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SMALL MODULAR REACTOR (SMR) ROADMAP

ASME STANDARDS TECHNOLOGY, LLC

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FOREWORD

This American Society of Mechanical Engineers (ASME) SMR Roadmap was funded and developed by ASME. Advanced Systems Technology and Management, Inc. (AdSTM) engaged various stakeholders -- particularly Small Modular Reactor (SMR) vendors, the U.S. Nuclear Regulatory Commission (NRC), and ASME Code Committees -- to solicit information that would be used as a basis for the conclusions and recommendations in this SMR Roadmap.

ASME recognizes that critical SMR design information was provided by B&W mPower, NuScale, and Holtec to support the analysis of the ASME Nuclear Codes and Standards in this SMR Roadmap. Finally, ASME acknowledges the NRC staff for its informal input on how the existing Codes and Standards might present potential licensing issues for SMR designs and configurations.

The authors acknowledge, with deep appreciation, the activities of the Peer Review Group (PRG) that consist of Richard Porco, Ralph Hill and Rick Swayne, and the ASME staff and volunteers who have provided valuable technical input, advice and assistance with review of, commenting on, and editing of, this document.

Established in 1880, ASME is a professional not-for-profit organization with more than 135,000 members and volunteers promoting the art, science and practice of mechanical and multidisciplinary engineering and allied sciences. ASME develops Codes and Standards that enhance public safety, and provides lifelong learning and technical exchange opportunities benefiting the engineering and technology community. Visit www.asme.org for more information.

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ABBREVIATIONS AND ACRONYMS

ACI	American Concrete Institute
ANS	American Nuclear Society
ANS	American National Standards
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASME ST-LLC	ASME Standards Technology, LLC
ASTM	American Society for Testing and Materials International
AWS	American Welding Society
BNCS	Board on Nuclear Codes and Standards
BPV	Boiler and Pressure Vessel
BPVC	Boiler & Pressure Vessel Code
CFR	Code of Federal Regulations
CNF	Cranes for Nuclear Facilities
CNRM	Committee on Nuclear Risk Management
COL	Construction and Operating License
CONAGT	Committee on Nuclear Air and Gas Treatment
CORDEL	Cooperation in Reactor Design Evaluation and Licensing
DSRS	Design-Specific Review Standard
DOE	Department of Energy
ECCS	Emergency Core Cooling System
FOAK	First of a Kind
FMEA	Failure Modes and Effects Analysis
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronics Engineers
iPWR	Integral Pressurized Water Reactor
ISI	In-Service Inspection
IST	In-Service Testing
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
IWE	Inservice Welding Examination
LWR	Light-Water-Reactor
MWe	Megawatt Electric
NESCC	Nuclear Energy Standards Coordination Collaborative
NIST	National Institute of Standards Technology
NPS	Nominal Pipe Size
NQA	Nuclear Quality Assurance
NRC	U.S. Nuclear Regulatory Commission
OM Code	ASME Operation and Maintenance of Nuclear Power Plants Code
PRA	Probabilistic Risk Assessment
PRG	Peer Review Group
QME	Qualification of Mechanical Equipment
RIM	Reliability and Integrity Management
RTNSS	Regulatory Treatment of Non-Safety Systems
SDO	Standards Development Organization
Section III	BPVC, Section III-Rules for Construction of Nuclear Facility Components
Section XI	BPVC, Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components
SMR	Small Modular Reactor
SRP	Standard Review Plan
SSC	Structures, Systems and Components

ABSTRACT

Small Modular Reactor (SMR) designs are currently scheduled for NRC licensing reviews in the next 1-2 years. Although there is no definitive schedule for deployment of SMRs, it is currently anticipated that the first commercial deployment will be in approximately the 2022-2024 timeframe. With anticipated domestic and international interest in SMR designs and the near-term licensing schedule, ASME developed this SMR Roadmap to engage all stakeholders -- including SMR vendors, NRC and ASME Code Committees -- to identify any potential ASME Codes and Standards issues that may affect the effective and timely SMR licensing. Generally, the SMR vendors believe that their designs can comply with current ASME Codes and Standards because they are based on existing and licensed light-water-reactor (LWR) technology. However, this SMR Roadmap discusses Code areas in the ASME Boiler & Pressure Vessel Code (BPVC) and ASME Operation and Maintenance of Nuclear Power Plants Code (OM Code) where potential differences between vendors and NRC regarding the proper interpretation and application these Code requirements may present licensing issues. Particularly, this SMR Roadmap discusses potential issues in BPVC, Section III-Rules for Construction of Nuclear Facility Components (Section III), BPVC, Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components (Section XI), and OM Code that may result from certain unique features of the SMR designs. Potential issues that are identified include:

Section III:

- The acceptability of the Section III fracture toughness requirements exemption (paragraph NB-2311) for small parts used for Class 1 components should be reviewed by the Section III for applicability to SMR designs.
- The rules of Subsection NE for the construction of metal containment vessels (Class MC) may need to be revisited for applicability to certain SMR designs.

Section XI:

- The Section XI Inservice Inspection (ISI) exemption for small Class 1 components and piping should be evaluated by Section XI for their applicability to SMRs.
- The inspection of SMR reactor vessels may be problematic in some designs due to compactness of design and limited accessibility.
- The soon-to-be-published (2015) Section XI, Division 2, “Reliability and Integrity Management” (RIM) program may benefit SMR ISI programs. However, reliance on the Division 2 methodology might cause an initial delay in the licensing process since it is a new approach to ISI not yet approved by the NRC.
- SMR pressure vessels clad on both sides may present issues for application of Section XI, Subsections IWE and NB.

OM Code:

- Periodic testing requirements of the OM Code presents an issue to the (a) NuScale design since opening the reactor vessel valves would produce a loss-of-coolant accident, and (b) mPower design which has an extended fuel cycle.

To address these potential issues, this SMR Roadmap recommends that the vendors more thoroughly evaluate their designs against both BPVC and OM Code, and NRC requirements, and engage the ASME Code Committees early in the process to develop appropriate requirements if issues need resolution. This would provide a technical basis, developed through ASME’s American National Standards Institute (ANSI)-approved Code consensus process that could be used to support their positions when engaging with the NRC during the design certification licensing process. In addition, some of these potential issues can be addressed through the development of ASME Code Cases. Currently, Code Cases are being developed that will address SMR extended fuel cycle issues.

1 SMR INTRODUCTION AND PURPOSE

NRC licensing reviews of several SMR designs will commence within the next 1-2 years upon the filing of Design Certification applications. Pre-application reviews between vendors of SMR designs and the NRC have been ongoing for over 5 years. While these near-term SMR designs are based on proven LWR technology, several new design features and systems may present challenges to their licensing or deployment. Critical to effective and timely licensing, SMR vendors and designers must appropriately interpret and apply ASME Codes and Standards that are NRC licensing requirements.

This SMR Roadmap project was initiated to start a dialogue and interactions between vendors, NRC, ASME and other stakeholders to determine how ASME Nuclear Codes and Standards will be interpreted and applied. These interactions will provide information, insights and strategies to facilitate SMR design development, NRC licensing, and ultimately commercial deployment of these new reactor technologies. This SMR Roadmap identifies critical technology and process issues that are covered by ASME Codes and Standards and may present barriers or unique challenges to effective NRC licensing. This SMR Roadmap identifies those specific ASME Nuclear Codes and Standards that are included by reference in the NRC's regulations and are therefore legally-binding regulatory requirements. This SMR Roadmap discusses areas in these Codes and Standards that might present licensing challenges to all the SMR designs under consideration, as well as unique design-specific challenges. Strategies to address identified challenges for NRC licensing are discussed. This SMR Roadmap does not identify any research and development needed at this time to support Codes or Standards modifications. Included in these strategies is the option for license applicants to propose alternatives to ASME Codes and Standards' requirements to the NRC for their review and approval.

This SMR Roadmap identifies the following potential issues that should be addressed and resolved for effective licensing:

(a) Section III

(1) SMR suppliers have indicated that they are able to comply with the current Section III requirements. Section III, however, was written for large LWRs. Therefore, it may be determined that certain areas of Section III may not be applicable to SMRs, most notably the exemptions in Section III for small components. It is recommended that Section III articulate the bases for these exemptions in order that they can be properly applied by component designers. This also would be useful to the regulator.

(b) Section XI

(1) The requirements in Section XI for Inservice Welding Examination (IWE) for examination of metallic containments were developed for large LWRs and are not applicable to certain of the SMR designs, such as the NuScale design. It is recommended that Section XI consider developing IWE requirements for other containment designs if they are requested by the SMR vendors as their designs are more completely developed.

(2) SMR designs may have unique examination requirements which may require additions to Section XI. It is recommended that SMR vendors engage Section XI to develop examination requirements for unique SMR containment designs.

(3) IWB-1220 provides rules for exempting certain small components from volumetric, surface and visual examination. These exemptions may not be applicable to certain SMR designs. It is recommended that Section XI consider revisiting these examination exemptions for applicability to SMR designs.

(4) Section XI, IWB-2500 contains other exemptions for the examination of small components. It is recommended that Section XI consider revisiting these examination exemptions for applicability to SMR designs.

Another purpose of this SMR Roadmap is to provide a framework to address other nuclear code or standards issues that might be (a) applicable to other SMR or advanced reactor designs that are not considered "near-term" licensing challenges, and (b) the responsibility of other Standards Development Organizations (SDOs) such as ANS, IEEE, AWS, ASTM, etc. The process used to develop this SMR Roadmap can be used as a model or template to expand the scope of this SMR Roadmap to address, for example, SMR designs from countries outside the U.S., or be used as a model to develop other SMR or advanced reactor roadmaps. As with all roadmaps, information and interactions between all stakeholders are critical to understanding the issues and providing a framework and strategy for resolution. Since nuclear energy remains a critical component of the U.S. energy policy, nuclear technology and licensing issues must be timely addressed and resolved by all stakeholders to properly advance the nuclear energy option for domestic and international applications.

Finally, to the extent that this SMR Roadmap discusses how these new SMR designs interpret and apply ASME Codes and Standards for NRC licensing reviews, it will provide information and insights as to how ASME Nuclear Codes and Standards can be used for licensing purposes by other countries where these Codes and Standards are adopted and referenced to support nuclear power plant designs and licensing.

2 ASME NUCLEAR CODES AND STANDARDS – INTRODUCTION AND ORGANIZATION

ASME is the leading international developer of Codes and Standards associated with the art, science, and practice of mechanical engineering. Starting with the first issuance of its legendary Boiler & Pressure Vessel Code in 1915, ASME's Codes and Standards have grown to nearly 600 offerings currently in print. These international offerings cover a breadth of topics, including pressure technology, nuclear plants, elevators and escalators, construction, engineering design, standardization, and performance testing. Dedicated volunteers — engineers, scientists, government officials, and others — contribute their technical expertise to protect public safety, while reflecting best practices of industry. The results of their efforts are being used in over 100 nations; thus setting the standard for the development of international Codes and Standards.

The first BPVC (1914 edition) published in 1915 was one book, 114 pages long. The 2010 edition of the BPVC is more than 16,000 pages and contains 12 Sections. The 12 Sections of the BPVC either provide the rules for constructing pressure-retaining components, or are “service codes”, such as Materials (Section II, Parts A through D), Nondestructive Examination (Section V), and Welding and Brazing Qualifications (Section IX) or, provide guidelines for care and operation of boilers. Sections VI and VII provide guidelines for the care and operation of power and heating boilers. (Section VI: Recommended Rules for the Care and Operation of Heating Boilers Section VII: Recommended Guidelines for the Care of Power Boilers). The so-called service codes are referenced by both nuclear and nonnuclear Sections of the BPVC. Code Cases provide rules that permit the use of materials and alternative methods of construction that are not covered by existing BPVC rules.

ASME develops and revises standards based on market needs through a consensus process whose meetings dealing with standards-related actions are open to all members of the public. ASME consensus committees are comprised of volunteer subject matter experts from a diverse range of interests, including manufacturers, users, government, and general interest. ASME standards and subsequent revisions are based upon review of reliable technical data by the consensus committee and its sub-tier committees.
The ASME is accredited by the ANSI. ANSI facilitates the development of American National Standards (ANS) by accrediting the procedures of SDOs. Accreditation by ANSI signifies that the procedures used by the standards body in connection with the development of American National Standards meets ANSI’s essential requirements for openness, balance, consensus and due process.

The ASME process includes a broad public review for all of its standards actions. Any interested member of the general public may review and comment on proposed ASME standards or revisions, as well as initiate an appeal based on previously submitted concerns. ASME's voluntary standards may be adopted by jurisdictional authorities as a means of complying with their governing regulatory requirements.

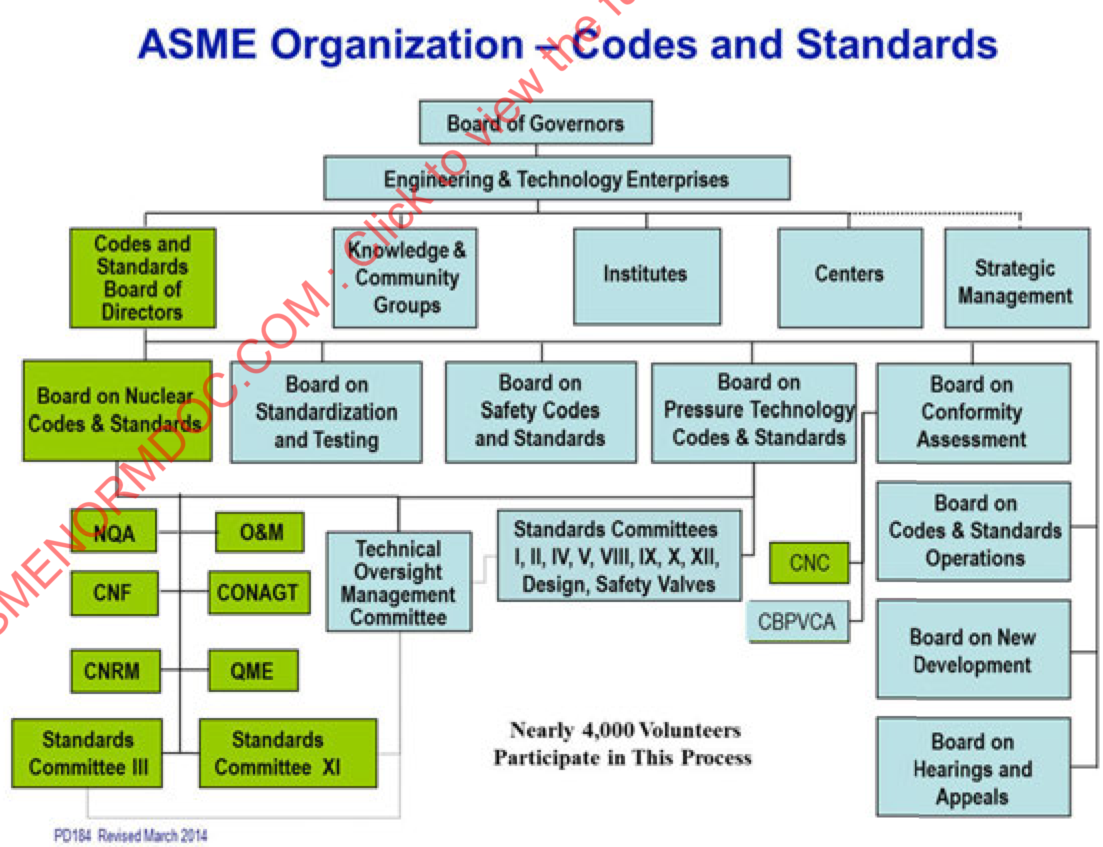
The NRC, the federal agency responsible for issuing construction permits, operating licenses, or combined (construction and operating) licenses for new nuclear power plants, requires conformance with certain ASME Codes and Standards in its regulations. Therefore, to obtain a license to construct or operate a nuclear power plant, a plant owner and its subcontractors designing and supplying nuclear components must meet the requirements of these codes. The NRC incorporates by reference certain industry Codes and Standards including Section III and Section XI of the BPVC and the OM Code into its regulations.

Section III of the BPVC, “Rules for Construction of Nuclear Facility Components,” provides rules for the materials selection, design, fabrication, installation, examination, and testing of nuclear components. Section III was first published in 1964 to address the larger and more complex designs of the emerging commercial nuclear power industry.

To address the safe operation of nuclear reactors, ASME developed and published Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," and the OM Code, "Code for Operation and Maintenance of Nuclear Power Plants," to ensure that continued safe operation is maintained over the life of the plant. These two codes are also required by NRC regulations, making the periodic inspection and testing of components and meeting specified acceptance standards a federal requirement for maintaining a license to continue operation. This gives the NRC and the public a level of confidence that any degradation of the plant during the period of operation will be detected early, adequately corrected, and will not reduce safety below an acceptable level. These and other standards incorporated by reference in the NRC's regulations are treated like any other properly issued regulation and have the force of law.

In addition to Sections III and XI of the BPVC and the OM Code, ASME has other nuclear-related Codes and Standards. All of the ASME nuclear-related Codes and Standards committees report to the ASME Board on Nuclear Codes and Standards (BNCS). These committees include: Nuclear Quality Assurance (NQA), Cranes for Nuclear Facilities (CNF), Committee on Nuclear Air and Gas Treatment (CONAGT), Committee on Nuclear Risk Management (CNRM), and Qualification of Mechanical Equipment (QME). Under each of these committees are one or more standards. For example, there are two standards for nuclear cranes. NUM-1, "Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type)" and NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)." The Figure 1 illustrates the ASME Codes and Standards Organization. The Committees highlighted in Green are nuclear-related.

Figure 2-1: ASME Codes and Standards Organization (Committees Highlighted in Green are Nuclear-Related)



3 STAKEHOLDERS FOR ROADMAP OUTREACH

To support the information and insights in this SMR Roadmap, the following entities were engaged in discussions:

- NuScale Power
- B&W mPower
- Holtec
- U.S. Nuclear Regulatory Commission
- Members of ASME Board on Nuclear Codes and Standards (BNCS) Committee provided extensive information.

4 SMR TECHNOLOGY AND LICENSING OVERVIEW

4.1 Introduction to SMR Technology and Licensing

The concept and technology of SMRs is not new. Some of the earliest concepts were developed in the 1970s for merchant ship propulsion and industrial process heat applications. Others were derived from early reactor designs or concepts from U.S. Department of Energy (DOE) national laboratories or academia. Concepts keep evolving around the world. The International Atomic Energy Agency (IAEA) reports that more than 50 SMR concepts were under development during the past fifty years [1]. The IAEA uses the abbreviation SMRs to refer to small- and medium-sized reactors. Small reactors are considered by the IAEA as those having capacities of <300 MWe and medium reactors are considered as those having capacities of < 700 MWe. In the U.S., SMR is an abbreviation for Small Modular Reactors typically having capacities of < 300 MWe.

Today's SMR design concepts are driven both by technology and financial considerations. Advanced technical considerations include: enhanced safety; improved security; and increased flexibility in siting and application. Financial considerations that favor new SMRs include: lower upfront capital costs; greater quality and consistency by production in factory settings; an easier ability to deploy additional modules to meet projected electrical demand; lower construction cost and schedule by "plug and play" fabrication at the site; and better compatibility with local electrical grid infrastructure. After a large earthquake and resulting tsunami in Japan destroyed 4 of the 6 units of the Fukushima Daiichi nuclear plant in 2011, the promise of enhanced safety and plant resilience of SMRs has become a critical technical and financial consideration in all countries seeking nuclear power as an energy option. The use of SMRs is particularly important for countries seeking a nuclear power option that do not have a robust electric grid where the installation of a large nuclear power plant could result in potential grid instabilities if the nuclear power plant suddenly shuts down and its large source of power is removed from the grid.

In response to the international interest in SMRs, vendors with new designs have begun interaction with the NRC regarding regulatory licensing of their designs. The licensing and regulation of SMRs in the U.S. will be done by the NRC. SMR licensing in other countries will be performed by the regulatory bodies having jurisdiction in that country. Regardless of licensing, the commercial deployment of SMRs will be a global enterprise that requires interactions and collaborations between countries. Vendors will apply for licensing approval of their designs in the country of design origin (i.e., where the vendor is located). SMR vendors external to the U.S. may also wish to obtain an NRC license for their designs to enhance marketability in the U.S. and other countries. The approved SMR designs will then be manufactured largely in the country of origin, marketed globally, and licensed for operation in the country of deployment. In the U.S., SMR vendors will apply to the NRC for approval of their designs under the provisions of Title 10 of the Code of Federal Regulations (10 CFR) either Part 50 or Part 52. These licensing processes and their Codes and Standards implications are a substantial consideration in this SMR Roadmap.

If a U.S. SMR design receives regulatory approval from the NRC, it will likely be marketed and deployed both in the U.S. and internationally. Internationally, a country's specific regulatory requirements and the specific information required to be provided by the SMR vendor to obtain regulatory approval for SMR systems and components might present a challenge to the host countries' regulatory authorities in assuring that their regulatory requirements are met. The licensing/regulatory authority must be able to license and regulate SMRs in a manner that adequately assures all safety, environmental, regulatory and policy issues are addressed and resolved, particularly in the post-Fukushima environment. Importantly, the licensing authority must be able to assess the enhanced safety characteristics of SMR designs to support approval or certification of these advanced reactor technologies and their subsequent licensing. SMR designs having enhanced safety features and significantly reduced risk to the public may afford the licensing authority the ability to use a risk-informed or a graded approach to review and approve SMR designs. A graded approach to licensing when used in this context permits the licensing authority to alter the scope and depth of its licensing review based on local safety or environmental considerations; or unique features of the SMR design under consideration. If the SMR design has been previously approved by a competent regulatory authority such as the NRC, the host regulatory authority might alter the scope and depth of its licensing review based on that prior regulatory review and approval.

While this SMR Roadmap only focuses on SMR designs being developed in the U.S., reactor developers world-wide are seeking to develop SMR designs to meet the large anticipated market demand. Designs are being developed by both traditional reactor vendors and new start-up companies. Some SMR concepts are being developed by research organizations, typically characterized by advanced fuels, materials and coolants, and often with unique first-of-a-kind design features that may require decades to develop and qualify for commercial application. All of the SMRs within the scope of this SMR Roadmap are actively being developed by companies in the U.S. and these companies have had some level of engagement with potential customers and the NRC. These designs are considered to have the potential to be licensed and deployed within the next 10-15 years, depending on developer resource commitment and customer interest. Because of the near-term commercial potential, it is important to address and resolve, in a timely manner, all Code and Standards issues that may impede licensing or commercial development and deployment of SMRs in both domestic and international markets.

4.2 Near-Term Commercial SMRs

SMRs are defined broadly to include a range of technologies, particularly a range of diverse fuels and coolants. U.S. SMR reactor technology currently chosen for NRC licensing and near-term commercial deployment is based on proven LWR technology. Most of these SMRs will be integral pressurized water reactors (iPWRs) that contain both the reactor and the steam generator in the same containment vessel.[1] LWRs have a well-established framework of regulatory requirements, a technical basis for these requirements, and supporting regulatory guidance that provides acceptable approaches for meeting NRC requirements. The NRC uses a Standard Review Plan (SRP), NUREG-0800, to review licensing applications for these reactor designs. The SRP is a guidance document used by the NRC Staff that provides regulatory consistency in the review of new applications submitted for licensing. Since the SMR designs are different than the current U.S. operating fleet of large LWRs, some of the guidance in the SRP may not be applicable to SMRs.

NUREG-0800 was revised in January 2014 to provide general review guidance for SMRs. This revision incorporates the review philosophy and framework for the staff's review of light-water SMRs licensing applications filed under 10 CFR Part 52, and how the staff will use risk-insights (described later). The NRC will require design-specific review standards (DSRSs) for the licensing of each SMR. These DSRSs are under development by the NRC and SMR vendors and it is expected that the mPower DSRS will be published in mid-2014. This DSRS will be used to support the NRC review of the mPower SMR licensing application which currently is scheduled to be submitted in 2015. Further, some of the regulatory required criteria in Section III, Section XI and OM Code may not be applicable or acceptable to the NRC for SMR designs. This is discussed further below.

A description of these SMRs and their basic characteristics are contained in Appendix A. Additionally, the NRC has a well-established set of validated analytical codes for LWR technologies that may be applicable to SMRs and a well-established infrastructure for conducting safety research needed to support its

[1] The Holtec SMR design is not technically an iPWR, but the design is considered by Holtec to be an "integrated" PWR because of its unique coupling of the reactor to the steam generators. Other SMRs are being developed in the U.S. for licensing and deployment such as the GE PRISM and the Gen4 reactors. The PRISM is a sodium-cooled fast reactor and the Gen4 is a lead-bismuth cooled reactor. These reactors are not included in the scope of this Roadmap because they are not scheduled for near-term licensing or commercialization.

independent safety review of a nuclear power plant design and the technical adequacy of a licensing application. It should be emphasized that the SMRs, particularly the iPWRs, will be subject to the same NRC licensing processes as other large LWRs. The safety requirements and standards imposed on SMRs by the regulator will likely be similar to those imposed on the new large LWRs, such as the AP-1000, with possible modification to address the SMR designs and enhanced safety features. The result will be no diminution of public health and safety.

4.3 NRC Licensing of SMRs

New SMRs can be licensed by NRC under either of two existing regulatory approaches. The first approach is the traditional "two-step" process described in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", which requires first a construction permit (CP) and then a separate operating license (OL). The second approach is the new "one-step" licensing process described in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," which incorporates a combined construction and operating license (COL) approach to licensing. It should be noted that the both the SMR vendor B&W and the proposed license applicant TVA, have chosen the two-step 10 CFR Part 50 process for licensing of the mPower SMR at the Clinch River, Tennessee site. This two-step process was chosen by TVA and approved by the NRC for this first-of-a-kind (FOAK) design to reduce the risk of delay in obtaining a final design certification under the 10 CFR Part 52 process. It is believed that B&W, the vendor for the mPower design, will apply for design certification under 10 CFR Part 52 subsequent to TVA being granted an operating license under the NRC's 10 CFR Part 50 process. It is anticipated, based on discussions with the other current SMR vendors that all other U.S. applicants for FOAK SMRs will adhere to the 10 CFR Part 52 NRC licensing process.

Key to the licensing of any new reactor design, including SMRs, is the approach that the regulatory authority will utilize in assessing the potential effects on public health and safety for the design of the reactor and its associated safety systems. Regardless of the licensing process chosen, the licensing authority will need to consider the relative merits of a deterministic versus a risk-informed performance-based approach to assess and approve the safety case for SMRs. It may initially be a challenge to use a risk-informed, performance-based licensing approach, since there is no similar history of operating experience for these new SMR designs which exists for large LWRs. However, iPWRs and large LWRs share some basic similarities in technology which could allow the use of risk-informed insights in some aspects of SMR design.

The NRC and the U.S. nuclear industry see significant advantages in the 10 CFR Part 52 process and it is the preferred licensing process for all applications that are not FOAK designs. Advantages to this process are:

- Standard Certified Design, approved by NRC rulemaking, provides regulatory stability.
- Early identification and resolution of all safety issues as part of the design certification process.
- Standard Design Certification is valid for 15 years from the date of issuance
- Public and transparent licensing process
- Predictable and efficient licensing process that reduces financial risk

4.3.1 Use of Deterministic or Risk-Informed Approaches for Licensing SMRs

New SMR LWR designs offer significantly enhanced safety, security and simplicity of design. Both the industry and the NRC recognize that many of the existing technical requirements would be applicable to these new designs. Traditionally, LWRs were licensed using a deterministic process with supporting analyses to prove the safety case and establish the licensing basis. However, with the significant improvements in safety design, the NRC may permit greater use of Probabilistic Risk Assessment (PRA) techniques and risk insights to establish the licensing basis for SMRs. In 2010, the NRC directed the staff to develop a design-specific, risk-informed review plan for each SMR [2]. The staff responded to this SRM in SECY-11-0024, "Use of Risk-Insights to Enhance the Safety Focus of Small Modular Reactor Reviews", dated February 18, 2011. The NRC approved the risk-informed framework and it is now incorporated in the SRP, NUREG-800 revision for SMRs noted above.

SMR license applicants that agree to participate in the risk-informed and integrated review framework described in the SRP, must develop and use design-specific PRAs to support their DSRS and licensing basis. The 10 CFR Part 52 licensing process requirements include the submittal of a description of the design specific probabilistic risk assessment and its results, and the information necessary to demonstrate how operating experience insights have been incorporated into the plant design.

While the NRC has indicated that it will permit greater use of risk-informed insights obtained from a PRA to support and establish a licensing basis, the use of the PRA would be commensurate with the quality and completeness of the design and PRA results presented with the application. Depending on the quality and completeness, the NRC might use the PRA and risk-insights to complement a deterministic analysis to establish a licensing basis (including the selection of licensing basis events). Or, it might rely more heavily on the PRA and use a deterministic process and supporting analyses to complement the PRA for some aspects of design and licensing basis. In the post-Fukushima environment, the quality of a design-specific PRA coupled with the use of deterministic acceptance criteria and analysis becomes crucial for the licensing of SMRs that either seek relief from traditional LWR safety requirements, or are used to support revised licensing requirements. The use of PRA risk-informed insights for licensing of SMR LWR designs will largely depend on the quality and completeness of the PRA described in the 10 CFR Part 52 application and the similarity of the SMR design to existing LWR designs. PRA results might also support SMR unique interpretations or applications of ASME Codes and Standards as discussed in this SMR Roadmap.

4.3.2 Specific NRC Licensing Requirements Referencing ASME

NRC regulation 10 CFR § 50.55a(b) incorporates requirements of Section III, Division 1, and Section XI, Division 1 of the BPVC ; and the OM Code. These Codes are legally binding, subject to specified limitations and modifications, on entities that have applied for or have been granted Combined Licenses, Construction Permits or Operating Licenses.

NRC provisions in 10 CFR § 50.55a specify the following ASME Code or Standard requirements:

- Section 50.55a(c), "Reactor Coolant Pressure Boundary," requires, in part, that components of the reactor coolant pressure boundary be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III, "Rules for Construction of Nuclear Power Plant Components," of the BPVC or equivalent quality standards.
- Section 50.55a(d), "Quality Group B" components must meet the requirements for Class 2 Components in Section III of the BPVC.
    - Guidance for quality group classifications of components which are to be included in the safety analysis reports pursuant to § 50.34(a) and § 50.34(b) may be found in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radiological-Waste-Containing Components of Nuclear Power Plants," and in Section 3.2.2 of NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants."
- Section 50.55a(e), "Quality Group C" components must meet the requirements for Class 3 components in Section III of the BPVC.
- Section 50.55a(f), "Inservice Testing Requirements," requires, in part, that Class 1, 2, and 3 components and their supports meet the requirements of the OM Code or equivalent quality standards.

- Section 50.55a(g), "Inservice Inspection Requirements," requires, in part, that Class 1, 2, 3, MC (metal containment), and CC (concrete containment) components and their supports meet the requirements in Section XI of the BPVC or equivalent quality standards.

Licensees must demonstrate and document how they will meet the above ASME and other NRC licensing requirements. The NRC issues a range of regulatory guides and other guidance to describe what methods the NRC staff considers acceptable for use in implementing specific parts of NRC regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs to review applications for permits and licenses. Regulatory guides are not substitutes for regulations and compliance with them is not required. However, regulatory guides are one way, acceptable to the NRC staff, to meet the underlying NRC regulation. Regulatory guides also may endorse, or endorse with exceptions, industry standards prepared by SDOs such as ASME. Typically commitments to a Regulatory Guide are referred to as Regulatory Commitments. Regulatory Commitments are not legally-binding and can be changed by the licensee or applicant.

Three regulatory guides are cited in 10 CFR § 50.55a: Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III"; Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"; and Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, OM Code". These regulatory guides when incorporated into 10 CFR § 50.55a establish pre-approved alternatives to the ASME requirements in Section III, Section XI and the OM Code. The Code Cases contained in these three regulatory guides are not required to be used as they are alternatives to the underlying Code. However, if an applicant or licensee commits to use a Code Case approved in one these regulatory guides, it then becomes legally binding on them.

The three regulatory guides incorporated in 10 CFR § 50.55a are atypical in that if Code Cases are committed to they become legally binding on the licensee or applicant as they represent an approved alternative to the legally-binding requirements of the underlying ASME Code. However, a license can propose methods and solutions that differ from those set forth in these regulatory guides. This increases the burden of proof. These alternative methods or solutions will be deemed acceptable if they provide an adequate basis for the findings required for the issuance of the proposed license application. Alternative methods to meet a requirement are different than seeking relief from the specific requirement as discussed below.

4.3.3 Relief from NRC Requirements

An NRC license applicant or licensee may apply for an exemption from a requirement cited in the Code of Federal Regulations when it finds that it is not possible or practical to meet part of or all of the requirements of a particular rule. Provisions governing a request for exemption from requirements by an NRC applicant are found in 10 CFR § 50.12 or 10 CFR § 52.7. The NRC staff must review exemption requests and will either approve or deny each one.

Guidance for development and review of exemption requests are provided in NRR Office Instruction LIC-103, *Requests for Exemption from NRC Regulations*. In essence, the applicant for an exemption must demonstrate that the exemption requested is authorized by law (that is, not otherwise prohibited), will not present an undue risk to public health and safety, and is consistent with common defense and security. Additionally, "special circumstances" as defined in 10 CFR § 50.12(a)(2) must exist to support granting the exemption. Particularly important for SMR license applicants seeking exemption from 10 CFR Part 50 requirements is that due to unique design features the applicants may be able to demonstrate that special circumstances exist such that:

> *"…(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule; or*
> *(iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated; or*
> *(iv) The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption…."*

SMR license applicants could argue "special circumstances" for an exemption by demonstrating that the substantially increased safety of SMR designs and operations were not contemplated by the requirements. For example, an exemption could be granted in circumstances where the application of the requirement would not serve the underlying safety purpose of the regulation or is not necessary to achieve the underlying purpose of the rule.

Generally speaking, the exemption process is a lengthy one and requires substantial justification and documentation to be submitted by the applicant or licensee. The NRC sets a high standard for the granting of exemptions to its regulations. Only the Commission may grant exemptions to NRC regulations.

10 CFR § 50.55a, the NRC regulation that requires the use of Sections III and XI of the BPVC and the OM Code, has an exemption process that differs from that in 10 CFR § 50.12 in that exemptions to Code requirements do not require Commission approval and can be granted by the Director, Office of Nuclear Reactor Regulation or the Director, Office of New Reactors. This process is specified in 10 CFR § 50.55a(a)(3). To obtain an exemption to Code requirements the applicant or licensee must demonstrate that:

> *"(i) The proposed alternatives would provide an acceptable level of quality and safety; or*
> *(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."*

Applicants for a NRC license or holders of an NRC license may request the use of alternatives to the ASME Code and other Standards incorporated by reference in 10 CFR § 50.55a. These are commonly referred to as "relief requests". The NRC staff must review and approve relief requests by issuing a safety evaluation. As noted above, the use of an alternative requirement requires approval by an Office Director. Procedures and guidance for development and review of relief requests are provided in NRR Office Instruction LIC-102, *Relief Request Reviews*.

NRC has granted exemptions to requirements both for the initial licensing of plants and during their operation. Exemptions for any given plant have been infrequent, particularly in recent years since changes to certain regulations have eliminated some of the more frequent occasions for exemptions. While exemptions have been granted, the NRC has sought to avoid the excessive use of 10 CFR § 50.12 and has preferred to revise the underlying regulation or develop more effective implementing guidance as the preferred alternative.

As an example, an exemption was granted from 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17, "Electric Power Systems" for the AP-1000 design. The underlying purpose of the requirement of GDC 17 to provide two offsite power sources to the plant is to ensure sufficient power to accomplish safety functions. The AP-1000 design does not rely on power from the offsite system to accomplish safety function. Therefore, the underlying purpose of the rule is met without the need for two independent offsite circuits. The staff concluded that special circumstances exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose of having two offsite power sources. This met the requirements for an exemption to GDC 17, as described in 10 CFR § 50.12. The staff concluded that an exemption to the requirements of GDC 17 for two physically independent offsite circuits was justified.

Moreover, exemption applications have generally been for changes that were limited in scope. For example, a plant might be exempted from inservice inspection for a specific pipe or weld but not for large portions of the plant. Many of the past exemptions were for minor changes to the schedules for containment leakage testing. The justifications for these exemptions were relatively straightforward and uncomplicated. There was no need to contemplate secondary impacts from approval of the request.

For SMR plants of LWR design, the number and complexity of needed exemptions may be modest. However, the exemptions needed for certification of non-LWR designs will deviate from these patterns in two respects -- they may be numerous, and they may be complex. This situation will create two difficulties for the licensing process: *(1)* the effort on the part of the NRC and the applicant to obtain approval of these exemptions will be significant, and *(2)* the public perception associated with the issuance of so many complex exemptions may be unduly negative.

While the exemption process probably can be used judiciously for SMRs of LWR design, the licensing of non-LWR designs may require a different approach. Any new licensing approach with revised requirements will require a time-consuming rulemaking change to the Code of Federal Regulations and will not be available in the near term.

4.4 NRC Manufacturing License

SMR modules will be manufactured in a factory-like setting and then the modules will be transported to the site for final fabrication and installation. Manufacturing in this manner offers advantages in quality and efficiency through replication, assembly-line construction, and maintaining a stable and skilled workforce. NRC requirements in 10 CFR § 52.167 permit the issuance of a manufacturing license with necessary and sufficient Inspections, Tests, Analyses, and Acceptance Criteria (ITAACs) to ensure the reactor will be manufactured and can be transported and operated in accordance with its license. The only manufacturing license issued by NRC was in 1982 to Offshore Power Systems for manufacture of eight floating nuclear power plants – none of which were manufactured because of market conditions prevailing at the time. Since the manufacturing license process has been used only once, many years ago, regulatory guidance for this process is minimal. Presently, none of the SMR vendors contacted in the development of this SMR Roadmap have any plans to use the manufacturing license process.

NRC requirements in 10 CFR Part 110 place limitations on the export of nuclear equipment and nuclear material. There are two types of export licenses: a General License; and a Specific License. A General License means an export or import license that is effective without the filing of a specific application with the Commission or the issuance of licensing documents to a particular person. A General License is a type of license issued through rulemaking by the NRC. A Specific License means an export or import license document issued to a named person and authorizing the export or import of specified nuclear equipment or materials.

A U.S. SMR vendor would be required to obtain an export license from the NRC. However, 10 CFR Part 110 is limited to the export of specified nuclear equipment -- it is silent on the export of an entire reactor fully assembled and ready for fuel load. The importing country and customer must meet all U.S. law and regulatory export requirements, including NRC's requirements in 10 CFR Part 110, "Export and Import of Nuclear Equipment and Material", and DOE's requirements in 10 CFR Part 810, "Assistance to Foreign Atomic Energy Activities." These requirements are in the process of revision and should be considered by both exporters of U.S. SMRs and importing countries and companies.

4.5 Role of Industry Codes and Standards for SMR Deployment

SMRs will be licensed and deployed in the global market. However, the deployment of SMRs as a global enterprise is not supported by an international licensing/certification framework that permits a "plug and play" environment similar to electronics such as TVs, computers and smart phones. The current nuclear licensing strategy requires any SMR design, regardless of the pedigree and robustness of licensing in country of origin, to be licensed also in the country of deployment (also referred to as the host country). Contrast this licensing process with that of aircraft approval. In essence, an aircraft approved or certified for air-worthiness by a competent approval authority is recognized worldwide through an international convention. While the international nuclear power community has recognized the need to harmonize licensing processes, technical acceptance criteria, and Codes and Standards to the extent practicable, the reactor design in the country of origin is licensed in accordance with that country's regulatory requirements and approved industry Codes and Standards. It is possible that the country of origin's required Codes and Standards may not be acceptable to the regulatory body of a host country. This may necessitate SMR design variations to comply with the regulatory requirements of the host country or the performance of a reconciliation of each host country's Codes and Standards and regulatory requirements with those of the country of origin.

The nuclear energy community and the SDOs recognize that the development of international nuclear Codes and Standards through the consensus process will facilitate worldwide licensing and deployment of SMRs. SDOs are spending much time and resources to support the development of international standards for all industries. The intent for world-wide consensus standards supporting SMR designs is that licensing authorities will adopt or reference these standards in their licensing basis for the designs. Adoption or reference to international standards as acceptable methods to meet a host country's licensing requirements will streamline the licensing process. It will also facilitate the end goal of a global enterprise for SMRs that can be interconnected and integrated in a way that vastly improves safe and secure operations through sharing of information.

A framework of international nuclear power plant licensing based in part on the adoption and use of international Codes and Standards is a laudable long-term goal. However, it is recognized that this goal may not be completely achievable because of historical nuclear licensing policy; and unique local, regional or country conditions and barriers. However, a fleet of SMRs deployed globally presents an opportunity to interconnect through the use of international standards.

As an example, the NRC references approximately 520 standards in its regulations, regulatory guides, and the staff's SRP. Over 160 NRC staff members participate in approximately 300 committees of SDOs. The NRC regularly reviews consensus standards developed by these SDOs and, if appropriate, incorporates them in its regulations, or endorses them in regulatory guides, or references them in the SRP. The NRC goal is that on a 5-year cycle, the approximately 425 regulatory guides, the most common source of referenced consensus standards, are reevaluated to determine whether they need updating, including the endorsement of new or revised consensus standards. More frequent revisions may occur based on technical evolutions and users' needs. 10 CFR § 50.55a is revised periodically to incorporate by reference later Editions of Sections III and XI of the BPVC and the OM Code. The latest Edition and Addenda of Sections III and XI of the BPVC incorporated by reference in 10 CFR § 50.55a is the 2007 Edition including the 2008 Addenda. The latest OM Code requirements incorporated by reference include: Subsections ISTA, ISTB, ISTC, and ISTD, Mandatory Appendices I and II, and Nonmandatory Appendices A through H and J, and include the 1995 Edition through the 2006 Addenda.

In the U.S., the nuclear industry, NRC and DOE recognized the need to identify:

(a) what new industry standards were developed or being developed that supported new nuclear technologies;
(b) what new or revised nuclear standards were needed;
(c) what industry Codes and Standards were referenced in NRC licensing documents and whether those references were up-to-date;
(d) how nuclear industry Codes and Standards referenced by NRC should be incorporated in a web-based database to support applicants and international use and licensing of nuclear technologies; and
(e) whether a nuclear global enterprise such as the deployment of SMRs would benefit from the development of international nuclear Codes and Standards.

In 2009, the Nuclear Energy Standards Coordination Collaborative (NESCC) was established under the sponsorship and coordination of the ANSI and the National Institute of Standards Technology (NIST), with the sponsorship of DOE and the NRC. NESCC provides a cross-stakeholder forum to bring together representatives of the nuclear industry, SDOs, subject matter experts, academia, and national and international governmental organizations to develop standards that support new nuclear designs, licensing, operation, fabrication and deployment. In addition, there are Codes and Standards activities in cross-cutting areas that are relatively technology neutral in that the standards involve new materials, techniques, or methods that are applicable to essentially all reactor technologies for use in new design or construction. Examples include high-density polyethylene piping, digital instrumentation and controls, composite concrete construction, and risk methodologies for advanced reactors. The NRC also recognized that its regulatory guidance documents needed further review and revision to ensure that they appropriately referenced current Codes and Standards. It proposed to develop a database of referenced standards. A high priority for the NESCC was to support the NRC in the development of its web-based database of standards for worldwide use.

The NESCC is not considering any new Codes or Standards that would be specifically applicable to the near-term SMRs under consideration in this SMR Roadmap. However, NESCC activities are germane to this SMR Roadmap in that the web-based database of standards being developed by NRC will provide information on what ASME Codes and Standards are referenced in NRC regulatory guidance documents. When this database is published (expected in mid-2014), it should be evaluated by ASME and other SDOs to ensure that the references are current, complete and accurately cited by the NRC.

Additionally, the NESCC might be an appropriate forum to assess and discuss the need for new nuclear Codes and Standards that might be applicable to advanced reactor designs. The NESCC could resolve conflicts between SDOs that might arise from the need to develop new nuclear Codes and Standards, and it might provide a source of government funding or other resources to support these SDO activities. Finally, the NESCC might provide government assistance to establish a framework for the international development and use of nuclear Codes and Standards to support a global fleet of SMRs or other reactor designs. As discussed below, it is a laudable goal to develop such a framework so that the licensing and deployment of nuclear power plants becomes a more unified global enterprise.

5 ROADMAP DISCUSSION AREAS FOR NEW OR REVISED ASME CODES AND STANDARDS

5.1 Section III, "Rules for Construction of Nuclear Facility Components"

All the SMR vendors contacted indicated that they do not see any problem with compliance with current Section III requirements. There is however one important caveat -- the NRC may conclude that the current Codes and Standards accepted and required for the current fleet of large LWRs may not be fully applicable to certain aspects of SMR designs.

While this is largely a Section XI and OM Code issue, there may be areas in Section III that are affected. For example, the Section III fracture toughness requirements for material used for Class 1 components, i.e., reactor coolant pressure boundary, might be a potential problem area. In paragraph NB-2311 of the 2013 Edition of Section III, all thicknesses of material for a pipe, tube, fittings, pumps, and valves with a nominal pipe size (NPS) of 6-inch (DN 150) and smaller are not required to be impact tested. (Impact testing is done to determine the material's susceptibility to brittle fracture.) This small parts exemption in Section III and other similar Section III exemptions would be subject to regulatory review. This exemption, while appropriate for large LWRs that have little if any Class 1 piping 6 inches and less, may not be acceptable to the regulator for SMRs that have more Class 1 piping that would fall under the exemption. The NRC could decide that this exemption may not be appropriate for SMRs. The point here is that although SMR vendors indicate they can meet current Section III requirements, the regulator may decide that existing Code requirements created for large LWRs may not be appropriate for certain SMR applications.

In one of the SMR designs, the reactor containment will be designed to Section III, Division 1, Subsection NB - Class 1 Components (NB) rules which is permissible by Section III, Subsection NCA – General Requirements for Division 1 and Division 2 (NCA). NCA-2134(c) allows this unique application of NB rules for metallic containment (Class MC) design instead of rules from Section III, Division 1, Subsection NE - Class MC Components (NE), provided the rules of NE-7000 are applied in lieu of the rules of NB-7000 for protection against overpressure. Typically, NB would be applied to reactor coolant pressure boundary components.

Without the exception allowed by NCA-2134(c), reactor containments for the new LWRs would be designed to NE for free-standing steel containments or Section III, Division 2 Code for Concrete Containments[2] for concrete containments. Failure modes and effects analysis (FMEA) of this containment are being developed to identify and compare failure modes to those of a reactor vessel. There is a possibility that the FMEA may indicate areas where use of the NB rules as currently written for containment may need to be revisited for this application of NB based on design pressures and containment configuration applied. Further, current Section XI inservice examination requirements are written for reactor coolant pressure boundary components. These examination requirements may need to be adjusted if the rules of NB are applied to a structure called the containment that is not considered reactor coolant pressure boundary.

Another potential issue for SMRs may be the applicability of the current NRC definition for the reactor coolant pressure boundary. NRC regulation 10 CFR § 50.55a(c)(2) exempts certain components from Subsection NB rules. 10 CFR § 50.55a(c)(2) exempts components from meeting the rules of Subsection NB (Class 1) if, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system. Depending on the SMR designs, the NRC may conclude that this definition may not be applicable for SMRs.

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[2] ASME Section III Division 2 is prepared by an ACI-ASME Technical Committee for Nuclear Service under the sponsorship of ACI and ASME.

5.2 Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"

In the area of Inservice Inspection, there are several areas identified where Section XI may need to be modified for SMRs. Section XI exempts components from examination based on their safety classification and nominal pipe size. For example, Class 1 components exempted from examinations include:

- Class 1 components and pipe segments 1-inch NPS and smaller (IWB-1220)
- Class 1 pressure retaining welds in piping less than 4-inch NPS - Table IWB-2500-1(B-J)
- All Class 1 Full Penetration Welded Nozzles in Vessels, Nozzle Inside Radius Section - Table IWB-2500-1 (B-D)
- Class 1 bolts and studs 2 inches and less in diameter - Table IWB-2500-1 (B-G-1, -2)

Since a majority of the SMR piping and components are smaller than indicated above, many of the components and pipe segments in an SMR would be excluded from in-service examination by current Section XI rules and regulatory requirements. In the current operating fleet of nuclear power plants these size exemptions are typically justifiable based on low safety consequences of failure of small components. Additionally, most components are larger than the specified exemption criteria and thus are subject to in-service inspection.

Further, the inspection of the reactor vessel in some of the SMR designs, such as the NuScale design, may be difficult to perform due to compactness of design and limited accessibility. The annular space between the outside surface of the reactor vessel and the reactor containment is small by design. It is suggested that for SMRs with these design constraints, development of alternative in-service examination requirements be considered and discussed with the NRC and the Section XI Standards Committee.

IWB-1220 in part exempts Class 1 components from volumetric, surface, VT-1 visual and VT-3 visual examinations if these components:

- Are connected to the reactor coolant system and are part of the reactor coolant pressure boundary;
- Are of such a size and shape that on postulated rupture the resulting flow of coolant from the reactor coolant system under normal plant operating conditions is within the capacity of makeup systems, excluding emergency core cooling systems that are operable from on-site emergency power;
- Are 1-inch nominal pipe size or smaller, except for steam generator tubing; or
- Are reactor vessel head connections and associated piping 2-inches and smaller, made inaccessible by control rod drive penetrations.

This relates to the discussion above regarding the NRC definition of reactor coolant pressure boundary and the components that are exempted from meeting Subsection NB by the NRC. Sections III and XI of the BPVC were developed for the fleet of operating reactors, mainly large LWRs. Should the NRC determine that the definition of reactor coolant pressure boundary is not appropriate for SMRs, a change in the Class 1 exemption requirements in 10 CFR § 50.55a(c)(2) may also affect the components exempted from examination by IWB-1220.

Section XI is currently being revised to include a new Division 2 for RIM. While Division 2 has not been finalized, the concept in the new Division 2 is to develop a program that will be broadly applicable to all reactor types. The RIM program addresses the plant Life Cycle and will select and implement a combination of design, fabrication, inspection, surveillance, operation, and maintenance requirements for passive metallic components that meet the plant level risk and reliability goals that are identified and selected for the RIM program by the Owner. Division 2 proposes that plant level reliability goals be derived from regulatory limits on the risks, frequencies, and radiological consequences of licensing basis events that are defined in the PRA. Further, the RIM program may be developed for plants that have either single or multiple reactor modules. This feature may benefit certain SMR designs.

Division 2 is scheduled to be published in the 2015 Edition of Section XI. The fact that this is a new approach for in-service examination may increase the time needed for review and acceptance of the 2015 Edition of Section XI by the NRC. Accordingly, NRC acceptance of the RIM process and incorporation by reference in 10 CFR Part 50 may take several years after ASME publication. Although the long-term objective of Section XI is to use RIM for ISI, in the interim it may be more expedient, until RIM is accepted by the regulator, for ASME to develop Code Cases that build on current practice to address specific topics in this area.

The NuScale SMR design has a containment design that is vastly different than reactor containment designs used in the current operating fleet. Section XI, Subsection IWE, addresses in-service inspection for metal reactor containments. IWE was based on current containment building designs. The NuScale containment vessel, unlike current designs, is a pressure vessel that is clad on both sides for corrosion resistance and submerged in a pool of borated water that also functions as the refueling pool and post-accident heat sink. Although the requirements of IWE could be used, they are not relevant to the NuScale SMR containment design. The modification of IWE requirements to address the NuScale SMR containment designs should be considered.

Another area that was identified by NuScale that could be addressed by Section XI was the examination of the containment vessel welded joints. The NuScale containment vessel is stainless steel clad on both sides due to its submergence in the reactor pool. Most containment vessel examinations specified in Section XI are either VT-1 or VT-3 visual examinations. This is appropriate for the containments in the current U.S. operating fleet. However, the NuScale containment is being designed as a Class MC vessel in accordance with the rules in Section III, Subsection NB, as permitted by Section III, NCA-2134(c). Volumetric examination is specified in Section XI, IWB-2500-1 (B-B) for in-service examination of pressure retaining welds in Class 1 vessels other than reactor vessels. Section XI, Table IWE-2500-1 (E-A) which are applicable to Class MC, requires only visual examination of containment surfaces. The unique design of the NuScale containment may require revised rules for examination of welded joints. In addition, since the annulus between the reactor vessel and the containment in the NuScale design is small, examination of the interior containment surfaces may present a challenge. The examination of welds in a Class MC containment is not required.

Other areas identified by NuScale where revisions to Section XI would be desirable include:

- Examination requirements for small lines.
- Interior examination requirements for the integral reactor vessel (pressurizer and steam generator are inside reactor vessel in both the mPower and the NuScale design.).
- Containment examination rules -- possibly a combination of MC and Class 1 rules.
- Examination exemption for small diameter bolts.
- Exemption from internal examination of small diameter vessel nozzles.

Section XI was developed for use by the current operating fleet which is comprised of large LWRs that have approximately 18-month fuel cycles. The mPower SMR has a 4-year fuel cycle. This presented a problem with the inspection intervals defined Section XI. Section XI specifies 10-year inspection intervals. During each 10 year inspection interval all the specified in-service inspections must be completed. Since the mPower SMR would only shut down for refueling every four years completion of their in-service inspection program they would have to complete their in-service inspection program in 8 years or 12 years. The 4-year fuel cycle, while economically beneficial from an operational point of view, was inconsistent with Section XI requirements.

To address the issue of extended fuel cycles, Section XI issued Code Case N-842, "Alternate Inspection Program for Longer Fuel Cycles," that was approved on January 28, 2014. Code Case N-842, in part, specifies that the length of the inspection interval shall be 8 years and the maximum interval shall not extend more than 1 year beyond the original pattern of 8-year intervals and shall not exceed 9 years in length and each inspection period shall be 4 years with certain tolerances specified by the BPVC. Code Case N-842 has not yet been accepted by the NRC in Regulatory Guide 1.147.

5.3 Operations and Maintenance Code

The NuScale SMR design incorporates passive safety features that use natural circulation as the motive force for reactor coolant during normal and abnormal operations. Natural circulation is created by the change in density of the water caused by heating and cooling. Passive safety features do not rely on components that use mechanical motion or some external motive force to perform their safety function, such as motor-operated valves or motor-driven pumps. Boiling and condensation mechanisms are used to remove decay heat during normal operations and are also relied on for emergency core cooling in the event of a reactor transient or accident. Therefore, since water will circulate through the reactor by natural circulation and there is an ample supply of water in the reactor pool in which the modules are located, there are no emergency core cooling pumps, containment heat removal pumps, or reactor coolant circulating pumps.

There are four valves on the NuScale reactor vessel that open to effect the Emergency Core Cooling System (ECCS) boiling and condensation mode of cooling. These valves cannot be tested during normal operation as opening them would produce a loss of coolant accident. ISTC-3521 of the OM Code addresses various scenarios when valve exercising for Category A and Category B valves is not practical either during plant operation or cold shutdown. Category A valves are valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function(s). Category B valves are valves for which seat leakage in the closed position is inconsequential for fulfillment of the required function(s). In these scenarios, the valves can be full-stroke tested during refueling outages. While the OM Code addresses this situation for the current operating fleet of reactors, it is predicated on an 18-month to 2-year fuel cycle. Periodic testing of these SMR valves is an item that may need to be considered by the OM Committee for those SMRs that have extended fuel cycles beyond that of the current operating reactors. For example, the mPower SMR design is proposing a 4-year fuel cycle. This extends the interval in which testing of certain valves that can only be tested during a reactor shutdown can be accomplished.

The OM Code currently is considering 3 Code Cases that will, when approved, address issues associated with extended operating cycles. The topics addressed in these proposed Code Cases for nuclear power plants with extended operating cycles are: testing of Class 1 pressure relief valves; testing of explosively actuated valves; and valve leak testing. These proposed Code Cases provide OM Code alternatives that will be useful for SMRs and other reactor designs with extended fuel cycles.

The OM Code is also preparing an Appendix to address the demonstration, availability and capability of safety significant components in non-safety systems for post-2000 nuclear power plants with passive post-accident heat removal systems. Nuclear power plants that have passive safety systems include for example, the Westinghouse AP-1000, and the NuScale SMR as described above. The NRC in Chapter 19.3 of NUREG-0800 [3], addressed the issue of regulatory treatment of non-safety systems (RTNSS) for passive advanced LWRs:

> "*The RTNSS process applies broadly to those nonsafety-related SSCs that perform risk significant functions and, therefore, are candidates for regulatory oversight. The RTNSS process uses the following five criteria to determine those SSC functions:*

A. SSC functions relied on to meet beyond design basis deterministic NRC performance requirements such as those set forth in Title 10 of the Code of Federal Regulations (10 CFR) 50.62 for mitigating Anticipated Transients Without Scram (ATWS) and in 10 CFR 50.63 for Station Blackout (SBO).

B. SSC functions relied on to ensure long-term safety (the period beginning 72 hours after a design basis event and lasting the following 4 days) and to address seismic events.

C. SSC functions relied on under power-operating and shutdown conditions to meet the Commission goals of a core damage frequency (CDF) of less than $1x10^{-4}$ each reactor year and a large release frequency (LRF) of less than $1x10^{-6}$ each reactor year.

D. SSC functions needed to meet the containment performance goal, including containment bypass, during severe accidents.

E. SSC functions relied on to prevent significant adverse systems interactions between passive safety systems and active nonsafety SSCs."

The RTNSS criteria above, particularly Criteria B and C, will apply to some of the SMR designs. Therefore, this is an important topic that is currently being addressed by the OM Code that will have application for some SMR designs.

NuScale is currently not pursuing a risk-based in-service test program as they believe they do not have sufficient information at this time to support using risk-informed insights.

NuScale was encouraged to interact with the OM Code Committee and to identify any areas where modification of the OM Code to address their design concept may be beneficial.

5.4 ASME Nuclear Crane Standards

No problems complying with the current ASME NUM and NOG standards for "nuclear cranes" were identified.

5.5 ASME Section V, "Nondestructive Examination"

Section V compliance issues have not yet been identified by any SMR vendor.

5.6 ASME AG-1, "Code on Nuclear Air and Gas Treatment"

AG-1 compliance issues have not yet been identified by any SMR vendor.

6 R&D ACTIVITIES FOR SMR STANDARDS

No specific areas have been identified at this time where research and development is necessary to support the development of Codes and Standards specific to the near-term SMRs.

7 GLOBAL COLLABORATIONS FOR INTERNATIONAL SMR STANDARDS

7.1 International Strategy and Framework for SMR Licensing

Energy needs, and particularly clean energy needs, across the globe have created a market and demand for SMRs. The promise of SMRs, both with enhanced safety and increased flexibility in applications and financing, has created a global market and potential enterprise that must be supported by more streamlined processes for licensing and deployment. Embarking countries that desire or require nuclear energy will probably demand, solicit and procure a nuclear technology that is proven and has been licensed by a competent regulatory authority such as the NRC. Embarking nations are risk averse and resource limited – they want a proven technology and a previously licensed/approved design to mitigate a financial constraint. Embarking nations also have limited regulatory resources and capabilities – a certified/approved design will permit a licensing process that can leverage the proven and licensed design. The international nuclear energy community must work together to identify and develop a more effective framework for licensing of these previously approved/certified SMRs. The World Nuclear Association's (WNA) Working Group on Cooperation in Reactor Design Evaluation and Licensing (CORDEL) established an SMR ad hoc working group to develop a position paper on an International Certification process for standardized SMRs. ASME will coordinate its recommendations in this SMR Roadmap with CORDEL as appropriate.

7.2 Development of International Codes and Standards

A first priority should be the development of international nuclear Codes and Standards that can be adopted and referenced by sovereign licensing authorities. The NRC database of Codes and Standards can be used as a starting point for this effort. Also it is crucial for each country with existing nuclear power and every embarking nation with nuclear power aspirations, to participate in the development of consensus nuclear standards that may provide a solid foundation for agreement on an international strategy and framework for licensing. As noted earlier, the NESCC could provide the U.S.– based forum to assess the benefits, strategy and the framework for such international collaborations. International collaboration on nuclear standards is an important safety and economic function. These standards exist to serve all stakeholders in the industry – manufacturers, suppliers, regulators, insurers, operators and the public.

7.3 ASME International Collaboration

Notwithstanding the lack of an international strategy and framework for the development of international nuclear standards, ASME (as well as other SDOs) recognizes the benefit of the development of nuclear Codes and Standards that can be adopted, or adopted with modifications, in countries that want to develop their own nuclear power plants, or want to license and deploy a nuclear power plant that has been licensed or approved using ASME Codes and Standards. There are almost 1300 volunteers participating in the various nuclear committees, subcommittees and working groups. They include approximately 100 international participants from eight countries besides the U.S. ASME Nuclear Codes and Standards are referenced in over 100 countries. ASME continues to seek broad international consensus in the development and use of Codes and Standards. Further acceptance of nuclear Codes and Standards in an international framework will help facilitate the licensing and deployment of reactors, such as SMRs, that are developed for international markets.

8 RECOMMENDATIONS

1) 10 CFR Part 50, Appendix A, "General Design Criteria" in its Introduction makes the following statements:

> *"These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units," and*

> *"There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified."*

Some of the issues identified above in the discussion of Section III, Section XI and the OM Code with regard to the applicability of the current Codes will be driven by the regulator. Currently SMR vendors indicate they can meet current Codes and Standards requirements. But the regulator, based on its review and technical evaluation, may conclude that certain current Codes and Standards requirements, developed for the current operating fleet of large LWRs, are not applicable to SMR designs. Unfortunately, some of these issues may not be identified until the NRC conducts a detailed technical review of an application submitted by an SMR vendor.

SMR vendors as part of their design process should evaluate their designs against the current ASME Code and Standards and the NRC's regulations and reach their own decision as to their applicability. If SMR vendors conclude that the current Codes and Standards do not provide an adequate degree of protection for public health and safety or may be applied differently by the NRC, they should engage the ASME Code Committees early in the process to develop appropriate requirements. This would provide a technical basis, developed through the ASME's ANSI-approved Code consensus process that could be used to support their position when engaging with the NRC during the application review and licensing process.

2) Except for the mPower SMR, the first of which is expected to be licensed using the 10 CFR Part 50 process, all other SMRs are expected to be licensed using the 10 CFR Part 52 process. 10 CFR § 50.55a states in its introductory paragraph:

> *Each combined license for a utilization facility is subject to the following conditions in addition to those specified in § 50.55, except that each combined license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section, but only after the Commission makes the finding under § 52.103(g) of this chapter."*

Paragraph (f) requires compliance with the OM Code, and Paragraph (g) requires compliance with Section XI of the BPVC. The § 52.103(g) decision is made by the Commission on satisfactory demonstration that of all the Inspections, Tests and Acceptance Criteria (ITAAC) have been met.

The acceptable § 52.103(g) decision allows the Combined License holder to load fuel and commence power operation. Therefore, for plants licensed using the 10 CFR Part 52 Combined License process, compliance with the OM Code and Section XI is not required prior to fuel load. Since ISI and Inservice Testing (IST) are operational programs they would be typically be excluded from a Part 52 design certification.

10 CFR § 50.55a(f) requires that pumps and valves which are classified as ASME Code Class 1, 2 or 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness.

10 CFR § 50.55a(g) requires components which are classified as ASME Code Class 1, 2 or 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 must be designed and be provided with access to enable the performance of inservice examination of these components.

The provision of access to enable inservice test and inservice inspection is an issue that needs to be addressed during the design of the plant. However, 10 CFR § 50.55(a) indicates that the provision for access is not required until the Commission makes the 10 CFR § 52.103(g) finding, i.e., grants permission to load fuel. Understanding what access is necessary to be provided is also a function the type of inservice inspection (non-destructive examination techniques) to be used. Since most would understand this concept, practically speaking it may not create an issue.

As part of the SMR design process, it is important that this provision of access for inservice examination and testing be addressed even though it may not be reviewed as part of a design certification process. If specialized inservice inspection techniques are to be employed due to limited space, such as the examination of the NuScale containment, Section XI and perhaps Section V of the BPVC should be engaged early in the design process. It is recommended that SMR vendors and ASME consider and address any potential ISI and IST issues prior to design certification reviews to ensure sufficient time for any design modifications.

3) Although the long-term objective of Section XI is to use RIM for ISI, in the interim it may be more expedient, until RIM is accepted by the regulator, for ASME to develop Code Cases that build on current practice to address specific topics in this area.

4) The NuScale SMR design employs a containment design that is vastly different than other LWRs currently in operation. Consequently, the rules for Inservice Examination contained in Section XI, Subsection IWE are not applicable to the NuScale containment. It is recommended that NuScale work with the Section XI Committee to develop rules for inservice inspection of their containment.

5) Many of SMR designs because of their size and the size of their components will be able to take advantage of certain size exemptions presently contained in Section III and Section XI. As indicated above, there are certain exemptions in Section III based on component size. For example, Class 1 components exempted from examination include:

- Class 1 components and pipe segments 1-inch NPS and smaller (IWB-1220)
- Class 1 pressure retaining welds in piping less than 4 inches NPS - Table IWB-2500-1(B-J)
- All Class 1 Full Penetration Welded Nozzles in Vessels, Nozzle Inside Radius Section -Table IWB-2500-1 (B-D)
- Class 1 bolts and studs 2 inches and less in diameter - Table IWB-2500-1 (B-G-1, -2)