NEI TECHNICAL REPORT

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Advanced Reactor Codes and Standards Needs Assessment

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Executive Summary

The future of nuclear power in the United States and the re-establishment of U.S. global leadership in nuclear energy depends, in part, on the successful deployment of new and advanced reactor designs. Currently, over thirty U.S. companies are actively developing advanced designs that differ markedly from the light water reactors (LWRs) operating in the U.S. today. For these designs to be deployed commercially in the U.S., they must be approved by the Nuclear Regulatory Commission (NRC). Because the NRC's regulatory process is light water reactor-focused, modifications to the existing regulatory and technical guidance are needed to support the timely and efficient licensing of advanced technologies.

The NRC's regulatory processes for LWRs and the associated rules, regulatory and industry guidance documents, and codes and standards have been developed over the past 50 years. Near term development of a similar infrastructure to support the design, licensing and deployment of advanced reactors within the timelines is not realistic. Designers are able to move forward to complete their designs and license applications utilizing existing codes and standards with design specific analysis and justification. However, efficiencies can be achieved through the development of additional codes and standards applicable to advanced reactors.

The purpose of this document is to identify codes and standards that could provide the greatest benefit for the advanced reactor design types being developed today, and to prioritize them so that the most beneficial codes and standards are developed first. Prioritization is based on the benefit to potential NRC applicants in terms of facilitating the licensing process and reducing design, component fabrication, facility construction and plant operating costs. Prior activities by the Oak Ridge National Laboratory (ORNL), American Nuclear Society (ANS) and NRC identified technical areas that warrant additional research and development to support standards development activities and a lengthy list of standards that need levels of revision to support the deployment of advanced reactors. Building on those activities, the tables contained herein list prioritized codes and standards and include descriptions of their content to explain the rationale of the specific changes needed to facilitate application to advanced reactors.

In developing the lists, over 800 codes and standards were identified. Of these, 18 have been evaluated as high priority with the potential to provide the greatest benefit for near term development. This list is summarized 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 the Department of Energy (DOE) and the NRC is needed.

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LIST OF ABBREVIATIONS

ACI	American Concrete Institute
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	Boiler and Pressure Vessel Code
DOE	Department of Energy
GAIN	Gateway for Accelerated Innovation in Nuclear
HTGR	High Temperature Gas Reactor
IEEE	Institute of Electrical and Electronics Engineers
ISA	International Society of Automation
JCNRM	ANS/ASME Joint Committee on Nuclear Risk Management
LMFR	Liquid Metal Fast Reactor
LWR	Light Water Reactor
MSR	Molten Salt Reactor
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIA	Nuclear Innovation Alliance
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
RG	Regulatory Guide
SDO	Standards Development Organization
SFR	Sodium Fast Reactor

1 OVERVIEW

The future of nuclear power in the U.S. and the re-establishment of U.S. global leadership in nuclear energy will require the successful deployment of new and advanced reactor designs. Currently, there are numerous U.S. companies engaged in advanced reactor design development activities. These designs offer safety enhancements by “designing out” the potential for many types of accident scenarios that challenge existing LWRs and by using passive safety features that do not rely on alternating current or direct current power. The designs being developed are significantly different from the large LWRs in operation in the U.S. today in that they use coolants such as helium, liquid metal, and molten salts, and operate at much higher temperatures.

To realize the benefits of enhanced economic viability and safety, these advanced reactor designs must first be demonstrated to have enhanced safety and obtain Nuclear Regulatory Commission (NRC) approval. NRC’s existing regulatory processes are focused on large LWR technology. Changes are being developed to support the licensing of advanced non-LWRs. NRC, the Department of Energy (DOE) through the National Laboratories, the Nuclear Energy Institute (NEI), the Nuclear Innovation Alliance (NIA) and the standards development organizations (SDOs) have begun work to facilitate the development of the technical and licensing infrastructure for advanced non-LWRs.

Because the domestic and international deployment of advanced reactors depends on economic viability as well as the robustness of the safety of the design, additional codes and standards will help minimize unnecessary conservatism in the designs and will help make the regulatory process more timely, efficient and certain.

The purpose of this document is to identify and prioritize the codes and standards that should be developed or modified to facilitate the deployment of advanced reactors. To help explain the rationale behind the needed change, the subject matter for each guidance document, rule, or code and standard is described.

2 BACKGROUND

Consensus codes and standards have been employed in the design, licensing, construction and operation of commercial nuclear power plants since the first demonstration plants were built and operated. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) provided the design criteria for the reactor pressure boundary and listed materials suitable for the fabrication of pressure boundary components, building on experience gained with power boilers and naval reactor components. The early design criteria were very conservative to address the level of sophistication in the available analysis tools and the experiences with power boilers. Over time, as analysis methods improved and construction methods evolved, the Code endorsed the more sophisticated analysis and construction methods and reduced the conservatisms necessary to maintain margins of safety that were required.

The ASME BPVC is one key example of consensus standards used by the nuclear industry and endorsed by the Nuclear Regulatory Commission through regulation or regulatory guidance. However, consensus standards from a wide range of consensus standards bodies are routinely used by the industry and the NRC in ensuring the safe design, construction and operation of commercial nuclear power facilities. The subjects addressed by these standards include virtually every aspect of plant safety, and the standards development organizations involved with the development and promulgation of these standards include ASME, ASCE, ACI, ANS, ASTM, IEEE, ISA and NFPA, just to mention a few of the key organizations.

The standards organizations involve members representing a balance of interests including designers, operators, suppliers, construction organizations, consultants and government organizations. This ensures that the standards reflect interests from all relevant organizations and are not dominated by a single party.

Consensus standards that are developed or revised to address advanced reactor technologies can provide significant support to the design and licensing of advanced reactors, and support component fabrication, facility construction and eventual facility operation.

3 OBJECTIVES

The overall objective of this document is to identify the advanced reactor codes and standards that could be the most beneficial for supporting the design, licensing, construction and operation of advanced reactors. The assessment is technology-inclusive, so it can be applicable to all designs being developed by advanced reactor vendors. The results of this assessment should be helpful in the prioritization and scheduling of the development of key standards that could be most beneficial to advanced reactor developers.

4 PRIORITIZED LIST OF CODES AND STANDARDS NEEDS

4.1 DOE-sponsored Review of Applicable Codes and Standards

Since all commercial nuclear power plants in operation today in the U.S. are LWR designs, the regulatory infrastructure of rules, regulatory guidance and policy is currently focused on LWRs. Consensus codes and standards are endorsed in NRC's regulations and in Regulatory Guides (RGs) and play a key role in the LWR regulatory infrastructure. These consensus codes and standards are developed and maintained by standards development organizations, such as ASME and the Institute of Electrical and Electronics Engineers (IEEE). Since the codes and standards are also LWR focused, many do not apply to non-LWR technologies. To understand the magnitude and scope of work required to establish codes and standards for advanced reactors, the Oak Ridge National Laboratory (ORNL) under DOE funding performed a scoping study focused on codes and standards for SFRs that provided [35]:

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.

The first step in estimating the magnitude and scope of the effort was to obtain a list of all standards cited in Regulatory Guides. This step identified 865 standards.

The second step was to narrow the number of standards for an in-depth review to assess their potential application to SFRs. This step identified 114 standards; however, the 43 standards from the IEEE were reviewed separately because they were expected to be technology neutral (this assumption was confirmed by a separate review). Of the remaining 71 standards, 11 were duplicates, leaving 60 standards for detailed review.

The third step was to review the 60 standards with respect to their applicability to SFRs and to identify the need for new standards unique to SFRs. This step identified 12 potential new standards that would be design specific. Of the 60 standards endorsed by RGs, 46 were voluntary consensus standards and 14 were initiated by the nuclear industry. Of the 46 consensus standards 17 will not require any changes. Of the remaining standards, 13 will require limited changes, 13 will require significant changes, and 3 lacked sufficient information to assess their applicability. Since this review focused on codes and standards needed for SFRs alone, it is clear that a substantial level of effort is needed by the SDOs, nuclear industry and the NRC to support the development and endorsement of the full range of codes and standards that would be beneficial to all advanced reactor designs.

4.2 Prioritization of Research and Development Needs

During the May 2, 2018 ANS /NRC Workshop to Develop a Strategic Vision for Advanced Reactor Standards, technology working groups identified the top technical areas for continued research and development. Additionally, the need for development or revision of six consensus standards was identified beyond those identified in the ORNL review. The technical areas where the need for continued R&D was identified are summarized in Table 1.

Table 1. Top Technical Areas for R&D by Technology

Technology	Technical Area / Issues to be Addressed	Lead Organization
HTGR	ASME Sections III and VIII cyclic loads for high temperature design. For example, metallic reactor core support structures typically may be exposed to temperatures in excess of 500°C (930°F) in normal operation, with short-term excursions to 670°C (1240°F) [39]. Alloy 800H core support components may be exposed long-term to 800 to 850°C (1472 to 1562°F) for some types of reactors. Other components, such as compact heat exchangers, may see normal operating temperatures in local areas up to 850°C (1560°F), with short-term excursions to 900°C (1650°F). 1. Develop initial loading and cyclic stress-strain curves for all materials to be used for BPV III-5 and VIII construction for the complete temperature range of interest if these data are needed for the analysis methods selected. 2. Develop improved design methodology for creep-fatigue evaluation by analysis. This approach should take full advantage of modern analysis tools, such as elastic-plastic finite element analysis with creep strain capability. Note that R&D is recommended to develop the necessary material models.	DOE/Vendor/GAIN Vouchers
HTGR	Fiber optic (specifically) and qualification of instrumentation and control (I&C) for high temperatures in general: For example [40],	DOE/Vendor

Technology	Technical Area / Issues to be Addressed	Lead Organization
	1. No temperature sensors currently available to measure the pebble temperature distribution in the core directly in pebble bed reactors. 2. Distributed fiber-optic Bragg thermometry is susceptible to photo-bleaching in high radiation, high temperature environments. 3. No suitable in-core neutron flux measurement sensor is commercially available that functions reliably at temperatures above ~550°C.	
HTGR	ASME Section XI high-temperature ISI: For example, identify non-destructive examination (NDE) technologies of advanced monitoring, diagnostics and prognostics systems (e.g., acoustic emission, ultrasonic, advanced material characterization) applicable to components of HTGRs for in-service inspection [41].	Vendor/GAIN Voucher
LMFR, other fast reactors	Structural alloys, cladding and coating materials for the temperature ranges and fluences of interest: Qualification of metallic materials for advanced high-temperature and high fast flux applications is to be based on creep behavior, fatigue properties, structural stability and corrosion resistance.	DOE
LMFR, other fast reactors	Concrete considerations at high temperature and fluence: Higher fluence may have detrimental effects on concrete compressive and tensile strength and modulus of elasticity. For example, for fast neutron fluence between 7×10^{18} and 3×10^{19} n/cm^2 the modulus of irradiated concrete was between 10% and 20% less than that of non-irradiated unheated concrete [42].	DOE
LMFR, other fast reactors	I&C - Spectral, material, temperature, and lifetime considerations: For example, a neutron flux monitoring system is required to provide measurements that aid in reactor start-up and enable efficient plant control, to monitor reactivity changes, and to detect reactor abnormal conditions. The overall neutron flux varies by 12-14 orders of magnitude from reactor shutdown to full-power operation conditions. R&D is needed to design and test the neutron flux monitoring system in a high-temperature sodium environment [43].	DOE/Vendor/GAIN Voucher
LMFR, other fast reactors	Fuel and material handling variations: For example, for LMFR fuel transportation, loading, unloading, storage, and cleaning, the fuel handling system consists of the rotational plug, in-vessel fuel handling machine, ex-vessel fuel transportation device, in-vessel storage or/and ex-vessel interim storage, fuel cleaning station, etc.	Vendor/GAIN Vouchers

Technology	Technical Area / Issues to be Addressed	Lead Organization
	Even though fuel handling experience was developed at the previously operated fast reactors both domestically and internationally, each fuel handling system is tailored for the specific geometry of the plant and there is no standard design for fuel handling systems [43].	
LMFR, other fast reactors	Decay heat: A standard that differs from ANSI/ANS 5.1-1979 LWR decay heat curve is needed to address fast spectrums, different coolants, fuel management practices including conversion ratios, and fuel configurations.	DOE/Vendor/GAIN Vouchers
MSR	Material options (metallic, graphite, etc.) for core components in a high fast neutron flux environment: For example, the advanced austenitic stainless steel alloy 709 has been under development for a number of years and is currently being code-qualified as a replacement material for the 300-series austenitic stainless steels (304, 316). The advanced ferritic-martensitic stainless steel modified 9Cr-1Mo has a significant strength advantage over the low-chrome ferritic steel Fe-2¼Cr-1Mo and can be used, with some additional R&D, as a construction material for a 60-year, non-replaceable steam generator design, and it can also be considered as a candidate material for an IHX design with a full 60-year design life [43].	DOE
MSR	Design, materials, and fabrication of structural components clad or lined with corrosion-resistant materials: For example, alloys high in iron and chromium such as inconel 106 and type 316 stainless corrode readily and lose significant amounts of mass within a few thousand hours [44]. Alloys with high nickel, molybdenum, niobium and silicon show better corrosion resistance [45].	DOE/Vendor/GAIN vouchers
MSR	Refractory alloys: welding, fabrication and joining techniques, and better understanding of embrittlement and fracture behavior are needed.	DOE/Vendor/GAIN vouchers
All designs	In general, additional material options such as high strength nickel alloys are needed to broaden the approved material choices for high temperature structural applications in many advanced reactor designs.	DOE/Vendors

4.3 Prioritization of Non-LWR related Codes and Standards Development Needs

Given the large number of codes and standards identified by ORNL and in the ANS/NRC workshop that need to be developed or revised for advanced reactors, they were prioritized based on the criteria discussed below.

Criteria utilized to rank importance include:

1. Standard will support design efforts;
2. Standard will support licensing review;
3. Standard will reduce component fabrication time and costs;
4. Standard will reduce facility construction time and costs; and
5. Standard will reduce O&M costs.

Standards were then scored to determine priority based on the following:

1. Standards that (1) support design and licensing, (2) support either design or licensing and at least two other criteria, or (3) standards where three of the other criteria are satisfied, were ranked as "High."
2. Standards that satisfied two of the criteria not specifically related to design or licensing were ranked as "Medium".
3. Standards that satisfied one of the criteria not specifically related to design or licensing were ranked as "Low."

Table 2 summarizes the codes and standards identified in the ORNL and ANS/NRC activities that are affected along with a proposed prioritization. The numbers in the priority column relate to the relevant ranking criteria. A total of 36 standards are included in the table, which includes new standards on advanced manufacturing methods that were not identified by either ORNL or ANS/NRC.

Table 2. Prioritization of Codes and Standards

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
High 1,2,4	Equivalent to ANSI/ANS 6.4-2006, Concrete for Passive Heat Removal Systems - Irradiation and Thermal Limits	This standard would contain methods and data needed to calculate the concrete thickness required for radiation shielding in advanced nuclear power plants. Specific recommendations are to be made regarding radiation attenuation calculations, shielding design and standards of documentation. The standard would provide guidance to architect-engineers, utilities and	New / substantive effort	ANS

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
		reactor vendors who are responsible for the shielding design of advanced reactor designs.		
Medium 2,3	Equivalent to ASME QME-1, Qualification of Passive Equipment NOTE: This standard was identified in the ORNL report, Section 4.3, but was not included in standards listed in Appendix A of that report.	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.	New / substantive effort	ASME
High 1,2	ANS-30.1-201x Risk-informed Performance-based Principles and Methods	This standard describes the objectives of incorporating risk-informed and performance-based (RIPB) information into the safety design of advanced reactors. This standard will demonstrate high-level uses of RIPB methods.	New / on-going	ANS
High 1,2,5	ANS-30.2-201x Categorization and classification of SSCs	This standard provides a single technology neutral categorization and classification process for SSCs for advanced reactors that is, where possible, RIPB. This process will then be used to determine special treatment of SSCs to meet the safety basis. It would provide a complete (i.e., necessary and sufficient) repeatable logical process based upon RIPB objectives.	New / on-going	ANS
Not rated. This standard is on hold.	ANS-20.1-201x, Nuclear Safety Criteria and Design Criteria for Fluoride Salt-cooled High-temperature Reactors	This standard will establish the nuclear safety criteria, and design requirements for the fluoride salt-cooled high-temperature reactor nuclear power plants. RIPB criteria are used wherever possible. It describes the design process to be followed to establish those criteria and perform structures, systems, and component classifications.	New	ANS
High 1,2,5	ANSI/ANS 53.1, Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants, 2011	This standard provides a process for establishing top-level safety criteria; safety functions; top-level design criteria; licensing-basis events; design-basis accidents; safety classification of systems, structures and components (SSCs); safety analyses; defense-in-depth; and adequate	Limited	ANS

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
		assurance of special treatment requirements for safety-related SSCs throughout the operating life of the plant. The standard does not provide detailed guidance for design; other standards cover that.		
Medium 1	ANS-20.2 Nuclear Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten-Salt Reactor Nuclear Power Plants	This standard establishes the nuclear safety design criteria and functional performance requirements for liquid-fuel molten-salt reactor nuclear power plants. The document uses RIPB criteria wherever possible. It also describes the design process to be followed to establish those criteria and perform SSC classifications. This standard has been drafted.	New / on-going	ANS
High 1,2	ASME/ANS RA-S-1.4-2013 PRA for Non-LWRs (trial use)	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. This standard has been issued for trial use and has been subjected to many trial use applications. Lessons learned from the trial use and other parallel activities are currently being used by the JCNRM to provide an update for ANS status in 2020.	Finalize	ASME/ANS
High 1,2,4	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.	Substantive	ASME
Medium 1	ANSI/ISA-67.02.01-2014, Safety-related Instrument Sensing Lines	For advanced reactor designs, 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	Substantive	ISA

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
		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. Advanced reactors may also use optical sensors.		
Low 5	ASME BPVC Section XI, In-service Inspection of Components	This Section contains Division 1, “Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing of Components of Light-Water-Cooled Plants” and Division 2 “Rules for Inservice Inspection of Nuclear Power Plant Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants.” Some adaptation of flaw evaluation requirements to advanced reactors may be necessary.	Limited	ASME
High 1,2	New standards equivalent to SFR-DC 73 for sodium leak detection and mitigation (RG 1.232)	A new standard will be required to define 1. the means to detect sodium leakage into inert 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.	New	ANS
High 1,2,5	ANS 56.2-1984 (ANSI N271-1976), Containment Isolation for Fluid Systems	This standard specifies minimum design, actuation, testing, and maintenance requirements for the containment isolation of fluid systems after a LOCA in LWRs. These fluid systems penetrate the primary containment and include piping systems (including instrumentation and control) for all fluids entering or leaving the containment. Electrical systems are not included. The provisions for containment isolation impose additional requirements which are not required for the	Substantive	ANS

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
		fluid system function. Changes are necessary to address the varied advanced reactor designs, coolants as well as functional containment concepts.		
High 1,2,3,5	ASME AG-1-2009, Air and Gas Treatment	This standard considers the design, fabrication, inspection and testing of air cleaning and conditioning components and appurtenances used in safety-related systems in nuclear facilities. Changes are necessary to address the varied advanced reactor designs as well as functional containment concepts.	Substantive	ASME
High 1,2,3	ASME BPVC Section III, Construction of plant components	This Section contains requirements for the material, design, fabrication, and examination of supports which are intended to conform to the requirements for Classes 1, 2, 3, and metal containment construction. Nuclear power plant supports for which rules are specified in this Subsection are those metal supports which are designed to transmit loads from the pressure retaining barrier of the component or piping to the load carrying building structure. Changes are necessary to address the varied advanced reactor designs as well as functional containment concepts.	Substantive	ASME
High 1,2,3	New Standards for Advanced Manufacturing Techniques, including Additive Manufacturing NOTE: These types of standards were not included in either the ORNL report or the ANS/NRC summary.[1]	Additive manufacturing offers a new paradigm for engineering design and manufacturing by enabling unique macro-structural design of components and micro/nano-structural design of the material. There is a need to develop and publish standards applicable to the manufacturing of components suitable for use in safety-related and other plant applications.	New	ASME
High	ASME BPVC Section III Division 1, Subsection	Subsection NE pertains to Class MC components while Division 2 pertains to concrete reactor vessels and containments. Changes are	Substantive	ASME

[1] An NEI report was issued on May 13, 2019 entitled "Roadmap for Regulatory Acceptance of Advanced Manufacturing Methods in the Nuclear Energy Industry." The report identified the high priority manufacturing methods of greatest interest to the industry in the next 5 years.

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
1,2,4	NE, and Division 2, Containment	necessary to address the varied advanced reactor designs as well as functional containment concepts. Updates should consider including steel-plate composite (SC) construction for containment structures. Note that AISC N690 allows for SC walls for safety-related structures other than containment.		
High 1,2	ASME BPVC Section III, Division 5, High Temperature Reactors	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.	On-going	ASME
High 1,2,4	ASME N509-2002, Air Cleaning Units and Components	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.	Substantive	ASME
High 1,2,3	ASME QME-1-2007, Qualification of Active Mechanical Components	This standard describes the requirements and guideline for qualifying active mechanical equipment, such as pumps, valves, and dynamic restraints, used in nuclear facilities. The requirements and guidelines presented include the principles, procedures, and methods of qualification. Changes are necessary to address	Substantive	ASME

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
		the varied advanced reactor design components and high temperature applications.		
Low 4	ASTM D3911-16, Test Method for Evaluating Coatings	This test method is designed to provide a uniform test to determine the suitability of Coating Service Level 1 coatings used inside primary containment of LWR facilities under simulated design basis accident (DBA) conditions. This test method is intended only to demonstrate that under DBA conditions, the coatings will remain intact and not form debris which could unacceptably compromise the operability of engineered safety systems. 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.	Substantive	ASTM
Low 4	ASTM D7491-08, Management of Non-conforming Coatings	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.	Substantive	ASTM
High 1,2,4	NFPA 251, Methods of Tests of Fire Resistance of Building Construction and Materials	This standard specifies methods for determining the fire-resistive abilities of building members and assemblies, including tests of: 1. Bearing and nonbearing walls and partitions 2. Columns 3. Floor and roof assemblies 4. Loaded restrained beams 5. Performance of protective membranes in wall, partition, floor, or roof assemblies. Changes are necessary to address the varied advanced reactor design SSCs.	Substantive	NFPA

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
High 1,2,4	ACI 349.1R-07, Reinforced Concrete Design for Thermal Effects	This standard provides general considerations in designing reinforced concrete structures for nuclear power plants subject to thermal effects. Thermal effects are defined to be the exposure of a structure or component thereof to varying temperature at its surface or temperature gradient through its cross section; the resulting response of the exposed structure is a function of its age and moisture content, temperature extreme(s), duration of exposure, and degree of restraint. The terms “force,” “moment,” and “stress” apply and are used in this report where a structure is restrained against thermally induced movements. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	Limited	ACI
High 1,2,4	ACI 349-2013, Nuclear Safety-related Concrete Structures and Structural Members	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.	Limited	ACI
Low 2	ANSI/ANS 3.1-2014, Selection, Qualification and Training of Personnel	This standard provides criteria for the selection, qualification, and training of personnel for nuclear power plants. The qualifications of personnel in the operating organizations appropriate to safe and efficient operation of a nuclear power plant are addressed in terms of the minimum education, experience and training requirements. Limited changes are necessary mostly to address specific language in the	Limited	ANS

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
		standard so as to be applicable to the varied advanced reactor designs.		
Low 2	ANSI/ANS 3.2-2012, Managerial and Administrative Controls	This standard provides requirements and recommendations for managerial and administrative controls to ensure that activities associated with operating a nuclear power plant are carried out without undue risk to the health and safety of the public. This standard provides requirements for implementing managerial and administrative controls consistent with requirements of 10 CFR 50, Appendix B. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	Limited	ANS
Low 2	ANSI/ANS 3.4-1996, Physical and Mental Health Requirements for Reactor Operators and Senior Operators	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.	Limited	ANS
Low 2	ANSI/ANS 3.5-2009, Plant Simulators for Use in Operator Training and Examination	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.	Limited	ANS
Medium 1,2	ANSI/ANS 6.3.1-1987 (R2007), Testing Radiation Shields	This standard defines calculational requirements and discusses measurement techniques for estimates of dose rates near LWR nuclear power plants due to direct and scattered gamma rays from contained sources on-site. On-site locations outside plant buildings and locations in the off-site unrestricted area are considered. The requirements for measurements and data interpretation of measurements are given. The	Limited	ANS

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
		standard includes normal operation and shutdown conditions but does not address accident or normal operational transient conditions. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.		
Low 4	ASME N510-2007, Testing of Air-treatment Systems	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.	Limited	ASME
Low 5	ASME N511-2007, In-service Testing of HVAC Systems	This standard covers the requirements for in-service testing of nuclear safety-related air treatment, heating, ventilating and air conditioning systems in nuclear facilities. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.	Limited	ASME
Low 4	ASTM D4286-08, Coating Contractor Qualifications	This document 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 LWRs. The qualification criteria and requirements address the essential basic capability of a contractor to execute nuclear 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.	Limited	ASTM

Priority	Standard/Guidance	Description of Content and Issues to be Addressed for Advanced Reactors	Type of Change	Organization
Low 4	ASTM D4538-05, Terminology for Protective Coatings	This terminology covers terms and their definitions 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.	Limited	ASTM
Low 5	ASTM D5144-08, Use of Protective Coatings	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.	Limited	ASTM
Low 5	ASTM D7167-05, Monitoring Performance of Safety-related Coatings	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.	Limited	ASTM

5 SCHEDULE

DOE, with stakeholder input, has developed a vision and goal for supporting the development and ultimate deployment of advanced reactor technology as part of a broader federal commitment to energy security, economic prosperity and national security. The vision is that by 2050, advanced reactors will provide a significant and growing component of the nuclear energy mix both domestically and globally. In support of this, a goal has been established that by the early 2030s, at least two non-light water advanced reactor concepts will have reached technical maturity, demonstrated safety and economic benefits, and completed licensing reviews sufficient to allow construction to commence [36]. In the time since DOE's vision and goal was established a greater sense of urgency for being able to utilize advanced reactors has emerged. As a result, efforts are focused on accelerating activities to support non-light water demonstrations in the 2020s which increases the importance of prioritizing and coordinating codes and standards development efforts.

Codes and standards should be developed by the various standards development organizations and endorsed by the NRC in a phased manner based on the prioritization discussed in Section 4 of this report. 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 help assure the efficient design, licensing, construction and operation of advanced reactors.

6 CONCLUSIONS AND RECOMMENDATIONS FOR PATH FORWARD

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. Table 2 provides a prioritized list of the relevant codes and standards that should be addressed to support the deployment of advanced reactors in the U.S.

Consensus standards organizations are voluntary activities and the pace of activities is generally dictated by the availability of the participants. Additionally, often R&D is needed to provide the technical basis for the code development or revision. Coordination, prioritization and funding for these activities can be difficult and even when funding is obtained, the pace of the R&D can be dictated by the level of available funding.

It is recommended that public-private partnerships be formed to provide financial support for those situations where R&D or other special needs is essential to developing or revising codes and standards on a schedule that supports the deployment of advanced reactors.

Figure 1 illustrates a process under which public-private partnerships would engage SDOs to support their activities in developing or revising codes and standards.

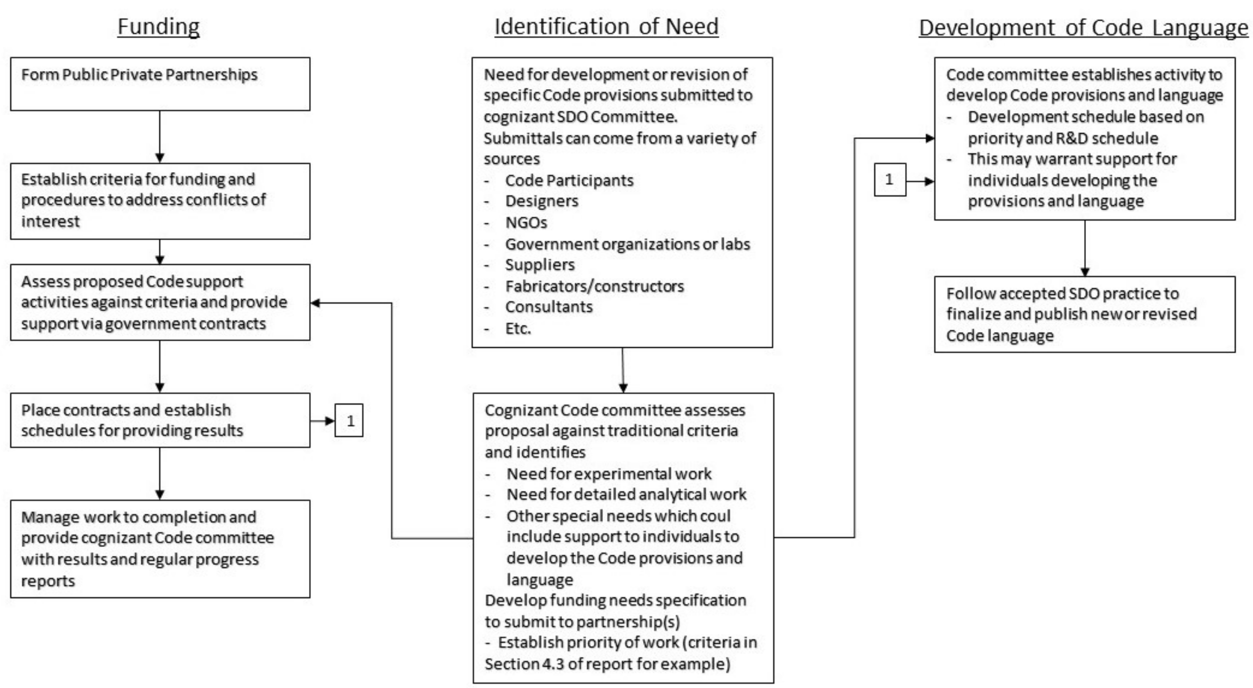

Figure 1. Consensus Standards Funding Process

Consensus codes and standards have been an important contributor to the design, licensing, component fabrication, facility construction, and operation for the existing fleet of operating reactors and for new LWRs. It is anticipated that they will make similar contributions to the deployment of advanced reactor technologies. Providing adequate support to timely development and revision of these standards is expected to facilitate the deployment of these important new technologies.

7 REFERENCES

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APPENDIX A: HIGHEST PRIORITY CODES AND STANDARDS FOR NEAR-TERM DEPLOYMENT

Document	Responsible Organization	Description of Content and Issues to be Addressed
Equivalent to ANSI/ANS 6.4-2006, Concrete for Passive Heat Removal Systems - Irradiation and Thermal Limits	ANS	This new standard would contain methods and data needed to calculate the concrete thickness required for radiation shielding in advanced nuclear power plants. Specific recommendations are to be made regarding radiation attenuation calculations, shielding design, and standards of documentation. The standard would provide guidance to architect-engineers, utilities, and reactor vendors who are responsible for the shielding design of advanced reactor designs.
ANS-30.1-201x Risk-informed Performance-Based Principles and Methods	ANS	This standard describes the objectives of incorporating risk-informed and performance based (RIPB) information into the safety design of advanced reactors. This standard will demonstrate high-level uses of RIPB methods.
ANS-30.2-201x Categorization and classification of SSCs	ANS	This standard provides a single technology neutral categorization and classification process for SSCs for advanced reactors that is, where possible, RIPB. This process will then be used to determine special treatment of SSCs to meet the safety basis. It would provide a complete (e.g., necessary and sufficient) repeatable logical process based upon RIPB objectives.
ASME/ANS RA-S-1.4-2013 PRA for Non-LWRs (trial use)	ASME/ANS	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. This standard has been issued for trial use and has been subjected to many trial use applications. Lessons learned from the trial use and other parallel activities are currently being used by the JCNRM to provide an update for ANS status in 2020.

Document	Responsible Organization	Description of Content and Issues to be Addressed
ANSI/ANS 53.1 Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants, 2011	ANS	This standard provides a process for establishing top-level safety criteria; safety functions; top-level design criteria; licensing-basis events; design-basis accidents; safety classification of systems, structures and components (SSCs); safety analyses; defense-in-depth; and adequate assurance of special treatment requirements for safety-related SSCs throughout the operating life of the plant. The standard does not provide detailed guidance for design; other standards cover that.
ASME BPVC Division 1 and 2, Subsection NCA, Containment Barrier	ASME	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.
New standards equivalent to SFR-DC 73 for sodium leak detection and mitigation (see NRC RG 1.232)	ANS	A new standard is needed to define: 1. the means to detect sodium leakage into inert 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.

Document	Responsible Organization	Description of Content and Issues to be Addressed
ANS 56.2-1984 (ANSI N271-1976), Containment Isolation for Fluid Systems	ANS	This standard specifies minimum design, actuation, testing, and maintenance requirements for the containment isolation of fluid systems after a LOCA in LWRs. These fluid systems penetrate the primary containment and include piping systems (including instrumentation and control) for all fluids entering or leaving the containment. Electrical systems are not included. The provisions for containment isolation impose additional requirements which are not required for the fluid system function. Changes are necessary to address the varied advanced reactor designs, coolants as well as functional containment concepts.
ASME AG-1-2009, Air and Gas Treatment	ASME	This standard considers the design, fabrication, inspection, and testing of air cleaning and conditioning components and appurtenances used in safety-related systems in nuclear facilities. Changes are necessary to address the varied advanced reactor designs as well as functional containment concepts.
ASME BPVC Section III, Construction of plant components	ASME	This Section contains requirements for the material, design, fabrication, and examination of supports which are intended to conform to the requirements for Classes 1, 2, 3, and metal containment construction. Nuclear power plant supports for which rules are specified in this Subsection are those metal supports which are designed to transmit loads from the pressure retaining barrier of the component or piping to the load carrying building structure. Changes are necessary to address the varied advanced reactor designs as well as functional containment concepts.
New Standards for Advanced Manufacturing Techniques, including Additive manufacturing	ASME	Additive manufacturing offers a new paradigm for engineering design and manufacturing by enabling unique macro-structural design of components and micro/nano-structural design of the material. There is a need to develop and publish

Document	Responsible Organization	Description of Content and Issues to be Addressed
NOTE: This standard was not included in either the ORNL report or the ANS/RES summary.		standards applicable to the manufacturing of components suitable for use in safety-related and other plant applications. These standards also need to be applicable to manufacturing replacement parts for operating plants and to manufacturing components for advanced reactors, addressing the higher temperature and non-light water operating environments.
ASME BPVC Section III Division 1, Subsection NE and 2, Containment	ASME	Subsection NE pertains to Class MC components while Division 2 pertains to concrete reactor vessels and containments. Changes are necessary to address the varied advanced reactor designs as well as functional containment concepts. Updates should consider including steel-plate composite (SC) construction for containment structures. Note that AISC N690 allows for SC walls for safety-related structures other than containment.
ASME BPVC Section III, Division 5, High Temperature Reactors	ASME	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.
ASME N509-2002, Air Cleaning Units and Components	ASME	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-

Document	Responsible Organization	Description of Content and Issues to be Addressed
		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.
ASME QME-1-2007, Qualification of Active Mechanical Components	ASME	This standard describes the requirements and guideline for qualifying active mechanical equipment, such as pumps, valves, and dynamic restraints, used in nuclear facilities. The requirements and guidelines presented include the principles, procedures, and methods of qualification. Changes are necessary to address the varied advanced reactor design components and high temperature applications.
NFPA 251, Methods of Tests of Fire Resistance of Building Construction and Materials	NFPA	This standard specifies methods for determining the fire-resistive abilities of building members and assemblies, including tests of: 1. Bearing and nonbearing walls and partitions 2. Columns 3. Floor and roof assemblies 4. Loaded restrained beams 5. Performance of protective membranes in wall, partition, floor, or roof assemblies. Changes are necessary to address the varied advanced reactor design SSCs.
ACI 349.1R-07, Reinforced Concrete Design for Thermal Effects	ACI	This standard provides general considerations in designing reinforced concrete structures for nuclear power plants subject to thermal effects. Thermal effects are defined to be the exposure of a structure or component thereof to varying temperature at its surface or temperature gradient through its cross section; the

Document	Responsible Organization	Description of Content and Issues to be Addressed
		resulting response of the exposed structure is a function of its age and moisture content, temperature extreme(s), duration of exposure, and degree of restraint. The terms “force,” “moment,” and “stress” apply and are used in this report where a structure is restrained against thermally induced movements. Limited changes are necessary mostly to address specific language in the standard so as to be applicable to the varied advanced reactor designs.
ACI 349-2013, Nuclear Safety-related Concrete Structures and Structural Members	ACI	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.