Summary of ORNL Standards Activities

Dr. Michael Muhlheim
Oak Ridge National Laboratory

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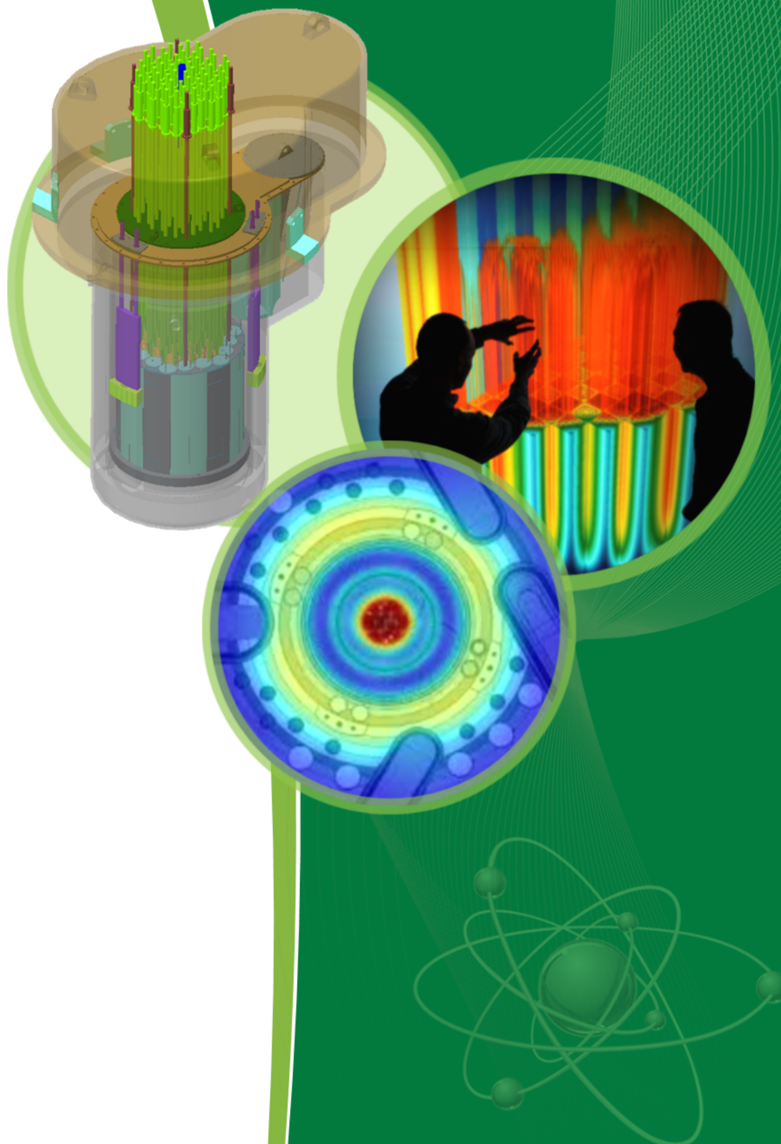

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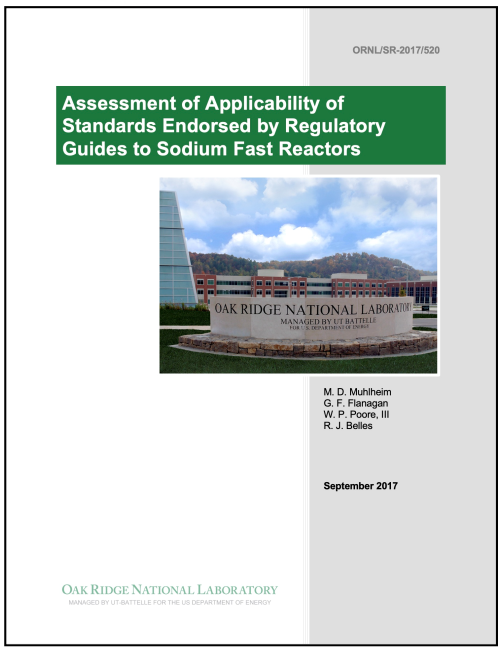
DOE initiated a scoping study to understand the size and scope of expanding the NRC's LWR-specific regulatory framework to SFRs and later to MSRs

The scope had to be reduced to a manageable amount

- 865 standards, industry reports, journal articles cited in 486 Div. 1-10 RGs
- 3,364 standards, industry reports, and journal articles in cited the Standard Review Plan (SRP) (NUREG-0800)
- 19 standards required by the Code of Federal Regulations (CFR)



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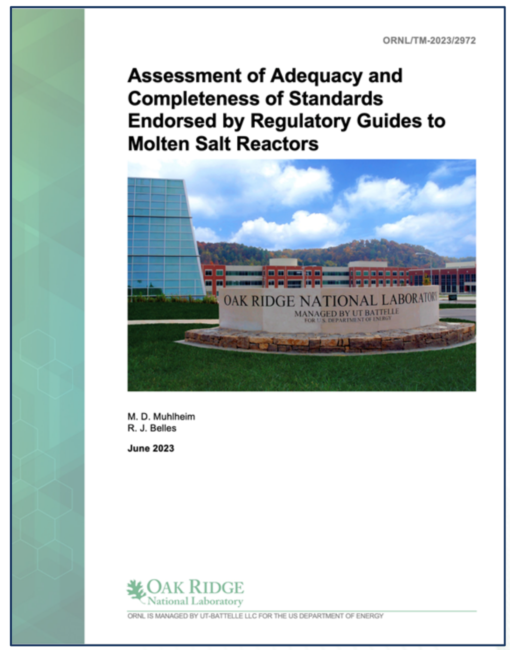

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Level of Effort (LOE) required to modify / update an endorsed standard for applicability

	SFR	MSR
LOE	No.	No.
1	57	141
2	13	24
3	**13**	**19**
4	4	6
5	1	8
Total	88 standards, 7 SDOs	197 standards, 23 SDOs
6	**12**	**14**

SFR
Endorsed standards in Div. 1 RGs

Level of effort
1 = None
2 = Limited changes
3 = Substantive changes needed
4 = Insufficient design info
5 = Not applicable
6 = New design-specific requirement

MSR
Endorsed standards in Div. 1, 3, and 5 RGs, approved for use in SRP, required by CFR

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

"Top four" standards recommended for revision

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.

Conclusions

- 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

- MSRs will have the same issues as SFRs
  - High energy spectrum
  - High temperature
  - Coolant
  - Materials

- Standard selection should be based on providing benefit to the most designs (e.g., fuel characterization) and timeliness of need (e.g., design)

- Ideal would be 1 standard that addresses multiple technologies (i.e., applicable to MSRs, FSRs, HTGRs, etc.)

Backup slides

Number of standards requiring significant revisions or new development by SDO

SDO	SFR		MSR	
	LOE 3	LOE 6	LOE 3	LOE 6
ANS	2	7	5	13
ASME	7	5	7	1
ASTM	2		3	
ISA	1		1	
ISO			1	
NFPA	1		2	
Total No. of standards	**13**	**12**	**19**	**14**

Level of effort
3 = Substantive changes needed
6 = New design-specific requirement

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Summary of results

<table>
<tr><th>SFRs</th><th>MSRs</th></tr>
<tr><td>13 endorsed standards will likely need substantive changes for applicability to SFRs</td><td>19 endorsed standards will likely need substantive changes for applicability to MSRs</td></tr>
<tr><td>
➢ Protective coatings and test methods for protective coatings may differ
➢ Temperatures in SFRs may exceed concrete and steel limits in standards
➢ Types of steel, concrete, and source terms may differ greatly for SFRs compared to LWRs
➢ Those components required to function during a DBA (PA) will be different for SFRs and will require modification to some standards (e.g., seismic, dynamic qualifications)
➢ Containments will be different from current plants
➢ Fire issues (fire-induced failures, testing, etc.)
➢ Presence of sodium affects EQ, habitability, fire, …
</td><td>
➢ 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
➢ Those components required to function during a DBA (postulated accident) will be different and will require modification to some standards (e.g., seismic, dynamic qualifications)
➢ Containments will be different from current plants
➢ Fire issues and fire suppression (e.g., fire-induced failures, testing) will differ
➢ Criticality safety considerations for handling and storage may be different for MSR liquid or solid fuel compared to LWR solid fuel
➢ Dry cask storage and aging management may be different for MSR liquid, or solid fuel compared to LWR solid fuel
</td></tr>
<tr><td>12 new consensus standards for SFRs are recommended</td><td>19 new consensus standards for MSRs are recommended</td></tr>
<tr><td>
➢ 10 SFR-DCs (70–79) identified in RG 1.232,
➢ Passive cooling
➢ Passive equipment
</td><td>
➢ Different operating environment
➢ Passive cooling
➢ Passive equipment
</td></tr>
</table>

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Results specific to SFRs

SFR LOE=3 standards		
ANS	ANS 56.2-1984 (ANSI N271-1976)	Containment Isolation Provisions for Fluid Systems
ANS	ANSI/ANS 6.4-2006	Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants
ASME	ASME AG-1-2009	Code on Nuclear Air and Gas Treatment
ASME	ASME BPVC Division 1 and 2, Subsection NCA	General Requirements for Division 1 and Division 2
ASME	ASME BPVC Section III	Rules for Construction of Nuclear Power Plant Components
ASME	ASME BPVC Section III Division 2, 2001 edition through 2003 Addenda	Rules for Construction of Nuclear Power Plant Components
ASME	ASME BPVC Section XI	Rules for Inservice Inspection of Nuclear Power Plant Components
ASME	ASME N509-2002	Nuclear Power Plant Air-Cleaning Units and Components
ASME	ASME QME-1-2007	Qualification of Active Mechanical Equipment Used in Nuclear Power Plants
ASTM	ASTM D3911-16	Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design-Basis Accident (DBA) Conditions
ASTM	ASTM D7491-08	Standard Guide for Management of Non-Conforming Coatings in Coating Service Level I Areas of Nuclear Power Plants
ISA	ANSI/ISA-67.02.01-2014	Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants
NFPA	NFPA 251	Standard Methods of Tests of Fire Resistance of Building Construction and Materials

SFR LOE=6 new standards
NEW-- Nuclear Analysis and Design of Concrete for Passive Heat Removal Systems. Higher energy neutrons and photons may affect the characteristics of the concrete. That is, the radiation and thermal environment of SFRs 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 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.
SFR-DC 70 addresses the design of the intermediate heat transfer loop. Although the intermediate heat transport system contains non-radioactive coolant, it should be monitored and inspected in areas where a sodium leak and any subsequent chemical reaction with air, concrete, or water might interfere with the safety function of equipment.
SFR-DC 71 addresses the need for maintaining purity of primary sodium coolant and cover gas. Although sodium is not a corrosive coolant, it can interact with trace impurities in heat transfer surfaces over time. Therefore, maintaining its purity is important to prevent chemical attack and to prevent buildup of reaction products which might lead to fouling or plugging of coolant channels.
SFR-DC 72 addresses the fact that sodium melts at 98°C and is a solid at room temperature. After startup, core residual heat is sufficient to keep sodium in the liquid state. However, heating may be required during initial filling operations, in cases of extended periods of shutdown, and to prevent sodium freezing in some sample or instrument lines. This criterion requires that a heating system be provided to assure that sodium freezing does not occur in safety related systems and components which contain or could be required to contain sodium. The criterion also requires the heating system be designed and controlled so as not to exceed safety design limits of these safety systems while in operation.
SFR-DC 73 requires sodium leak detection and mitigation of reactions between sodium and air or concrete in the event of a leak to assure that safety functions of SSCs that could be affected by the leak are maintained.
SFR-DC 74 addresses the issue of potential sodium-water reactions. This criterion requires the designer to minimize the possibility of a steam generator leak and to mitigate the effects should a leak occur to assure the function of SSCs important to safety is not compromised.
SFR-DC 75 is similar to GDC 30 in 10 CFR Part 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed, fabricated, and tested using quality standards and controls sufficient to ensure that failure of the intermediate system would be unlikely.
SFR-DC 76 is similar to GDC 31 in 10 CFR Part 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed to avoid brittle and rapidly propagating facture modes.
SFR-DC 77 is similar to GDC 32 in 10 CFR Part 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed to avoid brittle and rapidly propagating fracture modes.
SFR-DC 78 addresses the consequence of leakage between the primary coolant system and a heat removal system (i.e. RHR system, intermediate coolant system) is more significant for primary coolant system (potentially impacting the fuel design limits or integrity of the primary coolant boundary) than it is for the heat removal system (coolant drawdown or introduction of radioactive sodium).
SFR-DC 79 is similar to GDC 33 in 10 CFR Part 50, Appendix A. GDC 33 focuses on the effects of primary coolant (sodium) loss. A leak in a SFR primary coolant system may expel the cover gas rather than the primary coolant. The cover gas in the SFR performs an important to safety function by protecting the sodium coolant from chemical reactions.

Results specific to MSRs

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

MSR LOE=6 new standards		
ANS	High	A new standard to address the higher energies imparted on concrete being used for passive heat removal.
ANS	High	Tritium is produced within the heat exchanger tubes, which impacts corrosion and potential for release, and that this could be addressed through material standards.
ANS	High	Systems shall be provided as necessary to maintain the composition of the fuel salt within specified limits. Fuel salt properties are determined by its composition, which must be maintained within acceptable limits. (ANS 20.2, Criterion 71)
ANS	High	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 unrestricted environment. (Evaluation Findings in ORNL/TM-2020/1478)
ANS	High	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. (Evaluation Findings in ORNL/TM-2020/1478)
ANS	High	New criteria reflective of the importance of preventing freezing of salt and thermally damaging fuel salt contacting containment layers. (ANS 20.2, Criterion 72)
ANS	High	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. (SFR-DC 78, ANS 20.2, Criterion 74)
ASTM	High	The information on the reactor fuel should include a description of the required characteristics. A standard similar to ASTM C776-89, Part 45 should be developed for molten salt. (Evaluation Findings in ORNL/TM-2020/1478)
ANS	High	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, and the design-basis pressure are maintained. (Evaluation Findings in ORNL/TM-2020/1478)
ANS	High	A new standard should be developed that addresses materials being exposed to tritium and the susceptibility to corrosion.
ASME	Medium	A new standard based on review of ASME QME-1 should be developed for the qualification of passive equipment.
ANS	Low	Some MSR designs use dissolved fuel rather than the dispersed fuel approach This represents a novel approach to reactor fueling (and defueling) that is not directly addressed by existing experience and could lead to new standard requirements.
ANS	Low	Sufficient information should be provided to show that the functional and safety-related design bases can be achieved by the control elements designs. (Evaluation Findings in ORNL/TM-2020/1478)
ANS	Low	Each nuclear reactor should contain a neutron startup source that ensures the presence of neutrons during all changes in reactivity. Acceptance criteria for information on the neutron startup source should be developed. (Evaluation Findings in ORNL/TM-2020/1478)