rather than the typical LWR-type containment. ASME BPVC Section III may not be applicable and if it is, it will have to be updated to reflect this significant difference. Substantive changes will be needed or a new standard developed specifically for MSRs.	Pressure Vessel Code, Section III, Division 1, subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig.
1.08702	ASME BPVC Section III Division 5	High Temperature Reactors	1.87-2	—	—	—	ASME	1	Low	BPVC Section III Division 5 was endorsed by the revision to RG 1.87 in January 2023.	There are various activities being undertaken and on-going to supplement Division 5. Activities such as extending the qualified lifetimes of Class A materials to support a 60-year design life, qualification of additional materials, development of analysis methods to simplify the Division 5 design analyses, development of design rules for integrally cladded components with weld overlay on Class A materials to support molten salt reactor applications, incorporation of graphite irradiation data to support graphite

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											design rules, and incorporation of ceramic composite design rules.
1.03401	ASME BPVC Section IX	Welding and Brazing Qualifications	1.34-1 1.43-1 1.50-1 1.71-1	SRP 5.2.3 SRP 5.3.1 SRP 5.4.2.1 SRP 6.1.1 SRP 6.2.1.1.C SRP 5.2.3 SRP 5.3.1 SRP 5.4.2.1 SRP 5.2.3 SRP 5.3.1 SRP 5.4.2.1 SRP 6.1.1 SRP 10.2 SRP 10.3.6 SRP 5.2.3 SRP 5.4.2.1 SRP 6.1.1 SRP 10.2 SRP 10.3.6	SRP 4.5.2 SRP 5.2.3 SRP 5.4.2.1 SRP 5.3.1 SRP 6.1.1 SRP 10.2 SRP 10.3.6	—	ASME	1	Low	This Section contains requirements for the qualification of welders, welding operators, brazers, brazing operators, plastic fusing operators, and the material joining processes they use during welding, brazing, and fusing operations for the construction of components under the rules of the ASME Boiler and Pressure Vessel Code, the ASME B31 Codes for Pressure Piping, and other Codes, standards, and specifications that reference this Section. This Section is divided into four parts. (1) Part QG contains general requirements for all material-joining processes. (2) Part QW contains requirements for welding. (3) Part QB contains requirements for brazing. (4) Part QF contains requirements for plastic fusing.	This standard likely covers most if not all of the brazing and welding qualifications.
3.02701	ASME BPVC Section V	Nondestructive Examination	3.27-1 3.37-0	—	—	—	ASME	1	Low	This Section contains requirements and methods for nondestructive examination which are referenced and required by other BPVC Sections. It also includes manufacturer's examination responsibilities, duties of authorized inspectors and requirements for qualification of personnel, inspection and examination. Examination methods are intended to detect surface and internal discontinuities in materials, welds, and fabricated parts and components.	
3.60000	ASME BPVC Section VIII	Pressure Vessels	3.60-0	—	SRP 3.2.2 SRP 3.9.4	—	ASME	1	Low	This Division of Section VIII provides requirements applicable to the design, fabrication, inspection, testing, and certification of pressure vessels operating at either internal or external pressures exceeding 15 psig.	
1.14720	ASME BPVC Section XI	Rules for Inservice Inspection of Nuclear Power Plant Components	1.147-20 1.84-39	SRP 3.1.2 SRP 3.1.3 SRP 3.2.1 SRP 3.2.2 SRP 3.8.1 SRP 3.8.2 SRP 4.5.2 SRP 5.2.1.2 SRP 5.2.2 SRP 5.2.3 SRP 5.4.2.1 SRP 6.1.1 SRP 10.2 SRP 10.3.6 SRP 3.8.1 SRP 3.8.2 SRP 5.2.1.2 SRP 5.2.4	MANY sections in SRP 3 SRP 5 SRP 6 SRP 9 SRP 10 SRP 13 SRP 14	10 CFR 50.55a 10 CFR 50, Appendix G	ASME	3	Low	RG 1.147 is ISI of BPVC Section XI, Division 1 components. The rules and requirements for those components and systems of this plant type that contain other fluids are provided by references to Articles or portions thereof in Division 1 of Section XI, on the basis that these Division 1 rules are appropriate and applicable to Liquid-Metal-Cooled Nuclear Power Plants; otherwise such rules and requirements are provided in Division 3. Section XI consists of three Divisions, as follows: Division 1 = Rules for Inspection and Testing of Components of Light-Water-Cooled Plants Division 2 = Rules for Inspection and Testing of Components of Gas-Cooled Plants Division 3 = Rules for Inspection and Testing of Components of Liquid-Metal-Cooled Plants	
1.24400	ASME BTH-1–2017	Design of Below-the-Hook Lifting Devices	1.244-0	—	—	—	ASME	1	Low	There have been many formal requests for interpretation of the limited structural design criteria stated within ASME B30.20, Below-the-Hook Lifting Devices, a safety standard. As a consequence, industry has for quite some time expressed a need for a comprehensive design	

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										standard for below-the-hook lifting devices that would complement the safety requirements of ASME B30.20. The provisions defined in this Standard address the most common and broadly applicable aspects of the design of below-the-hook lifting devices.	
1.05204	ASME N509-2002 (SRP accepts N509-1989)	Nuclear Power Plant Air-Cleaning Units and Components	1.52-4 1.140-3	SRP 6.1.1 SRP 6.2.3 SRP 6.3 SRP 6.4 SRP 6.5.1 SRP 6.5.3 SRP 6.5.5 SRP 9.1.2 SRP 9.1.3 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5 SRP 11.3 SRP 11.5 SRP 12.3-12.4 SRP 15.6.5 SRP 6.1.1 SRP 6.5.1 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5 SRP 11.1 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 12.3-12.4 SRP 14.2	SRP 6.5.1	—	ASME	3	High	This standard covers requirements for the design, construction, and qualification and acceptance testing of the air-cleaning units and components which make up Engineered Safety Feature (ESF) and other High efficiency air and gas treatment systems used in nuclear power plants. The standard does not cover sizing of a complete nuclear air treatment system, redundancy, or single-failure requirements. It applies only to systems which employ particulate filtration, ambient-temperature adsorption, or both, as the principal functional mechanism. Changes are necessary to address the varied advanced reactor designs.	Because ASME AG-1 supplements ASME N509-2002, it is this relationship that should be reviewed more closely. AG-1 will require substantial changes because needed sections in Division III, Process Gas Treatment are not complete. (Section GE in AG-1, Hydrogen Recombiners, is complete but it is likely to be N/A for an MSR.) Molten salt aerosols and byproducts may affect the adequacy and completeness of this standard.
1.01302	ASME N510-2007	Testing of Nuclear Air-Treatment Systems	1.13-2 1.52-4 1.140-3	SRP 9.1.1 SRP 9.1.21 SRP 9.1.3 SRP 9.1.5 SRP 9.4.2 SRP 9.5.1.1 SRP 9.5.2 SRP 11.5 SRP 14.2 SRP 6.1.1 SRP 6.2.3 SRP 6.3 SRP 6.4 SRP 6.5.1 SRP 6.5.3 SRP 6.5.5 SRP 9.1.2 SRP 9.1.3 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5	SRP 6.5.1	—	ASME	2	Low	This standard provides a basis for the development of test programs for air-treatment systems and does not include acceptance criteria except where the results of one test influence the performance of other tests. Acceptance criteria shall be developed based on the design/function in accordance with ASME N509. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	Tests and Inspections addressed in ASME N510 include visual inspection; duct leak test; housing leak test; airflow capacity; air-aerosol mixing uniformity; in-service leak test, HEPA filters; in-service leak test, adsorbers; system bypass; air heater performance; and laboratory tests of adsorbent. These test and inspections are applicable to MSRs. Completeness of tests and inspections need to be reviewed for a molten salt environment (e.g., does testing cover aerosols).

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				SRP 11.3 SRP 11.5 SRP 12.3-12.4 SRP 15.6.5 SRP 6.1.1 SRP 6.5.1 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5 SRP 11.1 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 12.3-12.4 SRP 14.2							
1.05204	ASME N511-2007	In-Service Testing of Nuclear Air Treatment, Heating, Ventilating, and Air-Conditioning Systems	1.52-4	SRP 6.1.1 SRP 6.2.3 SRP 6.3 SRP 6.4 SRP 6.5.1 SRP 6.5.3 SRP 6.5.5 SRP BTP 6-3 SRP 9.1.2 SRP 9.1.3 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5 SRP 11.3 SRP 11.5 SRP 12.3-12.4 SRP 14.2 SRP 15.6.5	—	—	ASME	2	Low	The purpose of this Standard is to provide requirements for in-service testing, the results of which are used to verify that the nuclear air treatment, heating, ventilating, and air- conditioning systems perform their intended function.	Completeness of tests and inspections need to be reviewed for a molten salt environment (e.g., does testing cover aerosols).
1.24400	ASME NML-1–2019	Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities	1.244-0	—	—	—	ASME	1	Low	This Standard applies to all lifting and handling operations at nuclear facilities, including the training and certification of personnel, and the maintenance, inspection, testing, and rework and modification of overhead handling systems and other lifting devices. The application of this Standard shall begin at the point of initial fuel load at the affected unit under construction.	
1.24400	ASME NOG-1–2020	Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)	1.244-0	—	SRP 9.1.5	—	ASME	1	Low	This Standard covers electric overhead and gantry multiple girder cranes with top running bridge and trolley used at nuclear facilities and components of cranes at nuclear facilities. The items qualified by this Standard are the bridge wheels up through the crane bridge and trolley.	
1.02805	ASME NQA-1-2008	Quality Assurance Requirements for Nuclear Facility Applications	1.28-5 1.8-3 1.97 3.48-1 3.75-0	SRP 12.5 SRP 13.2.1 SRP 13.2.2 SRP 2.5.4 SRP 2.5.5 SRP 5.4.2.1 SRP BTP 7-14 SRP 9.2.7	SRP Ch 4.5.1 SRP 17.5 SRP BTP 7-14	10 CFR 50.55a	ASME	2	Low	NQA-1 is a multipart Standard that provides includes requirements and nonmandatory guidance to establish and implement a QA program for any nuclear facility application. Part I contains QA program requirements for the siting, design, construction, operation, and decommissioning of nuclear facilities. Part II contains QA requirements for the planning and conducting of the fabrication, construction, modification, repair, maintenance, and testing of systems, components, or	The NRC staff performed a review and identified that differences exist between the previously endorsed guidance (NQA-1-2008 and NQA-1a-2009 addenda) and the most recently issued guidance (NQA-1b-2011, NQA-1-2012 and NQA-1-2015). Additional time and resources are required to understand the impact of these changes. The NRC staff

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				SRP 13.1.2 SRP 17.5 SRP 3.10 SRP 3.11 SRP 6.2.1.1 SRP BTP 7-10 SRP 9.3.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 12.4 SRP 12.5 SRP 13.3 SRP 14.3.7 SRP 14.3.9 SRP 14.3.11 SRP 18.0						activities for nuclear facilities. Part III contains nonmandatory guidance. Part IV contains NQA position papers and other quality program information.	continues to endorse the previous guidance and is not aware of any issues that would preclude its use.
1.02805	ASME NQA-1a-2009 (Addenda to ASME NQA-1-2008)	Quality Assurance Requirements for Nuclear Facility Applications	1.28-5	SRP 2.5.4 SRP 2.5.5 SRP 5.4.2.1 SRP BTP 7-14 SRP 9.2.7 SRP 13.1.2 SRP 17.5	SRP BTP 7-14	—	ASME	2	Low	NQA-1 is a multipart Standard that provides includes requirements and nonmandatory guidance to establish and implement a QA program for any nuclear facility application. Part I contains QA program requirements for the siting, design, construction, operation, and decommissioning of nuclear facilities. Part II contains QA requirements for the planning and conducting of the fabrication, construction, modification, repair, maintenance, and testing of systems, components, or activities for nuclear facilities. Part III contains nonmandatory guidance. Part IV contains NQA position papers and other quality program information.	The NRC staff performed a review and identified that differences exist between the previously endorsed guidance (NQA-1-2008 and NQA-1a-2009 addenda) and the most recently issued guidance (NQA-1b-2011, NQA-1-2012 and NQA-1-2015). Additional time and resources are required to understand the impact of these changes. The NRC staff continues to endorse the previous guidance and is not aware of any issues that would preclude its use. It there are existing code cases that an MSR applicant wants to use for their design they would have to get ASME to review them for that purpose. Code cases are specifically focused on an aspect of the design.
1.10004	ASME QME-1-2017	Qualification of Active Mechanical Equipment Used in Nuclear Power Plants	1.100-4	SRP 3.10 SRP 5.4.12 SRP BTP 7-10	SRP 3.10	—	ASME	3	High	This Standard provides the requirements and guidelines for the qualification of active mechanical equipment whose function is required to ensure the safe operation or safe shutdown of a nuclear facility. In addition to requirements and guidelines put forth in this Standard, the active mechanical equipment shall comply with the requirements of the applicable design and construction codes and standards. Advanced Light Water Reactor (ALWR) First-of-a- Kind Engineering Project on Equipment Seismic Qualification, Advanced Reactor Corporation (ARC), April 1995 and NUREG/CR-6464, "An Evaluation of Methodology for Seismic Qualification of Equipment, Cable Trays, and Ducts in ALWR Plants by Use of Experience Data." USNRC, 1997 are cited as non-mandatory references for active mechanical equipment.	Some of the requirements and guidance provided in this standard are not applicable to the non-LWRs because the qualification requirements and guidelines are for active components that must function to ensure safe operation, safe shutdown, or operation during design basis events. Other components, such as dynamic restraints are applicable. The standard should be updated to reflect only those applicable portions. Cooling water systems should be changed to structural and equipment cooling systems.
1.20003	ASME RA-Sa-2009	Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications	1.200-3	SRP Appendix 7.1-A SRP BTP 7-12 SRP 9.5.1.2 SRP 19.0 SRP 19.1 SRP 19.2	SRP 19.0 SRP 19.1	—	ASME	1	Low	This standard establishes requirements for a PRA for advanced non-light water reactor nuclear power plants. The requirements in this standard were developed for a broad range of PRA scopes including operating states, hazard types, and different end states.	

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1.04401	ASTM A-262-1970	Detecting Susceptibility to Intergranular Attack in Stainless Steels	1.44-1	SRP 4.5.1 SRP 4.5.2 SRP 5.2.3 SRP 5.3.1 SRP 5.4.2.1 SRP 5.4.13 SRP 6.1.1	SRP 4.5.1 SRP 5.2.3 SRP 5.3.1 SRP 6.1.1	—	ASTM	1	Low	Control of the application and processing of stainless steel to avoid severe sensitization is needed to diminish the numerous occurrences of intergranular stress-corrosion cracking in sensitized stainless steel components of nuclear reactors. Test data demonstrate that sensitized stainless steel is significantly more susceptible to intergranular stress-corrosion cracking than is nonsensitized (solution heat-treated) stainless steel. Of specific concern in this guide are the unstabilized austenitic stainless steels, which include American Iron and Steel Institute (AISI) Types 304 and 316, normally used for components of the reactor coolant system and other safety-related systems. Low carbon grade stainless steel (i.e., 304L and 316L) should be used where the material comes in contact with the reactor coolant.	
1.13604	ASTM A-370-2005	Standard Test Methods and Definitions for Mechanical Testing of Steel Products	1.136-4 NOT endorsed in R3	SRP 3.8.1 SRP 3.8.3 SRP 3.8.4	SRP 5.4.1.1 SRP 10.2 SRP 10.2.3	—	ASTM	1	Low	ASTM A370-22 covers procedures and definitions for the mechanical testing of steels, stainless steels, and related alloys. The various mechanical tests herein described are used to determine properties required in the product specifications. Variations in testing methods are to be avoided, and standard methods of testing are to be followed to obtain reproducible and comparable results. In those cases in which the testing requirements for certain products are unique or at variance with these general procedures, the product specification testing requirements shall control.	
3.03700	ASTM A262-70	Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels	3.37-0	—	—	—	ASTM	4	Low	Use of this RG assumes MSR fuel fabrication may be implemented at a facility on or adjacent to the reactor site. Materials for onsite fuel preparation system not defined.	May not be an MSR material issue
5.01201	ASTM B-117 -07a	Standard Practice for Operating Salt Spray (Fog) Apparatus	5.12-1	—	—	—	ASTM	1	Low	Physical Protection Standard is applicable to MSRs	Most recent version is 2019. Reference list refers to 2007 version.
5.02701	ASTM C1112-99-2005	Standard Guide for Application of Radiation Monitors to the Control and Physical Security of Special Nuclear Material	5.27-1	—	—	—	ASTM	1	Low	Physical Protection Standard is applicable to MSRs	Most recent version is 2017. Reference list refers to 2005 version.
5.02701	ASTM C1189-11	Standard Guide to Procedures for Calibrating Automatic Pedestrian SNM Monitors	5.27-1	—	—	—	ASTM	1	Low	Physical Protection Standard is applicable to MSRs	—
5.02701	ASTM C1269-97-2012	Standard Practice for Adjusting the Operational Sensitivity Setting of In-Plant Walk-Through Metal Detectors	5.27-1	—	—	—	ASTM	1	Low	Physical Protection Standard is applicable to MSRs	Most recent version is 2021. Reference list refers to 2012 version.
5.02701	ASTM C1270-97-2012	Standard Practice for Detection Sensitivity Mapping of In-Plant Walk	5.27-1	—	—	—	ASTM	1	Low	Physical Protection Standard is applicable to MSRs	Most recent version is 2021. Reference list refers to 2012 version.

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		Through Metal Detectors									
5.02701	ASTM C1309-97-2012	Standard Practice for Performance Evaluation of In-Plant Walk-Through Metal Detectors	5.27-1	—	—	—	ASTM	1	Low	Physical Protection Standard is applicable to MSRs	Most recent version is 2022. Reference list refers to 2012 version.
—	ASTM C776-89, Part 45	Standard Specification for Sintered Uranium Dioxide Pellets	—	—	SRP 4.2	—	ASTM	5	Low	Fuel salt is an essential element of providing adequate containment, heat removal, and reactivity control. Fuel salt properties are determined by its composition, which must be maintained within acceptable limits. The closest standard approved for use (SRP 4.2) is ASTM C776-89, Part 45, which specifies the chemical, nuclear, and physical characteristics of UO2 pellets. Because the pellets are so different than molten salt this standard is deemed N/A with a standard specific to MSRs being developed as a new standard.	
1.05403	ASTM D 5162-15	Standard Practice for Discontinuity (Holiday) Testing of Nonconductive Protective Coating on Metallic Substrates	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	4		The chemical resistance of coatings and linings must be tested for use. Tests are for short term and long-term exposures. This test method is intended to be used as a screening test to evaluate coatings and linings on steel and concrete substrates. Long-term tests require immersing the lining test specimens in the appropriate test solution for a minimum of 180 days. Unknown for MSRs is what the coatings and linings will be exposed.	
1.05204	ASTM D3803-1991	Standard Test Methods for Nuclear-Grade Activated Carbon	1.52-4 1.140-3	SRP 6.1.1 SRP 6.2.3 SRP 6.3 SRP 6.4 SRP 6.5.1 SRP 6.5.3 SRP 6.5.5 SRP 9.1.2 SRP 9.1.3 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5 SRP 11.3 SRP 11.5 SRP 12.3-12.4 SRP 15.6.5 SRP 6.1.1 SRP 6.5.1 SRP 9.4.1 SRP 9.4.2 SRP 9.4.3 SRP 9.4.4 SRP 9.4.5 SRP 11.1 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 12.3-12.4 SRP 14.2	SRP 9.4.2 SRP 9.4.3 SRP 9.4.4	—	ASTM	3	High	The test method in ASTM D3803 is a very stringent procedure for establishing the capability of new and used activated carbon to remove radio-labeled methyl iodide from air and gas streams. The conditions employed in the standard were selected to approximate operating or accident conditions of a nuclear reactor which would severely reduce the performance of activated carbons. Guidance for testing new and used carbons using conditions different from the test method in ASTM D3803 is offered in Annex A1 of the standard. The appropriateness of the test method will need to be evaluated when a more detailed design is available.	The 30°C, 95 % relative humidity methyl iodide test is the most reliable test method to establish the methyl iodide removal efficiency of any adsorbent. However, nuclear facilities often require test parameters (temperature, humidity, etc.) which are based on different operating conditions. When tests are required to be performed either under Test Method D3803 or any other conditions following the ASTM test procedure, the parameter tolerances need to be tightened for both new and used carbon testing
1.05403	ASTM D3843-16	Standard Practice for Quality	1.54-3	SRP 6.1.1 SRP 6.1.2	—	—	ASTM	2	Low	ASTM D3843-16 (R2021) is most recent and broadly applicable standard to cover coatings.	ASTM D3843-16 (R2021) supersedes ANSI N101.4-72 cited in RG 3.30-0.

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		Assurance for Protective Coatings Applied to Nuclear Facilities		SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3						This standard references: ASTM D4227 Concrete Coatings ASTM D4228 Steel Coatings ASTM D4537 Personnel Coating Inspectors ASTM D4538 Terminology ASTM D5144 Coating Use in NPPs ANSI N45.2 QA for NPPs ASME NQA-1 QA for NPPs Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	This standard cites additional standards listed - None of the ASTM standards on coatings are cited in a Division 3 RG. All are applicable to MSRs
1.05403	ASTM D3911-16	Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design-Basis Accident (DBA) Conditions	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	SRP 6.1.1 SRP 6.1.2	—	ASTM	3	Low	The objective of this guide is to provide a common basis on which protective coatings for the surfaces of nuclear power generating facilities may be qualified and selected by reproducible evaluation tests. This guide also provides guidance for application and maintenance of protective coatings. All Coating Service Level I coatings must be resistant to the effects of radiation and must be DBA qualified. Service Level III coatings must be evaluated for use in accordance with the requirements of plant licensing commitments and the job specifications. This document covers coating work on previously coated surfaces as well as bare substrates. This guide applies to all coating work in Coating Service Level I and III areas (that is, safety-related coating work). Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs. Deviations in actual surface preparation and in application and curing of the coating materials from qualification test parameters require an engineering evaluation to determine if additional testing is required. Changes are necessary to address the varied advanced reactor design SSCs and coating applications.	The designer of light water-moderated nuclear reactor systems must consider the possibility of a DBA and the subsequent events which might lead to the release or expulsion of a fraction of the fission-product inventory of the core to the reactor containment facility. Under the environmental operating and accident conditions of nuclear power generation facilities, encompassing PWRs and BWRs, coating performance may be affected by exposure to anyone, all, or a combination of the following conditions: ionizing radiation; contamination by radioactive nuclides and subsequent decontamination processes; chemical and water sprays; High-temperature High-pressure steam; and abrasion or wear.
1.05403	ASTM D3912-10	Standard Test Method for Chemical Resistance of Coatings and Linings for Use in Nuclear Power Plants	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	4		The chemical resistance of coatings and linings must be tested for use. Tests are for short term and long-term exposures. This test method is intended to be used as a screening test to evaluate coatings and linings on steel and concrete substrates. Long-term tests require immersing the lining test specimens in the appropriate test solution for a minimum of 180 days. Unknown for MSRs is what the coatings and linings will be exposed.	This test method in ASTM D3912-16 establishes procedures for the evaluation of the chemical resistance of coatings and linings for tanks, vessels, and similar facilities. (Coatings for water immersion applications should use ASTM D7230.) This test method is intended to be used as a screening test to evaluate coatings and linings on steel and concrete substrates. This test method addresses two exposure intervals: short term (typically 5 days) and long term (typically 180 days).
3.07300	ASTM D4015-2000	Standard Test Methods for Modulus and Damping of Soils by Fixed-Base Resonant Column Devices	3.73-0	—	—	—	ASTM	1	Low	These test methods cover the determination of shear modulus and shear damping as a function of shear strain amplitude for solid cylindrical specimens of soil in intact and reconstituted conditions by torsional vibration using resonant column devices. The vibration of the specimen may be superposed on a controlled static state of stress in the specimen. The vibration apparatus and specimen may be enclosed in a triaxial chamber and subjected to an all-around pressure and axial load. In addition, the specimen may be subjected to other controlled conditions (for	

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										example, pore-water pressure, degree of saturation, temperature).	
1.05403	ASTM D4082-10	Standard Test Method for Effects of Gamma Radiation on Coatings for Use in Nuclear Power Plants	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	1	Low	This test method covers a standard procedure for evaluating the lifetime radiation tolerance of coatings to be used in nuclear power plants. This test method is designed to provide a uniform test to assess the suitability of coatings, used in nuclear power facilities, under radiation exposure for the life of the facilities, including radiation during a DBA.	Based on the assessed lifetime radiation of coating and radiation during a DBA, the irradiation dose rate, irradiation accumulated dose, and radiation source will need to be revised. For an MSR, DBA may be the Postulated Accident. The suitability of coatings is based on lifetime exposure, including radiation during a DBA. The DBAs will be different for MSRs. As such, even though the standard says, "unless otherwise specified," the irradiation dose rate, irradiation accumulated dose, and radiation source will likely be different. Specific plant radiation exposure may exceed or be less than the amount specified in 7.2 of this standard. Nevertheless, the test methods are appropriate.
1.05403	ASTM D4227-05	Standard Practice for Qualification of Coating Applicators for Application of Coatings to Concrete Surfaces	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	1	Low	This practice provides a standard qualifying method for coating applicators to verify their proficiency and ability to attain the required quality for application of specified coatings to <u>steel surfaces</u> including those in safety-related areas in a nuclear facility. It is the intent of this practice to judge only the ability of the coating applicator to apply specified coatings with the proper tools and equipment.	It is the intent of this practice to judge only the ability of the coating applicator to apply specified coatings with the proper tools and equipment.
1.05403	ASTM D4228-05	Standard Practice for Qualification of Coating Applicators for Application of Coatings to Steel Surfaces	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	2	Low	This practice provides a standard qualifying method for coating applicators to verify their proficiency and ability to attain the required quality for application of specified coatings to <u>steel surfaces</u> including those in safety-related areas in a nuclear facility. It is the intent of this practice to judge only the ability of the coating applicator to apply specified coatings with the proper tools and equipment.	It is the intent of this practice to judge only the ability of the coating applicator to apply specified coatings with the proper tools and equipment.
1.05403	ASTM D4286-08	Standard Practice for Determining Coating Contractor Qualifications for Nuclear Powered Electric Generation Facilities	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	2	Low	The qualification criteria and requirements address the essential basic capability of a contractor to execute nuclear coating work. Determining the qualifications of contractors is independent of reactor type. This standard is applicable to MSRs. Change scope of the standard from "light-water nuclear power plants" to "nuclear power plants."	This standard provides a criteria guide and procedural method to assist utility owners, architects, engineers, constructors, and other selection agencies in determining the overall qualifications of a coating contractor to execute coating work for the primary containment and other safety-related facilities of light- water nuclear power plants. The selection of a contractor and contractor evaluation worksheet is applicable to non-LWRs.
1.05403	ASTM D4537-12	Standard Guide for Establishing Procedures To Qualify and Certify Personnel Performing Coating Work Inspection in Nuclear Facilities	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	2	Low	This guide delineates the requirements for development of procedures for the qualification and certification of personnel who perform inspection of coating and lining work. It is the intent of this guide to provide a recommended basis for qualification and certification, not to mandate a singular basis for all qualifications.	To assure satisfactory performance of the inspections and to avoid compromising safety-related coating systems.
1.05403	ASTM D4538-15	Standard Terminology Relating to Protective Coating and Lining Work	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4	—	—	ASTM	2	Low	ASTM D4538 provides the terminology and their definitions relevant to the use of protective coatings in nuclear power plants. This terminology covers terms and their definitions	The referenced documents and terminology (definitions) in ASTM D4538 are LWR-specific. Referenced documents and terminology should be expanded to include non-LWR

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		for Power Generation Facilities		SRP 11.5 SRP 17.3						relevant to the use of protective coatings in nuclear power plants. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	documents and terminology. A LOCA is defined. For an MSR this should be replaced with the definition of a postulated accident.
1.05403	ASTM D4541-09	Standard Test Method for Pull-Off Strength of Coatings Using Portable Adhesion Testers	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	1	Low	This test method covers a procedure for evaluating the pull-off strength (commonly referred to as adhesion) of a coating system from metal substrates. The test determines either the greatest perpendicular force (in tension) that a surface area can bear before a plug of material is detached, or whether the surface remains intact at a prescribed force (pass/fail). The procedure in this standard was developed for metal substrates but may be appropriate for other rigid substrates such as plastic and wood.	The pull-off strength of a coating is an important performance property that has been used in specifications.
1.05403	ASTM D5139-12	Standard Specification for Sample Preparation for Qualification Testing of Coatings to Be Used in Nuclear Power Plants	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	1	Low	This specification defines the size, composition, surface preparation, and coating application variables for preparing samples for evaluating coatings and linings over various substrates. Substrates include steel panels and miscellaneous materials such as aluminum, galvanized steel, and other metals, and concrete blocks and castable materials such as grout, fireproofing, and other castables.	This specification provides uniform requirements for the preparation of test samples used for testing of coatings and linings to be used in nuclear power plants.
1.05403	ASTM D5144-08	Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	SRP 6.1.1 SRP 6.1.2	—	ASTM	2	Low	The objective of this guide is to provide a common basis on which protective coatings for the surfaces of nuclear power generating facilities may be qualified and selected by reproducible evaluation tests. This guide also provides guidance for application and maintenance of protective coatings. All Coating Service Level I coatings must be resistant to the effects of radiation and must be DBA qualified. Service Level III coatings must be evaluated for use in accordance with the requirements of plant licensing commitments and the job specifications. References are for LWR technology and should be updated. Emergency diesel generators will not be used. Testing, coating materials, surface preparation, and QA (san references) are applicable to non-LWRs. The QA requirements are applicable to MSRs; however, the DBAs in D5144 apply to LWRs and thus D5144 will require changes.	The designer of light water-moderated nuclear reactor systems must consider the possibility of a DBA and the subsequent events which might lead to the release or expulsion of a fraction of the fission-product inventory of the core to the reactor containment facility. Under the environmental operating and accident conditions of nuclear power generation facilities, encompassing PWRs and BWRs, coating performance may be affected by exposure to anyone, all, or a combination of the following conditions: ionizing radiation; contamination by radioactive nuclides and subsequent decontamination processes; chemical and water sprays; High-temperature High-pressure steam; and abrasion or wear.
1.05403	ASTM D5163-16	Standard Guide for Establishing a Program for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	1	Low	This test method covers the procedure for assessing the adhesion of coating films to substrate by using a knife. This test method is used to establish whether the adhesion of a coating to a substrate or to another coating (in multi-coat systems) is at a generally adequate level. NOTE 1—The term "substrate" relates to the basic surface on which a coating adheres (may be steel, concrete, etc. or other coating).	Coatings, to perform satisfactorily, must adhere to the substrates on which they are applied. This test method has been found useful as a simple means of assessing the adhesion of coatings.
1.05403	ASTM D5498-12a	Standard Guide for Developing a Training Program for Personnel Performing Coating Work Inspection	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	1	Low	This guide is intended to assist those responsible for developing a program for the indoctrination and training of personnel performing coating and lining inspection work for nuclear facilities.	Personnel trained for coating and lining work inspection are required to perform examination/inspection tasks to verify conformance of coating and lining work to written requirements.

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		for Nuclear Facilities									
1.05403	ASTM D6677-07	Standard Test Method for Evaluating Adhesion by Knife	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	1	Low	This test method covers procedures for evaluating the pull-off adhesion strength of a coating on concrete. The pull-off adhesion strength and mode of failure of a coating from a concrete substrate are important performance properties that are used in specifications. This test method serves as a means for uniformly preparing and testing coated surfaces and evaluating and reporting the results. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	
1.05403	ASTM D7108-12	Standard Guide for Establishing Qualifications for a Nuclear Coatings Specialist	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	2	Low	This guide delineates recommendations for development of procedures and criteria for designation of an individual as a Nuclear Coatings Specialist involved in coating work in nuclear facilities. The Nuclear Coatings Specialist is responsible for the technical aspects of the safety-related coatings program in a nuclear facility or organization, which includes establishing processes and quality control requirements.	Only those personnel within their respective organizations who meet the requirements of this guide are designated as Nuclear Coatings Specialists. This guide describes the general duties and responsibilities of a Nuclear Coatings Specialist; education, training and experience qualifications; and maintenance of qualification.
1.05403	ASTM D7167-12	Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	2	Low	This guide covers procedures for establishing a program to monitor the performance of Coating Service Level III lining (and coating) systems in operating nuclear power plants. Monitoring is an ongoing process of evaluating the condition of the in-service lining systems. This document covers procedures for establishing a program to monitor the performance of Coating Service Level III lining (and coating) systems in operating nuclear power plants. Monitoring is an ongoing process of evaluating the condition of the in-service lining systems. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	Establishment of an in-service linings monitoring program permits planning and prioritization of lining maintenance work as needed to maintain lining integrity and performance in nuclear Coating Service Level III systems. Coating Service Level III lining systems subject to this guide are generally those applied to metal substrates comprising raw water, condensate-quality water, or fuel oil wetted (that is, full or intermittent immersion) surfaces. The establishing procedures to monitor the performance applies to MSRs and the scope should be expanded.
1.05403	ASTM D7234-012	Standard Test Method for Pull-Off Adhesion Strength of Coatings on Concrete Using Portable Pull-Off Adhesion Testers	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	2	Low	This test method covers procedures for evaluating the pull- off adhesion strength of a coating on concrete. The pull-off adhesion strength and mode of failure of a coating from a concrete substrate are important performance properties that are used in specifications. This test method serves as a means for uniformly preparing and testing coated surfaces and evaluating and reporting the results.	
1.05403	ASTM D7491-08	Standard Guide for Management of Non-Conforming Coatings in Coating Service Level I Areas of Nuclear Power Plants	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	ASTM	3	High	This guide provides the user with guidance on developing a program for managing non-conforming coatings in Coating Service Level I areas of a nuclear power plant. Non-conforming coatings include degraded previously DBA-qualified or acceptable coatings, unqualified coatings, unknown coatings, and unacceptable coatings. Changes are necessary to address the varied advanced reactor design SSCs and coating applications. The use of the plant corrective action program for non-conforming coatings are LWR (BWR)-specific such as ECSS suction strainer, safety related SSC performance	The key to ensuring plant safety is to manage the amount of non-conforming coatings so that it does not exceed the amount assumed in calculations that support plant operation. There may be significant work to develop safety-related protective coatings (such as the EPRI Report 1003102 referenced), and they may find that over time initially acceptable coatings may be found to be

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										after a LOCA, MSLB, etc. Examples should be provided to inform users that D7491 is applicable to MSRs or to eliminate LWR examples.	unacceptable. Thus, there may be considerable work managing coatings found to be not compatible with requirements.
1.13702	ASTM D975-13	Standard Specification for Diesel Fuel Oils	1.137-2	SRP 9.5.4	—	—	ASTM	1	Low	This specification covers seven grades of diesel fuel oils suitable for various types of diesel engines. Correct sampling procedures are critical to obtaining a representative sample of the diesel fuel oil to be tested. The recommended procedures or practices provide techniques useful in the proper sampling or handling of diesel fuels.	
—	ASTM E 185-73	Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels	—	—	—	10 CFR 50, Appendix H	ASTM	2	Low	This practice covers procedures for designing a surveillance program for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in light-water moderated nuclear power reactor vessels. New advanced light-water small modular reactor designs with a nominal design output of 300 MWe or less have not been specifically considered in this practice. This practice includes the minimum requirements for the design of a surveillance program, selection of vessel material to be included, and the initial schedule for evaluation of materials.	10 CFR 50, Appendix H requires ASTM E-185-73, ASTM E 185-79, and ASTM E 185-82. This standard is issued under the fixed designation E185; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (´) indicates an editorial change since the last revision or reapproval. The title of these standards is "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."
—	ASTM E-208-95a (Re-approved 2000)	Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels	—	—	SRP 5.4.1.1 SRP BTP 5-3 SRP 10.2 SRP 10.2.3	—	ASTM	1	Low	The ASTM E208 drop weight test is used to determine the nil-ductility transition (NDT) temperature of ferritic steels 5/8 inch thick (15.9 mm) and over. The NDT temperature is the temperature at which the fracture mode of steel changes from ductile to brittle. At temperatures above the NDT a piece of steel will stretch or deform in a ductile manner before fracturing when loaded to its ultimate tensile strength. At temperatures lower than the NDT that same piece of steel will fail in a brittle manner when loaded to its yield strength (roughly half of its ultimate tensile strength). Once a brittle failure begins the crack will continue to grow until it runs out of material, the pressure is released, or it encounters steel that is more ductile.	
1.19901	ASTM E488/E488M-15	Standard Test Methods for Strength of Anchors in Concrete Elements	1.199-1	SRP 3.8.3 SRP 3.8.4	—	—	ASTM	1	Low	The test methods in ASTM E488 address the tensile and shear strengths of post-installed and cast-in-place anchors in test members made of cracked or uncracked concrete. Loadings include quasi-static, seismic, fatigue and shock. Environmental exposures include freezing and thawing, moisture, decreased and elevated temperatures and corrosion. These test methods provide basic testing procedures for use with product-specific evaluation and acceptance standards and are intended to be performed in a testing laboratory.	These test methods are intended for use with post-installed and cast-in-place anchors designed for installation perpendicular to a plane surface of a test member. This standard prescribes separate procedures for static, seismic, fatigue and shock testing. Nothing in this standard, however, shall preclude combined tests incorporating two or more of these types of loading (such as seismic, fatigue and shock tests in series).
5.01205	ASTM F883	Standard Performance Specification for Padlocks	5.12-1	—	13.6.1 13.6.2	—	ASTM	1	Low	Physical Protection Standard is applicable to MSRs	Current revision is 2022. Reference list refers to 2004 version.
—	AWS D1.1-1981	Structural Welding Code	—	—	SRP 6.1.1	—	AWS	1	Low	The code was specifically developed for welded steel structures that use carbon or low allow steels that are 1/8 in [3 mm] or thicker with a minimum specified yield strength of 100 ksi [690 MPa] or less.	

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3.05403	DIN 25463-1	Calculation of the decay power in nuclear fuels of light water reactors - Part 1: Uranium oxide nuclear fuel for pressurized water reactors	3.54-3	—	—	—	DIN	5	Low	Not applicable to MSRs. This is a German National Standard that will not be updated for MSRs	Need decay heat calculations for the constituents of molten fuel salts
1.18002	IEC 61000-3	Electromagnetic Compatibility (EMC) - Part 3: Limits	1.180-2	SRP 3.11 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP 9.5.2	SRP Appendix 7.1-A	—	IEC	1	Low	Selected MIL-STD-461G and IEC 61000 test methods are endorsed by RG 1.180, R2. Operating envelopes specified in the standards with clarifications on application and conditions for omission provided.	
1.18002	IEC 61000-4	Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques	1.180-2	SRP 3.11 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP 9.5.2	SRP Appendix 7.1-A	—	IEC	1	Low	Selected MIL-STD-461G and IEC 61000 test methods are endorsed by RG 1.180, R2. Operating envelopes specified in the standards with clarifications on application and conditions for omission provided.	
1.18002	IEC 61000-6	Electromagnetic Compatibility (EMC) - Part 6: Generic Standards	1.180-2	SRP 3.11 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP 9.5.2	SRP Appendix 7.1-A	—	IEC	1	Low	Selected MIL-STD-461G and IEC 61000 test methods are endorsed by RG 1.180, R2. Operating envelopes specified in the standards with clarifications on application and conditions for omission provided.	
1.17103	IEEE 1008 ANSI/IEEE 1008-1987	IEEE Standard for Software Unit Testing	1.171-3	SRP 7.0-A SRP Appendix 7.1-A Table 7-1 SRP BTP 7-14	SRP Appendix 7.1-A SRP BTP 7-14 SRP Table 7-1	—	IEEE	1	Low	Software unit testing is a process that includes the performance of test planning, the acquisition of a test set, and the measurement of a test unit against its requirements.	This standard defines an integrated approach to systematic and documented unit testing. The approach uses unit design and unit implementation information, in addition to unit requirements, to determine the completeness of the testing. This standard describes a testing process composed of a hierarchy of phases, activities, and tasks and defines a minimum set of tasks for each activity.
1.16802	IEEE 1012-1998	IEEE Standard for Software Verification and Validation	1.168-2	SRP BTP 7-14 SRP BTP 7-21 SRP Appendix 7.0-A SRP Appendix 7.1-A SRP Appendix 7.1-D SRP Table 7-1	SRP Appendix 7.1-A SRP BTP 7-14 SRP Appendix 7.1-D SRP Table 7-1	—	IEEE	1	Low	The scope of V&V processes encompasses systems, software, and hardware, and it includes their interfaces. This standard applies to systems, software, and hardware being developed, maintained, or reused [legacy, commercial off-the-shelf (COTS), nondevelopmental items]. The term software also includes firmware and microcode, and each of the terms system, software, and hardware includes documentation. V&V processes include the analysis, evaluation, review, inspection, assessment, and testing of products.	Verification and validation (V&V) processes are used to determine whether the development products of a given activity conform to the requirements of that activity and whether the product satisfies its intended use and user needs. V&V life cycle process requirements are specified for different integrity levels.
1.16802	IEEE 1028-1997	IEEE Standard for Software Reviews and Audits	1.168-2	SRP BTP 7-14 SRP BTP 7-21 SRP Appendix 7.0-A SRP Appendix 7.1-A SRP Appendix 7.1-D SRP Table 7-1	SRP Appendix 7.1-A SRP Table 7-1	—	IEEE	1	Low	The purpose of this standard is to define systematic reviews and audits applicable to software acquisition, supply, development, operation, and maintenance. This standard describes how to carry out a review. Software reviews can be used in support of the objectives of project management, system engineering, verification and validation, configuration management, quality assurance, and auditing.	This standard is concerned only with the reviews and audits; procedures for determining the necessity of a review or audit are not defined, and the disposition of the results of the review or audit is not specified. Types included are management reviews, technical reviews, inspections, walk-throughs, and audits.

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1.16901	IEEE 1042 ANSI/IEEE 1042-1987	IEEE Guide to Software Configuration Management	1.169-1	SRP 7.0-A SRP Appendix 7.1-A SRP Appendix 7.1-D Table 7-1 SRP BTP 7-14	—	—	IEEE	1	Low	This guide describes the application of configuration management (CM) disciplines to the management of software engineering projects. Software configuration management (SCM) consists of two major aspects: planning and implementation.	For those planning SCM activities, this guide provides insight into the various factors that must be considered.
1.18002	IEEE 1050-1996	Guide for Instrumentation and Control Equipment Grounding in Generating Stations	1.180-2 1.204-0	SRP 3.11 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP 9.5.2 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP BTP 7-11 SRP 8.1 SRP 8.2 SRP 8.3.1	SRP 7.1-A SRP 8.2 SRP 8.3.1	—	IEEE	1	Low	This application guide was developed to identify I&C equipment grounding methods to achieve both a suitable level of protection for personnel and equipment, and suitable electric noise immunity for signal ground references. Grounding design is normally based on the concept of two separate grounding systems—the equipment ground and the signal reference ground. The concepts of equipment grounding are covered in other IEEE standards. The concepts of grounding of instrument chassis, cable shields, signal pairs, and other related instrumentation and control items require special care in order to ensure that both personnel working on equipment are adequately protected from electrical shock and that interference signals are not inadvertently coupled into signal circuits.	The typical environment in a generating station provides many sources of electrical noise such as the switching of large inductive loads, High fault currents, electronic drives, and High-energy, High-frequency transients associated with switching at transmission voltage levels. The increasing use of solid-state equipment and microprocessor-based control systems in these applications introduces a number of specific concerns with respect to electrical noise control.
1.17303	IEEE 1074-1995	IEEE Standard for Developing Software Life Cycle Processes	1.173-3	SRP Appendix 7.1-A SRP 7.0-A SRP Appendix 7.1-A SRP Table 7-1	SRP Appendix 7.1-A SRP BTP 7-14 SRP Table 7-1	—	IEEE	1	Low	This standard provides a process for creating a software project life cycle process (SPLCP). This methodology begins with the selection of an appropriate software project life cycle model (SPLCM) for use on the specific project. It continues through the definition of the software project life cycle (SPLC), using the selected SPLCM, the activities provided in Annex A, and the portion of the software life cycle that is relevant to the project. The methodology concludes with the augmentation of the software life cycle with organizational process assets (OPAs) to create the SPLCP.	This standard defines the process by which an SPLCP is developed. It can be used where software is the total system or where software is part of a larger system.
1.18904	IEEE 1202	IEEE Standard for Flame Testing of Cables for Use in Cable Trays in Industrial and Commercial Occupancies	1.189-4	SRP 3.2.1 SRP Appendix 7.1-A SRP 7.4 SRP 9.5.1 SRP 9.5.1.1 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP BTP 11-3 SRP 11.4 SRP 11.5 SRP 13.1.2-13.1.3 SRP 17.5	SRP 9.5.1.2	—	IEEE	1	Low	This standard provides a protocol for exposing cables to a theoretical 20 kW (70 000 Btu/h) flaming ignition source for a 20 min test duration. The test determines the flame propagation tendency of single-conductor and multi-conductor cables intended for use in cable trays. The purpose of this standard is to establish test protocols and performance criteria to determine the flame propagation tendency and optionally, smoke generation, of power, signal and fiber cables.	
1.18904	IEEE 242-2001	Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems (Buff Book).	1.189-4	SRP 3.2.1 SRP Appendix 7.1-A SRP 7.4 SRP 9.5.1 SRP 9.5.1.1 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP BTP 11-3 SRP 11.4 SRP 11.5 SRP 13.1.2-13.1.3 SRP 17.5	SRP 8.2 SRP 8.3.1 SRP BTP 8-3	—	IEEE	1	Low	IEEE Std 242-2001, commonly known as the IEEE Buff Book, is published as a reference source to provide a better understanding of the purpose for and techniques involved in the protection and coordination of industrial and commercial power systems. This publication presents in a step-by-step, simplified, yet comprehensive, form the principles of system protection and the proper application and coordination of those components that may be required to protect industrial and commercial power systems against abnormalities that could reasonably be expected to occur in the course of system operation. The objectives of electrical system protection and coordination are to limit the extent and duration of service interruption whenever equipment failure, human error, or adverse natural events occur on any portion of	

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										the system and to minimize damage to the system components involved in the failure.	
—	IEEE 279–1971	Criteria for Protection Systems for Nuclear Power Generating Stations	—	SRP 6.1.1 SRP 6.3 almost all of 7 SRP Table 7-1 SRP 8.1 SRP 8.3.1 SRP 8.3.2 SRP Appendix 8-A SRP BTP 8-5 SRP BTP 8-7 many sections in 7 8 9 15 BASED ON IEEE 603	Most of SRP Chapter 7 SRP 8.1 SRP 8.3.1 SRP 8.3.2 SRP Appendix 8-A SRP BTP 8-1 SRP BTP 8-6 SRP BTP 8-7 SRP BTP 8-9	10 CFR 50.55a	IEEE	5	Low	For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements in IEEE Std 279–1968, or the requirements in IEEE Std 279–1971, or the requirements in IEEE Std 603–1991, and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.	New plants or upgrades to I&C systems in existing plants follow IEEE 603 and not IEEE 279.
1.03203	IEEE 308-2001	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations	1.32-3 1.128-2	SRP 8.1 SRP 8.2 SRP 8.3.1 SRP 8.3.2 SRP BTP 8-2 SRP BTP 8-9	SRP 8.2 SRP 8.3.1 SRP 8.3.2 SRP 8.4	—	IEEE	5	Low	This standard applies to the Class 1E portions of the following systems and equipment in single-unit and multiunit nuclear power generating stations: AC power systems, DC power systems, and Instrumentation and control (I&C) power systems. Class 1E power systems shall be designed to provide that no design basis event causes the following: A loss of electric power to a number of engineered safety features, surveillance devices, or protection system devices so that a required safety function cannot be performed, A loss of electric power to equipment that could result in a reactor transient capable of causing significant damage to the fuel cladding or to the reactor coolant pressure boundary.	The working group for IEEE Std 308-2001 determined that no significant changes were required for application to newer plant designs. Several minor changes have been made. Diesel generator is replaced with standby power supply throughout the standard to allow for prime movers other than diesel engines. The requirement to have a Class 1E ac power system is removed for passive reactor designs that use natural forces to respond to accidents and operational events instead of using large ac equipment. Recognizing the importance of batteries to passive reactor designs during event response with loss of offsite power, a requirement was added to provide for reliable permanent or temporary power to reenergize battery chargers prior to the end of the battery discharge cycles.
1.06303	IEEE 317-1983	IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	1.63-3	SRP 3.11 SRP 8.1 SRP 8.3.1 SRP 8.3.2	SRP 3.11 SRP 8.1 SRP 8.3.1	—	IEEE	4		The containment in an MSR may not be leak tight so this would change the penetration requirements. Because the containments are designed and operate differently from LWR containments, there is insufficient design information to know if containment penetration requirements are applicable. For example, the standard states that “The electric penetration assembly including aperture seal(s) shall be designed to have a total gas-leak rate not greater than 1 × 10-2 std cm3/s using dry nitrogen at design pressure and ambient temperature after installation and after any DBEs (excluding direct steam jet impingement).” MSRs may not have or need this requirement, or it may be relaxed.	This standard prescribes the requirements for the design, construction, qualification, test, and installation of electric penetration assemblies in nuclear containment structures for stationary nuclear power generating stations.
1.08901	IEEE 323-1974	IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations	1.89-1 1.147-20	SRP Appendix 7.1-A SRP 11.1 SRP 12.2 SRP 12.3-12.4 SRP 13.2.2 SRP 15.0.1	—	10 CFR 50.49	IEEE	1	Low	The normal and abnormal service conditions for the equipment shall be specified. These conditions shall include the nominal values and their expected durations, as well as extreme values and their expected durations. Examples include, but are not limited to, pressure and temperature, humidity, radiation, seismic operating basis	This standard describes the basic requirements for qualifying Class 1E equipment and interfaces. The qualification requirements, when met, demonstrate and document the ability of equipment to perform safety

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				SRP 3.11 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Appendix 7.1-D SRP Table 7-1						earthquake (OBE) and nonseismic vibration, operating cycles, electrical loading and signals, condensation, chemical spray, and submergence, and EMI/RFI and power surges. The postulated design basis event (DBE) conditions during or after which the equipment is required to perform its safety function(s), shall be specified. Equipment shall be qualified for the duration of its operational performance requirement for each applicable design basis event condition, including any required post design basis event operability period.	function(s) under applicable service conditions including design basis events, reducing the risk of common-cause equipment failure. This standard does not provide environmental stress levels and performance requirements. A qualified life is not required for equipment located in a mild environment and which has no significant aging mechanisms.
1.04001	IEEE 334-2006	IEEE Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations	1.40-1	SRP 3.11	SRP 3.11	—	IEEE	1	Low	Common methods currently in use for seismic qualification by test are presented. Two approaches to seismic analysis are described, one based on dynamic analysis and the other on static coefficient analysis. Two approaches to experienced-based seismic evaluation are described, one based on earthquake experience and the other based on test experience.	Recommended practices are provided for establishing procedures that will yield data to demonstrate that the Class 1E equipment can meet its performance requirements during and/or following one safe shutdown earthquake event preceded by a number of operating basis earthquake events. This recommended practice may be used to establish tests, analyses, or experienced-based evaluations that will yield data to demonstrate Class 1E equipment performance claims or to evaluate and verify performance of devices and assemblies as part of an overall qualification effort.
1.03000	IEEE 336-1971 (ANSI N45.2.4-1972)	IEEE Recommended Practice for Installation, Inspection, and Testing for Class 1E Power, Instrumentation, and Control Equipment at Nuclear Facilities	1.30-0 1.128-2 3.27-1 3.37-0	SRP 14.2 SRP 17.1 SRP 8.1 SRP 8.3.2 SRP 14.2	—	—	IEEE	1	Low	The recommendations set forth in this recommended practice apply to the work of organizations that participate in the installation of new equipment or equipment modifications, inspections, and testing, or modification of power, instrumentation, and control equipment and systems in a nuclear facility from the time the equipment is turned over for installation until it is declared operable for service.	This recommended practice provides considerations for the pre-installation, installation, inspection, and testing of Class 1E power, instrumentation, and control equipment and systems of a nuclear facility while in the process of installing, inspecting, and testing during new construction, modification, and maintenance.
1.11803	IEEE 338-1987	Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,	1.118-3	SRP 7.2 SRP 7.3 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP BTP 7-17 SRP 8.1 SRP 8.3.1 SRP 8.3.2	SRP BTP 5-4 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP BTP 7-17 SRP 8.1 SRP 8.3.1	—	IEEE	1	Low	The safety systems shall be designed to be testable during operation of the nuclear power generating station and/or during those intervals when the station is shut down. This testability shall permit the independent testing of redundant channels and load groups while (1) maintaining the capability of these systems to respond to bona fide signals, (2) tripping the output of the channel being tested, if required, or (3) bypassing the equipment consistent with safety requirements and limiting conditions for operation. Annex C, Evaluation process for surveillance test changes, provides BWR and PWR examples. Because this Annex is informative it does not require modification.	The standard provides criteria for the performance of periodic testing of nuclear power generating station safety systems. The scope of periodic testing consists of functional tests and checks, calibration verification, and time response measurements, as required, to verify that the safety system performs its defined safety function. Post-maintenance and post-modification testing are not covered by this document.
1.10004	IEEE 344-2013	IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations	1.100-4 1.128-2	SRP 3.9.3 SRP 3.10 SRP 5.4.12 SRP BTP 7-10	SRP 3.7.3 SRP 3.8.1 SRP 3.10 SRP 5.4.12	—	IEEE	1	Low	Common methods currently in use for seismic qualification by test are presented. Two approaches to seismic analysis are described, one based on dynamic analysis and the other on static coefficient analysis. Two approaches to experienced-based seismic evaluation are described, one based on earthquake experience and the other based on test experience.	Recommended practices are provided for establishing procedures that will yield data to demonstrate that the Class 1E equipment can meet its performance requirements during and/or following one safe shutdown earthquake event preceded by a number of operating basis earthquake events. This recommended practice may be used to establish tests, analyses, or experienced-based evaluations that will

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											yield data to demonstrate Class 1E equipment performance claims or to evaluate and verify performance of devices and assemblies as part of an overall qualification effort.
1.05302	IEEE 379-2000	Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems	1.53-2	SRP 7.2 SRP 7.3 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Appendix 7.1-D SRP Table 7-1 SRP BTP 7-17 SRP BTP 7-19 SRP 8.1 SRP 8.3.1 SRP 8.3.2 SRP 15.1.1-15.1.4 SRP 15.2.1-15.2.5 SRP 15.2.6 SRP 15.2.7 SRP 15.3.1-15.3.2 SRP 15.4.4-15.4.5 SRP 15.5.1-15.5.2 SRP 15.6.1	SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Appendix 7.1-D SRP 7.2 SRP 7.3 SRP 7.9 SRP 8.3.1 SRP 8.3.2	—	IEEE	1	Low	The safety systems shall perform all required safety functions for a design basis event in the presence of the following: a) Any single detectable failure within the safety systems concurrent with all identifiable, but nondetectable failures. b) All failures caused by the single failure. c) All failures and spurious system actions that cause, or are caused by, the design basis event requiring the safety function. The single failure could occur prior to, or at any time during, the design basis event for which the safety system is required to function.	This standard covers the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power generating station safety systems.
1.07300	IEEE 382-1972 (ANSI N41.6)	IEEE Trial Use Guide for Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations	1.73-0	SRP 3.11	—	—	IEEE	1	Low	The primary objective of qualification is to demonstrate with reasonable assurance that safety-related actuators for which a qualified life or condition has been established can perform their safety function(s) without experiencing common-cause failures before, during, and after applicable design basis events. Safety-related actuators, with their interfaces, must meet or exceed the equipment specification requirements. This continued capability is ensured through a program that includes, but is not limited to, design control, quality control, qualification, installation, maintenance, periodic testing, and surveillance.	This standard establishes criteria for qualification of safety-related actuators, and actuator components, in Nuclear Power Generating Stations in order to demonstrate their ability to perform their intended safety functions.
1.18904	IEEE 383-2003	IEEE Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations	1.189-4 1.211-0	SRP 3.2.1 SRP Appendix 7.1-A SRP 7.4 SRP 9.5.1 SRP 9.5.1.1 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP BTP 11-3 SRP 11.4 SRP 11.5 SRP 13.1.2-13.1.3 SRP 17.5	SRP 3.11 SRP 9.5.1.2	—	IEEE	1	Low	Degradation with time (aging), followed by exposure to the environmental extremes of temperature, pressure, humidity, radiation, mechanical stress, or chemical spray or a combination of these resulting from DBEs present a potential for common-cause failures of Class 1E cable and splices. A qualified life is not required for cables and splices located in a mild environment, if the cables and splices are operated within the limits established by applicable specifications and standards. Qualification by analysis alone is not acceptable.	This standard provides general requirements and methods for qualifying electric cables, and splices for nuclear facilities. An objective of qualification is to establish a qualified life for cables and splices that are installed in environmentally harsh areas and must perform a safety function during and following a DBE.
1.07503	IEEE 384-1992	Standard Criteria for Independence of Class 1E Equipment and Circuits	1.75-3 1.128-2	SRP 7.2 SRP 7.3 SRP 7.5 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP BTP 7-10	SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP BTP 7-11 SRP 8.3.1	10 CFR 50, Appendix R	IEEE	1	Low	Physical separation and electrical isolation shall be provided to maintain the independence of Class 1E circuits and equipment so that the safety functions required during and following any design basis event can be accomplished.	This standard describes the independence requirements of the circuits and equipment comprising or associated with Class 1E systems. It sets forth criteria for the independence that can be achieved by physical separation and electrical isolation of circuits and equipment that are

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				SRP BTP 7-11 SRP Table 7-1 SRP 8.1 SRP 8.3.1 SRP 8.3.2 10 CFR 50, Appendix R							redundant but does not address the determination of what is to be considered redundant.
1.00904	IEEE 387-1995	IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations	1.9-4	SRP 8.1 SRP 8.3.1 SRP 8.4	SRP 8.3.1	—	IEEE	1	Low	The purpose of this standard is to provide the principal design criteria, the design features, testing, and qualification requirements for the individual diesel-generator units that enable them to meet their functional requirements as a part of the standby power supply under the conditions produced by the design basis events cataloged in the Plant Safety Analysis.	This standard describes the criteria for the application and testing of diesel-generator units as Class 1E standby power supplies in nuclear power generating stations. Diesel generators in the new plant designs are not safety-related.
1.12902	IEEE 450-2002	Recommended Practice for Maintenance, Testing and Replacement of Vented Lead-Acid Batteries for Stationary Applications	1.129-2 1.128-2 5.44-3	SRP 8.1 SRP 9.3.2	SRP 8.3.2	—	IEEE	1	Low	This document provides recommended maintenance, test schedules, and testing procedures that can be used to optimize the life and performance of permanently-installed, vented lead-acid storage batteries used in standby service. It also provides guidance to determine when batteries should be replaced. This recommended practice is applicable to standby service stationary applications where a battery charger normally maintains the battery fully charged and provides the dc loads. Physical Protection Standard is applicable to MSRs	The purpose of this recommended practice is to provide the user with information and recommendations concerning the maintenance, testing, and replacement of vented lead-acid batteries used in stationary applications. Most recent version is 2020. RG refers to 1987 version.
1.12802	IEEE 484-2002	Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications	1.128-2	SRP 8.1 SRP 8.3.2 SRP 14.2	SRP 8.3.2	—	IEEE	1	Low	This recommended practice provides recommended design practices and procedures for storage, location, mounting, ventilation, instrumentation, preassembly, assembly, and charging of vented lead-acid batteries. Required safety practices are also included. This recommended practice is applicable to full float stationary applications where a battery charger normally maintains the battery fully charged and provides the direct current (dc) loads.	This recommended practice is meant to provide organizations with criteria to be used for storage, location, mounting, ventilation, instrumentation, preassembly, assembly, and charging of vented lead-acid batteries.
1.21201	IEEE 485-2010 (IEEE 485-1997)	IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications	1.212-1	—	SRP 8.3.2 SRP 8.4	—	IEEE	1	Low	Some factors relating to cell selection are provided for consideration. Installation, maintenance, qualification, testing procedures, and consideration of battery types other than lead-acid are beyond the scope of this recommended practice. Design of the dc system and sizing of the battery charger(s) are also beyond the scope of this recommended practice.	This recommended practice describes methods for defining the dc load and for sizing a lead-acid battery to supply that load for stationary battery applications in full float operations.
1.09705	IEEE 497-2016	IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations	1.97-5	SRP 3.10 SRP 3.11 SRP 6.2.1.1 SRP BTP 7-10 SRP 9.3.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 12.4 SRP 12.5 SRP 13.3 SRP 14.3.7 SRP 14.3.9 SRP 14.3.11 SRP 18.0	SRP 7.5 SRP Appendix 7.1-A SRP BTP 7-10 SRP 11.5	—	IEEE	1	Low	The purpose of this standard is to establish selection, design, performance, qualification and display criteria for accident monitoring instrumentation. It provides guidance on the use of portable instrumentation and examples of accident monitoring display configurations.	This standard contains the functional and design criteria for accident monitoring instrumentation for nuclear power generating stations. This standard is intended for new plant designs and for operating nuclear power generating stations desiring to perform design modifications.
1.15801	IEEE 535-1986	IEEE Standard for Qualification of Class 1E Lead	1.158-1	SRP 3.11	—	—	IEEE	1	Low	The users of Class 1E lead storage batteries are required to provide assurance that such equipment meets or exceeds its design specifications throughout its installed	This standard describes qualification methods for Class 1E vented lead acid batteries and racks to be used in nuclear

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		Storage Batteries for Nuclear Power Generating Stations								life. This is accomplished through a quality assurance program that includes design, qualification, production, quality control, installation, maintenance, and periodic testing.	power generating stations outside primary containment.
1.15601	IEEE 572-2006	IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations	1.156-1	SRP 3.11	—	—	IEEE	1	Low	It is required that Class 1E Connection Assemblies meet or exceed the specified performance requirements throughout their installed life. This is accomplished through a quality assurance program. It is the degradation with time (aging), followed by exposure to the environmental extremes of temperature, pressure, humidity, radiation, vibration, or chemical spray resulting from design basis events (DBE), or a combination of these, which presents a potential for causing common-mode failures of Class 1E Connection Assemblies. For these reasons it is necessary to establish a qualified life and qualified condition for Connection Assemblies required to function during and/or following a DBE.	This standard provides basic requirements, direction, and methods for qualifying Class 1E Connection Assemblies for service in nuclear power generating stations. These include connectors, terminations, and environmental seals in combination with related cables or wires as assemblies. This standard does not apply to containment electric penetrations, fire stops, in-line splices, or components for service within the reactor vessel. The qualification requirements in this standard, when met, demonstrate and document the ability of the equipment to perform safety function(s) under applicable service conditions (including design basis events) reducing the risks of common cause-equipment failures.
1.04701	IEEE 603-1991	IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations	1.47-1 1.53-2 1.153-1	SRP 6.1.1 SRP 6.3 SRP 7.2 SRP 7.5 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP BTP 7-2 SRP BTP 7-3 SRP BTP 7-17 SRP Table 7-1 SRP 8.1 SRP 8.3.1 SRP 8.3.2 SRP Appendix 8-A SRP BTP 8-5 SRP BRP 8-7 SRP 7.2 SRP 7.3 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Appendix 7.1-D SRP Table 7-1 SRP BTP 7-17 SRP BTP 7-19 SRP 8.1 SRP 8.3.1 SRP 8.3.2 SRP 15.1.1-15.1.4 SRP 15.2.1-15.2.5 SRP 15.2.6 SRP 15.2.7 SRP 15.3.1-15.3.2	SRP 5.2.1.1 SRP BTP 5-2 SRP Chapter 7 SRP Chapter 8 SRP 9.2.2 SRP 9.2.7 SRP 14.3 SRP 14.3.5	10 CFR 50.55a	IEEE	2	Low	A specific basis shall be established for the design of each safety system of the nuclear power generating station. The design basis shall also be available as needed to facilitate the determination of the adequacy of the safety system, including design changes. The safety systems shall, with precision and reliability, maintain plant parameters within acceptable limits established for each design basis event. The power, instrumentation, and control portions of each safety system shall be comprised of more than one safety group of which any one safety group can accomplish the safety function. The Appendixes, which are not a part of IEEE Std 603-1991, provide examples for illustration purposes using LWR DBEs.	This standard establishes minimum functional design criteria for the power, instrumentation, and control portions of nuclear power generating station safety systems. These criteria are established to provide a means for promoting safe practices for design and evaluation of safety system performance and reliability.

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				SRP 15.4.4-15.4.5 SRP 15.5.1-15.5.2 SRP 15.6.1 SRP Appendix 7.1-C SRP 8.1 SRP 8.3.21 SRP 8.3.2 SRP Appendix 8-A SRP BTP 8-1 SRP BTP 8-5 SRP BTP 8-6 SRP BTP 8-7 SRP BTP 8-9							
1.15102	IEEE 622-1987	IEEE Recommended Practice for the Design and Installation of Electric Heat Tracing Systems for Nuclear Power Generating Systems	1.151-2	SRP 3.2.1 SRP 3.2.2 SRP Appendix 7.1-A SRO Appendix 7.1-B SRP Appendix 7.1-C SRP 7.5 SRP 7.7 SRP 11.5	SRP Table 7.1	—	IEEE	1	Low	The use of electric heat tracing systems to prevent the temperature of fluids from dropping to or below the freezing point of the fluid in important or critical piping systems, which could be very important at MSRs. The purpose IEEE 622 is to provide recommendations that may be used in the design, installation, and maintenance of electric heat tracing systems as applied to mechanical piping systems. These recommendations are intended to ensure that the piping systems will be maintained at specified operating temperatures, which in turn will ensure that the piping systems' fluids will be available not only during station operation but also during normal and emergency shutdown.	One reference and one citation in an Informative Appendix refer to light-water cooled reactors. No change needed for this.
1.21300	IEEE 649-2006	ç	1.213-0	—	—	—	IEEE	1	Low	The manufacturers and users of Class 1E motor control centers are required to provide assurance that such equipment can meet or exceed its specific performance requirements throughout its installed life. This is accomplished through a quality assurance program that includes, but is not limited to, design, qualification, production quality control, installation, maintenance, surveillance, and periodic testing. This standard treats only the qualification portion of the program.	This standard describes the basic principles, requirements, and methods for qualifying Class 1E motor control centers for both harsh and mild environment applications in nuclear power generating stations.
1.21000	IEEE 650-2006	IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations	1.210-0	—	—	—	IEEE	1	Low	Specifications for the equipment to be qualified include the equipment identification, the Class 1E performance characteristics, the input power supply, the environmental conditions, and the effect of changes in input power supply and environmental conditions upon the Class 1E performance characteristics. If the equipment specification includes margins, their values shall be identified.	This standard describes methods for qualifying static battery chargers and inverters for Class 1E installations outside containment in nuclear power generating stations.
1.20400	IEEE 665-1995	IEEE Guide for Generating Station Grounding	1.204-0	SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP BTP 7-11 SRP 8.1 SRP 8.2 SRP 8.3.1	SRP Table 7-1 SRP 8.1 SRP 8.2 SRP 8.3.1	—	IEEE	1	Low	IEEE 665-1005 provides a guide for the design of generating station grounding systems and for grounding practices applied to generating station indoor and outdoor structures and equipment, including the interconnection of the station and substation grounding systems. Guidance for the grounding of control and instrumentation equipment in generating stations can be found in IEEE Std 1050-1989.	This guide was developed to identify grounding practices that have generally been accepted by the electric utility industry as contributing to effective grounding systems for personnel safety and equipment protection in generating stations.
1.20400	IEEE 666-1991	IEEE Design Guide for Electrical Power Service Systems for Generating Stations	1.204-0	SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP BTP 7-11 SRP 8.1 SRP 8.2 SRP 8.3.1	SRP Table 7-1 SRP 8.1 SRP 8.2 SRP 8.3.1	—	IEEE	1	Low	When electric power for auxiliary loads is supplied from the power grid, the service system begins at the point where the tap from the power grid terminates, either at a station service bus or at the terminals of the transformer that supplies the bus. This guide contains a listing of typical power plant auxiliary loads and criteria for their power service and examples of single-line diagrams for a typical plant. It also includes tables of typical power service parameters to illustrate the range of typical values	This design guide applies to station service systems that supply electric power to auxiliary loads for generating stations that produce electric power.

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										for each parameter, and it identifies the approximate effect of the minimum and maximum value of each parameter on the load. The standard discusses on-site ac power sources, diesel generators, turbine generators, etc. but only recognizes there use and does not require their use.	
1.15203	IEEE 7-4.3.2-2003	IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations	1.152-3 5.71-1	SRP 7.1 SRP 7.2 SRP 7.3 SRP 7.4 SRP 7.5 SRP 7.6 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Appendix 7.1-D SRP Appendix 7.0-A SRP Table 7-1 SRP BTP 7-11 SRP BTP 7-12 SRP BTP 7-14 SRP BTP 7-17 SRP BTP 7-18 SRP BTP 7-19 SRP BTP 7-21 SRP 14.3 SRP 17.3	SRP 7.1 SRP Appendix 7.1-D SRP 7.2 (II) SRP 7.3 (II) SRP 7.5(II) SRP Ch 7.4 Ref. 3 SRP Ch 7.6 Ref. 3 SRP Ch 7.0 Ref. 3 SRP Ch 7.0-A Ref. 7 SRP BTP 7-9 SRP BTP 7-14 Ref. 5	—	IEEE	1	Low	This standard serves to amplify criteria in IEEE Std 603-1998 to address the use of computers as part of safety systems in nuclear power generating stations. The criteria contained herein, in conjunction with criteria in IEEE Std 603-1998, establish minimum functional and design requirements for computers used as components of a safety system. Physical Protection Standard is applicable to MSRs	This standard specifies additional computer-specific requirements (incorporating hardware, software, firmware, and interfaces) to supplement the criteria and requirements of IEEE Std 603-1998. Most recent version is 2016. Reference list refers to 2003 version.
1.06303	IEEE 741-1997 IEEE 741-2022 is latest (IEEE 741-1986)	Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations	1.63-3	SRP 3.11 SRP 8.1 SRP 8.3.1 SRP 8.3.2	SRP 8.2 SRP 8.3.1 SRP 8.3.2	—	IEEE	1	Low	The electrical penetration assemblies installed as part of the containment structure may require special consideration in the selection of their protection. This special consideration arises where the potential exists for a fault inside containment to result in a penetration seal failure, such that a breach of containment could occur. Where a penetration assembly can indefinitely withstand the maximum current available due to a fault inside containment, no special consideration is required.	The criteria for protection to protect against the maximum current is applicable.
1.16901	IEEE 828-1990	IEEE Standard for Configuration Management Plans	1.169-1	SRP 7.0-A SRP Appendix 7.1-A SRP Appendix 7.1-D Table 7-1 SRP BTP 7-14	SRP Appendix 7.1-A SRP BTP 7-14 SRP Appendix 7.1-D SRP Table 7-1	—	IEEE	1	Low	SCM planning information shall be partitioned into the six classes: Introduction-Describes the Plans purpose, scope of application, key terms, and references SCM management-(Who?) Identifies the responsibilities and authorities for accomplishing the planned activities SCM activities-(What?) Identifies all activities to be performed in applying to the project SCM schedules-(When?) Identifies the required coordination of SCM activities with the other activities in the project SCM resources-(How?) Identifies tools and physical and human resources required for execution of the Plan SCM plan maintenance-Identifies how the Plan will be kept current while in effect	This standard establishes the minimum required contents of a Software Configuration Management (SCM) Plan (the Plan). It is supplemented by IEEE Std 1042-1987, which provides approaches to good software configuration management planning. This standard applies to the entire life cycle of critical software, e.g., where failure would impact safety or cause large financial or social losses. It also applies to noncritical software and to software already developed.
1.17003	IEEE 829-2008 ANSI/IEEE 829-1983	IEEE Standard for Software Test Documentation	1.170-3	SRP 7.0-A SRP Appendix 7.1-A Table 7-1 SRP BTP 7-14	SRP Appendix 7.1-A SRP BTP 7-14 SRP Appendix 7.1-D SRP Table 7-1	—	IEEE	1	Low	This standard supports all software life cycle processes, including acquisition, supply, development, operation, and maintenance.	This standard applies to all software-based systems. It applies to systems and software being developed, acquired, operated, maintained, and/or reused [e.g., legacy, modified, Commercial-Off-the-Shelf (COTS), Government-Off-the-Shelf (GOTS), or Non-Developmental Items (NDIs)].
1.17203	IEEE 830-1993	IEEE Recommended Practice for	1.172-3	SRP 7.0-A SRP Appendix 7.1-A	SRP Appendix 7.1-A	—	IEEE	1	Low	This recommended practice describes recommended approaches for the specification of software requirements. A good SRS should provide several specific benefits,	The content and qualities of a good software requirements specification (SRS) are described, and several

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		Software Requirements Specifications		Table 7-1 SRP BTP 7-14	SRP BTP 7-14 SRP Table 7-1					such as the following: Establish the basis for agreement between the customers and the suppliers on what the software product is to do, Reduce the development effort, Provide a basis for estimating costs and schedules, Provide a baseline for validation and verification, Facilitate transfer of the software product to new users or new machines, and Serve as a basis for enhancement.	sample SRS outlines are presented. This recommended practice is aimed at specifying requirements of software to be developed but also can be applied to assist in the selection of in-house and commercial software products.
1.10004	IEEE C37.98-2013	IEEE Standard for Seismic Qualification Testing of Protective Relays and Auxiliaries for Nuclear Facilities	1.100-3 no revision	SRP 3.9.3 SRP 3.10 SRP 5.4.12 SRP BTP 7-10	SRP 3.10	—	IEEE	1	Low	IEEE C37.98 describes the methods and conditions for seismic qualification of protective relays and auxiliaries such as test and control switches, terminal blocks, and indicating lamps for use in nuclear facilities. Earlier standards had an emphasis on fragility testing of relays. The primary intent of this standard is to focus on seismic qualification, also known as proof testing (either to generic levels or specific levels), rather than fragility testing.	
1.20400	IEEE C62.23-1995	IEEE Application Guide for Surge Protection of Electric Generating Plants	1.204-0	SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP BTP 7-11 SRP 8.1 SRP 8.2 SRP 8.3.1	SRP Table 7-1 SRP Appendix 7.1-A SRP 8.1 SRP 8.2 SRP 8.3.1	—	IEEE	1	Low	To provide an understanding for consistent and comprehensive surge protection and to reduce interference, the power generating plant has been divided in this guide into four subareas: the power lines, the switchyard, the power plant, and the remote ancillary systems. Within each subarea, the “surge environment” in which the associated equipment and systems are required to operate is addressed in terms of the common overvoltage and electromagnetic interference sources identified as: — Direct lightning strokes, — Incoming surges, — Internally generated surges, — Ground potential rise, — Electromagnetic interference	Surge overvoltages can cause equipment damage, system malfunction, or power interruptions at electric power generating plants if plants are not adequately protected against them. Excessive surge voltages have to, therefore, be controlled or reduced to permissible levels. These overvoltage surges in power generating plants may be generated by lightning or by system events such as switching, faults, load rejections, or by some combinations of these.
1.18002	IEEE C62.41.1-2002 (IEEE C41.1-1991 is inactive)	IEEE Guide on the Surge Environment in Low-Voltage (1000 V and Less) AC Power Circuits	1.180-2	SRP 3.11 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP 9.5.2	SRP Appendix 7.1-A SRP Table 7-1	—	IEEE	1	Low	IEEE Std C62.41.1-2002 the first of a Trilogy of three IEEE standards addressing surges in Low-voltage ac power circuits, focuses on the surge environment and on the TOV environment. This part provides readers with basic information on the occurrence of surges, as a database for the second document of the Trilogy, IEEE Std C62.41.2-2002 where recommendations are presented on the selection of representative surge parameters to be considered in assessing equipment immunity and performance of SPDs. The third document of the Trilogy, IEEE Std C62.45-2002, presents recommendations on surge testing procedures for obtaining reliable measurements and enhancing operator safety.	This is a guide describing the surge voltage, surge current, and TOV environment in Low-voltage [up to 1000 V root mean square (rms)] ac power circuits. This scope does not include other power disturbances, such as notches, sags, and noise.
1.18002	IEEE C62.41.2-2002 (IEEE C62.41-1991 is inactive)	IEEE Recommended Practice on Characterization of Surges in Low-Voltage (1000 V and Less) AC Power Circuits	1.180-2	SRP 3.11 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP 9.5.2	SRP Appendix 7.1-A SRP Table 7-1 SRP BTP 7-11	—	IEEE	1	Low	IEEE Std C62.41.1-2002 the first of a Trilogy of three IEEE standards addressing surges in Low-voltage ac power circuits, focuses on the surge environment and on the TOV environment. This part provides readers with basic information on the occurrence of surges, as a database for the second document of the Trilogy, IEEE Std C62.41.2-2002 where recommendations are presented on the selection of representative surge parameters to be considered in assessing equipment immunity and performance of SPDs. The third document of the Trilogy, IEEE Std C62.45-2002, presents recommendations on surge testing procedures for obtaining reliable measurements and enhancing operator safety.	The scope of this recommended practice is to characterize the surge environment at locations on ac power circuits described in IEEE Std C62.41.1-2002 by means of standardized waveforms and other stress parameters.
1.18002	IEEE C62.45-1992 (IEEE C62.45)	IEEE Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Circuits	1.180-2	SRP 3.11 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C	SRP Appendix 7.1-A	—	IEEE	1	Low	IEEE Std C62.41.1-2002 the first of a Trilogy of three IEEE standards addressing surges in Low-voltage ac power circuits, focuses on the surge environment and on the TOV environment. This part provides readers with basic information on the occurrence of surges, as a database for the second document of the Trilogy, IEEE	The scope of this recommended practice is the performance of surge testing on electrical and electronic equipment connected to Low-voltage ac power circuits, specifically using the

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				SRP Table 7-1 SRP 9.5.2						Std C62.41.2-2002 where recommendations are presented on the selection of representative surge parameters to be considered in assessing equipment immunity and performance of SPDs. The third document of the Trilogy, IEEE Std C62.45-2002, presents recommendations on surge testing procedures for obtaining reliable measurements and enhancing operator safety.	recommended test waveforms defined in IEEE Std C62.41.2™-2002.
3.05403	ISO 10645:1992	Nuclear Energy -- Light Water Reactors -- Calculation of The Decay Heat Power in Nuclear Fuels	3.54-3	—	—	—	ISO	3	Medium	LWR-specific. Require calculation of decay heat in molten salts.	Updated 2022 Need decay heat calculations for the constituents of molten fuel salts
3.07103	ISO 11311:2011	Nuclear Criticality Safety - Critical Values for Homogeneous Plutonium-Uranium Oxide Fuel Mixtures Outside of Reactors	3.71-3	—	—	—	ISO	1	Low	Applicable to MSRs	
3.07103	ISO 11320:2011	Nuclear Criticality Safety - Emergency Preparedness and Response	3.71-3	—	—	—	ISO	1	Low	Applicable to MSRs	
3.07103	ISO 14943:2004	Nuclear Fuel Technology - Administrative Criteria Related to Nuclear Criticality Safety	3.71-3	—	—	—	ISO	1	Low	Applicable to MSRs	
—	ISO 1496 (1978)	General Cargo Containers	—	—	—	10 CFR 73.26	ISO	1	Low	10 CFR 73.26 requires that *Shipment by sea* shall be made only on container-ships. The strategic special nuclear material container(s) shall be loaded into exclusive use cargo containers conforming to American National Standards Institute ANSI) Standard MH5.1--"Basic Requirements for Cargo Containers" (1971) or International Standards Organization (ISO) 1496, "General Cargo Containers" (1978). Locks and seals shall be inspected by the escorts whenever access is possible.	
3.07103	ISO 16117:2013	Nuclear Criticality Safety - Estimation of The Number of Fissions of a Postulated Criticality Accident	3.71-3	—	—	—	ISO	1	Low	Applicable to MSRs	
3.07103	ISO 1709:1995	Nuclear Energy -- Fissile Materials -- Principles of Criticality Safety in Storing, Handling and Processing	3.71-3	—	—	—	ISO	1	Low	Applicable to MSRs	Current version is 2018
3.07103	ISO 27467:2009	Nuclear Criticality Safety - Analysis of A Postulated Criticality Accident	3.71-3	—	—	—	ISO	1	Low	Applicable to MSRs	
3.07103	ISO 27468:2011	Nuclear Criticality Safety - Evaluation of Systems Containing PWR	3.71-3	—	—	—	ISO	1	Low	Applicable to MSRs	

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		UOX Fuels - Bounding Burnup Credit Approach									
—	ISO 389	Standard Reference Zero for the Calibration of Puritone Audiometer	—	—	—	10 CFR 73, Appendix B	ISO	1	N/A	Hearing: (a) Individuals shall have no hearing loss in the better ear greater than 30 decibels average at 500 Hz, 1,000 Hz, and 2,000 Hz with no level greater that 40 decibels at any one frequency by ISO 389 ANSI S3.6-1969.	
3.07103	ISO 7753:1987	Nuclear Energy -- Performance and Testing Requirements for Criticality Detection and Alarm Systems	3.71-3	—	—	—	ISO	1	Low	Applicable to MSRs	
3.02100	MIL-C-81302B	Type I and Type II TRICHLOROTRIFLUOROETHANE	3.21-0	—	—	—	MIL	4		Cleaning solvents; unknown if this would apply to MSR materials.	Although a MIL-STD may be considered as a standard from an Industry Group (Non-SDO), they are cited as standards and included in this review.
5.01201	MIL-DTL-29181	Hasp, High Security, Shrouded, for High and Medium Security Padlock	5.12-1	—	—	—	MIL	1	Low	Physical Protection Standard is applicable to MSRs	Most recent version is Rev D, 2019. Reference list refers to 1998 version.
5.01201	MIL-DTL-43607H	Padlock, Key Operated, High Security, Shrouded Shackle	5.12-1	—	—	—	MIL	1	Low	Physical Protection Standard is applicable to MSRs	Most recent version is Rev J, 2010, MIL-DTL-43607H is dated 1998 in reference list.
1.18002	MIL-STD-461G	Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment	1.180-2	SRP 3.11 SRP 7.9 SRP Appendix 7.1-A SRP Appendix 7.1-B SRP Appendix 7.1-C SRP Table 7-1 SRP 9.5.2	SRP Appendix 7.1-A	—	MIL	1	Low	This standard establishes interface and associated verification requirements for the control of the electromagnetic interference (EMI) emission and susceptibility characteristics of electronic, electrical, and electromechanical equipment and subsystems designed or procured for use by activities and agencies of the Department of Defense (DoD). The emissions and susceptibility and associated test procedure requirements in this standard are designated in accordance with an alphanumeric coding system. Conducted emissions, Radiated emissions, Conducted susceptibility, and Radiated susceptibility.	Although a MIL-STD may be considered as a standard from an Industry Group (Non-SDO), they are cited as standards and included in this review.
5.01201	NFPA 101-2012	Life Safety Code	5.12-1	—	—	—	NFPA	1	Low	Changes also require addressing MSR-specific fire issues.	Most recent version is 2021. Reference list refers to 2012 version.
—	NFPA 232	Standard for the Protection of Records	—	—	SRP 17.1	—	NFPA	1	Low	This standard provides requirements for records protection equipment and facilities and records-handling techniques that safeguard records in a variety of media forms from the hazards of fire and its associated effects.	
1.18904	NFPA 251	Standard Methods of Tests of Fire Resistance of Building Construction and Materials	1.189-4	SRP 3.2.1 SRP Appendix 7.1-A SRP 7.4 SRP 9.5.1 SRP 9.5.1.1 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP BTP 11-3 SRP 11.4 SRP 11.5	—	—	NFPA	3	High	This standard provides methods of fire tests for the fire-resistive properties of building members and assemblies.	This standard specifies methods for determining the fire-resistive abilities of building members and assemblies. Changes are necessary to address the varied advanced reactor design SSCs.

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				SRP 13.1.2-13.1.3 SRP 17.5							
1.18904	NFPA 600	Standard on Industrial Fire Brigades	1.189-4	SRP 3.2.1 SRP Appendix 7.1-A SRP 7.4 SRP 9.5.1 SRP 9.5.1.1 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP BTP 11-3 SRP 11.4 SRP 11.5 SRP 13.1.2-13.1.3 SRP 17.5	—	—	NFPA	1	Low	This standard contains minimum requirements for organizing, operating, training, and equipping facility fire brigades when responding to fires in industrial, commercial, institutional, and similar properties. This standard provides minimum requirements for the occupational safety and health of facility fire brigade members while performing firefighting and related response activities.	
3.01502	NFPA 801	Standard for Fire Protection for Facilities Handling Radioactive Materials	3.15-2	—	—	—	NFPA	2	Low	Use of this RG assumes MSR fuel fabrication may be implemented at a facility on or adjacent to the reactor site. Changes require addressing MSR-specific fire issues.	May need to consider alternatives to water-based fire suppression systems due to High temperatures required to maintain salt in liquid form - steam explosion potential.
1.19101	NFPA 805	Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants	1.191-1 1.205-2 3.15-2	SRP 9.5.1.1 SRP 9.1.5.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 9.5.1.1 SRP 9.5.1.2 SRP 19.1	SRP 9.5.1.1 SRP 9.5.1.2 SRP 19.1	10 CFR 50.48(c)	NFPA	3	High	The four goals of NFPA 805, and thus NEI 04-02, are: the nuclear safety goal, the radioactive release goal, the life safety goal, and the plant damage/business interruption goal. Many fire issues are specific/involve BWR and PWR specific designs. Changes require addressing MSR-specific fire issues.	10 CFR 50.48 endorses with exceptions the NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," as a voluntary acceptable approach for demonstrating compliance with 10 CFR 50.48 Section (b) and Section (f). NEI 04-02 provides guidance for implementing the requirements of this rule, and to the degree endorsed by the NRC, represents methods acceptable to the NRC for implementing in whole or in part a risk-informed, performance-based fire protection program. Definitions used in NEI 04-02 are contained in Chapter 3 of NFPA 805.
1.18904	NFPA E119	Standard Test Methods for Fire Tests of Building Construction and Materials	1.189-4	SRP 3.2.1 SRP Appendix 7.1-A SRP 7.4 SRP 9.5.1 SRP 9.5.1.1 SRP 9.5.2 SRP 11.2 SRP 11.3 SRP BTP 11-3 SRP 11.4 SRP 11.5 SRP 13.1.2-13.1.3 SRP 17.5	—	—	NFPA	1	Low	The fire test methods described in NFPA E119 are applicable to assemblies of masonry units and to composite assemblies of structural materials for buildings, including loadbearing and other walls and partitions, columns, girders, beams, slabs, and composite slab and beam assemblies for floors and roofs. They are also applicable to other assemblies and structural units that constitute permanent integral parts of a finished building.	
5.00701	NILECJ-STD-0601.00	Walk-Through Metal Detectors for Use in Weapons Detection	5.7-1	—	—	—	NILECJ	1	Low	Physical Protection Standard is applicable to MSRs	Superseded by NIJ Standard–0601.01
5.07101	NIST SP 800-37 (2004)	Guide to Certification and Accreditation of Federal Information Systems	5.71-1	—	—	—	NIST	1	Low	Security guidance applicable to MSRs	Although a standard from NIST (NBS) is not a standard from an SDO, they are cited as standards and included in this review.
5.07101	NIST SP 800-53, Rev. 3 (2009)	Recommended Security Controls	5.71-1	—	—	—	NIST	1	Low	Security guidance applicable to MSRs	Although a standard from NIST (NBS) is not a standard from an SDO, they are

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		for Federal Information Systems									cited as standards and included in this review.
5.07101	NIST SP 800-64, Rev. 2 (2008)	Security Considerations in the System Development Life Cycle	5.71-1	—	—	—	NIST	1	Low	Security guidance applicable to MSRs	Although a standard from NIST (NBS) is not a standard from an SDO, they are cited as standards and included in this review.
5.07101	NIST SP 800-82 (2008)	Guide to Industrial Control Systems Security	5.71-1	—	—	—	NIST	1	Low	Security guidance applicable to MSRs	Although a standard from NIST (NBS) is not a standard from an SDO, they are cited as standards and included in this review.
5.07101	NIST SP 800-86 (2006)	Guide to Integrating Forensic Techniques into Incident Response	5.71-1	—	—	—	NIST	1	Low	Security guidance applicable to MSRs	Although a standard from NIST (NBS) is not a standard from an SDO, they are cited as standards and included in this review.
5.07101	NIST SP-50	Building an Information Technology Security Awareness and Training Program	5.71-1	—	—	—	NIST	1	Low	Security guidance applicable to MSRs	Although a standard from NIST (NBS) is not a standard from an SDO, they are cited as standards and included in this review.
5.02101	NSRDS-NBS 29, 1969	Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 keV to 100 GeV	5.21-1	—	—	—	NBS	1	Low	Assay guidance may be applicable to MSRs	Although a standard from NIST (NBS) is not a standard from an SDO, they are cited as standards and included in this review.
1.05403	SSPC PA 2 (Society for Protective Coatings (SSPC))	Procedure for Determining Conformance to Dry Coating Thickness Requirements	1.54-3	SRP 6.1.1 SRP 6.1.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.3	—	—	SSPC	1	Low	This standard describes a procedure for determining shop or field conformance to a specified coating dry film thickness (DFT) range on ferrous and non-ferrous metal substrates using two types of nondestructive coating thickness gages (Type 1, magnetic pull-off, and Type 2, electronic) described in ASTM D7091. This standard defines a procedure to determine whether coatings conform to the minimum and the maximum thickness specified.	
5.00902	TID 20893 Rev 3 1969 published by NIST	Standard Nuclear Instrument Modules	5.9-2	—	—	—	NIST	1	Low	Standard provides a common footprint for electronic nuclear instrument modules or NIMs.	Although a standard from NIST (NBS) is not a standard from an SDO, they are cited as standards and included in this review.
5.01201	UL 1034-2000	Burglary-Resistant Electric Locking Mechanisms	5.12-1	—	—	—	UL	1	Low	Physical Protection Standard is applicable to MSRs	Current revision is 2020. Reference list refers to 2000 version.
5.01201	UL 437-2004	Key Locks	5.12-1	—	—	—	UL	1	Low	Physical Protection Standard is applicable to MSRs	Current revision is 2017. Reference list refers to 2004 version.
3.03200	UL 586	UL Standard for Safety High-Efficiency, Particulate, Air Filter Units	3.32-0	—	—	—	UL	1	Low	Use of this RG assumes MSR fuel fabrication may be implemented at a facility on or adjacent to the reactor site.	UL Standard would be applicable to NPPs, but is called out by Design Guidance for Fuel Reprocessing
—	UL 752	Standard for Bullet-Resisting Equipment	—	—	SRP 13.6.2	—	UL	1	Low	Per UL 752, this test method is conducted to determine whether "…protection is provided against complete penetration, passage of fragments of projectiles, or spalling (fragmentation) of the protective material to the degree that injury would be caused to a person standing directly behind the bullet-resisting barrier." Materials, devices, and fixtures, as well as building components and	

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										electrically-operated equipment, used to form bullet-resisting barriers which protect against robbery, holdup, or armed attack.	
5.01201	UL 768-2006	Standard for Combination Locks	5.12-1	—	—	—	UL	1	Low	Physical Protection Standard is applicable to MSRs	Current revision is 2018. Reference list refers to 2006 version.

APPENDIX B. WITHDRAWN STANDARDS

RG	Standard	Title	Current Rev	Status	Comments
5.71	ANSI ISO/IEC 17799 (2005)	Information Technology Security Techniques: Code of Practice for Information Security Management	2005	Inactive	
5.71	ANSI ISO/IEC TR 13335-1-1996	Information Technology: Guidelines for the Management of IT Security—Part 1: Concepts and Models	2004	Inactive	Superseded by ISO/IEC 13335-1:2004 Withdrawn
5.71	ANSI ISO/IEC TR 13335-3-1998	Information Technology: Guidelines for the Management of IT Security—Part 3: Techniques for the Management of IT Security	1998	Inactive	Superseded by ISO/IEC 27005:2018 Under Development as ISO/IEC 27005
5.71	ANSI ISO/IEC TR 13335-4-2000	Information Technology: Guidelines for the Management of IT Security—Part 4: Selection of Safeguards	2000	Inactive	Superseded by ISO/IEC 27005:2018 Under Development as ISO/IEC 27005
5.71	ANSI ISO/IEC TR 13335-5-2001	Information Technology: Guidelines for the Management of IT Security—Part 5: Management Guidance on Network Security	2001	Inactive	Superseded by ISO/IEC 18028-1:2006 Withdrawn
3.30	ANSI N101.2-72	Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities	1972	Inactive	Withdrawn 2013
3.30	ANSI N101.4-72	Quality Assurance for Protective Coatings Applied to Nuclear Facilities	1972	Inactive	Superseded by ASTM D3843-16(2021)e1
5.48 5.58	ANSI N15.18 (ANSI N15.18-1975)	Mass Calibration Techniques for Nuclear Materials Control	1988	Inactive	
5.48 5.58	ANSI N15.19 (ANSI N15.20-1975)	Volume Calibration Techniques for Nuclear Materials Control	1989	Inactive	
5.09 5.11 5.34 5.53 5.58	ANSI N15.20-1975	Guide to Calibrating Nondestructive Assay Systems	1987	Inactive	
3.50	ANSI N299-76	Administrative and Managerial Controls for Operation of Nuclear Fuel Reprocessing Plants	1976	Inactive	
3.03 3.21	ANSI N45.2-1971	Quality Assurance Program Requirements for Nuclear Power Plants	1977	Inactive	Superseded by ASME NQA-1 and NQA-2
3.37	ANSI N45.2.1-1973	Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants	1980	Inactive	Superseded by ASME NQA-1 and NQA-2

RG	Standard	Title	Current Rev	Status	Comments
10 CFR 50, Appendix J	ANSI N45.4-1972	Leakage Rate Testing of Containment Structures for Nuclear Reactors	—	withdrawn	This standard was withdrawn and superseded by: ANS 56.8-2020, which is not endorsed or approved for use.
3.30	ANSI N512-1974	Protective Coatings (Paints) for the Nuclear Industry	1974	Inactive	
3.60	ANSI/ANS 2.19-1981	Guidelines for Establishing Site-Related Parameters for Site Selection and Design of an Independent Spent Fuel Storage Installation (Water-Pool Type)	R1990	Inactive	
1.97 R2 1.97 R3 SRP	ANSI/ANS 4.5-1980	Criteria For Accident Monitoring Instrumentation for Nuclear Power Plants	1980 (R1988)	Inactive	No replacement
3.60	ANSI/ANS 57.9-1984	Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)			
3.48 3.62	ANSI/ANS 57.9-1984	Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)	2000	Inactive	
SRP 9.4.2 SRP 9.4.3 SRP 9.4.4	ANSI/ANS 59.2-1985	Safety Criteria for Nuclear Power Plant HVAC Systems Located Outside Primary Containment	2013	withdrawn	—
5.11 5.34	ANSI/ASTM C 853-79	Standard Test Methods for Nondestructive Assay of Special Nuclear Materials Contained in Scrap and Waste	1982	Inactive	
5.09	ANSI/IEEE 301-1976	Test Procedures for Amplifiers and Preamplifiers for Semiconductor Radiation Detectors for Ionizing Radiation	1988	Inactive	
5.09	ANSI/IEEE 325-1971 (R1977)	Test Procedures for Germanium Gamma-Ray Detectors	1986	Inactive	
5.09	ANSI/IEEE 645-1977	Test Procedures for High-Purity Germanium Detectors for Ionizing Radiation	1977	Inactive	
SRP 9.3.1	ANSI/ISA S7.3-R1991	Quality Standard for Instrument Air - Renumbered ISA-7.0.01	2010	withdrawn	ISA S7.0.1 superseded ISA S7.3, which was withdrawn in 2010. ISA S7.0.1 is not endorsed, approved, or required.
5.27	ASTM C1236-99-2005	Standard Guide for In-Plant Performance Evaluation of Automatic Vehicle SNM Monitors	2005	Inactive	
5.27	ASTM C1237-99-2005	Standard Guide to In-Plant Performance Evaluation of Hand-Held SNM Monitors	2014	Inactive	

RG	Standard	Title	Current Rev	Status	Comments
5.27	ASTM C993-97-2012	Standard Guide for In-Plant Performance Evaluation of Automatic Pedestrian SNM Monitors	2012	Inactive	
3.73	ASTM D3999-91	Standard Test Methods for the Determination of the Modulus and Damping Properties of Soils Using the Cyclic Triaxial Apparatus	2011	Inactive	Superseded by ASTM D3999/D3999M, which is also inactive
3.73	ASTM D5311-96	Standard Test Method for Load Controlled Cyclic Triaxial Strength of Soil	2011	Inactive	Superseded by ASTM D5311/D5311M
10 CFR 50.73	IEEE 803-1983	Recommended Practice for Unique Identification in Power Plants and Related Facilities--Principles and Definitions	2006	withdrawn	—
5.58	ISO Guide 6-1977	Mention of reference materials in International Standards	1977	Inactive	
3.32	MIL-F-51068D	Filters, Particulate (High-Efficiency Fire-Resistant)	1995	Inactive	Withdrawn 1995
3.32	MIL-F-51079B	Filter Medium, Fire-Resistant, High-Efficiency	1998	Inactive	Withdrawn 1998
5.12	QPL-FF-P-110	Padlock, Changeable Combination (Resistant to Opening by Manipulation and Surreptitious Attack)	1971	Inactive	

APPENDIX C. IDENTIFICATION OF NON-SDO RELATED DOCUMENTS

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
10 CFR 50 Appendix K	AECL-3281	An Investigation of Heat Transfer in the Liquid Deficient Regime	AECL	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	ANL-6548, page 7, May 1962	Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction	ANL	—	—	—	10 CFR 50 Appendix K
5.80	DOE/DP-0035	Safeguards Seal Reference Manual	DOE	5.80	—	—	—
3.13	EPA 600-R-95-051	Seismic Design Guidance for Municipal Solid Waste Landfill Facilities	EPA	3.13	—	—	—
3.67	EPA 400-R-92-001	Manual of Protective Action Guides and Protective Actions for Nuclear Incidents	EPA	3.67	—	—	—
1.01203	EPRI TR-100082	Standardization of the Cumulative Absolute Velocity	EPRI	1.12-3	SRP 3.7.4	—	—
1.06804	EPRI 1008219	PWR Primary-to-Secondary Leak Guidelines	EPRI	1.68-4	SRP 6.1.1 SRP 6.3 SRP 7.0 SRP 9.2.7 SRP 9.3.1 SRP 10.2 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 13.1.1 SRP 14.2 SRP 14.2.1 SRP 14.3	SRP 11.5	—
1.14716	EPRI NSAC-202L-R2	Recommendations for an Effective Flow Accelerated Corrosion Program	EPRI	1.147-16	SRP 3.8.1 SRP 3.8.2 SRP 5.2.1.2 SRP 5.2.4	SRP 10.2 SRP 10.3.6	—
1.14720	EPRI NP-3944	Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines	EPRI	1.147-20 (GL 89-03)	SRP 3.8.1 SRP 3.8.2 SRP 5.2.1.2 SRP 5.2.4	SRP 10.2 SRP 10.4.7	—
1.16400	EPRI TR-106439	Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications	EPRI	1.164-0 1.168-2 1.169-1	— SRP Appendix 7.0-A SRP Appendix	SRP Appendix 7.0-A SRP Appendix 7.1-D SRP 9.5.2	—

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
					7.1-A SRP Appendix 7.1-D SRP Table 7-1 SRP BTP 7-14 SRP BTP 7-21 SRP Appendix 7.0-A SRP Appendix 7.1-A SRP Appendix 7.1-D SRP Table 7-1 SRP BTP 7-14	SRP BTP 7-14 SRP BTP 7-18	
1.16400	EPRI TR-107330	Generic Requirements Specification for Qualifying a Commercially Available PLC for Safety-Related Applications in Nuclear Power Plants	EPRI	1.164-0	—	SRP Appendix 7.1-D SRP BTP 7-18	—
1.16400	EPRI 3002002982	EPRI 3002002982, Revision 1 to EPRI NP-5652 and TR-102260, “Plant Engineering: Guideline for the Acceptance of Commercial-Grade Items in Nuclear Safety-Related Applications	EPRI	1.164-0	—	—	—
1.16701	EPRI NP-6695 (EPRI 1025288)	Guidelines for Nuclear Plant Response to an Earthquake	EPRI	1.167-1 WITHDRAWN	—	—	—
1.20502	EPRI 1011989 (NUREG/CR-6850)	Fire PRA Methodology for Nuclear Power Facilities	EPRI	1.205-2	SRP 9.5.1.2 SRP 11.2 SRP 11.3 SRP 11.4	SRP 19.0 SRP 9.5.1.1 SRP 9.5.1.2	—
1.23100	EPRI NP-5652	Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications (NCIG-07)	EPRI	1.231-0 (GL 89-03)	—	SRP 17.3	—
1.23100	EPRI 1025243 R1 (EPRI 3002002289)	Plant Engineering: Guideline for the Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Nuclear Safety-Related Applications	EPRI	1.231-0	—	—	—
1.23100	EPRI TR-1025243	Plant Engineering: Guideline for the Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Nuclear Safety-Related Applications	EPRI	1.231-0	—	—	—
3.73	EPRI NP-4726	Seismic Hazard Methodology for the Central and Eastern United States, Volumes 1-3, Revision 1	EPRI	3.73	—	—	—

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
3.73	EPRI TR-102261-V1	The Earthquakes of Stable Continental Regions: Assessment of Large Earthquake Potential	EPRI	3.73	—	—	—
—	EPRI 1008224	PWR Secondary Water Chemistry Guidelines	EPRI	—	—	SRP 11.5	—
—	EPRI 1009684	CEUS Ground Motion Project Final Report	EPRI	—	—	SRP 2.5.2	—
—	EPRI 1012965	Use of CAV in Determining Effects of Small Earthquakes on Seismic Hazard Analysis	EPRI	—	—	SRP 2.5.2	—
—	EPRI 1013420	Pressurized-Water Reactor Primary Water Zinc Application Guidelines	EPRI	—	—	SRP 11.5	—
—	EPRI 1018644	Guidelines for Operating an Interim on Site Low Level Radioactive Waste Storage Facility, Revision 1	EPRI	—	—	SRP 11.4	—
—	EPRI ALWR Utility Requirements Document, Volume II, Chapter 11, Revision 6	EPRI ALWR Utility Requirements Document, Volume II, "Evolutionary Plants," Chapter 11, "Electric Power Systems," Revision 6, December 1993, Electric Power Research Institute.	EPRI	—	—	SRP 11.5	—
—	EPRI NP-5930	A Criterion for Determining Exceedance of the Operating Basis Earthquake	EPRI	—	—	SRP 3.7.4	—
—	EPRI Report Series, “BWR Water Chemistry Guidelines.”	BWR Water Chemistry Guidelines	EPRI	—	—	SRP 5.4.8 SRP 9.3.2 SRP 10.2 SRP 10.4.6	—
—	EPRI Report Series, “PWR Secondary Water Chemistry Guidelines.”	PWR Secondary Water Chemistry Guidelines	EPRI	—	—	SRP 10.2 SRP 10.4.6 SRP 11.5	—
—	EPRI TR-100370	Fire-Induced Vulnerability Evaluation (FIVE)	EPRI	—	—	SRP 19.0	—
—	EPRI TR-102293	Guidelines for Determining Design Basis Ground Motions	EPRI	—	—	SRP 2.5.2	—
—	ERPI TR-1002988	Seismic Fragility Application Guide	EPRI	—	—	SRP 19.0	—
10 CFR 50.55a	EPRI MRP–335, Revision 3–A	Materials Reliability Program: Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement ()	EPRI	10 CFR 50.55a	many	—	10 CFR 50.55a
10 CFR 50 Appendix K	NEDO-10329	Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors	GE	—	—	—	10 CFR 50 Appendix K
5.12	GSA FF-P-2827	Padlock, Key Operated, General Field Service	GSA	5.12	—	—	—
5.12	QPL-FF-L-2890	Lock Extension (Pedestrian Door, Deadbolt)	GSA	5.12	—	—	—
5.12	QPL-FF-L-2937	Combination Locks, Mechanical	GSA	5.12	—	—	—
5.75	INPO-AP-921	Principles of Training System Development	INPO	5.75	—	—	—
3.54	JAERI-M 91-034	Recommended Values of Decay Heat Power and Method to Utilize the Data	Japan Atomic	3.54	—	—	—

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
			Energy Research Institute				
3.73	UCRL-ID-115111	Eastern US seismic hazard characterization update	LLNL	3.73	—	—	—
1.10106	NEI 99-01	Development of Emergency Action Levels for Non-Passive Reactors	NEI	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	SRP 13.3	—
1.10106	NEI 06-04, Appendix A	Recommended Drill and Exercise Objectives," to NEI 06-04, "Conducting a Hostile Action-Based Emergency Response Drill	NEI	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	—	—
1.10106	NEI 07-01	Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors	NEI	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	—	—
1.10106	NEI 10-05	Assessment of On-Shift Emergency Response Organization Staffing and Capabilities	NEI	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	—	—
1.10106	NEI 13-01	Reportable Action Levels for Loss of Emergency Preparedness Capabilities	NEI	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	—	—
1.10106	NEI white paper	Implementing A 24-Month Frequency for Emergency Preparedness Program Reviews	NEI	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	—	—
1.12600	NEI 12.06 Rev 4	Diverse and Flexible Coping Strategies (FLEX) Implementation Guide	NEI	1.226-0	—	—	—
1.12702	NEI 95-10 Rev. 6	Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule	NEI	1.127-2	SRP 3.8.4 SRP 3.8.5 SRP 9.5.1.1	SRP 9.5.1.1	—
1.14904	NEI 09-09	Nuclear Power Plant-Referenced Simulator Scenario Based Testing Methodology	NEI	1.149-4	SRP 13.2.1 SRP 13.2.2	—	—
1.16300	NEI 94-01	Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J	NEI	1.163-0	SRP 6.1.1 SRP 6.2.6	SRP 6.1.1 SRP 6.2.6	—
1.18100	NEI 98-03	Guidelines for Updating Final Safety Analysis Reports	NEI	1.181-0	SRP 1.0	SRP 1.0	—
1.18600	NEI 97-04 Appendix B	Guidelines and Examples for Identifying 10 CFR 50.2 Design Bases	NEI	1.186-0	—	—	—
1.18703	NEI 96-07	Guideline for Implementation of Change Control Processes for New Nuclear Power Plants Licensed under 10 CFR Part 52	NEI	1.187-3	SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP 17.6	SRP 17.6 SRP 18.0	—

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
					SRP 18.0 □		
1.18904	NEI 00-01 Rev. 4	Guidance for Post-Fire Safe-Shutdown Circuit Analysis	NEI	1.189-4	SRP 3.2.1 SRP 11.2 SRP 11.3 SRP 11.4 SRP 11.5 SRP BTP 11-3 SRP 9.5.1.1 SRP 9.5.1.2 SRP Appendix 7.1-A SRP 7.4 SRP Table 7-1 SRP 13.1.2-13.1.3 SRP 13.2.1 SRP 17.5 SRP 9.5.1.2 SRP 11.2 SRP 11.3 SRP 11.4	SRP 9.5.1.2	—
1.19601	NEI 99-03	Control Room Habitability Assessment Guidance	NEI	1.196-1	SRP 4.2 SRP 6.1.1 SRP 6.4	—	—
1.20003	NEI 00-02, Revision A3	Probabilistic Risk Assessment (PRA) Peer Review Process Guidance	NEI	1.200-3	SRP Table 7-1 SRP BTP 7-12 SRP Appendix 7.1-A SRP 9.5.1.2 SRP 19.0 SRP 19.1 SRP 19.2	SRP 19.1	—
1.20003	NEI 17-07	Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard	NEI	1.200-3	SRP Appendix 7.1-A SRP BTP 7-12 SRP Table 7-1 SRP 9.5.1.2 SRP 19.0	—	—

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
					SRP 19.1 SRP 19.2		
1.20101	NEI 00-04	10 CFR 50.69 SSC Categorization Guideline	NEI	1.201-1	SRP 3.9.5 SRP 17.4	—	—
1.20502	NEI 04-02 Rev. 3	Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)	NEI	1.205-2	SRP 9.1.5.2 SRP 11.2 SRP 11.3 SRP 11.4	SRP 9.5.1.2	—
1.21502	NEI 08-01 Rev. 5 - Corrected	Industry Guideline for the ITAAC Closure Process under 10 CFR Part 52	NEI	1.215-2	SRP 5.2.1.2 SRP 9.2.7 SRP 11.3 SRP 11.4 SRP 11.5	—	—
1.21700	NEI 07-13 Rev. 8	Methodology for Performing Aircraft Impact Assessments for New Plant Designs	NEI	1.217-0	SRP 19.5	SRP 19.5	—
1.22700	NEI 12-02	Industry Guidance for Compliance with NRC Order EA-12-051, 'To Modify Licenses RG 1.226, Page 2 with Regard to Reliable Spent Fuel Pool Instrumentation	NEI	1.227-0	—	—	—
1.23300	NEI 18-04	Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development	NEI	1.233-0	—	—	—
1.23400	NEI 14-09	Guidelines for Implementation of 10 CFR Part 21 Reporting of Defects and Noncompliance	NEI	1.234-0	—	—	—
1.23900	NEI 15-03, Revision 3	Licensee Actions to Address Nonconservative Technical Specifications	NEI	1.239-0	—	—	—
1.24000	NEI 12-16	Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants	NEI	1.240-0	—	—	—
3.72	NEI 12-04 Rev 2	Guidelines for 10 CFR 72.48 Implementation, Revision 2	NEI	3.72	—	—	—
3.72	NEI 96-07 Rev 1	Guidelines for 10 CFR 50.59 Evaluations, Revision 1	NEI	3.72	—	—	—
3.76	NEI 14-03 Rev 2	Format, Content and Implementation Guidance for Dry Cask Storage Operations Based Aging Management	NEI	3.76	—	—	—
5.66	NEI-03-01	Nuclear Power Plant Access Authorization Program	NEI	5.66	—	—	—
5.71	NEI 04-04	Cyber Security Program for Power Reactors	NEI	5.71	—	—	—
5.73	NEI 06-11, Rev 1	Managing Personnel Fatigue at Nuclear Power Reactor Sites	NEI	5.73	—	—	—
5.75	NEI 03-12	Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan,	NEI	5.75	—	—	—

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
		[and Independent Spent Fuel Installation Security Program]					
5.79	NEI 03-01	Nuclear Power Plant Access Authorization Program	NEI	5.79	—	—	—
5.84	NEI 06-06	Fitness-for-Duty Program Guidance for New Nuclear Power Plant Construction Sites	NEI	5.84	—	—	—
—	NEI 03-12	Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, [and Independent Spent Fuel Storage Installation Security Program]	NEI	—	—	SRP 13.6.1 SRP 13.6.4 SRP 13.6.6	—
—	NEI ODCM Template 07-09A	Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description	NEI	—	—	SRP 11.5	—
1.07802	NUREG 0696	Functional Criteria for Emergency Response Facilities	NRC	1.78-2 1.101-6	SRP 6.1.1 SRP 6.4 SRP 9.4.1 SRP 14.2 SRP 11.5 SRP 13.3 SRP 14.3.10	SRP 13.3 SRP 14.3.10 SRP 18.0	—
1.07802	NUREG 2244	HABIT 2.2: Description of Models and Methods	NRC	1.78-2	SRP 6.1.1 SRP 6.4 SRP 9.4.1 SRP 14.2	—	—
1.09103	NUREG/CR-3330	Vulnerability of Nuclear Power Plant Structures to Large External Fires	NRC	1.91-3	—	—	—
1.10106	NUREG 0396/EPA 520/1-78-016	Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants	NRC	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	SRP 13.3	—
1.10106	NUREG 0654 Appendix 1/FEMA-REP-1, Rev. 1	Acceptable Deviations from Appendix 1 of NUREG-0654 Based Upon the Staff's Regulatory Analysis of NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels	NRC	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	SRP 13.3 SRP 14.3.10	—
1.10106	NSIR/DPR-ISG-01	Interim Staff Guidance: Emergency Planning for Nuclear Power Plants	NRC	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	—	—
1.10106	NSIR/DPR-ISG-02	Interim Staff Guidance: Emergency Planning Exemption Request for Decommissioning Nuclear Power Reactors	NRC	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	—	—

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
1.10106	NUREG/CR-7002	Criteria for Development of Evacuation Time Estimate Studies	NRC	1.101-6	SRP 11.5 SRP 13.3 SRP 14.3.10	—	—
1.20701	NUREG/CR-5704	Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels	NRC	1.207-1	SRP 3.12 SRP 5.4.2.1	—	—
1.20701	NUREG/CR-6583	Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels	NRC	1.207-1	SRP 3.12 SRP 5.4.2.1	—	—
1.20701	NUREG/CR-6609, Rev. 1	Effect of LWR Water Environments on Fatigue Life of Reactor Materials	NRC	1.207-1	SRP 3.12 SRP 5.4.2.1	—	—
—	NUREG 0588	Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment	NRC	—	—	SRP 6.2.1.1.B SRP 6.2.1.1.C	—
—	NUREG 0661	Mark I Containment Long Term Program	NRC	—	—	SRP 6.2.1.1.C	—
—	NUREG 0718	Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License	NRC	—	—	SRP 6.2.1.1.A SRP 6.2.1.1.B	—
—	NUREG 0737	Clarification of TMI Action Plan Requirements	NRC	—	—	SRP 6.2.1.1.A SRP 6.2.1.1.B	—
—	NUREG 0763	Guidelines for Confirmatory In-plant Tests of Safety-Relief Discharge for BWR Plants	NRC	—	—	SRP 6.2.1.1.C	—
—	NUREG 0783	Suppression Pool Temperature Limits for BWR Containments	NRC	—	—	SRP 6.2.1.1.C	—
—	NUREG 0802	Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments	NRC	—	—	SRP 6.2.1.1.C	—
—	NUREG 0808	Mark II Containment Program Load Evaluation and Acceptance Criteria	NRC	—	—	SRP 6.2.1.1.C	—
—	NUREG 0978	Mark III LOCA-Related Hydrodynamic Load Definition	NRC	—	—	SRP 6.2.1.1.C	—
—	NUREG 2115	Central and Eastern United States Seismic Source Characterization for Nuclear Facilities	NRC	—	—	SRP 2.5.2	—
10 CFR 50.55a	NUREG 2228	Weld Residual Stress Finite Element Analysis Validation: Part II—Proposed Validation Procedure	NRC	—	many	—	10 CFR 50.55a
1.155	NUMARC 8700	Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors	NUMARC (NEI)	1.155	SRP 6.1.1 SRP 6.2.4 SRP 6.2.5 SRP 6.3 SRP 8.1 SRP 8.2 SRP 8.3.1 SRP 8.3.2 SRP 8.4 SRP BTP 8-8 SRP 9.2.2	SRP 8.1 SRP 8.4	—

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
					SRP 9.2.6 SRP 9.2.7 SRP 9.3.1 SRP 9.3.4 SRP 9.4.1 SRP 9.4.5 SRP 10.2 SRP 10.3 SRP 10.4.9 SRP 17.5		
1.16004	NUMARC 93-01, Rev 4A	Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	NUMARC (NEI)	1.160-4	SRP 8.1 SRP 8.2 SRP 8.3.1 SRP 8.3.2 SRP 8.4 SRP 9.2.7 SRP 11.5 SRP 12.5 SRP 17.4 SRP 17.6	SRP 8.2 SRP 8.3.1 SRP 17.4 SRP 17.6	—
1.20101	NUMARC 91-06	Guidelines for Industry Actions to Assess Shutdown Management	NUMARC (NEI)	1.201-1	SRP 3.9.5 SRP 17.4	SRP 19.0	—
10 CFR 50 Appendix K	combination of the Thom correlation and the Martinelli-Nelson correlation	Prediction of Pressure Drop During Forced Circulation Boiling of Water Prediction of Pressure Drop During Forced Circulation Boiling of Water	paper	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	E. D. Hughes	A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia	paper	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	J. M. Healzer, J. E. Hench, E. Janssen, S. Levy	Design Basis for Critical Heat Flux Condition in Boiling Water Reactors	paper	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek	Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water	paper	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	J.B. McDonough, W. Milich, E.C. King	An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube	paper	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	L. S. Tong	Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution	paper	—	—	—	10 CFR 50 Appendix K

ID	Non-SDO document	Document title	Non-SDO	RG endorsing non-SDO document	RG cited in SRP section	SRP acceptance of document (SRP section)	CFR
10 CFR 50 Appendix K	modified Baroczy correlation A Systematic Correlation for Two-Phase Pressure Drop	A Systematic Correlation for Two-Phase Pressure Drop	paper	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	Moody model Trans American Society of Mechanical Engineers, 87, No. 1, February, 1965	Maximum Flow Rate of a Single Component, Two-Phase Mixture	paper	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	P. G. Barnett	A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles	paper	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	R. V. Macbeth	An Appraisal of Forced Convection Burnout Data	paper	—	—	—	10 CFR 50 Appendix K
3.13	EM 1110-2-1100	Coastal Engineering Manual	U.S. Army Corps of Engineers	3.13	—	—	—
3.13	EM 1110-2-1902	Engineering and Design Slope Stability	U.S. Army Corps of Engineers	3.13	—	—	—
3.13	ER 1110-2-106	Recommended Guidelines for Safety Inspection of Dams	U.S. Army Corps of Engineers	3.13	—	—	—
—	WCAP-8575	Augmented Startup and Cycle 1 Physics Program	Westinghouse	—	—	SRP BTP 4.1	—
10 CFR 50 Appendix K	USNRC Docket RM-50-1	Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation	Westinghouse	—	—	—	10 CFR 50 Appendix K
10 CFR 50 Appendix K	WCAP-7665	PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report	Westinghouse	—	—	—	10 CFR 50 Appendix K

APPENDIX D. DETAILS OF REVIEWS OF THE SOURCES OF THE STANDARDS

Standards could be endorsed via a regulatory guide (RG), approved for use via the standard review plan (SRP), or required for use by the Code of Federal Regulations (CFR). The standards selected for review could have been selected from one or more of these sources. For example, National Fire Protection Association (NFPA) 805 is endorsed by RG 1.191-1, RG 1.205-2, and RG 3.15-2; approved for use in SRP 9.5.1.1, SRP 9.5.1.2, and SRP 19.1; and required in 10 CFR 50.48(c).

The selection process identified only unique standards. The process first reviewed those standards endorsed by an RG, starting with Division 1 and then adding Division 3 and Division 5. The next step identified unique standards approved for use by the SRP. The last step identified unique standards required by the CFR.

A breakdown of the lessons learned at each step in the process is provided below.

Power Reactor Regulatory Guides (Division 1)

A total of 113 unique standards from SDOs were designated for review from the Division 1 RGs. Table D-1 and Figure D-1 show the number of consensus or industry standards endorsed by an RG in Division 1 (Power Reactors) by a standards development organization (SDO) and Figure D-2 shows the number of documents endorsed for a non-SDO (i.e., industry group).

Table D-1. Number of standards endorsed by RGs by SDO/industry group

SDO or industry group		Number of standards/documents
American Concrete Institute	ACI	3
American Institute of Steel Construction	AISC	1
American Nuclear Society	ANS	11
American National Standards Institute	ANSI	1
American Society of Mechanical Engineers	ASME	19
American Society for Testing and Materials	ASTM	25
International Electrotechnical Commission	IEC	3
Institute of Electrical and Electronic Engineers	IEEE	42
Instrumentation Society of America	ISA	2
Department of Defense (MIL standards)	MIL	1
National Fire Protection Association	NFPA	4
The Society for Protective Coatings	SSPC	1
Total SDOs		113
Electric Power Research Institute	EPRI	12
Nuclear Energy Institute	NEI	26
Nuclear Regulatory Commission	NRC	11
Nuclear Management and Resources Council	NUMARC	3
Total industry groups		49
Overall total		162

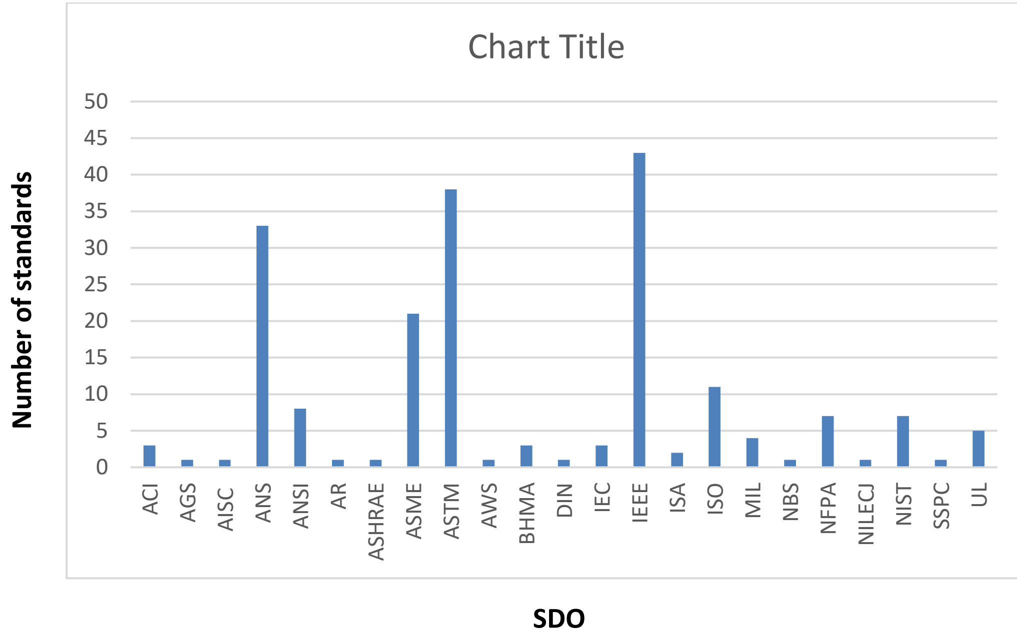

Figure D-1. Number of standards by SDO.

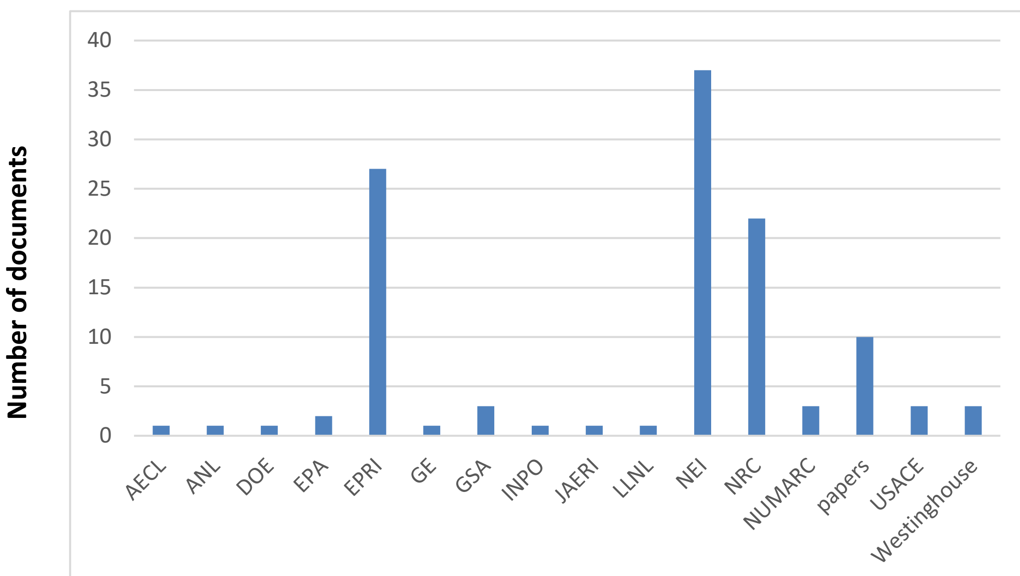

Figure D-2. Number of documents by non-SDO.

The American Society of Mechanical Engineers (ASME), ASTM International, and the Institute of Electrical and Electronic Engineers (IEEE) are the most cited SDOs, and the Electric Power Research Institute (EPRI), the Nuclear Energy Institute (NEI), and the US Nuclear Regulatory Commission (NRC) are cited frequently for documents from non-SDOs.

Fuels and Material Facilities Regulatory Guides (Division 3)

This portion of the review focused on the adequacy and completeness of the standards identified in the Fuels and Material Facilities RGs (Division 3) to an MSR. It is outside the scope of this review to prioritize the development activities of a standard by NRC or an SDO or to relate its development to the NRC mission.

A total of 75 standard citations among the 77 RGs listed in Division 3 (includes RG 3.11.1). Of the listed RGs, 41 are active and not included in the selection from Division 1, and 12 have been withdrawn. Of the 75 standard citations identified (standards cited in multiple RGs), 67 were unique. One additional ASTM standard for coatings (ASTM D3843-16) supersedes ANSI N101.4-72 cited in RG 3.30. ASTM D3843-16 references multiple ASTM standards for coatings and should be included in the MSR standard review for Division 3 RGs. Therefore, a total of 68 unique standards are associated with fuels and material facilities.

The results from the down-select process for fuels and material facilities were that 68 unique standards and standard-like documents were designated for review—41 consensus standards from 8 SDOs and 12 industry standard-like documents from 6 organizations. Table D-1 and Figure D-1 show the number of consensus or industry standards endorsed by an RG in Division 3 (Fuels and Material Facilities) by SDO or industry group.

Table D-1. Number of standards endorsed by Division 3 RGs by SDO/industry group

SDO or industry group		Number of endorsed standards/documents
American Global Standards	AGS	1
American National Standards Institute	ANSI	20
American Society of Mechanical Engineers	ASME	4
American Society for Testing and Materials	ASTM	3
International Organization for Standardization	ISO	10
National Fire Protection Association	NFPA	2
Underwriters Laboratories	UL	1
Total SDO		41
Department of Defense	DoD	1
Environmental Protection Agency	EPA	2
Electric Power Research Institute	EPRI	2
Japan Atomic Energy Research Institute	JAERI	1
Nuclear Energy Institute	NEI	3
US Army Corps of Engineers	USACE	3
Total industry group		12
TOTAL		53

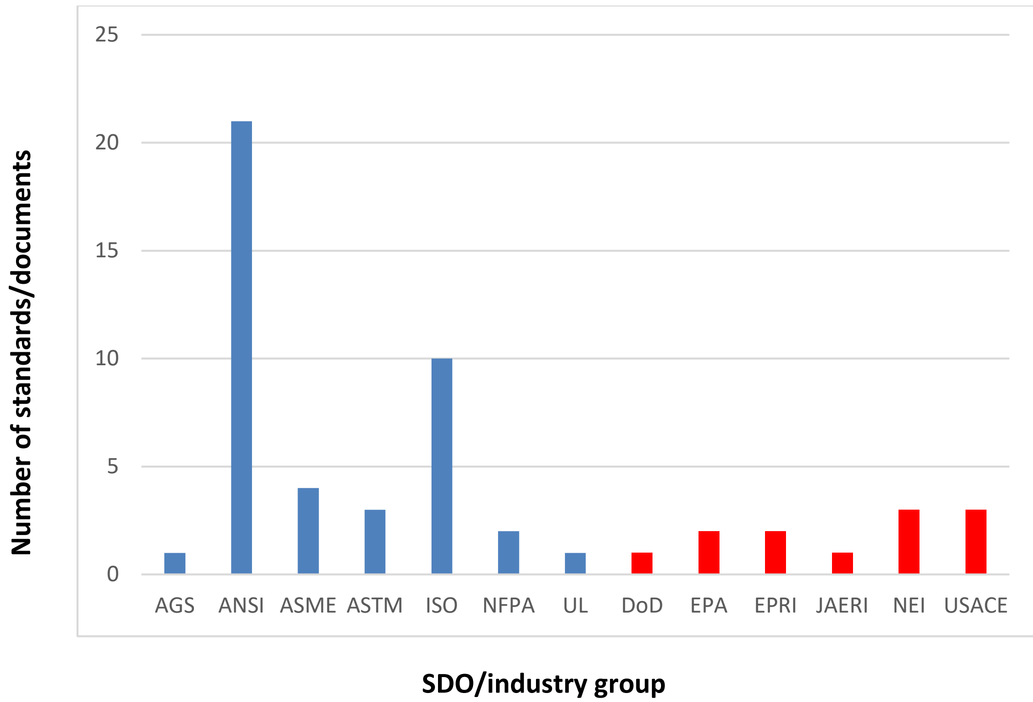

Figure D-1. Number of standards/documents endorsed by Division 3 RGs by SDO/industry group.

Materials and Plant Protection Regulatory Guides (Division 5)

This portion of the review focused on the adequacy and completeness of the standards identified in the Materials and Plant Protection RGs (Division 5) to an MSR. It is outside the scope of this review to prioritize the development activities of a standard by NRC or an SDO or to relate its development to the NRC mission.

There were 64 standard citations among the 89 RGs listed in Division 5. Of the listed RGs, 57 are active or in draft form while the remaining 32 have been withdrawn. Not all the active RGs were available for review because some included safeguards information or other official use only information. Six Division 5 RGs met this criterion. Three of the RGs that were reviewed are currently listed as drafts. Of the 64 standard citations identified (standards cited in multiple RGs), 55 were unique. An additional 5 standards were endorsed directly in SRP Chapter 13, for a total of 60 unique standards associated with materials and plant protection.

The results from the down-select process for materials and plant protection were that 60 unique standards and standard-like documents were designated for review—41 consensus standards from 9 SDOs and 19 industry standard-like documents from 4 organizations. Initial review of the 41 SDO consensus standards indicated that 27 are currently active from 8 SDOs. Likewise, initial review of the 19 industry standard-like documents indicated that 18 are currently active from 4 organizations. Table D-1 and Figure D-1 show the number of consensus or industry standards endorsed by an RG in Division 5 (Materials and Plant Protection) or SRP Chapter 13 (Conduct of Operations) by SDO or industry group.

Table D-1. Number of standards/documents endorsed by Division 5 RGs and SRP Chapter 13 by SDO/industry group

SDO or industry group		Number of endorsed standards/documents
American National Standards Institute	ANSI	6
American Society of Mechanical Engineers	ASME	1
American Society for Testing and Materials	ASTM	7
U.S. General Services Administration	GSA	4
Institute of Electrical and Electronics Engineers	IEEE	2
National Fire Protection Association	NFPA	1
National Institute of Justice	NIJ	2
Underwriters Laboratories	UL	4
Total SDO		27
Department of Defense	DoD	4
Department of Energy	DOE	2
Nuclear Energy Institute	NEI	5
National Institute of Standards and Technology	NIST	7
Total industry		18
TOTAL		45

Figure D-1. Number of standards/documents endorsed by Division 5 RGs and SRP by SDO/industry group.

Standard Review Plan

The SRP is directed at reactors (specifically LWRs). The Division 3 RGs provide guidance for fuels and material facilities (general fuel cycle facilities). Some MSR technologies or adapters may opt to colocate other portions of the fuel cycle with the reactor, such as fuel fabrication or used fuel processing. This configuration is the basis of including the Division 3 RGs in the standard review for MSRs. However, no link exists between the standards endorsed by Division 3 RGs and the SRP.

Table D-1 specifies chapter numbers and titles in the SRP.

Table D-1. SRP Table of Contents

SRP chapter	Title
1	Introduction and General Description of Plant
2	Site Characteristics
3	Design of Structures, Components, Equipment, and Systems
4	Reactor
5	Reactor Coolant System and Connected Systems
6	Engineered Safety Features
7	Instrumentation and Controls
8	Electric Power
9	Auxiliary Systems
10	Steam and Power Conversion System
11	Radioactive Waste Management
12	Radiation Protection
13	Conduct of Operations
14	Initial Test Program and ITAAC-Design Certification
15	Accident Analysis
16	Technical Specifications
17	Quality Assurance
18	Human Factors Engineering
19	Severe Accidents

Of interest is what SRP chapters cite an RG or standard. The SRP may cite the RG, approve the standard for use, or both. Except for Chapters 1, 12, and 16, all chapters of the SRP cited endorsed standards or the endorsing RG (Figure D-1).

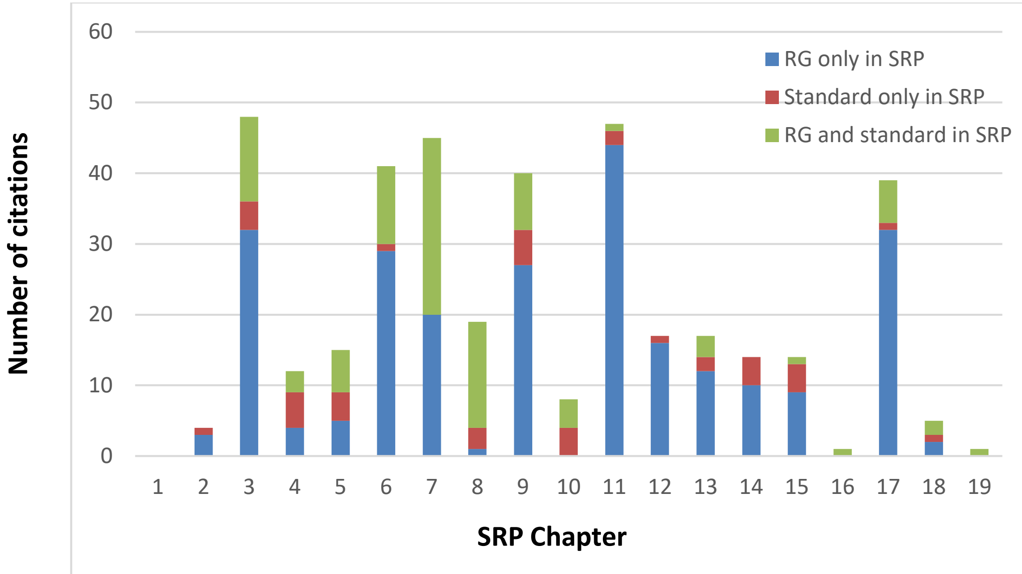

Figure D-1. Number of standards and/or RGs cited in each SRP chapter.

The system-based chapters—Chapters 5, 6, 7, 8, 9, and 10—are well represented by endorsed, approved, or required standards. These chapters account for 46% of the standards.

The design-based chapters—Chapters 2, 3, and 4—are well represented by endorsed, approved, or required standards (15%). Chapter 3, alone accounts for almost 12% of the standards cited.

Six RGs ***and*** the endorsed standards are not cited in the SRP (Table D-2).

Table D-2. RGs and endorsed standards not cited in the SRP

Regulatory guide	Title	Endorsed standard	Title
RG 1.87, R2	Acceptability Of ASME BPVC Section III, Division 5, "High Temperature Reactors"	ASME BPVC Section III Division 5	High Temperature Reactors
RG 1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants	ANSI/ANS 3.4-1996	Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants
RG 1.210, R0	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	IEEE 650-2006	IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations
RG 1.213, R0	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	IEEE 649-2006	IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations
RG 1.244, R0	Control of Heavy Loads at Nuclear Facilities	ASME BTH-1–2017	Control of Heavy Loads at Nuclear Facilities
RG 1.244, R0	Control of Heavy Loads at Nuclear Facilities	ASME NML-1–2019	Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities

Only SRP Chapter 13, Conduct of Operations, includes materials and plant protection standards or an endorsing Division 3 RG or NUREG.

Only a limited number of Division 5 RGs are endorsed in one or more sections SRP Chapter 13. Others are security related and are restricted and thus not available for review.

Not surprisingly, 10 Division 5 RGs and their 15 endorsed standards are not cited in the SRP.

Code of Federal Regulations

A review of the CFR identified 19 standards required by the CFR with 8 unique standards not endorsed by an RG or approved for use in the SRP. The 19 standards were from the SDOs listed in Table D-1.

Table D-1. Consensus standards references in the Code of Federal Regulations [41]

10 CFR section	SDO or coordinating organization
34.20	NBS
50.34	ASME
50.48	NFPA
50.49	IEEE
50.55a	ASME
50.55a	IEEE
50.61	ASME
50.73	IEEE
50 App G	ASME
50 App H	ASTM
50 App J	ANSI
50 App K	ANS
50 App R	IEEE
73.26	ANSI
73.26	ISO
73 App B	ANSI
73 App B	ISO

Of the eight unique standards required by the CFR, none required substantive changes or were of high priority for updating. However, six standards identified by the RG/SRP review required substantive changes (ANS 5, ASME BPVC Section III Division 1, ASME BPVC Section III Division 2, ASME BPVC Section XI, NFPA 805, IEEE 603-1991), and one standard was identified as having a high priority for revision (ASME BPVC Section II).