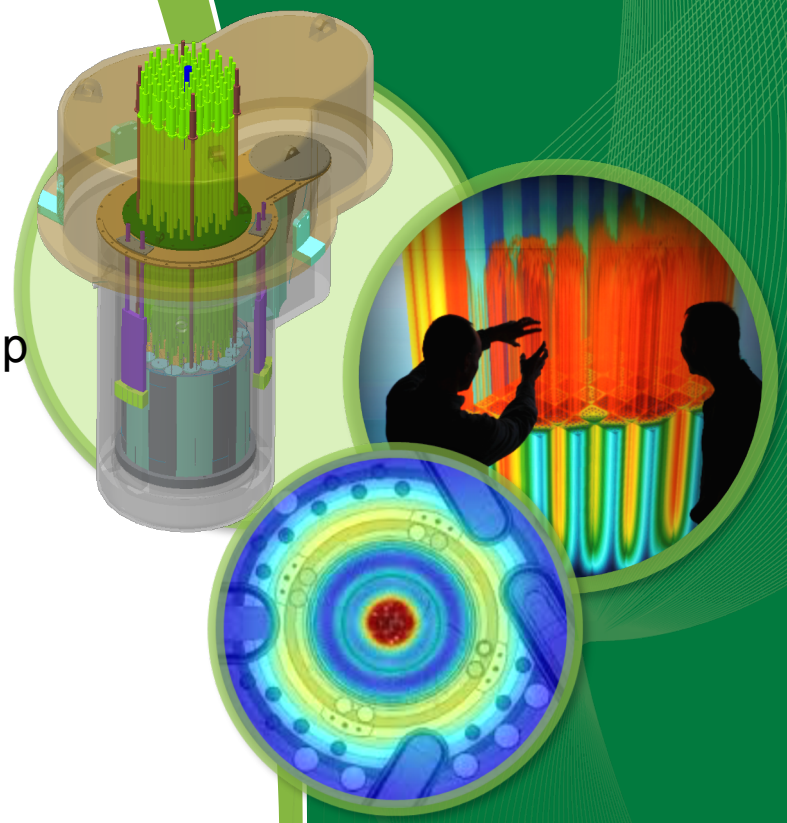
Reactor Technology Standards Support

Dr. Michael Muhlheim

Dr. George Flanagan
Advanced Reactor Systems and Safety Group
Reactor and Nuclear Systems Division
Oak Ridge National Laboratory



NRC-DOE Workshop on Advanced Reactors, Bethesda, MD

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IAP—"NRC Advanced Reactor Vision and Strategy: Mid-Term and Long-Term Implementation Action Plans" (ML17054D483)

- Contributing Activity No. 1 – Continue efforts to facilitate development of industry codes and standards
- Contributing Activity No. 2 – Develop regulatory guides and conduct rulemaking, as needed, to endorse industry codes and standards
- The outcome from these activities is to have available consensus codes and standards endorsed by the NRC to improve the effectiveness and efficiency of the licensing and regulation of non-LWR technologies
  - The review and NRC endorsement of codes and standards (with possible clarifications and exceptions) can only follow the development and issuance of the codes and standards by SDOs

DOE initiated a scoping study to understand the process, scope, and gaps of developing voluntary consensus standards for an SFR

1. Obtain a list of all standards cited in RGs
   - Standards include consensus standards and industry standards
2. From this list, select a few standards for an in-depth review to assess their potential application for non-LWR technologies
   - Down select the number of standards for review to endorsed standards **(HOW MANY)**
   - Assess the standards **applicability to a sodium fast reactor (SFR)** (i.e., technology specific or technology neutral)
   - Categorize the level of effort required to develop or revise the standard for applicability to an SFR **(HOW MUCH EFFORT)**
3. Describe the process for developing, approving, and endorsing a consensus standard
   - Discuss and estimate the timelines for modifying a standard through the standards committees
   - Discuss the process of citing or endorsing a standard by the NRC

A report on the outcome of this scoping study will completed in Sept 2017

Number of standards assessed*

Standards	Std org	RGs	Coverage
865 citations		486	Div 1-10 RGs
817 citations		225	Div 1 RGs (Power Reactors)
	30	179	Div 1 RGs, Active RGs
114 citations	9	67	Div 1 RGs, Active RGs, Endorsed active standards
71 citations	8	36	Div 1 RGs, Active RGs, Endorsed active standards, no IEEE standards**
60 standards	8	35	Remove duplicate standards

as is	limited	extensive	unknown	N/A	new	Assess standards

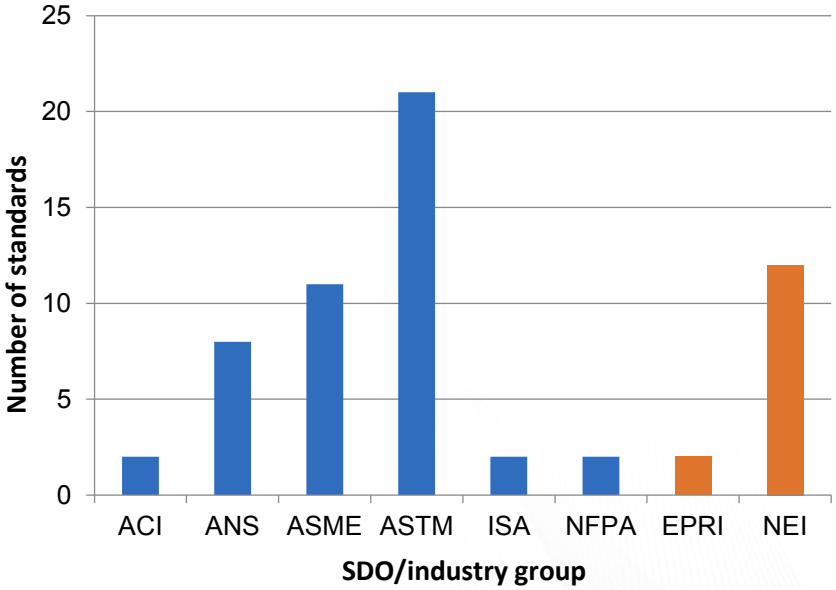
*Database distributed by NRC at the Nuclear Energy Standards Coordinating Collaborative (NESCC) circa 2012 (unpublished)

**IEEE standards are typically technology neutral

How many—60 standards endorsed in 35 RGs

- 6 SDOs, 46 standards
- 2 industry groups, 14 standards

SDO or industry group	No. endorsed standards	Total
ACI	2	46
ANS	8	
ASME	11	
ASTM	21	
ISA	2	
NFPA	2	
EPRI	2	14
NEI	12	
TOTAL	60	



Five "level of effort" categories were used to determine how much effort would be required to revise the standard for applicability to an SFR

A	B	C	D	E	F	G	H	I	J	K	L	M
ID	RG-rev	RG title	GDC	RG cited in SRP section	Standards	Standard title	SDO	Standard cited in SRP section	Change Summary	Level of Effort	Key Technical Issues	Comments, Notes

1 = none

- e.g., grades of fuel oil

2 = limited changes

- e.g., although applicable to all types of NPPs, specifically cites LWRs

3 = substantive changes needed

- e.g., use of sodium presents temperature and level measurement problems

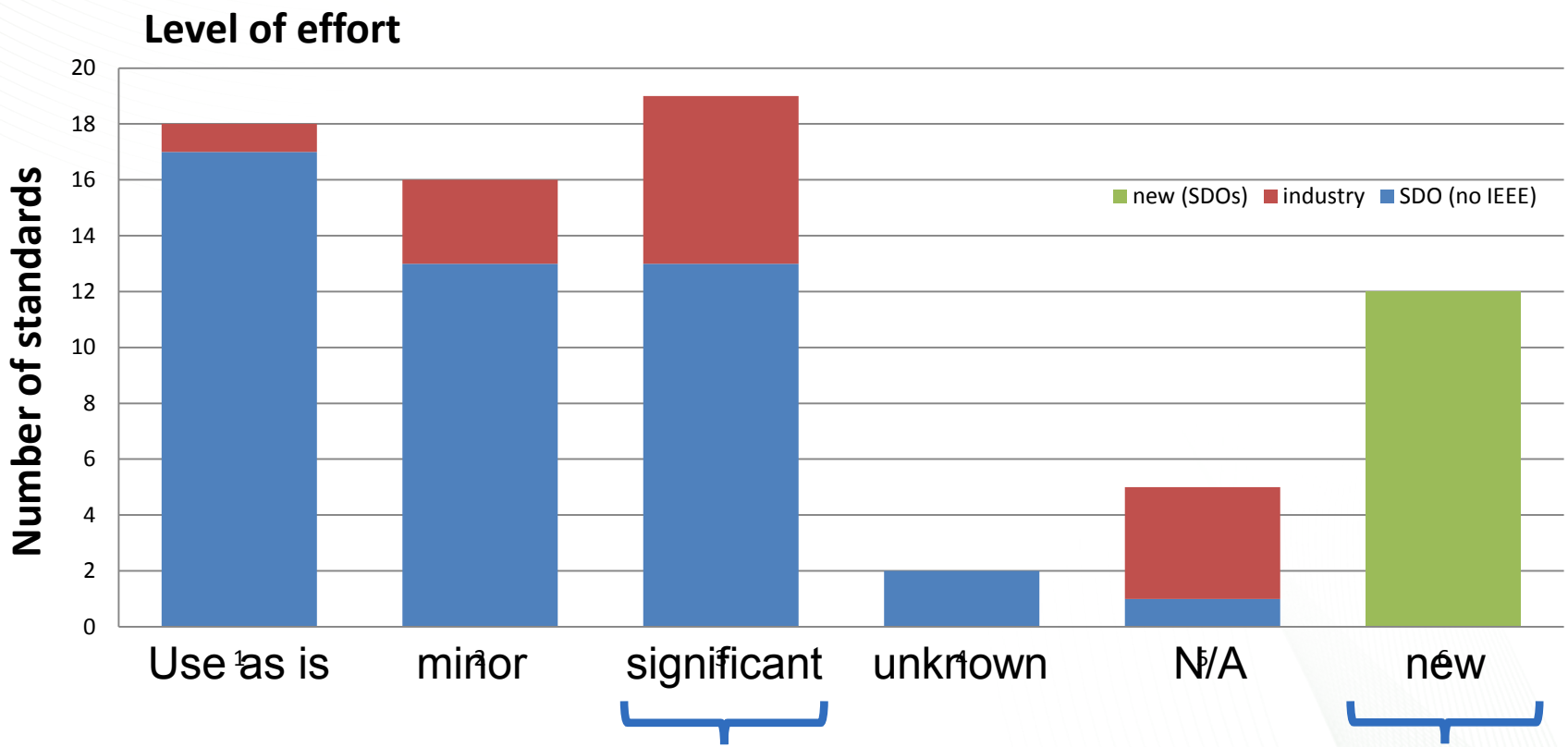
4 = insufficient design info

- e.g., conditions for testing of new and used carbons based on LWR accident conditions

5 = not applicable (N/A)

- e.g., requirements for EDGs

How much effort—35 of the 60 active standards endorsed by RGs will require revisions (i.e., 16 minor, 19 significant) 12 new standards identified



19 significant revisions

12 New

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Examples of “No changes (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 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	—	Quality assurance, 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.
EPRI 102543 R1 **(EPRI 3002002289)**	—	RG 1.231 endorses EPRI 1025243. However, EPRI 3002002289 supersedes EPRI 1025243. The "Level of Effort" is based on RG 1.231 endorsing the new EPRI document. The EPRI Technical Report 1025243 guidance was specifically developed to guide the technical evaluation and acceptance of commercial-grade design and analysis computer programs1. It incorporates knowledge of industry standards and operational experience in the formulation of QA guidance supporting both operating nuclear plant operations and upgrades, and new nuclear plant design and construction. The commercial grade dedication should be the same for SFRs.

Examples of "Limited Changes (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 SRO and 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 SFRs and the scope should be expanded to include SFRs.	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.
NEI 00-04	The process for evaluating and identifying Risk Informed Safety Classifications (RISC) SSCs is applicable to SFRs. The examples are all LWR specific and would have to be updated to provide guidance for SFRs.	The objective of this regulatory initiative is to adjust the scope of equipment subject to special regulatory treatment (controls) to better focus licensee and NRC attention and resources on equipment that has safety significance. This guideline addresses the use of risk insights to define the scope of equipment that should be subject to NRC special treatment provisions as defined in §50.69.

Examples of "Substantive changes (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 SFRs. 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 BPVC 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.
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.	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. In an SFR, the RCPB is the primary coolant boundary.

Examples of “Unknown (4)”

Standard	Change Summary	Key Technical Issues
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 SFR, 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.
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.

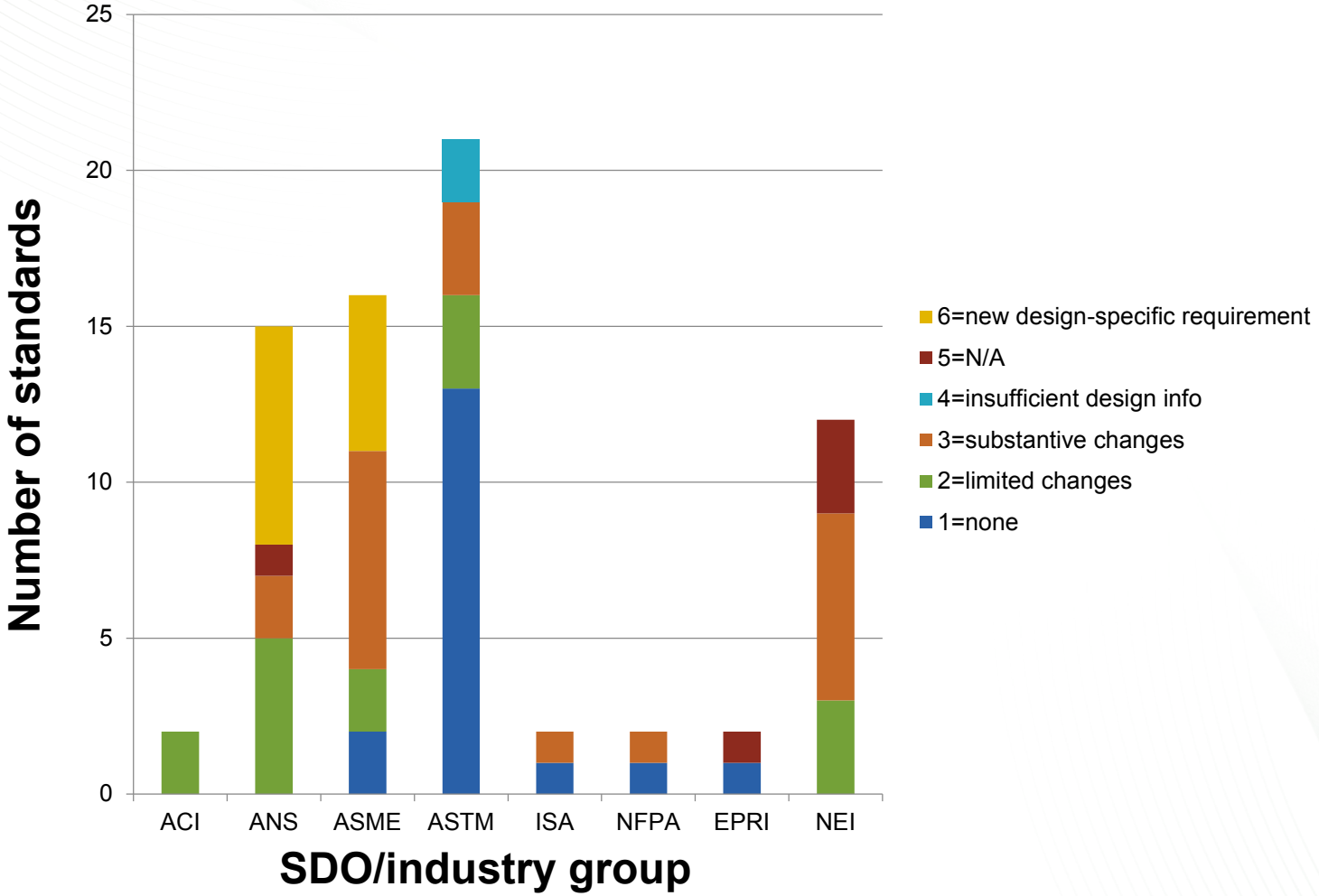
Examples of "N/A (5)"

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, SFRs will not use EDGs. 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 SFR-specific set of plant conditions would not be necessary.
NEI 98-03	—	UFSARs provide a description of each plant and, per the Supplementary Information for the FSAR update rule, serve as a "reference document to be used for recurring safety analyses performed by licensees, the Commission, and other interested parties." The UFSAR is used by the NRC in its regulatory oversight of a nuclear power plant, including its use as a reference for evaluating license amendment requests and in the preparation for and conduct of inspection activities. The N/A assessment is based not only on applicability to an SFR but to applicability during the design phase. This condition was applied to prioritize standards that will be needed now and with the understanding that guidance and knowledge will change from now until a plant is built.
EPRI NP-6695 (EPRI 1025288)	—	Due to seismic events that resulted in the shutdown of nuclear power plants in Japan and the United States, the IAEA (IAEA, 2011) and EPRI (EPRI, 2012) have developed and updated guidance documents on the response and restart of nuclear plants following a seismic event. This updated guidance and NRC staff experience associated with restart of the North Anna nuclear power plant need to be reflected in an updated regulatory guide. The N/A assessment is based not only on applicability to an SFR but to applicability during the design phase. This condition was applied to prioritize standards that will be needed now and with the understanding that guidance and knowledge will change from now until a plant is built.

Examples of "New Standards Needed (6)"

Standard	Change Summary	Key Technical Issues
–	NEW standard for the design and analysis 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 the 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.	A requirement of the qualification of passive equipment is needed.
–	10 NEW standards based on the 10 new SFR-DCs identified in DG-1330. SFR-DC 70: Intermediate coolant system SFR-DC 71: Reactor building design basis SFR-DC 72: Sodium heating systems SFR-DC 73: Sodium leakage detection and reaction prevention and mitigation SFR-DC 74: Sodium/water reaction prevention/mitigation SFR-DC 75: Quality of the intermediate coolant boundary SFR-DC 76: Fracture prevention of the intermediate coolant boundary SFR-DC 77: Inspection of the intermediate coolant boundary SFR-DC 78: Primary coolant system interfaces SFR-DC 79: Cover gas inventory maintenance	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. A new standard will be required to define 1. the means to detect sodium leakage in inerted or air environments, 2. the extent to which sodium-air and sodium-concrete reactions are limited and controlled, 3. the degree to which the effects of fires are mitigated, and 4. the means for evaluating the effectiveness of special features or conditions containing sodium to ensure that the safety functions of SSCs important to safety are maintained.

Of the SDOs with standards endorsed by a RG, all of the SDOs/industry groups (except EPRI) will need to be involved in revising or developing at least one standard for applicability to an SFR



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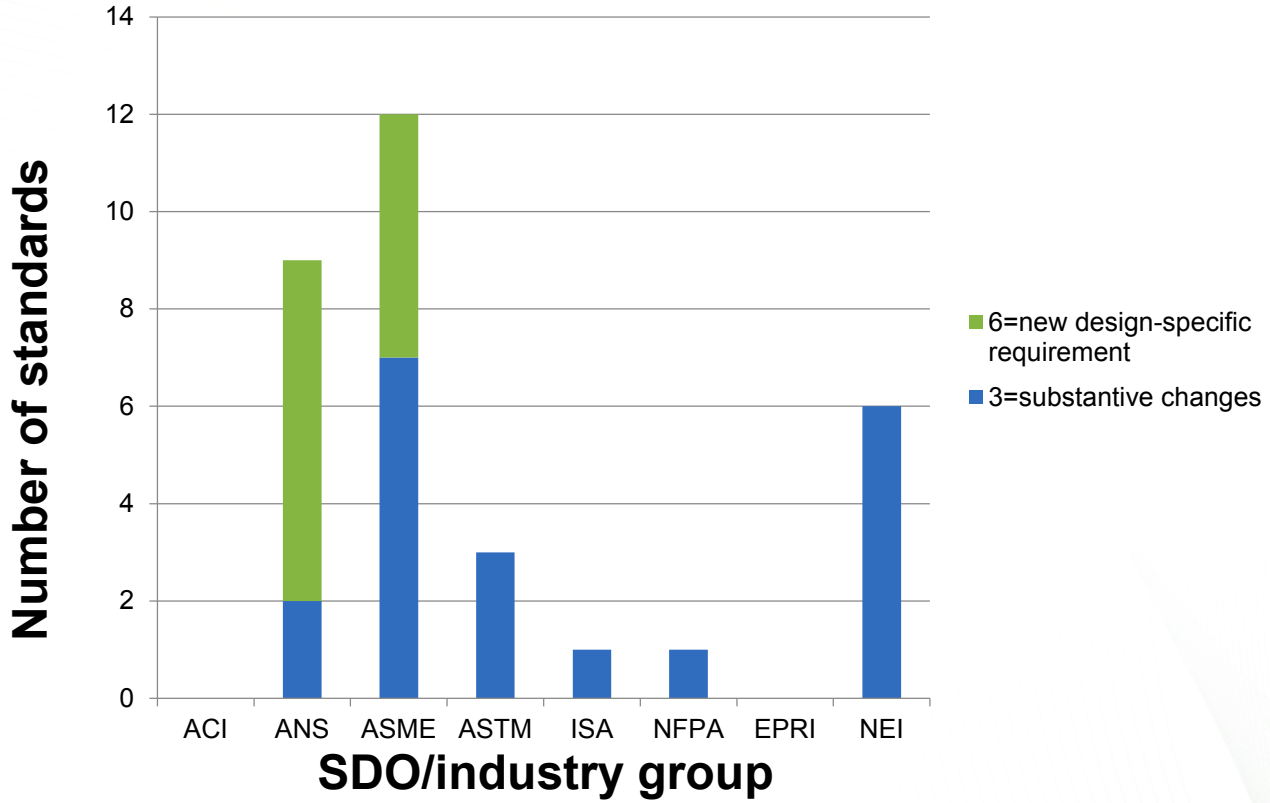
Impact could be significant on
SDOs/industry groups
Plant design
Start of operations



1. Time for minor changes to a standard to be approved (LOE = 2): 0.5–2 years
2. Time for significant changes to a standard to be approved (LOE = 3): 1–3 years
3. Time for the development and approval of a new standard (LOE = 6): 2–8 years

LOE = level of effort

Summary of results of reviews

- Of the 60 voluntary consensus standards and industry standards endorsed by RGs that have been reviewed, 19 will likely need substantive changes
  - 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, …
- 12 new consensus standards will be required
  - 10 SFR-DCs (70–79) identified in DG-1330
  - Passive cooling
  - Passive equipment
- The IEEE standards are technology neutral

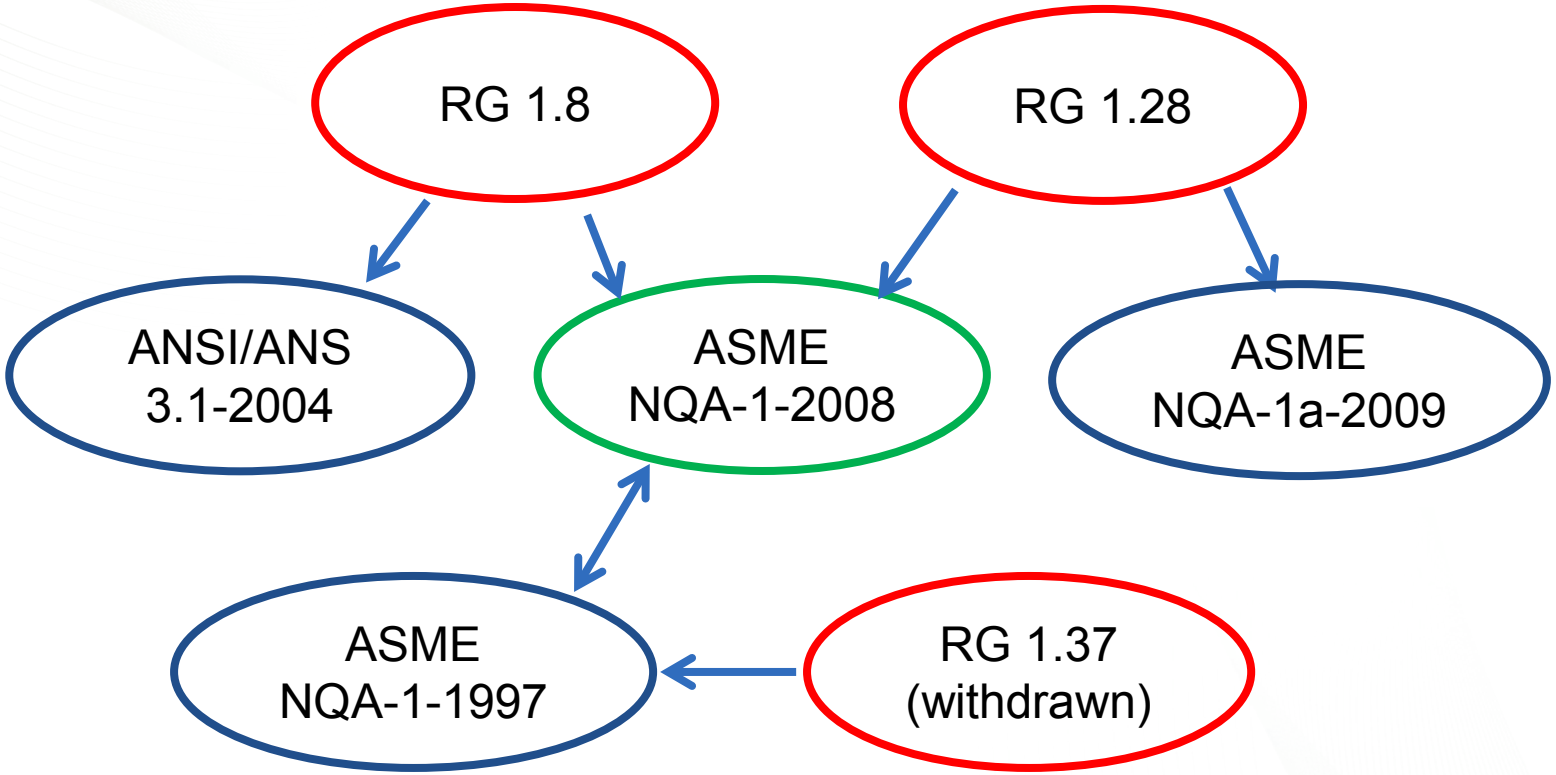
Next steps for the SFR technology…

- Rank the 31 standards needing significant changes (19) or are new (12)
  - Applicability to other reactor types
  - Data exists or research is needed to collect data
  - Impact on SDO—the number and a staggered submittal of requests will be necessary to prevent overwhelming an SDO
- Complete the gap assessment and provide detailed assessments and inputs to the revision of existing consensus standards and the development of new standards
  - Identify and assess those standards approved in the SRP (similar to endorsed standards in RGs)
  - Perform an assessment on the standards enforced by regulations (e.g., 10 CFR 50)
  - Develop a draft of the 12 new standards and identify revisions to those standards that will require significant changes
- Designs can proceed without approved standards; however, the benefits of approved standards include
  - Approved standards can help multiple licensees. This review focused on SFRs and should be expanded to other reactor technologies (e.g. MSRs, gas, etc.)
  - The use of standards is an integral part of the NRC's strategy to improve its readiness to regulate non-LWR technologies (IAP)

Backup

Duplicate standards removed from count

A RG may endorse more than 1 standard
A standard may be endorsed by more than 1 RG



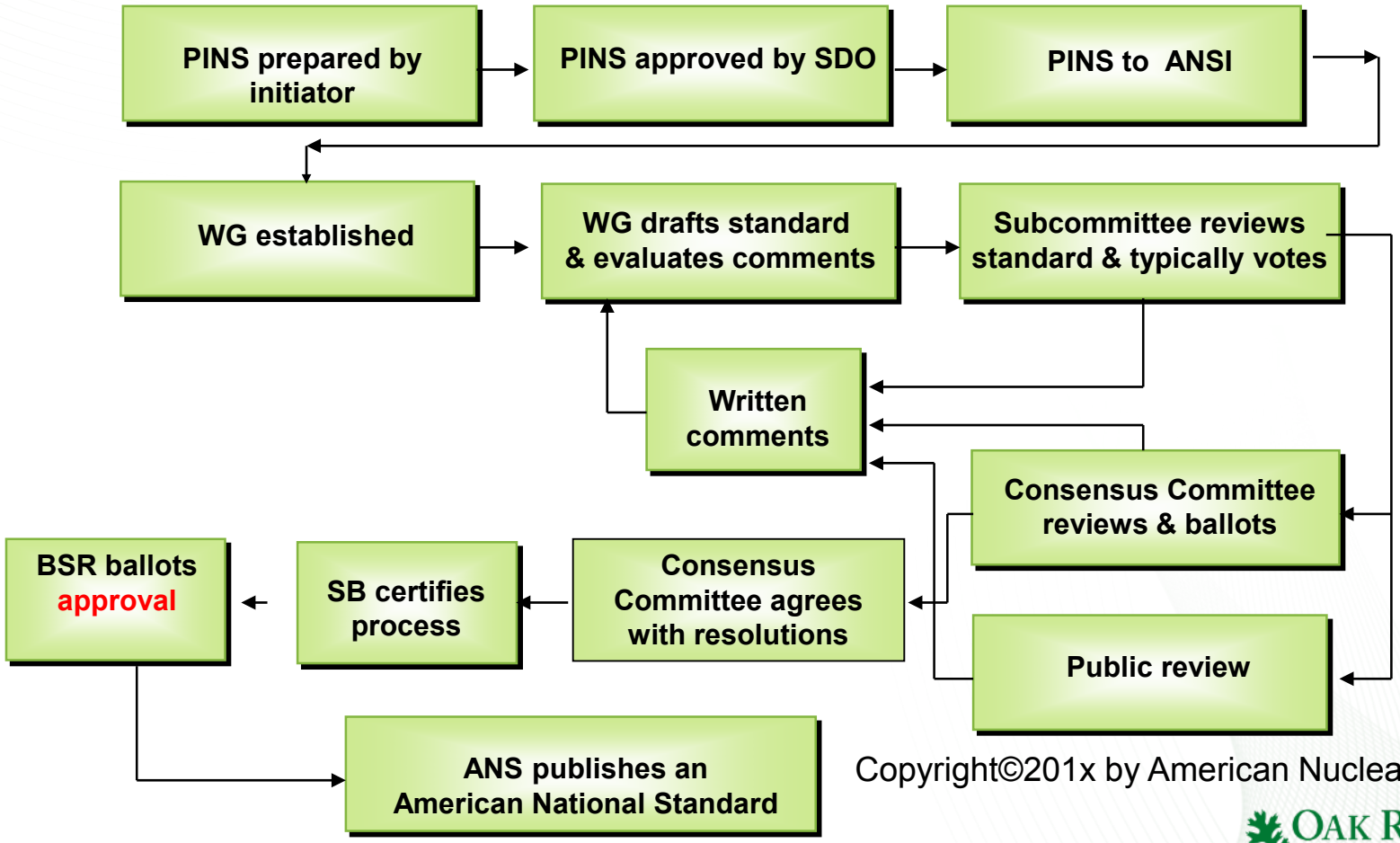
RG 1.8 Qualification and Training of Personnel for NPPs
RG 1.28 QA Program Criteria (Design and Construction)
RG 1.37 QA Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled NPPs
ANSI/ANS 3.1-2004 Selection, Qualification, and Training of Personnel for NPPs
ASME NQA-1-2008 QA Requirements for Nuclear Facility Applications
ASME NQA-1a-2009 Addenda to ASME NQA-1-2008

Developing and approving a voluntary consensus standard is a long and involved process

—the example below is provided by the ANS Standards Committee—

Voluntary Consensus Standards Development Process

American Nuclear Society Standards Committee



Standards selected for review (LOE = 3)

ACI	ACI 349.1R-07	Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures
ACI	ACI 349-2013	Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary
ANS	ANS 56.2-1984 (ANSI N271-1976)	Containment Isolation Provisions for Fluid Systems
ANS	ANSI/ANS 3.1-2014 based on periodic review	Selection, Qualification, and Training of Personnel for Nuclear Power Plants
ANS	ANSI/ANS 3.2-2012	Managerial, Administrative, and Quality Assurance Controls for Operational Phase of Nuclear Power Plants
ANS	ANSI/ANS 3.4-1996	Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants
ANS	ANSI/ANS 3.5-2009	Nuclear Power Plant Simulators for Use in Operator Training and Examination
ANS	ANSI/ANS 59.51-1997	Fuel Oil Systems for Safety-Related Emergency Diesel Generators
ANS	ANSI/ANS 6.3.1-1987 (R2007)	Program for Testing Radiation Shields in Light Water Reactors (LWR)
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 N510-2007	Testing of Nuclear Air-Treatment Systems
ASME	ASME N511-2007	In-Service Testing of Nuclear Air Treatment, Heating, Ventilating, and Air-Conditioning Systems
ASME	ASME NQA-1-2008	Quality Assurance Requirements for Nuclear Facility Applications
ASME	ASME NQA-1a-2009 (Addenda to ASME NQA-1-2008)	Quality Assurance Requirements for Nuclear Facility Applications
ASME	ASME QME-1-2007	Qualification of Active Mechanical Equipment Used in Nuclear Power Plants
ASTM	ASTM D3803-1991	Standard Test Methods for Nuclear-Grade Activated Carbon
ASTM	ASTM D3843-16	Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities
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 D3912-10	Standard Test Method for Chemical Resistance of Coatings and Linings for Use in Nuclear Power Plants
ASTM	ASTM D4082-10	Standard Test Method for Effects of Gamma Radiation on Coatings for Use in Nuclear Power Plants
ASTM	ASTM D4227-05	Standard Practice for Qualification of Coating Applicators for Application of Coatings to Concrete Surfaces
ASTM	ASTM D4228-05	Standard Practice for Qualification of Coating Applicators for Application of Coatings to Steel Surfaces
ASTM	ASTM D4286-08	Standard Practice for Determining Coating Contractor Qualifications for Nuclear Powered Electric Generation Facilities
ASTM	ASTM D4537-04a	Standard Guide for Establishing Procedures To Qualify and Certify Personnel Performing Coating Work Inspection in Nuclear Facilities
ASTM	ASTM D4538-05	Standard Terminology Relating to Protective Coating and Lining Work for Power Generation Facilities
ASTM	ASTM D4541-09	Standard Test Method for Pull-Off Strength of Coatings Using Portable Adhesion Testers
ASTM	ASTM D5139-10	Standard Specification for Sample Preparation for Qualification Testing of Coatings To Be Used in Nuclear Power Plants
ASTM	ASTM D5144-08	Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants
ASTM	ASTM D5163-08	Standard Guide for Establishing a Program for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants
ASTM	ASTM D5498-09	Standard Guide for Developing a Training Program for Personnel Performing Coating Work Inspection for Nuclear Facilities
ASTM	ASTM D6677-07	Standard Test Method for Evaluating Adhesion by Knife
ASTM	ASTM D7108-05	Standard Guide for Establishing Qualifications for a Nuclear Coatings Specialist
ASTM	ASTM D7167-05	Standard Guide for Establishing Procedures To Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant
ASTM	ASTM D7234-05	Standard Test Method for Pull-Off Adhesion Strength of Coatings on Concrete Using Portable Pull-Off Adhesion Testers
ASTM	ASTM D7491-08	Standard Guide for Management of Non-Conforming Coatings in Coating Service Level I Areas of Nuclear Power Plants
ASTM	ASTM D975-13	Standard Specification for Diesel Fuel Oils
EPRI	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	EPRI NP-6695 (EPRI 1025288)	Guidelines for Nuclear Plant Response to an Earthquake
ISA	ANSI/ISA-67.02.01-2014	Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants
ISA	ANSI/ISA-S67.04.01-2006	Setpoints for Nuclear Safety-Related Instrumentation
NEI	NEI 00-01 Rev. 2(b)	Guidance for Post-Fire Safe-Shutdown Circuit Analysis
NEI	NEI 00-04	10 CFR 50.69 SSC Categorization Guideline
NEI	NEI 04-02 Rev. 2	Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)
NEI	NEI 07-13 Rev. 8	Methodology for Performing Aircraft Impact Assessments for New Plant Designs
NEI	NEI 08-01 Rev. 5 - Corrected	Industry Guideline for the ITAAC Closure Process under 10 CFR Part 52
NEI	NEI 09-09	Nuclear Power Plant-Referenced Simulator Scenario Based Testing Methodology
NEI	NEI 94-01	Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J
NEI	NEI 95-10 Rev. 6	Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule
NEI	NEI 96-07	Guidelines for 10 CFR 50.59 Evaluations
NEI	NEI 97-04 Appendix B	Guidelines and Examples for Identifying 10 CFR 50.2 Design Bases
NEI	NEI 98-03	Guidelines for Updating Final Safety Analysis Reports
NEI	NEI 99-03	Control Room Habitability Assessment Guidance
NFPA	NFPA 251	Standard Methods of Tests of Fire Resistance of Building Construction and Materials
NFPA	NFPA 600	Standard on Industrial Fire Brigades

Consensus standards referenced in the Code of Federal Regulations (CFR)

10 CFR Section	Standards Developing Organization or coordinating organization
34.20	ANSI
50.55a	ASME
50.55a	IEEE
50.61	ASME
50 App G	ASME
50 App H	ASTM
50 App J	ANSI
50 App J	ANS
50 App K	ANS
50 App R	IEEE
73.26	ANSI
73.26	ISO
73 App B	ANSI
73 App B	ISO

http://www.nrc.gov/about-nrc/regulatory/standards-dev/consensus.html#rg