SANDIA REPORT
SAND2013-3683
Unlimited Release
Printed May 2013

Evaluation of the Applicability of Existing Nuclear Power Plant Regulatory Requirements in the U.S. to Advanced Small Modular Reactors

Jeffrey LaChance, Gregory Baum, Felicia Durán, Kathy Ottinger Farnum, Sabina Jordan, Bobby Middleton, Timothy Wheeler

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Issued by Sandia National Laboratories, operated for the United States Department of Energy by Sandia Corporation.

Printed in the United States of America. This report has been reproduced directly from the best available copy.

Available to DOE and DOE contractors from
U.S. Department of Energy
Office of Scientific and Technical Information
P.O. Box 62
Oak Ridge, TN 37831

Telephone: (865) 576-8401
Facsimile: (865) 576-5728
E-Mail: reports@adonis.osti.gov
Online ordering: http://www.osti.gov/bridge

Available to the public from
U.S. Department of Commerce
National Technical Information Service
5285 Port Royal Rd.
Springfield, VA 22161

Telephone: (800) 553-6847
Facsimile: (703) 605-6900
E-Mail: orders@ntis.fedworld.gov
Online order: http://www.ntis.gov/help/ordermethods.asp?loc=7-4-0#online

SAND2013-3683
Unlimited Release
Printed May 2013

Evaluation of the Applicability of Existing Nuclear Power Plant Regulatory Requirements in the U.S. to Advanced Small Modular Reactors

Jeffrey LaChance, Timothy Wheeler, and Cathy Ottinger Farnum
Risk & Reliability Department

Bobby Middleton
Advanced Nuclear Concepts

Sabina Jordan
Policy and Decision Analytics

Felicia Durán
Security Systems Analysis

Gregory Baum
International Nuclear Security Engineering

Sandia National Laboratories
P.O. Box 5800
Albuquerque, New Mexico 87185

Abstract

The current wave of small modular reactor (SMR) designs all have the goal of reducing the cost of management and operations. By optimizing the system, the goal is to make these power plants safer, cheaper to operate and maintain, and more secure. In particular, the reduction in plant staffing can result in significant cost savings. The introduction of advanced reactor designs and increased use of advanced automation technologies in existing nuclear power plants will likely change the roles, responsibilities, composition, and size of the crews required to control plant operations. Similarly, certain security staffing requirements for traditional operational nuclear power plants may not be appropriate or necessary for SMRs due to the simpler, safer and more automated design characteristics of SMRs. As a first step in a process to identify where regulatory requirements may be met with reduced staffing and therefore lower cost, this report identifies the regulatory requirements and associated guidance utilized in the licensing of existing reactors. The potential applicability of these regulations to advanced SMR designs is identified taking into account the unique features of these types of reactors.

EXECUTIVE SUMMARY

Recently, advanced small modular reactors (SMRs) have received a great deal of attention as an alternative to large, capital-intensive light water reactors that are often difficult for even a large utility to finance. The current wave of small modular reactor designs all have the goal of reducing the cost of management and operations. In particular, the reduction in plant staffing can result in significant cost savings. By optimizing the system, the goal is to make these power plants safer, cheaper to operate and maintain, and more secure.

Existing nuclear power plants in the U.S. are large (on the order of 1000 MWe) light water reactors (LWRs) and would cost on the order of $10 billion to build today, an investment that can be highly risky for an electric utility. On the other hand, SMRs are designed to produce roughly 200 to 300 MWe and expected to cost substantially less (on the order of $2 billion), a less risky investment. The capital costs of SMRs and their potential use in servicing small electrical grids have resulted in interest in developing SMRs. They have the further advantage of being either built independently or as modules in a larger nuclear power complex. Modules can be developed at manufacturing facilities and shipped to power plants for use.

There are many types of SMR designs being formulated including light water and liquid metal reactors. The light water reactor designs (e.g., mPower, NuScale, the Westinghouse SMR, and the Holtec SMR-160) have the lowest technological and regulatory risk since they are similar to most operating reactors in the U.S. The designs under consideration in the U.S. are pressurized water reactors (PWRs) that utilize conventional fuels with higher burnup and are mostly integral designs (i.e., there are no coolant loops since the pressurizer and steam generators are located in the reactor vessel). These designs are generally referred to as integral PWRs (iPWRs). Small liquid metal reactors (LMRs) represent more of a technological risk as the safety case for them is more uncertain due the lack of LMR commercial use. They use liquid metals such as sodium, lead, or lead bismuth which have high thermal conductivity and operate at or near atmospheric pressure. Small LMR designs include the Integral Fast Reactor (ARC-100) which is a scaled down version of the Integral Fast Reactor, the GE-Hitachi PRISM design, and the Toshiba Super-Safe, Small and Simple (4S) design.

Generally, SMRs have several technological advantages that can affect the operation, safety, and security of the plant. Two examples are passive safety features that utilize gravity-driven or natural convection systems – rather than engineered, pump-driven systems – to supply backup during upset conditions and negative temperature moderator coefficients. Higher fuel burnup rates increases the refueling period and reduces the amount of waste. The smaller size can also potentially result in a reduced emergency planning zone to less than the 10 miles required for current operating plants and a smaller footprint for a security force to protect.

One area where SMRs are expected to provide a significant cost savings without compromising safety and security is in the area of plant staffing. Nuclear Regulatory Commission (NRC) regulations specify the operator, security, and emergency response requirements for licensed nuclear reactors. Current regulations are based on the staffing requirements for the large LWRs currently in operation. The reduced size of SMRs and the passive designs with inherent safety features utilized in most SMR designs can reduce the plant staffing. Operational staff may be reduced through the use of automated response features and shared control rooms. During upset conditions, passive safety features and lower levels of decay heat minimize the need for prompt operator actions to place the plant in a safe condition. Because the immediate response of the operators is to observe the response of the inherent and passive systems, there is a potential for reducing the operational staff.

Similarly, the design and operation of SMRs can result in reduced security staffing requirements. Specifically, the location of the reactors below ground in some SMR designs protects the plant against external threats such as large explosive weapons or aircraft impacts. Use of advanced physical security features allows for a more effective approach to protect the plant with potentially fewer security personnel. In addition, the passive safety features of SMR designs will allow for additional delay time for security forces to respond to an incident.

However, SMR designs face regulatory challenges since regulations have been built up around existing LWR technology. Some of the existing regulations may not be applicable to specific SMR designs (especially those utilizing cooling mediums other than water). In addition, use of passive safety systems can eliminate specific types of accidents and require less operator actions in response.

As a first step in a process to identify where regulatory requirements may be met with reduced staffing and therefore lower cost, this report identifies the regulatory requirements and associated guidance utilized in the licensing of existing reactors. The potential applicability of these regulations to advanced SMR designs is addressed taking into account the unique features of these types of reactors. Specific focus was placed on the potential for reduction in operating, security, and emergency response staffing. The following discussion presents some preliminary insights from this effort.

Operating Staff

The design features and concepts of operations for new generations of advanced reactors, as well as the introduction of new automated systems into existing plants, may require fewer licensed operators. Unless and until the NRC develops and establishes new regulations for MCR staffing for advanced reactors, applicants will have to submit exemption requests from the applicable regulations included primarily in 10 CFR 50.54(m). The exemptions may include variations in the prescribed number, composition, or qualifications of licensed operators. In SECY-02-0180, SECY-11-0098, and in NUREG-1791, the NRC has acknowledged that there may be a case for exemptions from MCR staffing regulations for advanced reactor designs, and that the regulations would not address all of the potential MCR configurations that may be proposed (e.g., more than two reactors controlled from the same MCR). The NRC staff has proposed a policy to address advanced reactor MCR staffing issue in both the near-term (i.e., no advanced SMR operating experience) and long term (i.e., subsequent to operational experience of licensed SMRs). This policy would involve evaluating requests for exemptions from 10CFR 54 (m) from both DC and CL applicants prior to the existence of SMR operating experience. Eventually, after operating experience of SMRs has been gained, the NRC would revise existing regulations and develop other regulations to provide specific control room staffing requirements for SMRs.

As outlined in SECY-11-0098, the NRC staff intends to pursue the short term strategy by revising the Standard Review Plan (NUREG-0800) and the two NUREG documents that deal directly with staff evaluation of control room staffing issues and related human engineering aspects, NUREG-0711 and NUREG-1791 (summarized in Section 3.2). The NRC intends to revise these staff guidance documents so that they identify and address to the extent possible differences in advanced reactor designs that might impact conduct-of-operations, operator performance and staffing levels.

Security

The NRC's physical security regulations for NPPs are generally technology neutral and therefore applicable for advanced SMRs. However, licensees for SMRs may wish to explore strategies to reduce cost and staffing while still achieving regulatory compliance. Each NPP is operated differently and the appropriate staffing level for the facility is determined considering: the number of personnel needed to ensure the protection of nuclear material and facilities; the number of personnel needed to maintain security programs to support the sustainability of a physical protection program, including the protective strategy; and the number of personnel needed to handle normal daily security operations. An overarching goal for reactor security is to establish and maintain a physical security program and infrastructure, to include a security organization capable of providing high assurance of the protection of nuclear material and facilities against adversary attack.

The NRC has stated expectations that advanced reactor designs will provide enhanced measures of safety and security. Additional research may be needed to assess the efficacy of any new security measures. In addition the NRC expects reactor designers to integrate security into the design and conduct a security assessment to evaluate the level of protection provided. The design approaches for SMR safety related systems potentially provide a larger measure of "intrinsic" security, although this would have to be demonstrated. In addition, depending on the particular SMR design, the plant footprint is likely to be smaller than that for a traditional LWR with a commensurate smaller number of targets and target sets and overall smaller footprint for security operations. This factor may in itself demonstrate cost savings for SMR security.

Emergency Response

While some of the emergency planning regulations, regulatory guides and other guidance are fully applicable to SMRs, there are some aspects that warrant evaluation with respect to the differences between SMRs and light water power reactors. These can include, but are not limited to, size of the emergency planning zone, notification times, shared facilities, collocation with other SMRs and other nuclear power reactors, number of staffing positions, and times for augmenting staffing and shared staffing. As an example, with the SMR passive safety features and the potential for reduced accident source terms and fission product releases, it may be appropriate for SMRs to develop reduced EPZ sizes using a dose/distance approach. There are precedents for this since the NRC has licensed several small reactors with an EPZ of 5 miles for plume (and 30 miles for ingestion).

NUREG-0654, Table B-1 (see Table 7-1) provides the specific guidance for staffing requirements for nuclear power plant emergencies. This table specifies a minimum of 10 on-shift responders in four functional areas and specifies 7 on-shift responders who perform response duties that may be performed by shift personnel in additional to their other assigned functions, with firefighting and site access control staffed on a site-specific basis. This type of shared staff function will be particularly relevant for SMRs with multiple reactors and shared control rooms. This table also specifies the number of augmenting responders within 30 and 60 minute timeframes. A number of current nuclear power plants have longer augmentation times than 30 and 60 minutes that have been approved on a case-by-case basis. For SMRs, with passive safety features, the time to augment the emergency staff will be relevant and depend upon the safety features and their impact in accident progression.

CONTENTS

FIGURES

TABLES

NOMENCLATURE

ACRS	Advisory Committee on Reactor Safeguards
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASSESS	Analytic System and Software for Evaluating Safeguards and Security
ATLAS	Advanced Time Line Analysis System
BPVC	Boiler Vessel Pressure Code
BWR	Boiling Water Reactor
CFR	United States Code of Federal Regulations
CL	Combined License
COL	Combined Construction and Operating License
CP	Construction Permit
DBA	Design Basis Accident
DBT	Design Basis Threat
DC	Design Certification
DOE	United States Department of Energy
DSRS	Design-Specific Review Standards
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EOF	Emergency Operations Facility
EP	Emergency Planning
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
ERDS	Emergency Response Data System
ESP	Early Site Permit
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HFE	Human Factors Engineering
HP	Health Physics
HTGR	High Temperature Gas-Cooled Reactor
iPWR	Integral Pressurized Water Reactor
ISG	Interim Staff Guidance
ITAAC	Inspections, Tests, Analyses, Acceptance Criteria
JCATS	Joint Conflict and Tactical Simulation
LMR	Liquid Metal Reactor
LWR	Light Water Reactor
MC&A	Material Control and Accountability
MCR	Main Control Room
ML	Manufacturing License
NEI	Nuclear Energy Institute
NLO	Non-Licensed Operator
NPP	Nuclear Power Plant
NRC	United States Nuclear Regulatory Commission

NUMARC	Nuclear Management and Resource Council
OL	Operating License
OSC	Operational Support Center
PAG	Protective Action Guideline
PAR	Protective Action Recommendation
PWR	Pressurized Water Reactor
REP	Radiological Emergency Preparedness
RG	Regulatory Guide
RO	Licensed Reactor Operator
SAR	Safety Analysis Report
SDA	Standard Design Approval
SDO	Standards Development Organization
SGI	Safe Guards Information
SM	Shift Manager
SMR	Small Modular Reactor
SNL	Sandia National Laboratories
SNM	Special Nuclear Material
SPDS	Safety Parameter Display System
SRO	Licensed Senior Reactor Operator
SSC	Structures, systems, and Components
SSNM	Strategic Special Nuclear Material
SRP	Standard Review Plan
STA	Shift Technical Advisor
TMI	Three Mile Island
TSC	Technical Support Center

1. INTRODUCTION

Recently, advanced small modular reactors (SMRs) have received a great deal of attention as an alternative to large, capital-intensive light water reactors that are often difficult for even a large utility to finance. The current wave of small modular reactor designs all have the goal of reducing the cost of management and operations. In particular, the reduction in plant staffing can result in significant cost savings. By optimizing the system, the goal is to make these power plants safer, cheaper to operate and maintain, and more secure.

However, SMR designs face regulatory challenges since regulations have been built up around large light water reactors (LWRs). Some of the existing regulations may not be applicable to specific SMR designs (especially those utilizing cooling mediums other than water). In addition, use of passive safety systems can eliminate specific types of accidents and require less operator actions in response. For example, with regard to conduct-of-operations and main control room staffing, the introduction of advanced reactor designs and increased use of advanced automation technologies in existing nuclear power plants will likely change the roles, responsibilities, composition, and size of the crews required to control plant operations. Current regulations regarding control room staffing, which are based upon the concept of operation for existing light-water reactors, may no longer apply. Similarly, certain security staffing requirements for traditional operational nuclear power plants may not be appropriate or necessary for SMRs due to the simpler, safer and more automated design characteristics of SMRs

As a first step in the process of identifying where regulatory requirements may be met with reduced staffing and therefore lower cost, this report identifies the regulatory requirements and associated guidance utilized in the licensing of existing nuclear power plants (NPPs). The potential applicability of these regulations to advanced SMR designs is identified taking into account the unique features of these types of reactors.

1.1. Background

Existing nuclear power plants in the U.S. are large (on the order of 1000 MWe) light water reactors (LWRs) and would cost on the order of $10 billion to build today, an investment that can be highly risky for an electric utility. On the other hand, small modular reactors (SMRs) are designed to produce roughly 200 to 300 MWe and expected to cost substantially less (on the order of $2 billion), a less risky investment. The capital costs of SMRs and their potential use in servicing small electrical grids have resulted in interest in developing SMRs. They have the further advantage of being either built independently or as modules in a larger nuclear power complex. Modules can be developed at manufacturing facilities and shipped to power plants for use.

There are many types of SMR designs being formulated including light water and liquid metal reactors. The light water reactor designs (e.g., mPower, NuScale, the Westinghouse SMR, and the Holtec SMR-160) have the lowest technological and regulatory risk since they are similar to most operating reactors in the U.S. The designs under consideration in the U.S. are pressurized water reactors (PWRs) that utilize conventional fuels with higher burnup and are mostly integral designs (i.e., there are no coolant loops since the pressurizer and steam generators are located in the reactor vessel). These designs are generally referred to as integral PWRs (iPWRs). Small liquid metal reactors (LMRs) represent more of a technological risk as the safety case for them is more uncertain due the lack of LMR commercial use. They use liquid metals such as sodium, lead, or lead bismuth which have high thermal conductivity and operate at or near atmospheric pressure. Small LMR designs include the Integral Fast Reactor (ARC-100) which is a scaled down version of the Integral Fast Reactor, the GE-Hitachi PRISM design, and the Toshiba Super-Safe, Small and Simple (4S) design.

Generally, SMRs have several technological advantages that can affect the operation, safety, and security of the plant. Two examples are passive safety features that utilize gravity-driven or natural convection systems – rather than engineered, pump-driven systems – to supply backup during upset conditions and negative temperature moderator coefficients. Higher fuel burnup rates increases the refueling period and reduces the amount of waste. The smaller size can also potentially result in a reduced emergency planning zone to less than the 10 miles required for current operating plants and a smaller footprint for a security force to protect.

One area where SMRs are expected to provide a significant cost savings without compromising safety and security is in the area of plant staffing. Nuclear Regulatory Commission (NRC) regulations specify the operator, security, and emergency response requirements for licensed nuclear reactors. Current regulations are based on the staffing requirements for the large LWRs currently in operation. The reduced size of SMRs and the passive designs with inherent safety features utilized in most SMR designs can reduce the plant staffing. Operational staff may be reduced through the use of automated response features and shared control rooms. During upset conditions, passive safety features and lower levels of decay heat minimize the need for prompt operator actions to place the plant in a safe condition. Because the immediate response of the operators is to observe the response of the inherent and passive systems, there is a potential for reducing the operational staff.

Similarly, the design and operation of SMRs can result in reduced security staffing requirements. Specifically, the location of the reactors below ground in some SMR designs protects the plant against external threats such as large explosive weapons or aircraft impacts. Use of advanced physical security features allows for a more effective approach to protect the plant with potentially fewer security personnel. In addition, the passive safety features of SMR designs will allow for additional delay time for security forces to respond to an incident.

1.2. Scope

The scope of this report includes a review of the existing regulations and associated sub-tier guidance documents used for licensing existing nuclear power plants in the U.S. Regulations and guidance related to operations staffing, safety evaluations, security, materials control and accountability, and emergency preparedness are identified and their general applicability to advanced SMRs (both LWRs and LMRs) is discussed.

1.3. Report Content

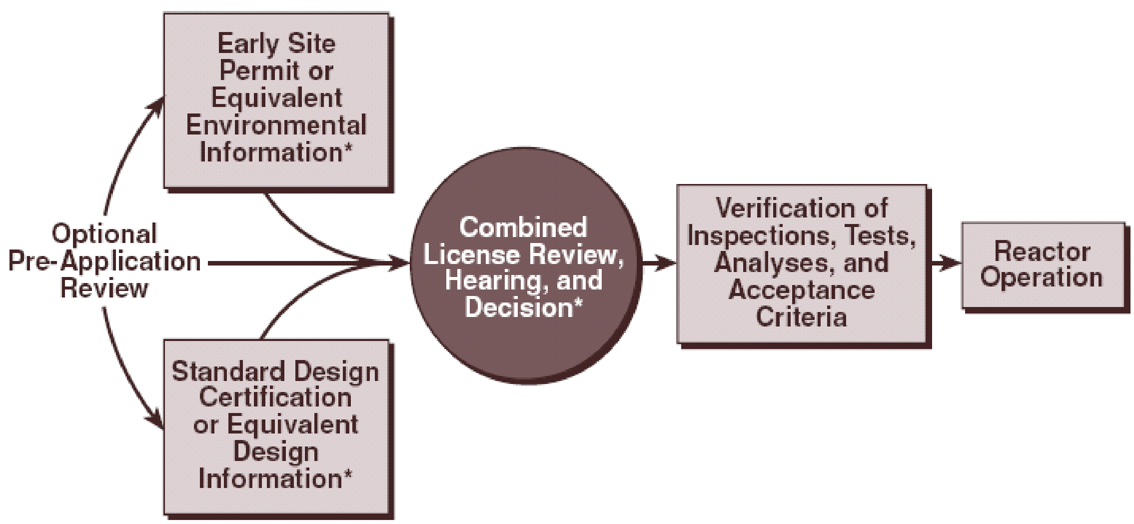
This report includes a general overview of the regulatory structure used in licensing NPPs in the U.S in Section 2. Sections 3 through 7 provide a summary of the actual regulations and sub-tier guidance requirements applicable to the following topics addressed in licensing an NPP: operation, safety, security, material safeguards, and emergency planning. Each of these sections also provides a summary of the applicability of these requirements to advanced SMRs. The appendices to the report provide detailed tables of the associated regulations and a brief discussion of their applicability to different types of advanced SMRs. Section 8 of the report provides a summary of the findings of this effort.

2. REGULATORY REQUIREMENTS FOR LICENSING NPPS IN THE U.S

This section identifies the current regulatory requirements pertinent to licensing a new reactor (including iPWRs and LMRs) by the NRC. It also discusses supporting documents that affect the existing licensing process such as regulatory guides, standard review plan, interim staff guidance, consensus codes and standards, and NRC policy statements.

2.1 NPP Licensing Process in the U.S

Commercial power reactors currently operating in the U.S. were licensed individually in accordance with the Title 10, Code of Federal Regulations [1], Part 50 (10 CFR Part 50). Under 10 CFR Part 50, each nuclear power plant was licensed in a two-step process with one permit to allow construction and a second permit to allow operation. In 1989, Part 52 was added to the CFR to streamline the application process by increasing licensing predictability, resolving environmental concerns prior to construction, and promoting standardized designs, which is the probable licensing course for all recent and future reactor designs. Part 52 specifies the regulations governing Early Site Permits (ESPs), Standard Design Certifications, and Combined Construction and Operating Licenses (COLs). Figure 2-1, taken from NUREG/BR-0298 Rev.2 [2], illustrates the possible interactions between the parts of the Part 52 process (i.e., the ESP, the design certification, and the CL).



*A combined license application can reference an early site permit, a standard design certification, both, or neither. If an application does not reference an early site permit and/or a standard design certification, the applicant must provide an equivalent level of information in the combined license application.

Figure 2-1. The Part 52 Licensing Process.

2.1.1 Early Site Permit

The NRC can issue an ESP for approval of one or more sites for one or more nuclear power facilities separate from the filing of an application for a construction permit or combined license in accordance with 10 CFR Part 52. The requirements for an ESP are provided in Subpart A of 10 CFR Part 52. No specific design is required, but a plant parameter envelope must be specified. This parameter envelope includes the number, type, and thermal power level of the facilities for which the site may be used; the boundaries of the site; the general proposed location of each facility on the site; the anticipated maximum levels of radiological and thermal effluents each facility will produce; the types of cooling systems, intakes, and outflows that may be associated with each facility; the seismic, meteorological, hydrologic, and geologic characteristics of the proposed site; the location and description of any nearby industrial, military, or transportation facilities and routes; and the existing and projected future population profile of the area surrounding the site.

An ESP is a partial construction permit and is, therefore subject to all procedural requirements in 10 CFR Part 52 that are applicable to construction permits. Applications for ESPs will be reviewed according to the applicable standards set out in 10 CFR Parts 50, 51, and 100 as they apply to applications for construction permits for nuclear power plants. ESPs are valid for 10 to 20 years and can be renewed for an additional 10 to 20 years. ESPs address site safety issues, environmental protection issues, and plans for coping with emergencies, independent of the review of a specific nuclear plant design.

2.1.2 Design Certification

The NRC may certify and approve a standard plant design under Subpart B of Part 52 through a rulemaking, independent of a specific site. The design certification (DC) is valid for 15 years. The issues that are resolved in a design certification have a more restrictive back fit requirement than issues that are resolved under other licensing processes. That is, the NRC cannot modify a certified design unless the modification is necessary to meet the applicable regulations in effect at the time of the design certification, or to assure adequate protection of the public health and safety. An application for a combined license under 10 CFR Part 52 can incorporate by reference a design certification and/or an ESP. The advantage of this approach is that the issues are resolved by the DC rulemaking process and those resolved during the ESP hearing process are precluded from reconsideration at the combined license stage.

NRC policy encourages early discussions (prior to a license application) between NRC and potential applicants such as utilities and reactor designers to offer licensing guidance and to identify and resolve potential licensing issues early in the licensing process. The Advisory Committee on Reactor Safeguards (ACRS) review on the safety aspects of the proposed design is required under Part 52.53. During this pre-application period for a design certification, the NRC holds public meetings with potential applicants to discuss advanced reactor designs to identify (1) major safety issues that could require Commission policy guidance to the staff, (2) major technical issues that the staff could resolve under existing regulations or NRC policy, and (3) any research needed to resolve identified issues.

A license applicant can also submit a proposed standard design for approval under Subpart E of 10 CFR Part 52. The process under this subpart is similar to the approval process followed for existing reactors under Part 50 and may involve approval for either the final design for the entire reactor facility or the final design of only major portions of the facility. The Subpart E process is also different than the process in Subpart B in that the design is not subject to rulemaking (issues resolved by the design review process can be reconsidered during licensing hearings). A standard design that is certified under Subpart B becomes an appendix to Part 52. A standard design that is approved under Subpart E is not included in Part 52.

2.1.3 Combined License

A combined license (CL), issued under Subpart C of 10 CFR Part 52, authorizes construction of the facility in a manner similar to a construction permit under 10 CFR Part 50. However, the CL will specify the inspections, tests, and analyses that the licensee must perform. It will also specify the acceptance criteria that, if met, are necessary to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license and the applicable regulations. Collectively, the inspections, tests, analyses, and acceptance criteria are referred to as the ITAAC. To obtain a CL, the application must include the technically relevant information required by 10 CFR 50.34 for an operating license.

After issuing a CL, the Commission will verify that the licensee completed the required inspections, tests, and analyses and that the acceptance criteria were met prior to operation of the facility. At periodic intervals during construction, the NRC will publish notices of these completions in the *Federal Register*. Then, not less than 180 days before the date scheduled for initial loading of fuel, the NRC will publish a notice of intended operation of the facility in the *Federal Register*. There is an opportunity for a hearing following construction, but the NRC will consider petitions for a hearing only if the petitioner demonstrates that the licensee has not met the acceptance criteria. Before a plant can operate, the Commission must determine that the acceptance criteria were met. In both licensing processes (10 CFR Part 50 and Part 52) the NRC maintains oversight of the construction and operation of a facility throughout its lifetime to assure compliance with the Commission's regulations for the protection of public health and safety and the environment.

2.1.4 Manufacturing License

A company may submit an application for a license to manufacturer nuclear power reactors under Subpart F of 10 CFR Part 52. Such a license does not allow construction, installation, or operation at the sites on which the reactors are to be operated. A nuclear power reactor manufactured under a manufacturing license may only be transported to and installed at a site for which either a construction permit under Part 50 or a combined license under Subpart C of Part 52 has been issued. A manufacturing license applicant may reference a standard design certification or a standard design approval in its application.

Manufacturing licenses may be utilized to manufacturer modular nuclear reactors such as the NuScale iPWR. Per 10 CFR Part 52, a modular design means a nuclear power station that consists of two or more essentially identical nuclear reactors (modules) and each module is a separate nuclear reactor capable of being operated independent of the state of completion or operating condition of any other module co-located on the same site, even though the nuclear power station may have some shared or common systems.

2.2 Regulatory Requirements

Title 10 of the Code of Federal Regulations CFR contains the regulations pertinent to the US NRC. Of these regulations, the following are pertinent to the licensing of nuclear power plants:

1. **Part 20 – Standards for Protection Against Radiation**. The regulations in this part establish standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the NRC. The contents of Part 20 are primarily procedural and thus are generally technology neutral.
2. **Part 50 – Domestic Licensing of Production and Utilization Facilities**. All operating LWRs in the US were licensed under Part 50 which contains many LWR-specific requirements. Paragraph 50.34, specifies the “Contents of applications; technical information” that must be supplied in a license submittal. Many sections of Part 50 are LWR specific. For example, § 50.46 provides the acceptance criteria for the emergency core cooling systems for LWRs. As such, many of the regulations in Part 50 are not applicable to LMRs. In addition, because of the unique nature of iPWRs, some of the existing regulations may not be applicable
3. **Appendix A of Part 50 – General Design Criteria**. This appendix establishes principal design criteria for safety of operating LWRs. However, some General Design Criteria (GDCs) can be usefully employed for other reactor technologies, such as LMRs (see Section 4.1.1). In addition, modification of some GDCs may be required for iPWRs.
4. **Part 51 – Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions**. This part contains environmental protection regulations applicable to NRC's domestic licensing and related regulatory functions. The contents of Part 51 are primarily procedural and thus are generally technology neutral. However, the requirements are specific to the type of license or certificates that are available.
5. **Part 52 – Licenses, Certifications, and Approvals for Nuclear Power Plants**. This part governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities. The contents of Part 52 are primarily procedural and thus are generally technology neutral (i.e., it applies to all types of reactors). However, the requirements are specific to the type of license or certificates that are available and thus are defined into associated subparts as discussed previously in Section 1.1.
6. **Part 73 – Physical Protection of Plants and Materials**. This part prescribes requirements for the establishment and maintenance of a physical protection system which will have capabilities for the protection of special nuclear material at fixed sites and in transit and of plants in which special nuclear material is used. These requirements are technology neutral.
7. **Part 74 – Material Control and Accounting of Special Nuclear Material**. This part has been established to contain the requirements for the control and accounting of special nuclear material at fixed sites and for documenting the transfer of special nuclear material. These requirements may be most applicable to LMRs utilizing plutonium fuel.

8. **Part 100 – Reactor Site Criteria**. The purpose of this part is to establish approval requirements for proposed sites for stationary power and testing reactors subject to part 50 or part 52. The contents of Part 100 are primarily procedural and thus are generally technology neutral.

Appendix A provides a list of regulations in Parts 50, 51, and 52 pertaining to the licensing process. The general applicability of these regulations to SMRs is also addressed in Appendix A (the applicability to an individual SMR would have to be addressed after the design is well established).

2.3 Standard Review Plan, Regulatory Guides, and Interim Staff Guidance

As part of the licensing process, guidance is provided to both the NRC staff and licensees on acceptable ways to meet the regulations. Several types of guidance documents are described in the following sections.

2.3.1 Standard Review Plan

The Standard Review Plan (SRP) provides guidance to NRC staff in performing safety reviews of construction permit (CP) or operating license (OL) applications (including requests for amendments) under 10 CFR Part 50 and early site permit (ESP), design certification (DC), combined license (CL or COL), standard design approval (SDA), or manufacturing license (ML) applications under 10 CFR Part 52 (including requests for amendments).

The SRP is intended to be a comprehensive and integrated document that provides NRC reviewers with guidance that describes methods or approaches that the staff has found acceptable for meeting NRC requirements. Implementation of the criteria and guidelines contained in the SRP by staff members in their review of applications provides assurance that a given design will comply with NRC regulations and provide adequate protection of the public health and safety. The SRP also makes the staff's review guidance for licensing nuclear power plants publicly available and is intended to improve industry and public stakeholder understanding of the staff review process. It should be noted that the SRP is not a substitute for NRC regulations, and compliance with the SRP is not required. The SRP generally describes an acceptable means of meeting the regulations, but not necessarily the only means, applications may deviate from the acceptance criteria in the SRP. A license applicant is required to identify differences between the design features, analytical methods, and procedural measures proposed for the facility and the SRP acceptance criteria, and evaluate how the proposed alternatives provide an acceptable method for complying with the NRC regulations.

The SRP was first issued in 1975 and has since been updated several times in NUREG-0800 [3]. The current version of the SRP is specific to the existing fleet of LWRs. However, it also includes new sections for evaluating new applications submitted under 10 CFR Part 52. A list of the current SRP sections is provided in Appendix B, Table B-1. Currently there is no specific set of SRPs for an LMR. For iPWRs, the NRC staff is establishing a set of Design-Specific Review Standards (DSRS) for each design that addresses the unique characteristics of the design and operation.

Currently there are draft DSRS for the mPower design that are out for public comment. Table B-1 addresses the general applicability of the existing SRPs to SMRs (the applicability to an individual SMR would have to be addressed after the design is well established).

2.3.2 Regulatory Guides

Regulatory Guides (RGs) are issued by the NRC to provide license applicants with a methodology, approach, and consensus technical standards that are broadly acceptable to the Commission in determining whether a proposed facility meets applicable NRC regulations. Regulatory Guides are advisory in nature, not regulations; applicants are free to suggest their own technical approaches toward complying with rules and regulations of the Commission. Currently there are 221 RGs for power reactors. Some of these are specific to operating LWRs, while others are technology-neutral and could apply to any reactor design. Appendix B, Table B-2 provides a list of the operating reactor RGs and addresses their general applicability to SMRs (the applicability to an individual SMR would have to be addressed after the design is well established).

Regulatory Guide 1.70 [4] provides the format and content of the Safety Analysis Report (SAR) that the license applicant for a new plant must provide to the NRC. The principal purpose of the SAR is to inform the Commission of the nature of the plant, the plans for its use, and the safety evaluations that have been performed to evaluate whether the plant can be constructed and operated without undue risk to the health and safety of the public. The SAR is the principal document for the applicant to provide the information needed to understand the basis on which this conclusion has been reached, it is the principal document referenced in the Construction Permit or Operating License that describes the basis on which the permit or license is issue, and it is the basic document used by NRC inspectors to determine whether the facility is being constructed and operated within the licensed conditions.

2.3.3 Interim Staff Guidance

Interim Staff Guidance (ISGs) are documents issued to clarify or to address issues not addressed in the SRP. The SRP and other guidance documents have been developed to enhance the NRC license review process. For new reactors, it is expected that guidance to NRC reviewers and license applicants may need to be modified to capture new insights or address emerging issues. In addition, in using the SRP or its references, the NRC staff, industry, or other stakeholders may discover guidance that is unclear, incorrect, or incomplete, or may find that new guidance is warranted. The issuance of ISGs provides a means to quickly address specific areas in guidance documents that need to be revised and serve as temporary guidance until the documents can be revised.

A list of ISGs associated with obtaining a reactor license for new reactors is provided in Table 2-1. Additional ISGs related to the use of digital instrumentation and control and for emergency response are also provided in Table 2-1. All of these ISGs are potentially applicable to SMRs.

Table 2-1. Interim Staff Guidance Associated with Licensing New Reactors.	
Interim Staff Guidance (ISG)	**Topic**
DC/COL-ISG-1	Interim Staff Guidance On Seismic Issues of High Frequency Ground Motion
DC/COL-ISG-2	Interim Staff Guidance on Financial Qualifications of Applicants For Combined License Applications
DC/COL-ISG-3	PRA Information to Support Design Certification and Combined License Applications
COL/ESP-ISG-4	Interim Staff Guidance on the Definition of Construction and on Limited Work Authorizations
DC/COL-ISG-5	GALE86 Code for Calculation of Routine Radioactive Releases in Gaseous and Liquid Effluents to Support Design Certification and Combined License Applications
DC/COL-ISG-6	Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications
DC/COL-ISG-7	Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures
DC/COL-ISG-8	Necessary Content of Plant-Specific Technical Specifications
DC/COL-ISG-010	Review of Evaluation To Address Adverse Flow Effects in Equipment Other Than Reactor Internals
DC/COL-ISG-011	Finalizing Licensing-basis Information
DC/COL-ISG-013	Assessing the Radiological Consequences of Accidental Releases of Radioactive Materials from Liquid Waste Tanks for Combined License Applications
DC/COL-ISG-014	Assessing the Radiological Consequences of Accidental Releases of Radioactive Materials from Liquid Waste Tanks in Ground and Surface Waters for Combined License Applications
DC/COL-ISG-015	Post-Combined License Commitments
DC/COL-ISG-016	Staff Guidance on Interim Staff Guidance DC/COL-ISG-016 – Compliance With 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d)
DC/COL-ISG-017	Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses
DC/COL-ISG-018	Interim Staff Guidance on Standard Review Plan, Section 17.4, Reliability Assurance Program
DC/COL-ISG-019	Gas Accumulation Issues in Safety Related Systems
DC/COL-ISG-020	Seismic Margin Analysis for New Reactors Based on Probabilistic Risk

Table 2-1. Interim Staff Guidance Associated with Licensing New Reactors.	
Interim Staff Guidance (ISG)	**Topic**
	Assessment
DC/COL-ISG-021	Review of Nuclear Power Plant Designs using a Gas Turbine Driven Standby Emergency Alternating Current Power System
DC/COL-ISG-022	Impact of Construction (Under a Combined License) of New Nuclear Power Plant Units on Operating Units at Multi-Unit Sites
DC/COL-ISG-024	Interim Staff Guidance Implementation of Regulatory Guide 1.221 on Design-Basis Hurricane and Hurricane Missiles DC/COL-ISG-024
DC/COL-ISG-025	Interim Staff Guidance COL-ISG-025 on Changes during Construction Under Title 10 of the Code of Federal Regulations Part 52
DI&C-ISG-01	Interim Staff Guidance on Digital Instrumentation and Control, Cyber Security, December 31, 2007
DI&C-ISG-02	Revision 2, Interim Staff Guidance on Diversity and Defense-in-Depth Issues, Revision 2, June 5, 2009
DI&C-ISG-03	Interim Staff Guidance on Review of New Reactor Digital Instrumentation and Control Probabilistic Risk Assessments
DI&C-ISG-04	Interim Staff Guidance on Highly-Integrated Control Rooms – Communications Issues (HICRc), Revision 1, March 2009
DI&C-ISG-05	Interim Staff Guidance on Highly Integrated Control Rooms - Human Factors Issues (HICR-HF), Revision 1
DI&C-ISG-06	Licensing Process Interim Staff Guidance**,** January 19, 2011
DI&C-ISG-07	Digital Instrumentation and Control Systems in Safety Applications at Fuel Cycle Facilities, December 1, 2010 Interim Staff Guidance on Digital Instrumentation and Control Systems in Safety Applications at Fuel Cycle Facilities, June 1, 2009
NSIR/DPR-ISG-01	Interim Staff Guidance on Emergency Planning for Nuclear Power Plants

2.4 Consensus Codes and Standards

In designing any nuclear reactor or fuel-cycle facility, designers have traditionally relied on a large number of codes and standards, many of which are consensus documents arrived at using the consensus process of the American National Standards Institute (ANSI) and other standards development organizations. These codes and standards have been developed and adopted by governmental agencies, either regulatory or developmental agencies like the NRC or Department of Energy (DOE).

The NRC works with standard development organizations (SDOs) to develop consensus standards for systems, equipment, and materials used in nuclear power plants. A list of SDOs, both domestic and international, that provide consensus standards used in the NRC regulatory process is provided in Table 2-2 (adapted from [5]). A standards development organization or SDO is a domestic or international organization that develops consensus standards. A standards committee can include technical experts from a wide variety of fields including academia, industry, government agencies, consultants, and national laboratories.

Table 2-2. List of Standard Development Organizations Cited in Regulatory Documents.

Standard Development Organization	Acronym
Acoustical Society of America	ASA
American Concrete Institute	ACI
American Institute of Steel Construction	AISC
American National Standards Institute	ANSI
American Nuclear Society	ANS
American Public Health Association	APHA
American Petroleum Institute	API
American Society of Civil Engineers	ASCE
American Society of Heating, Refrigerating, and Air Conditioning Engineers	ASHRAE
American Society of Mechanical Engineers	ASME
American Society of Non-Destructive Testing	ASNDT
American Society for Quality Control	ASQC
ASTM International (formerly, American Society for Testing and Materials)	ASTM
Applied Technology Council	ATC
American Water Works Association	AWWA
American Welding Society	AWS
Crane Manufacturers Association of America	CMAA
Diesel Engine Manufacturers Association	DEMA
Factory Mutual	FM
Health Physics Society	HPS
Heat Exchanger Institute	HEI

Table 2-2. List of Standard Development Organizations Cited in Regulatory Documents.

Standard Development Organization	Acronym
IEEE (formerly, Institute of Electrical and Electronic Engineers)	IEEE
Insulated Cable Engineers Association	ICEA
International Commission on Radiological Protection	ICRP
International Commission on Radiological Units and Measurements	ICRU
National Electrical Testing Association	NETA
International Organizations for Standardization	ISO
International Society of Automation (formerly, Instrument Society for Measurement and Control, Instrument Society of America)	ISA
Manufacturers Standardization Society	MSS
Military Standards	MIL
National Association of Corrosion Engineers	NACE
National Concrete Masonry Association	NCMA
National Council on Radiation Protection and Measurements	NCRP
National Electrical Manufacturers Association	NEMA
National Fire Protection Association	NFPA
National Institute of Standards and Technology	NIST
Nuclear Energy Institute	NEI
Underwriters Laboratories	UL

The NRC refers to a few voluntary consensus standards in NRC regulations (see Table 2-3 which is adapted from [5]) documented in the Code of Federal Regulations (CFR). There are several advantages to referencing consensus standards in regulations. The use of standards provides the level of regulatory certainty and predictability desired by stakeholders, they reflect a broad range of technical expertise and experience of the individuals who participate in the development of consensus standards, and they reduce the need to develop regulations to the level of detail comparable to that provided by consensus standards.

Table 2-3. Consensus Standards Required by the NRC for Nuclear Power Plant Licensing.

10 CFR Section	Standard			
	SDO	Number	Date	Title
50.55a	ASME	Boiler and Pressure Vessel Code (BPVC) Section III Division 1	1998 Ed. w/2000 Addenda	"Rules for Construction of Nuclear Power Plant Components" (Classes 1, 2, and 3)
50.55a	ASME	BPVC Section XI Division 1	1998 Ed. w/2000 Addenda	"Rules for Inservice Inspection of Nuclear Power Plant Components" (Classes 1, 2, 3, MC, and CC)
50.55a	ASME	OM Code	1998 Ed. w/2000 Addenda	"Code for Operation and Maintenance of Nuclear Power Plants"
50.55a(h)	IEEE	279	1971	"Criteria for Protection Systems for Nuclear Power Generating Stations"
50.55a(h)	IEEE	603	1991, including correction sheet 1/30/95	"Criteria for Safety Systems for Nuclear Power Generating Stations"
50.55a	ASME	BPVC Section III Code Cases	Rev. 32	Code Cases referenced in Regulatory Guides 1.84 and 1.85
50.55a	ASME	BPVC Section XI Code Cases	Rev. 13	Code Cases referenced in Regulatory Guides 1.147
50.61	ASME	BPVC Section III	1998 Ed. w/2000 Addenda	NB-2331, "Materials for Vessels"
50 App G	ASME	BPVC Section III Appendix G	1999 Ed. w/2000 Addenda	"Fracture Toughness Requirements"
50 App G	ASME	BPVC Section XI Appendix G	2000 Ed. w/2000 Addenda	"Fracture Toughness Requirements"
50 App H	ASME	E185	1982	"Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels
50 App J	ANSI	45.4	1972	"Leakage Rate Testing of Containment Structures for Nuclear Reactors"
50 App J	ANS	56.8	1987	"Containment Systems Leakage Testing Requirements"
50 App K	ANS	ANS-5	1971	"Decay Energy Release Rates Following Shutdowns of Uranium-Fueled Thermal Reactors"
50 App R	IEEE	384	1974	"IEEE Standard Criteria for Independence of Class1E Equipment and Circuits"

The NRC staff also issues documents such as regulatory guides that provide guidance on acceptable methods for complying with NRC regulations. These guidance documents frequently reference consensus standards as acceptable methods for compliance with NRC regulations. As mentioned previously, there are over 4000 citations of standards in NRC documents; over 300 are cited in the NRC Standard Review Plan (NUREG-0800) [3]. The NRC has updated the Standard Review Plan (SRP) in preparation for reviewing licenses for new reactors. As part of this SRP update and development program, a list of industry consensus codes and standards and other government and industry guidance referred to in the SRP and other regulatory documents (CFR, Regulatory Guides, NRC Bulletins, Information Notices, Circulars, Generic Letters, and Policy Statements) was developed in NUREG/CR-5973 [6]. A list of the consensus standards identified in the SRP is provided in Appendix C. The applicability of the standards provided in NUREG/CR-5973 to SMRs has not been determined as part of this report.

2.5 NUREGs

In addition to guidance provided in the SRP, RGs, and ISGs, the NRC provides guidance on specific issues in reports and brochures prepared by both the staff and by contractors. These are commonly referred to as NUREG reports. These are reports or brochures containing information on regulatory decisions, results of research, results of incident investigations, and other technical and administrative information. NUREGs can be referred to in other guidance documents such as the SRP. The depth and breadth of the subject matter covered in these documents is large and mostly is geared towards issues dealing with existing operating reactors. For that reason, only specific NUREGs dealing with the subject matter covered in this report are identified in Sections 3 through 7.

3. APPLICABILITY OF EXISTING OPERATION REQUIREMENTS TO ADVANCED SMRS

This section presents the regulatory requirements pertaining to the operating staff at nuclear power plants. The number of operators is generally determined by requirements for normal operation of the plant. The number of plant personnel required for emergency response following an accident is discussed in Section 7 of this report. NRC documents discussing staffing issues related to SMRs are also presented.

3.1 Regulations Related to NPP Operations

The following NRC regulations relate to operations and staffing of commercial power reactors, in particular, analyses and plans that must be carried out by the licensee to demonstrate to the NRC that the plant can be designed, constructed, operated, and decommissioned in a manner that provides for adequate protection of public health and safety and the safety of the workers and the environment:

1. 10 CFR Part 20 "Standards for Protection Against Radiation" as it relates to limits on radiation doses to members of the public and workers from normal operation of licensed facilities.
2. 10 CFR Part 50 Section 50.34(a)(1), and (a)(2) through (a)(8) "Contents of Applications; technical information" as it relates to the description and discussion of operation characteristics and a preliminary plan for the applicant's organization, training of personnel, and conduct of operations, that must be provided in the Preliminary Safety Analysis Report (PSAR) as part of the license application.
3. 10 CFR Part 50.36 "Technical specifications" as it relates to the establishment of technical specifications that set forth limits, operation conditions, and other requirements to be imposed upon the facility operations for the protection of public health and safety.
4. 10 CFR Part 50.40(b) "Common Standards" and 10 CFR Part 50.80 "Transfer of License," as they relate to establishing requirements that an applicant demonstrate suitable manpower estimates to support reactor operations.
5. 10 CFR Part 50.54 (i), (i-1), (j), (k) and (m) "Conditions of licenses" as it relates to the requirements that nuclear units must be operated by licensed operators and to staffing level requirements for control room operators at nuclear units.
6. 10 CFR Part 50 Appendix A "General Design Criteria" which establishes principal design criteria for safety of operating light water reactors (LWRs), as it relates to general design criteria that impose design requirements to ensure that specific systems and features can be inspected and tested, and that a control room be provided from which operators can operate the nuclear power unit safely under normal conditions and to maintain a safe condition under accident conditions.
7. 10 CFR Part 55 "Operators' Licenses" as it relates to the establishment of procedures and criteria for issuing licenses to operators and senior operators.

3.2 Sub-tier Guidance Documents Related to NPP Operations

Guidance for meeting the regulations cited above is provided in several documents discussed in the following sections. This includes Section 13 of the SRP and several NUREGs which are discussed in Section 3.2.1. NRC discussions on staffing in SMRs provided in SECYs to the Commission are summarized in Section 3.2.2.

3.2.1 Standard Review Plan and Other NUREGs

NUREG-0800: Standard Review Plan

As noted in Section 2.3.1, The Standard Review Plan (SRP), NUREG-0800 [3], provides guidance to NRC staff in performing safety reviews of construction permit (CP) or operating license (OL) applications (including requests for amendments) under 10 CFR Part 50 and early site permit (ESP), design certification (DC), combined license (CL), standard design approval (SDA), or manufacturing license (ML) applications under 10 CFR Part 52 (including requests for amendments). The SRP is intended to be a comprehensive and integrated document that provides NRC reviewers with guidance that describes methods or approaches that the staff has found acceptable for meeting NRC requirements. The SRP is not a substitute for NRC regulations, and compliance with the SRP is not required. However, the SRP generally describes an acceptable means of meeting the regulations, but not necessarily the only means. Applications may deviate from the acceptance criteria in the SRP.

SRP Section 13 "Conduct of Operations" provides guidance for reviewing an applicant's organizational positions and staffing plans for operating the unit, including:

1. Management and Technical Support Organization.
2. Operating Organization
3. Reactor operator Requalification Program; Reactor Operator Training
4. Non-Licensed Plant Staff Training
5. Emergency Planning
6. Operational Programs
7. Administrative Procedures
8. Physical Security

The most prescriptive staffing requirements are for reactor operators, both on site and in the control room during at-power operations, and for certain non-license personnel. The requirements for minimum staffing of licensed reactor operators and are established in 10 CFR Part 50.54 (m)(2)(i). Recommendations for minimum non-licensed personnel are established in II.C of SRP 13.1.2 – 13.1.3 according to guidance in NUREG-0737, *"Clarification of TMI Action Plan Requirements"* [7]. These are shown in Table 3.1. In addition to the personnel noted in Table 3.1, II.C of SRP 13.1.2 – 13.1.3 requires that a health physics technician and a radiation chemical technician be on site during power operations.

Table 3-1. Minimum Requirements[1] Per Shift for On-Site Staffing of Nuclear Power Units.

Number of nuclear power units operating[2]	Position	One Unit	Two units		Three units	
		One control room	One control room	Two control rooms	Two control rooms	Three control rooms
None	Senior Operator	1	1	1	1	1
	Operator	1	2	2	3	3
	Auxiliary (Non-Licensed) Operator	1	3	3		
One	Senior Operator	2	2	2	2	2
	Operator	2	3	3	4	4
	Auxiliary Operator	2	3	3		
Two	Senior Operator		2	3	3[3]	3
	Operator		3	4	5[3]	5
	Auxiliary Operator		3	4		
Three	Senior Operator				3	4
	Operator				5	6
	Auxiliary Operator				[4]	[4]

[1]Temporary deviations from the numbers required by this table shall be in accordance with criteria established in the unit's technical specifications.
[2]For the purpose of this table, a nuclear power unit is considered to be operating when it is in a mode other than cold shutdown or refueling as defined by the unit's technical specifications.
[3]The number of required licensed personnel when the operating nuclear power units are controlled from a common control room are two senior operators and four operators.
[4] Shift staffing of unlicensed personnel for special cases such as three units, operating from one or two control rooms, etc., will be determined case by case, based on the principles defined in item II.B.3 of SRP 13.1.2 – 3.1.3.

In contrast to the exactly prescribed control room staffing requirements of SRP 13.1.2 – 13.1.3, SRP 13.1.1 "Management and Technical Support Organization" gives guidance that an applicant's Safety Analysis Report (SAR) should *"give the approximate numbers of…each identified position or class of positions providing technical support for plant operations,…"* The SRP defines the following areas for which the SAR should give such information as:

- Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical, materials, and instrumentation and controls engineering.

- Plant chemistry,
- Health physics,
- Fueling and refueling operations support,
- Maintenance support,
- Operations support,
- Quality assurance,
- Training,
- Safety review,
- Fire protection,
- Emergency organization,
- Outside contractual assistance.

An example of how these staffing requirements and guidelines can be manifested in the license for a nuclear power plant is shown in Table 3-2 and 3-3, which are taken from the Vogtle Final Safety Analysis Report (FSAR) [8]. Table 3-2 shows compliance with the requirements for minimum required staffing during at-power operations for licensed operators as specified in 10 CFR 50.54 (m) and the guidance for non-licensed operators and other required personnel as specified in II.C of SRP 13.1.2 – 13.1.3. Table 3-3 shows the estimated operations department staff, in compliance with 10 CFR 50.40(b) and 10 CFR 50.80.

Table 3-2. Vogtle Electric Generating Plant, Units 3 & 4 CL Application Minimum On-Duty Operations Shift Organization for Two-Unit Plant.

Units Operating	Staffing Requirements for Two Units, Two Control Rooms
All units Shutdown	1 Shift Manager (Licensed Senior Reactor Operator) 2 Licensed Reactor Operators (RO) 3 non-Licensed Operators
One Unit Operating*	1 Shift Manager (Licensed Senior Reactor Operator) 1 Senior Reactor Operator (SRO) 3 Licensed Reactor Operators 3 non-Licensed Operators
Two Units Operating*	1 Shift Manager (Licensed Senior Reactor Operator) 2 Senior Reactor Operator 4 Licensed Reactor Operators 4 non-Licensed Operators

* Operating modes other than cold shutdown or refueling.

Notes to Table 3-2:

1. In addition, one Shift Technical Advisor (STA) is assigned per shift during plant operation. A shift manager or another Senior Reactor Operator (SRO) on shift, who meets the qualifications for the combined SRO/STA position as specified for option 1 of Generic Letter 86-04 may also serve as the STA. If this option is used for a shift, then the separate STA position may be eliminated for that shift.

2. In addition to the minimum shift organization above, during refueling a licensed SRO or limited Senior Reactor Operator (limited to fuel handling only) is required to directly supervise any core alteration activity.
3. A shift manager/supervisor (SRO licensed for each unit that is fueled), shall be on site at all times when at least one unit is loaded with fuel.
4. A radiation protection technician shall be onsite at all times when there is fuel in a reactor.
5. A chemistry technician shall be on site during plant operation in modes other than cold shutdown or refueling.
6. To operate, or supervise the operation of more than one unit, an operator (SRO or Reactor Operator) must hold an appropriate, current license for each unit.

Table 3-3. Vogtle Electric Generating Plant, Units 3 & 4 CL Application FSAR Estimated Operations Department Staffing Compliance with NUREG-0800 Section 13.1.1

Nuclear Function	Function Position (from ANSI/ANS-3.1-1993)	Positions for 1st unit.	Additional positions for 2nd unit.
Executive management	Chief executive officer	1	-
	Executive, nuclear generation and development	1	-
Nuclear support	Executive, operations support	1	-
Plant management	Executive	1	-
	Plant manager	1	-
Engineering	Executive	1	-
	Manager	1	-
System engineering	Functional manager	-	
	Systems engineer	23	10
Design engineering	Function manager	1	-
	Design engineer	15	2
Safety and engineering analysis	Functional manager	1	-
	Analysis engineer	3	-
Reactor engineering	Functional manager	1	-
	Reactor engineer	3	1
Engineering support	Functional manager	1	-
Maintenance	Manager	1	
Instrumentation and control	Functional manager	1	-
	Supervisor	4	4
	Technician	30	15
Mechanical	Functional manager	1	-
	Supervisor	2	-
	Technician	40	15

Table 3-3. Vogtle Electric Generating Plant, Units 3 & 4 CL Application FSAR Estimated Operations Department Staffing Compliance with NUREG-0800 Section 13.1.1

Nuclear Function	Function Position (from ANSI/ANS-3.1-1993)	Positions for 1st unit.	Additional positions for 2nd unit.
Electrical	Functional manager	1	-
	Supervisor	2	-
	Technician	20	10
Support	Functional manager	1	-
Operations	Manager	1	
Operations, daily	Functional manager	1	-
Operations, support	Functional manager	1	-
Operations, (on-shift)	Functional manager	5	5
	Supervisor	5	5
	Support Supervisor	5	5
	Licensed operator	10	10
	Non-licensed operator	30	30
	Shift technical advisor	5	5
Operations, outage	Functional manager	-	
Fire protection	Supervisor	1	-
Radiation protection	Functional manger	1	-
	Supervisor	5	-
	Technician	18	9
	ALARA specialist	3	1
Chemistry	Functional manager	1	-
	Supervisor	5	-
	Technician	12	12
Nuclear safety assurance	Manager	1	-
Licensing	Functional manger	1	-
	Licensing engineer	6	-
Corrective action	Functional manager	1	-
	Corrective action specialist	2	1
Emergency preparedness	Functional manager	1	-
	Planner	2	-
Training	Functional manager	1	-
	Supervisor operational training	1	1
	Operational training instructor	6	6
	Supervisor technical staff-maintenance training	1	-

NUREG-0737: Clarification of TMI Action Plan Requirements

NUREG-0737, released in 1980, contains post-TMI requirements that where approved by the Commission for implementation. With regard to staffing requirements, NUREG-0737 established the minimum staffing requirements for at-power reactors that are shown in Table 3.1. NUREG-0737 also established requirements on overtime for shift operators and certain non-licensed personnel.

NUREG-0694: TMI-Related Requirements for Operating Licenses

NUREG-0694 [9] establishes the requirements for minimum site staffing requirements that are clarified in NUREG-0737.

NUREG-1791: Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)

The NRC acknowledges in NUREG-1791 [10] that current regulations regarding control room staffing, which are based upon the concept of operation for existing light-water reactors, may no longer apply. Therefore, applicants for an operating license for an advanced reactor, and current licensees who have implemented significant changes to existing control rooms, may submit applications for exemptions from current staffing regulations. The NRC staff is responsible for reviewing the exemption requests and must determine whether the staffing proposals provide adequate assurance that public health and safety will be maintained at a level that is comparable to that afforded by compliance with the current regulations.

The purpose of the staffing plan review approach documented in this NUREG is to ensure that the applicant has systematically analyzed the requirements for the numbers of qualified personnel that are necessary to operate the plant safely under the operational conditions analyzed. That is, the staffing plan should answer the question, “How many individuals must be qualified and available to fill each job?”

The NRC contends that the approach in this NUREG is consistent with Chapter 18.0 “Human Factors Engineering” of the SRP and Rev. 2 of NUREG-0711 [11], “Human Factors Engineering Program Review Model.” It is also consistent with NUREG/IA-0137 [12], “A Study of Control Room Staffing Levels for Advanced Reactors.”

The types of information and data that the NRC expects to be considered in a regulatory review are outlined, and the reviewer is provided a systematic process to perform the review. As such, this NUREG provides useful guidance as well for the applicant. Eleven steps are defined and guidance and acceptance criteria for each step are given. NUREG-0711 is referenced in nine of these steps as providing more detailed guidance for specific aspects of those steps.

NUREG-0711: Human Factors Engineering Program Review Model

This document is used by the NRC staff to review the human factors engineering (HFE) programs of applicants for construction permits, operating licenses, standard design certifications, combined operating licenses, and for license amendments. The purpose of these reviews is to verify that accepted HFE practices and guidelines are incorporated into the applicant's HFE program, including staffing. Chapter 6 provides guidelines and criteria from SRP Section 13.1 and 10 CFR 50.54 for staff review of an applicant's analysis used to determine staffing levels in license applications.

NUREG/IA-0137: A Study of Control Room Staffing Levels for Advanced Reactors.

This report documents the results of an empirical study of operator and plant performance in simulator based settings. The simulator settings were designed to be representative of conventional and advanced plants. The advanced plant design employed passive systems. The control room architectures were also designed to represent both plant types. Two control room staffing configurations were employed in each plant setting: a staffing configuration reflecting the requirements of 10 CFR 50.54 (m); and a staffing configuration that involved a reduced number of control room operators. This report documents the study and discusses the implications and issues raised by this performance-based evaluation of control room staffing requirements for advanced passive plants.

NUREG-6838: Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)

The purpose of NUREG/CR-6838 [13] document is to describe the technical basis for a set of technically sound methods that the NRC staff may use for evaluating requests for exemption from current control room staffing regulations. These methods focus on the development and validation of staffing plans for personnel who are responsible for the safe operation of a nuclear power facility.

A sample of human performance measures is provided in the appendices of NUREG-6838 that may be used for staffing plan evaluation as well as a more in depth discussion of state-of-the-art methods for assessing situation awareness and cognitive workload.

3.2.2 NRC Documents On SMR Staffing

SECY-02-0180: Legal and Financial Policy Issues Associated with Licensing New Nuclear Power Plants

In SECY-02-0180 [14], the NRC staff acknowledges that current regulations do not address the possibility of more than two reactors being controlled from one control room, and that the safety implications of such an arrangement would have to be demonstrated. The staff also acknowledges that applicants could request an exemption to current staffing requirements to allow an alternate level of operator staffing for modular reactors, provided they address the safety implications.

SECY-10-0034: Potential Policy, Licensing, And Key Technical Issues For Small Modular Nuclear Reactor Designs

In public preapplication activities with the DOE and with individual SMR designers to discuss potential policy, licensing, and key technical issues for SMR designs, the NRC has identified the application of current on-site operator staffing requirements to small or multi-module facilities as a key licensing issue. This is acknowledged in SECY-10-0034 [15]. The NRC staff commits to studying this issue in SECY-10-0034 and state that if necessary, changes to existing regulator guidance concerning the operator staffing for SMRs would be developed.

No specific changes or resolutions are identified in SECY-10-0034, but the staff does acknowledge that current staffing requirements and guidance may need to be rethought for SMRs.

SECY-11-0098: Operator Staffing For Small Or Multi-Module Nuclear Power Plant Facilities

SECY-11-0098 [16] provides the Commission with the staff's proposed approach to resolving the issue of the appropriate number of on-site licensed operators and potential requests for exemptions from the on-site operator staffing requirements. The staff proposed to use a two-step approach to address operator staffing requirements for SMRs:

1. In the near-term, applicants could request exemptions to the current operator staffing requirements in 10 CFR 50.54(m) and the staff would review the request using existing or modified guidance.
2. Once experience is gained, the staff would initiate the long-term solution which is to revise the regulations to provide specific control room staffing requirements for SMRs.

To support this approach, the NRC established a separate internal working group to undertake the following activities:

- oversight of revisions to NUREG-0800, NUREG-0711, and NUREG-1791
- identification of differences in the advanced reactor designs that could impact operator performance and in turn impact staffing levels
- oversight of any new issues identified by the Commission, following their review of the Near-Term Task Force's "Recommendations for Enhancing Reactor Safety in the 21st Century" [17]
- oversight of proposed regulatory revisions

The staff committed to continued engagement with external stakeholders to discuss the near- and long-term solutions and will further develop the specific aspects involved in both solutions.

3.3 Summary of Operations Requirements Applicability to Advanced SMRs

3.3.1 Summary of Main Control Room Regulatory Issues

In summary, the NRC is faced with a situation in which the design features and concepts of operations for new generations of advanced reactors, as well as the introduction of new automated systems into existing plants, may lead applicants to request variations in the prescribed number, composition, or qualifications of licensed personnel. Unless and until the NRC develops and establishes new regulations for main control room (MCR) staffing, applicants will have to submit exemption requests from applicable regulations included primarily in 10 CFR 50.54(m).

The current requirements for operator staffing outlined in 10 CFR 50.54(m) prescribe the number of operators required per unit and per control room. As an example, for three operating nuclear power units, the minimum staffing table in 10 CFR 50.54(m)(2)(i) assumes there are at least two control rooms, and mandates a total of eight licensed operators. The number of licensed operating personnel that the rule prescribes are based on assumptions and operating experience from the operation of large light-water reactors. The staffing requirements of 10 CFR 50.54(m)(2)(i) are illustrated in Table 3.1, and an example of how these have been implemented at a large advanced reactor (AP1000) is shown from the Vogtle CL application in Table 3-2. Vogtle did not apply for an exemption from the regulations. Furthermore, the regulation does not address a situation where three or more units are controlled from a single control room. Such a configuration is a possibility with certain SMR designs.

3.3.2 Proposed NRC Strategy on Control Room Staffing

The NRC has acknowledged that there may be a case for exemptions from the MCR staffing regulations for advanced reactor designs, and that the regulations would not address all of the potential MCR configurations that may be proposed (e.g., more than two reactors controlled from the same MCR). This is documented in SECY-02-0180, SECY-11-0098, and in NUREG-1791. In SECY-11-0098, the NRC staff propose a policy to address this issue in both the near-term (i.e., no advanced SMR operating experience) and long term (i.e., subsequent to operational experience of licensed SMRs). This policy would involve evaluating requests for exemptions from 10CFR 54 (m) from both DC and CL applicants prior to the existence of SMR operating experience. Eventually, after operating experience of SMRs has been gained, the NRC would revise existing regulations and develop other regulations to provide specific control room staffing requirements for SMRs.

As outlined in SECY-11-0098, the NRC staff intends to pursue the short term strategy by revising the Standard Review Plan (NUREG-0800) and the two NUREG documents that deal directly with staff evaluation of control room staffing issues and related human engineering aspects, NUREG-0711 and NUREG-1791 (summarized above in Section 3.2). The NRC intends to revise these staff guidance documents so that they identify and address to the extent possible differences in advanced reactor designs that might impact conduct-of-operations, operator performance and staffing levels.

3.3.3 Issues Relevant to Proposed NRC Strategy on Control Room Staffing

In NUREG/IA-0137, the results of studies of variable crew sizes in different control room configurations (conventional versus advanced plant MCR designs). The study's conclusions are that advanced MCR designs minimum sized crews performed better than did normal sized crews. Hence, the report recommends that decisions about control room staffing should be based upon design features including function allocation, automation, integration, and plant-specific characteristics (e.g., passive system performance).

The Nuclear Energy Institute (NEI), in its position paper on control room staffing for SMRs [18] finds that the NRC's staff guidance for reviewing an applicant's Human Factors Engineering (HFE) program (NUREG-0711) provides a reasonable basis for analysis, design, verification and validation, and implementation of the HFE process. To work towards the evolution of new requirements for SMR control room staffing, the NEI advocates for the NRC to work with industry to establish standardized and accepted levels of operator workload by testing existing acceptable operating plants so that these standards of mental and physical workloads would be available for detailed analysis and evaluation of the task analysis required for new SMR designs.

One SMR vendor, NuScale, has released a regulatory gap analysis [19]. As part of that survey of NRC regulatory licensing requirements, NuScale proposes to apply for an exemption to the minimum licensed operator staffing requirements of 10CFR 50.54 (m)(2)(i) and (iii) based on a NuScale design specific staffing plan that will be developed using the guidance of NUREG-0711 and NUREG-1791.

In summary, both the NRC and the nuclear power industry illustrate an awareness that new technologies in both SMR reactor systems and in the design of new control rooms will lead to the need to seek reasonable assessments of realistic SMR staffing needs rather than blind adherence to the current regulations set for in 10CFR 50.54 (m).

4. APPLICABILITY OF EXISTING SAFETY REQUIREMENTS TO ADVANCED SMRS

To be licensed by the NRC, an SMR must meet the safety requirements in the Code of Federal Regulations. These requirements are primarily provided in 10 CFR 50. The safety case for a NPP is made in the FSAR which must contain the information specified in 10 CFR Part 50 Section 50.34(a)(1). Proof of compliance with radiation dose limits, during both normal operations and following an accident, specified 10 CFR Parts 20 and 100 is required as part of the licensing process. As indicated in Chapter 2, Parts 20 and 100 are applicable to all reactors and thus are not discussed further.

4.1 NPP Safety Regulations

The following NRC regulations relate to health and safety impacts of commercial power reactors, in particular, analyses that must be carried out by the licensee to demonstrate to the NRC that the plant can be designed, constructed, operated, and decommissioned in a manner that provides for adequate protection of public health and safety and the safety of the workers and the environment:

1. Part 50 – Domestic Licensing of Production and Utilization Facilities. All operating LWRs in the US were licensed under Part 50 which contains many LWR-specific requirements. Paragraph 50.34, specifies the “Contents of applications; technical information” that must be supplied in a license submittal. Many sections of Part 50 are LWR specific and as such, many of the regulations in Part 50 are not applicable to LMRs. For example, § 50.46 provides the acceptance criteria for the emergency core cooling systems for LWRs. In addition, because of the unique nature of iPWRs, some of the existing regulations may not be applicable
2. 10 CFR Part 50 Section 50.34(a)(1) “Contents of Applications; technical information” relates to the evaluation and analysis of the offsite radiological consequences of postulated accidents with a fission product release that must be provided in the Preliminary Safety Analysis Report (PSAR) as part of the license application.
3. 10 CFR Part 50 Appendix A “General Design Criteria” establishes principal design criteria for safety of operating LWRs. However, some General Design Criteria (GDCs) can be usefully employed for other reactor technologies, such as LMRs (see Section 4.1.1).

4.1.1 General Design Criteria

Appendix A to 10 CFR Part 50 list 55 GDCs subdivided into the following categories:

- General Requirements (5 criteria)
- Protection by Multiple Fission Product Barriers (10 criteria)
- Protection and Reactivity Control Functions (10 criteria)
- Fluid Systems (17 criteria)
- Reactor Containment (8 criteria)

- Fuel and Radioactivity Control (5 criteria)

The general applicability of the GDCs to SMRs is provided in Appendix A of this report, Table A-4. Since the existing GDCs were generated for LWRs, they are all applicable to iPWRs. However, most are technology neutral. For LMRs, ten of the existing GDCs in the Fluid Systems category can be excluded for LMRs because they are LWR-specific. Excluding these, the remainder should be broadly applicable to LMRs (in fact, to all reactor technologies) with some changes in wording as appropriate.

In the 1970s the American Nuclear Society developed a specific set of GDCs for LMRs that was published in a standard in 1989 [20]. The standard maintained the existing GDCs in 10 CFR 50 Appendix A if possible, but also incorporated changes and additions to reflect the unique characteristics of LMRs. Examples of additional LMR-specific criteria address issues such as protection against sodium and NaK reactions and sodium heating systems. Examples of modified GDCs include a criterion covering inspection and surveillance of the reactor coolant boundary that was expanded to include the boundary of the cover gas system.

Currently, the ANS and ANSI are working on an update of ANSI/ANS 54.1. The revised draft standard defines safety objectives, general design criteria, selection of accident which need to be mitigated, and classification of Systems Structures and Components (SSCs) that can be used by designers and regulators of sodium-cooled reactor nuclear power plants. It is intended to serve a purpose similar to the GDC promulgated by the U.S. NRC for LWRs in 10 CFR 50 Appendix A. It is expected that this draft standard will be useful to the U.S. NRC as it develops the regulatory guidance for sodium-cooled reactor nuclear power plants. This current updated draft standard recognizes changes in LMR technology, safety, and the use of PRA in achieving a risk-informed and performance-based approach to the safe design of a reactor.

For all SMRs, some of the GDCs could be changed or removed if a risk-informed, performance-based approach were to be employed in licensing advanced reactors (e.g., if reliability criteria replace the older single failure criterion) or if technology changes would require a need to consider newer methods of instrumentation and control. In addition, if risk assessment methods, such as PRA, play an important role in establishing the licensing basis of the plant - e.g., in the selection of design basis accidents (DBAs) and the safety classification of SSCs - any criteria related to PRA technical scope and acceptability and the use of risk information in making licensing decisions may necessitate additional GDCs. Examples of subject matter covered by potential new GDCs could include:

- Acceptable levels of plant risk including risks from internally initiated events, internal fires and floods, external hazards, and from all modes of operation;
- Consideration of the impact on risk by plant aging and degradation; and
- Reliability and availability criteria.

4.2 Sub-tier Guidance Documents Addressing NPP Safety

As part of the licensing process, guidance is provided to both the NRC staff and licensees on acceptable ways to meet the regulations. Several types of guidance documents exist including the Standard Review Plan in NUREG-0800, Regulatory Guides, Interim Staff Guidance, and NUREG reports. These types of documents are described in Section 2 of this report.

Many of the safety-related guidance documents are specifically related to staffing requirements for operations and emergency response, as well as for security. Since staffing requirements is an important aspect of operating the plant in a safe condition, specific discussions on the pertinent staffing guidance is provided in Sections 3, 5, 6, and 7.

There are many additional guidance documents that if followed should result in a safe plant design. The sheer number of SRP sections and RGs prevents a detailed discussion on their potential applicability to SMRs. However, lists of the existing SRP sections and RGs and their general applicability to SMRs are provided in Appendix B of this report. Specific chapters of the SRPs related to safety are identified below.

The SRP includes several chapters that address the safe design of the plant. Chapter 3 addresses the design of structures, systems, and components to withstand external hazards such as earthquakes, high winds, aircrafts, and external floods. The unique design of some SMR will make them less susceptible to some of these hazards. For example, siting the reactor and important equipment below ground will reduce the susceptibility of some SMRs to aircraft crashes and earthquakes. The design of the reactor, reactor coolant system, engineered safety features, and the instrumentation and control systems are discussed in Chapters 4, 5, 6, and 7 of the SRP. Since the design of these aspects of the SMRs is different than for LWRs, these SRP chapters will require major modifications. Similarly, the accident analysis and severe accident response for SMRs will be substantially different (addressed in Chapters 15 and 19 of the SRP, respectively). For example, large break LOCAs are not possible in some iPWR designs. As indicated in Section 2.3.1 of this report, the NRC staff is establishing a set of Design-Specific Review Standards (DSRS) for different SMR designs that addresses the unique characteristics of the design and operation.

Regulatory Guides cover a broad range of issues most of which in some way are related to safety. Most of the existing RGs address issues that are technology neutral and thus would apply to SMRs. However, there are many that are LWR specific and thus would not be applicable to LMRs. The general applicability of each RG to SMRs is identified in Table B-1. For those RGs that are applicable, the unique design of SMRs may require that some of the regulatory guidance be modified or that ISGs be generated. In addition, additional RGs may be required.

A list of current ISGs is presented in Table 2-1. Although these ISGs were generated to address issues associated with current guidance for LWRs, it is likely that additional ISGs would be required to address applicability issues with current RGs.

4.3 Summary of Safety Requirements Applicability to Advanced SMRs

Appendix B of this report provides lists of the current SRP and RGs and their potential applicability to SMRs. Although many of these will be applicable to iPWRs, many would not be pertinent to LMRs. Even for iPWRs, their unique design features would require modification of the guidance.

5. APPLICABILITY OF EXISTING SECURITY REQUIREMENTS TO ADVANCED SMRS

To be licensed by the NRC, an SMR must meet the domestic safeguards requirements that address (1) the physical protection (security) of nuclear material and facilities, and (2) material control and accountability (MC&A) for nuclear materials. These requirements are provided in 10 CFR 73 and 10 CFR 74, respectively. In addition, SMRs must obtain a license for possession and use of special nuclear material (SNM) in accordance with the requirements of 10 CFR 70, "Domestic Licensing of special Nuclear Material." NRC requires that licensees of existing operational NPPs and future plants, including SMRs, comply with these requirements. The requirements for physical security are addressed in this section; the requirements for possession and use of SNM and MC&A are addressed in Section 6.

5.1 Regulations for NPP Security

Several parts of the CFR provide regulations related to physical security for NPPs and are applicable also to SMRs. Reactor licensing under 10 CFR 50 and/or 10 CFR 52 include requirements in §§50.34, 50.54, and 52.79 regarding content of license applications and conditions of licenses that address security requirements. Specifically, §50.34(c), §50.34(d), §52.79.35 and §52.79.36 include requirements for reactor license applications to include security plans, including a physical security plan, safeguards contingency plan, training and qualification plan, and cyber security plan, that are prepared in accordance with the criteria in 10 CFR 73. Other parts of the CFR provide regulations related to physical security for NPPs and are applicable also to SMRs. These security plans are submitted to NRC for approval with license applications. In addition, 10 CFR 50 includes guidance for aircraft impact assessments and loss of large areas. These CFR requirements are summarized below.

1. **10 CFR Part 73 – Physical Protection of Plants and Materials.** For NRC facilities, 10 CFR Part 73, "Physical Protection of Plants and Materials," addresses physical security. This regulation prescribes the necessary requirements for the establishment and maintenance of a physical protection system, which will have capabilities for the protection of SNM at fixed sites and in transit, and of plants in which special nuclear material is used. Licensees are responsible for implementing these requirements for the protection of their nuclear material and facilities.

   Specifically applicable for physical protection of power reactors, including SMRs, in accordance with §73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," a licensee must implement these requirements through its NRC-approved Security Plans, including Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Cyber Security Plan. The requirements state the general performance objective and requirements that each licensee must establish and maintain a physical protection program with the objective of providing high assurance that activities involving SNM do not constitute an unreasonable risk to public health and safety. The program must protect against the design basis threat of radiological sabotage and be designed to prevent significant core damage and spent fuel sabotage. Additional details for cyber security are provided in §73.54, "Protection of

Computers and Communication Systems and Networks," requires each facility to provide high assurance that digital computer and communications systems and networks are adequately protected against cyber attacks, up to and including the design basis threat, of safety-related and important-to-safety functions, security functions, emergency preparedness functions, including offsite communications, and support systems and equipment. Cyber security protection is particularly applicable to SMR designs that use modern technology and will have fully digital instrumentation and control, and also impacts safety, operations, and MC&A. Appendix D provides a detailed listing of security topical areas for NPPs and where the topic is addressed in 10 CFR 73.

2. **10 CFR Part 50.150 – Aircraft Impact Assessment**. For NRC Facilities, §50.150, "Aircraft Assessment," is a guidance document requiring each facility to perform a design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft.
3. **10 CFR Part 50.54 – Conditions of Licenses (Loss of Large Areas)**. For NRC Facilities, 10 CFR Part 50.54, "Conditions of Licensees-Loss of Large Areas," is a guidance document requiring each facility to develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire.

5.2 Sub-tier Guidance Documents for NPP Security

Sub-tier guidance for NPP security includes SRP sections for license application security reviews and several regulatory guides that address security planning, system design and operations. Instead of ISGs, for security, the NRC issues Security Orders that contain requirements for licensees to implement interim compensatory security measures beyond that currently required by NRC regulations and as conditions of licenses. These three types of sub-tier guidance related to NPP security are summarized in this section.

5.2.1 SRP Guidance for NPP Security

Chapter 13 of the SRP addresses Conduct of Operations and includes several sections that address s security for an LWR reactor license application. The areas of review for physical security are provided in Section 13.6 and vary based on the type of licensing action. Guidance for the security review of each type of licensing action is assigned a separate sub-section of the SRP. In addition, the SRP provides review guidance for access authorization and cyber security planning. The security SRP sections are summarized in the following paragraphs.

For CL applications (SRP 13.6.1 Physical Security–Combined License Applications), the SRP addresses the review of security plans (physical security, training and qualification, and safeguards contingency plans collectively, the Security Plan), which together describe a comprehensive physical security program for CL applicants and operating reactor licensees. The appendices to this SRP section include templates for each of the three security plans developed by the Nuclear Energy Institute and endorsed by the NRC for reactor license applications. The review encompasses the regulatory requirements in 10 CFR 73.55, and Appendix B, "General Criteria for

Security Personnel," and Appendix C, "Nuclear Power Plant Safeguards Contingency Plans," to 10 CFR 73. It includes the physical security organization; access controls, including physical barriers; searches of personnel, materials, and vehicles; a means of detection, assessment, delay, and response; the selection of personnel for security purposes; training and qualifying security personnel; the response to contingency events; and arrangements with law enforcement authorities for assistance in responding to security threats. It also includes a review of the description for implementation of the security plan (implementation schedule) for the physical security program, including phases for a multiunit plant, where applicable.

For DC applications (SRP 13.6.2 Physical Security–Design Certification), the SRP addresses the evaluation of the physical security systems, components, and measures (referred to as "physical security elements") identified to be within the scope of an applicant's design. The DC security review encompasses the material intended to meet the general performance objective described in 10 CFR 73.55(b) as well as guidance provided in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)." The review also addresses CL action items and requirements and restrictions, such as interface requirements and site parameters. The review must include the physical security elements within the physical design of the power reactor and supporting systems that are included in the DC application and may also include a review of the voluntarily submitted physical security elements. The physical security elements that are required within the review of a DC application are the elements that, because of their inherent nature, are included within the physical design of the power reactor and supporting systems. These physical security elements are usually designed, located, and constructed to directly support the protection of equipment essential to the safe operation of the power reactor, and include specifically physical barriers (walls, ceilings, floors, doors, gratings and openings) and vital areas (including a bullet-resistant control room and spent fuel pool) and associated portals and emergency exits. Associated design descriptions for physical barriers and vital areas, a list of vital equipment by component and location in vital areas, and security inspections, tests, analyses, and acceptance criteria (ITAAC) are required. As applicable, the review of the DC application would also include other physical security elements included within the physical design of the power reactor and supporting systems (e.g., the central alarm station (CAS), the secondary alarm station (SAS), and bullet-resistant enclosures).

For ESP applications (SRP 13.6.3 Physical Security – Early Site Permits), the review involves the evaluation of the site characteristics, pursuant to 10 of the CFR 100.21(f) and 52.17(a)(x), to provide reasonable assurance that adequate security plans and measures can be developed to meet the applicable requirements under 10 CFR 73.55, as well as guidance provided in 4.7, Revision 2, "General Site Suitability Criteria for Nuclear Power Stations," issued April 1998.

A new draft SRP section (SRP 13.6.4 Access Authorization Operational Program) addresses the requirements to implement an Access Authorization program through revisions to a Commission-approved Physical Security Plan. The Access Authorization program is established, implemented and maintained in accordance with the requirements of 10 CFR 73.55(b)(7), which integrates the performance requirements contained within 10 CFR 73.56, "Personnel Access Authorization Requirements for Nuclear Power Plants," and the criminal history checks of 10 CFR 73.57, "Requirements for Criminal History Checks of Individuals Granted Unescorted Access to a Nuclear Power Facility or Access to Safeguards Information by Power Reactor Licensees."

For cyber security, (SRP 13.6.6 Cyber Security Plan), the SRP review evaluates the applicant/licensee's plan to provide high assurance that the digital computer and communication systems and networks associated with safety, security, and emergency preparedness functions, as well as support systems and equipment, which if compromised, would adversely impact safety, security, or emergency preparedness functions, are adequately protected against cyber attacks. This requirement to provide a cyber security plan to the NRC for review is in accordance with 10 73.54, "Protection of Digital Computer and Communication Systems and Networks." The applicant/licensees must identify those assets that must be protected against cyber attacks; establish, implement, and maintain a cyber security program for the protection of the assets; and ensure that the cyber security program is incorporated into the physical protection program. The cyber security program must implement security controls to protect Critical Digital Assets from cyber attacks, apply and maintain defense-in-depth protective strategies, mitigate the effects of cyber attacks, and ensure that the Critical Digital Assets are not adversely impacted by the cyber attacks. The scope of the review is programmatic and does not address design information.

5.2.2 Regulatory Guides for NPP Security

The NRC provides regulatory guides pertaining to physical security equipment, systems, and operations. The publically available guides are accessible on the NRC website. These are listed and summarized in Table 5-1. In addition, a few guides include security-related information deemed to be not for public release, but that would be provided to a licensee when their application is initiated. The number and title for these guides are included in Table 5-1, but no description is provided.

Additional guidance is provided in the "Nuclear Power Plant Security Assessment Technical Manual" [21] and its associated format and content guide [22]. The NRC's high assurance evaluation is described in these two guidance documents that are applicable to new NPPs and their spent fuel stored in a pool. These two documents provide guidance for voluntarily performed security assessments of these facilities. The purpose of the security assessment is to demonstrate that the physical protection system for a new reactor facility meets the requirements in 10 CFR 73.55. The technical manual provides a set of best practices for physical protection system design and evaluation and describes modeling and analysis tools currently available and under development for licensee use, including for example, an adversary path evaluation using the Analytic System and Software for Evaluating Safeguards and Security (ASSESS) or the Advanced Time Line Analysis System (ATLAS); response force evaluation using Joint Conflict and Tactical Simulation (JCATS); and table top and field exercises (force-on-force exercises) can be employed to gather data for security assessments.

Table 5-1. Summary of Regulatory Guides for Physical Security.

Guide Number	Regulatory Guide Title	Summary Description
5.7	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	This guide describes measures the NRC staff considers acceptable for implementing 10 CFR 73 requirements for entry/exit control into protected areas, vital areas, and material access areas at nuclear facilities, including NPPs. The physical protection system must control entry and exit of personnel, vehicles, and material to assure only authorized access and to prevent unauthorized access to these areas of an NPP. The measures include controlling access, searching vehicles, personnel, or packages, and detecting attempted theft or diversion of material.
5.12	General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials	This guide provides criteria acceptable to the NRC staff for the selection and use of commercially available locks in the protection of facilities and material. Locks are acceptable devices to be used in adhering to the physical protection requirements identified above to assist in controlling access to areas, facilities, and materials through doors, gates, container lids, and similar material or personnel access points, and are considered essential components of a physical barrier.
5.20	Training, Equipping, and Qualifying of Guards and Watchmen	10 CFR 73 includes requirements for licensees for nuclear facility, including NPPs, to provide trained and equipped guards and watchmen to physically protect their facilities and the SNM in their possession. This guide provides criteria acceptable to the NRC staff for training, equipping, and qualifying guards and watchmen.
5.43	Plant Security Force Duties	This guide describes the necessary requirements to maintain and follow written security procedures that document the structure of the licensee's security organization and that detail those duties of guards, watchmen, and other individuals responsible for security contained in the 10 CFR Part 73 regulations. This guide provides criteria acceptable to the regulatory staff relative to the organization of the plant security force and duties of guards, watchmen, and other individuals responsible for security.

Table 5-1. Summary of Regulatory Guides for Physical Security.

Guide Number	Regulatory Guide Title	Summary Description
5.44	Perimeter Intrusion Alarm System	This guide describes the functions of perimeter intrusion detection sensors and detection methods that are acceptable to the NRC staff for meeting the portions of the NRC's 10 CFR Part 73. It provides guidance on sensors and methods that can be integrated to form an effective perimeter intrusion detection system. This guide provides guidance on selecting perimeter intrusion detection systems and on applications for nuclear power reactors, independent spent fuel storage installations, and certain special nuclear material processing facilities.
5.54	Standard Format and Content and Safeguards Contingency Plans for Nuclear Power Plants	This guide defines a sample contingency plan for safeguards of NPP that outline how the plant will handle safeguards events. The appendices of SRP 13.6.1 include related guidance with templates for security plans.
5.62	Reporting of Safeguards Events	This guidance addresses the reporting safeguards events. This includes the timeframe for each type of incident, the proper reporting methods, etc. Draft Regulatory Guide DG-5019 "Reporting and Recording Safeguards Events" is a proposed Revision 2 to this regulatory guide that will apply to a range of NRC-licensed facilities, including NPPs.
5.65	Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	This guide describes the protection of items of vital equipment at nuclear power reactors are contained in 10 CFR Part 73 regulations. These requirements are aimed at protecting against sabotage that could cause a radiological release. These revised requirements were developed to clarify or modify certain existing physical protection requirements. The amendments were designed to foster plant safety while maintaining adequate security. This guide presents approaches that are acceptable to the NRC staff for implementing the amendments. Emphasis in the guide is on minimizing the safeguards impact on safety.

Table 5-1. Summary of Regulatory Guides for Physical Security.

Guide Number	Regulatory Guide Title	Summary Description
5.66	Access Authorization Program for Nuclear Power Plants	This guide describes a method that the NRC staff considers acceptable to implement the requirements of 10 CFR 73.56, "Personnel Access Authorization Requirements for Nuclear Power Plants," and 10 CFR 26, "Fitness for Duty Programs," related to an access authorization program. Each holder of an operating license under the provisions of 10 CFR 50 or a combined license under the provisions of 10 CFR 52 is required to establish, maintain, and implement the requirements in 10 CFR 73.56 before fuel is allowed on site (in a protected area). The NRC established these requirements to provide high assurance that individuals granted unescorted access and those certified for unescorted access authorization are trustworthy and reliable and do not constitute an unreasonable risk to public health and safety or the common defense and security, including the potential to commit radiological sabotage. This guide includes a discussion of NEI 03-01, "Nuclear Power Plant Access Authorization Program," (not publicly available) as an industry standard that is an acceptable method to meet 10 CFR 73.56.
5.68	Protection Against Malevolent Use of Vehicles at Nuclear Power Plants	This guide provides guidance acceptable to the NRC staff to address the requirements in 10 CFR 73 related to malevolent use of vehicles at NPPs, including protecting against attacks by persons using a four-wheel drive land vehicle for transport and as a vehicle bomb, and establishing vehicle control measures (vehicle barriers) to prevent unauthorized transport to gain proximity to vital areas. The requirements of 10 CFR 73 include design goals, criteria, and a process for using alternative measures for protection against a land vehicle bomb.
5.69	Guidance for the Application of the Radiological Sabotage Design-Basis Threat in the Design, Development and Implementation of a Physical Security Program that Meets 10 CFR 73.55 Requirements (not publicly available)	
5.71	Cyber Security Programs for Nuclear Facilities	This guide provides a framework to aid in the identification of those digital assets that must be protected from cyber-attacks. These identified digital assets are referred to as "critical digital assets." Licensees should address the potential cyber security risks of critical digital assets by applying the defensive architecture and the collection of security controls identified in this regulatory guide.

Table 5-1. **Summary of Regulatory Guides for Physical Security.**

Guide Number	Regulatory Guide Title	Summary Description
5.74	Managing the Safety/Security Interface	This guide provides an approach that the NRC staff considers acceptable for licensees to use in satisfying the requirements of 10 CFR Part 73, specifically 10 CFR 73.58, "Safety/Security Interface Requirements for Nuclear Power Reactors." This guide describes a method that the staff of the NRC considers acceptable for licensees to assess and manage changes to safety and security activities so as to prevent or mitigate potential adverse effects that could negatively impact either plant safety or security.
5.75	Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities	This guide described approaches and methodologies acceptable to the NRC staff for the training and qualifications of security personnel at NPPs. Guidance is provided for licensees and applicants to use to select, train, equip, test, qualify, and requalify armed and unarmed security personnel, watchpersons, and other members of their security organization to ensure that these individuals possess and maintain the knowledge, skills, and abilities required to carry out their assigned duties and responsibilities effectively.
5.76	Physical Protection Programs at Nuclear Power Reactors (not publicly available)	
5.77	Insider Mitigation Program (not publicly available)	
5.79	Protection of Safeguards Information	This guide provides guidance acceptable to the NRC staff to address the requirements in 10 CFR 73 for protection of Safeguards Information. These regulations require the establishment, implementation, and maintenance of an information protection system.

Table 5-1. Summary of Regulatory Guides for Physical Security.

Guide Number	Regulatory Guide Title	Summary Description
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	The NRC's policy statement on PRA encourages greater use of this analysis technique to improve safety decision-making and improve regulatory efficiency. This regulatory guide describes an acceptable method for the licensee and NRC staff to use in assessing the nature and impact of licensing basis changes when the licensee chooses to support, or is requested by the staff to support, the changes with risk information. This regulatory guide focuses on safety risk changes to core damage frequency, but an alternative risk-informed approach could be employed for security risk management [23].

5.2.3 Security Orders

Security orders contain requirements for licensees to implement interim compensatory security measures beyond that currently required by NRC regulations and as conditions of licenses. Currently ten security orders dated from February 2002 through June 2006 have been issued for implementation of interim compensatory security measures to address radiological protection mitigation strategies, updated adversary characteristic, fitness for duty, training enhancements, design basis threat, access authorization, and interim safeguards and security. The NRC website provides publicly available information associated with security orders [24], additional details protected as Safeguards Information are not publicly available.

5.3 Summary of Security Requirements Applicability to Advanced SMRs

The NRC's physical security regulations for NPPs are generally technology neutral. Therefore, all will generally be applicable for advanced SMRs. However, licensees may wish to explore strategies to reduce cost and staffing and while still achieving regulatory compliance. An overarching goal for reactor security is to develop a set of Commission-approved security plans that describe the measures that will be taken to meet the general performance objective, to implement requirements, and to establish and maintain an overall level of system performance that provides high assurance of protection against the design basis threat (DBT) for radiological sabotage. Performance-based justification using accepted physical security assessment methodologies would be needed to demonstrate compliance to acceptance criteria.

A few key assumptions were made to determine the applicability of current NPP security regulations and also to establish a justifiable methodology to support reasonable security cost for future SMR's. The assumptions are briefly explained below.

1. The major security concern is radiological sabotage to public safety and health (e.g., threats that lead to core damage and potential releases to the environment); economic sabotage (e.g. property damage) are not required to be addressed for a license application.

2. The fresh fuel used in iPWRs is low enriched uranium while for LMRs, the fuel can be higher enriched and contain plutonium. The spent fuel is somewhat self-protecting in all reactors (including SMRs) due to high radiation levels for some period after it is removed from the reactor core.

3. The proposed designs of iPWRs have a number of common features (e.g. the operations of the reactor, building layout) that are relevant to the physical security design. The iPWR designs includes underground siting of the core, steam generators that are integral to the reactor to eliminate large pipes, flooding of the reactors and containments for accident mitigation, and more reliance on passive safety.

4. The passive safety features for the iPWR designs will be designed for use of up to 72 hours reducing reliance on power and operator actions.

5. All buildings that house critical safety related systems are designed to protect against phenomena and design basis threats.

6. Generally, all critical safety related systems in iPWRs (e.g. DC power, control rooms, chemical volume control system) are located below-grade and protected by a reinforced concrete missile shield to protect against natural and hostile phenomena.

7. Nuclear material in use and in storage is protected below-grade by the reinforced concrete missile shield (e.g. the reactor core, fresh fuel vault, and the spent fuel pool).

8. Protection of structure, systems, and components from internally and externally produced missiles is accomplished by the following practices:
   a. Location of the system or components in a missile proof structure,
   b. Physical separation of redundant systems,
   c. Fire walls,
   d. Flood mitigation building designs, and
   e. Measures to prevent internally generated missiles.

Three areas relating to "Response Requirements" under 10 CFR 73.55 could potentially allow a licensee to identify design features, analytical techniques, and procedural measures for SMRs that could demonstrate how proposed strategies may be acceptable for complying with the NRC regulations.

1. For existing NPPs, the number of armed responders designated to implement the protective strategy shall not be less than 10. However, taking into account the new safety and security features of SMRs, this number potentially could be less because of fewer target and target sets that require protection. The licensee would need to conduct a site-specific security assessment to determine the effectiveness of their physical protection system, including the protection strategy with reduced armed responders.

2. For existing NPPs, the number of armed responders needs to be sufficient to effectively implement the protective strategy. They must be available at all times inside the protected area (which may include the protected area access control point), and they must not be assigned other duties or responsibilities that could interfere with their assigned response duties. Again, taking into account the new safety and security features of SMRs and the reduction of target and target sets that require protection, some of the armed responders could potentially take on additional duties to handle the normal daily security operations such as access control, key service, escorting, patrolling, etc. Utilizing armed responders in a dual-purpose role would allow the licensee to reduce staffing levels. This concept of operations has also been effectively adapted at facilities in the DOE Complex. An alternative concept of operations for security staff may include a smaller core group of dedicated armed responders and other (non-security) operations staff that take on additional duties to handle normal daily security operations. Again, the licensee will need to conduct a site-specific security assessment, including a time-line analysis to determine if a proposed operational strategy for armed response meets required response force times.

3. For existing NPPs, the armed security officers designated to strengthen the onsite response are physically onsite and available at all times to carry out their assigned duties. This requirement is still needed because the critical safety-related systems are below grade and the areas where the nuclear material will be stored are also below-grade protected by the reinforced concrete missile shield. An onsite response below grade is required to maintain security control to protect the facility areas where key safety-related systems and nuclear material are located. However, taking into account the substantial delay from underground siting, the passive safety features providing a 72-hour safety window, and the reduction of target and target sets that require protection, the armed responders that are above-ground may not necessarily need to be located on-site. The licensee may consider utilizing local law enforcement for an armed response as a tertiary response for the above-ground protection strategy while most of the onsite response is dedicated only for the below-ground response of the facility.

Though these requirements relating to security staffing could be explored further, it is very important for the licensee to always keep in mind the responsibility to establish and maintain a physical security program and infrastructure, to include a security organization capable of providing high assurance of the protection of nuclear material and facilities against adversary attack. As stated in 10 CFR 73.55, a security organization needs the appropriate resources to be adequately designed, staffed, trained, qualified, and equipped to implement the physical protection program, including the protective strategy. The primary purpose of a security organization is to ensure the protection of nuclear materials and facilities. Licensees need resources to implement security programs that support this primary purpose, including handling normal daily security operations.

Each NPP is operated differently and the appropriate staffing level for the facility is determined considering: the number of personnel to ensure the protection of nuclear material and facilities; the number of personnel needed to maintain security programs to support the sustainability of a physical protection program, including the protective strategy; and the number of personnel needed to handle normal daily security operations.

The NRC has stated expectations that advanced reactor designs will provide enhanced measures of safety and security [25]. In addition the NRC expects reactor designers “to integrate security into the design and conduct a security assessment to evaluate the level of protection provided. Additional research may be needed to assess the efficacy of any new security measures” [26]. Although the physical security requirements are technology neutral and therefore applicable to SMRs, the design approaches for SMR safety related systems potentially provide a larger measure of intrinsic security, although this would have to be demonstrated. In addition, depending on the particular SMR design, the plant footprint is likely to be smaller than that for a traditional LWR with a commensurate smaller number of targets and target sets and overall smaller footprint for security operations. This factor may in itself demonstrate cost savings for SMR security.

6. APPLICABILITY OF EXISTING MATERIAL SAFEGUARDS REQUIREMENTS TO ADVANCED SMRS

To be licensed by the NRC, an SMR must meet the domestic safeguards requirements that include MC&A for nuclear materials. These requirements are provided in 10 CFR 74. MC&A addresses two areas: *material control* which is the use of control and monitoring measures to prevent or detect loss when it occurs or soon afterward, and *material accounting*, which is the use of statistical and accounting measures to maintain knowledge of the quantities of special nuclear material in each are of a facility. In addition, SMRs must obtain licenses for possession and use of SNM and byproduct material in accordance with the requirements of 10 CFR 70, and 10 CFR 30, respectively. This section reviews the requirements and guidance for possession and use of nuclear material and MC&A and their applicability for SMRs.

6.1 Regulations for NPP Material Control and Accounting

Several parts of the CFR address the possession and use of SNM and its control and accounting. In addition to facility safety licenses that NPPs receive under the requirements of 10 CFR 50 and 10 CFR 52, possession and use of nuclear material at NPPs are additional licensing actions under 10 CFR 70. Regulations for MC&A are provided in 10 CFR 74. These CFR regulations are summarized below.

10 CFR 70 – Domestic Licensing of Special Nuclear Material. Under 10 CFR 70, the NRC regulates all ownership of nuclear material. This regulation establishes procedures and criteria for the issuance of licenses to receive title to, own, acquire, deliver, receive, possess, use, and transfer special nuclear material; and establish and provide for the terms and conditions upon which the NRC will issue such licenses. 10 CFR 70 also outlines the designated levels of special nuclear material (SNM) and defines the distinctions. In general, SNM means plutonium and uranium enriched in the isotopes 233 and/or 235, but other material may be determined to be SNM. In addition, SNM also includes any material artificially enriched by any of these isotopes, but does not include source material. Within this designation, three levels, or categories are defined; these material categories are presented in Table 6-1. 10 CFR 70 applies to power reactor operations that will possess and use material and product forms containing plutonium and fissile isotopes of uranium, including SNM associated with fuel assemblies. The regulations address requirements to keep records and provide for inspections of all activities under the license; report any changes in licensed material levels; prepare and maintain a safeguards contingency plan; submit emergency plans; and report any material loss or damage that could hinder the ability to properly control or account for material, or unplanned contamination or criticality event. New risk-informed, performance-based approaches in Subpart H, “Additional Requirements for Certain Licensees Authorized to Possess a Critical Mass of Special Nuclear Material” have been implemented since September 2000, with the goal of providing additional confidence in the margin of safety for these facilities. The requirements of 10 CFR 70 also invoke physical security and MC&A requirements of 10 CFR 73 and 10 CFR 74, respectively.

Table 6-1. Classification Levels of Special Nuclear Material in 10CFR Part 70.

Category	Category Title	Quantity Description
III	Special nuclear material of low strategic significance	• >15g but <1kg of ^{235}U in uranium enriched to 20% or more in ^{235}U, or • >15g but <500g of ^{233}U, or • >15g but <500g of Pu, or • >15g but <1kg of contained ^{235}U, ^{233}U, and Pu in any combination, or • >1kg but <10kg of ^{235}U in uranium enriched between 10% and 20% in ^{235}U, or • >10kg of ^{235}U in uranium enriched above natural but less than 10% in ^{235}U
II	Special nuclear material of moderate strategic significance	• >1kg but <5kg of ^{235}U in uranium enriched to 20% or more in ^{235}U, or • >500g but <2kg of ^{233}U, or • >500g but <2kg of Pu, or • >1kg but <5kg of contained ^{235}U, ^{233}U, and Pu in any combination, or • >10kg of ^{235}U in uranium enriched between 10% and 20% in ^{235}U
I	Formula quantity of special nuclear material	• >5kg of material computed by summing contained mass of ^{235}U and 2.5 times the masses of ^{233}U and Pu

10 CFR 74, Material Control and Accounting of Special Nuclear Material. The requirements of 10 CFR 74 provide the regulations for MC&A of SNM at fixed sites and for documenting the transfer of SNM. Under the requirements of 10 CFR 70, license applications for the possession and use of SNM must include a description of applicant's program for MC&A of SNM to show how the requirements of 10 CFR 74 will be met. The general reporting and recordkeeping requirements apply to any entity that possesses SNM in a quantity greater than one gram of contained ^{235}U, ^{233}U, or Pu. These requirements are applicable to all NRC facilities that use SNM, including SMRs; the exception being independent fuel storage facilities licensed under 10 CFR 72. 10 CFR 74 requires that each licensee report all loss, theft, attempted theft, or unauthorized production of SNM within one hour of occurrence. Additionally, each licensee must complete Material Balance and Nuclear

Material Transaction Reports concerning all SNM that the licensee has received, produced, possessed, transferred, consumed, disposed, or lost. Each licensee is responsible for performing independent tests on all material, no matter the location within the facility (including in-process), to ensure proper accounting. Records must be kept in accordance with this regulation.

For each category of material quantity that the licensee possesses, reporting requirements, measurement tolerances, and protections against loss and theft are prescribed. In essence the law states that each facility must control and account for all special nuclear material, and that each facility must keep proper records and notify the NRC promptly of any actual or potential material loss or theft. The level to which each licensee must know the specific make-up of material is also defined and is dependent on the material quantity category. Quality assurance and documentation of such is highlighted and required for each facility's monitoring capability.

6.2 Sub-tier Guidance Documents for Material Control and Accounting

Sub-tier guidance for MC&A includes several regulatory guides that address areas associated with MC&A. These are summarized in Table 6.2.

6.3 Material Control and Accounting Requirements Applicability to Advanced SMRs

The requirements in 10 CFR Parts 70 and 74 dictate the material licensing, and control and accounting of SNM. These requirements are applicable to all nuclear facilities, including current LWRs, and will generally be applicable to SMRs. SMRs will require a license under 10 CFR 70 to possess and use the nuclear material for reactor operations. Licensing for possession and use of the type and quantity of nuclear material used for iPWRs will be similar to that for traditional LWRs. The MC&A requirements for iPWRs will also be similar to that for current LWRs. Each facility must account for and control all SNM, providing documentation that they are doing so in accordance with these requirements.

Review of the relevant regulations and regulatory guides indicate that in terms of MC&A, the requirements are no different for different sizes of NPPs. All NPPs are required to perform the same control, measurements, record-keeping, and reporting; although SMRs may have operations on a smaller scale. Because of the potential for the nuclear material in reactor fuel to be used in various weapons, the safe handling and early detection of any diverted material is imperative.

MC&A for SMRs will likely be modeled after current NPPs. Staffing requirements are low and only specified for proper material movement and inventories and are implied for proper bookkeeping, management, and reporting. Additionally, impacts on other aspects of the plant, including safety, operations, and security, are limited.

In addition to the iPWR type SMRs, other LMR designs have the potential to go through the regulatory approval process in the near term. NRC has stated that "advanced reactor designs using LWR fuel assemblies at less than five percent enrichment can meet MC&A requirements established in NRC regulations by following the existing NRC guidance. HTGR fuels (e.g., TRISO) and LMR fuels require further evaluation to determine whether existing MC&A

requirements are applicable" [26]. LMRs are generally very compact reactor cores that operate with a high energy density. This energy density is achieved by having highly enriched fuel (sometimes up to 50%) or potentially using plutonium as the fissionable material in the fuel. The sodium coolant allows for an efficient removal of this energy from a small reactor core, making these designs possible. For LMRs with the higher fuel enrichment, levels of material will likely reach category I. At this level, biannual physical inventories are required after proving the performance of the MC&A system to the NRC using bimonthly physical inventories. Additionally, process monitoring must be able to detect 95% of all prompt losses greater than five formula kilograms of material within three days of occurrence from any accessible process location. Process quality tests must also ensure detection and subsequent investigation of anything outside of three standard deviations of the process estimator and/or greater than 25 grams of strategic special nuclear material (SSNM). Finally, trend analyses must be performed to detect any trends in losses that pose safeguards significance.

Item monitoring is also enhanced in for category I material, which would include fresh and spent fuel for LMRs. Under category I amount requirements, all items must be either tamper-safe or placed in a controlled access vault and must be independently verified to within 50 grams of SSNM. Loss detection must be able to perform at 99% confidence with quantities greater than 5 kg of SSNM at 30 to 60 days after occurrence depending on the physical location of the material.

Another possible difference with regard to material safeguards for SMRs is the implementation of monitoring to fulfill international safeguards requirements. International safeguards applied by the International Atomic Energy Agency have a goal to detect diversion of significant amounts of fissile nuclear materials by the state/facility operator through verification of the state's declaration. The NRC "has added expectations that reactor designers consider the threat of theft and requirements for implementing international safeguards monitoring early in the design phase" [25]. This section has focused on domestic NRC requirements for MC&A. The requirements for safeguards measures to meet domestic and international safeguards objectives have different technical approaches. While many of the techniques are useful for either safeguards system, implementation and reporting requirements will differ and advanced SMR designs will have to consider these differences in their designs. The NRC also expects that "research may be needed to review the MC&A safeguards provisions and their technical basis to determine whether they are acceptable" for new reactor designs [26].

Table 6-2. Summary of Regulatory Guides for Material Control and Accounting.

Guide Number	Regulatory Guide Title	Summary Description
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste	This regulatory guide outlines the acceptable practices for measuring, evaluating, and reporting radioactive material in the waste streams at light water cooled NPPs.
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	This guide gives a sample cost-benefit analysis of a radioactive waste system for light water reactors. An appendix for 10 CFR 50 requires that this be done for each plant and that it demonstrate the cost-benefit in terms of ALARA(as low as reasonably achievable) for the effluent of the plant. This will apply directly to SMRs (at least those that use light water), and must be performed during the licensing of the reactor. There is no special impact on the design or planning of the SMR.
5.13	Conduct of Nuclear Material Physical Inventories	This guide outlines the best practices for how to receive material, track material in-process, account for spent material, etc. This guidance applies for the general, day-to-day operations and inventories of the NPP and outlines the use of approaches to tamper-safe nuclear material, material balance areas, and the assignment of responsibilities in some detail. One note that impacts the staffing is the recommendation of supervisor and inventory team roles. The two-person rule for transfer and inventory operations is suggested to ensure continuity of ownership and proper checking of the potential insider threat to diversion of material.
5.26	Selection of Material Balance Areas and Item Control Areas	Although no longer part of the CFR, 10 CFR 70.58 had outlined the need for Material Balance Areas and Item Control Areas for the physical and administrative control of nuclear materials. This regulatory guide discusses the suggested number of such areas, their general description, and the capabilities for each type. These areas are designed to aid in the material accounting process at each facility, giving checkpoints throughout the process where material is counted. Additionally, these areas allow for greater detection of diverted or lost material.

Table 6-2. Summary of Regulatory Guides for Material Control and Accounting.

Guide Number	Regulatory Guide Title	Summary Description
5.27	Special Nuclear Material Doorway Monitors	This guide discusses the need for doorway monitors to ensure the detection of personally diverted material. This guide relates more to the physical protection and security aspects defined in 10 CFR 73, but aids in material control and diversion detection.
5.29	Nuclear Material Control Systems for Nuclear Power Plants	This guidance addresses the requirements to establish, maintain, and follow written MC&A procedures sufficient to enable a licensee to account for SNM in his possession under license. It was withdrawn in 1998, but a new Draft Regulatory Guide DG-5028 "Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants" is a proposed revision of RG 5.29 that describes SNM MC&A system requirements applicable to all NPPs.
5.49	Internal Transfers of Special Nuclear Material	This regulatory guide dives a step deeper into the suggested process and documentation for internal material transfers. This guide should be used for procedures for internal transfers and material movement.
5.51	Management Review of Nuclear Material Control and Accounting Systems	This guide gives an overview of how the management should review MC&A systems at all facilities. This guidance is applicable for defining operating procedures at any NPP.
5.57	Shipping and Receiving Control of Strategic Special Nuclear Material	Guidance is given on the proper shipping and receiving of strategic SNM. The main focus of the guide is the documentation, verification method, and suggested number of personnel. Guidance on staffing is 1 to 2 people from each site (the shipping and the receiving) to ensure proper documentation, verification, and oversight of all material transfers.
5.58	Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements	This guide gives the recommended traceability in all SNM measurement systems used for MC&A. This includes the calibration, certification and testing of all these systems to ensure the proper measurement and reporting.

Table 6-2. Summary of Regulatory Guides for Material Control and Accounting.

Guide Number	Regulatory Guide Title	Summary Description
5.79	Protecting Safeguards Information	As defined in the 10 CFR 74, this guide gives details on the proper handling of safeguards information (SGI).
5.80	Pressure-Sensitive and Tamper Indicating Device Seals for Material Control and Accounting of Special Nuclear Material	This guide deals with the suggested types of seals used to detect material diversion. These can be placed on canisters, boxes, etc., for the purposes of detecting an attempted or actual material theft. Bar coding, verification, and types of seals are discussed.

7. APPLICABILITY OF EXISTING EMERGENCY PLANNING REQUIREMENTS TO ADVANCED SMRS

This section identifies the current emergency planning (EP) regulatory requirements that are relevant to licensing a new reactor, including small modular reactors (SMR), by the NRC. This section also provides a discussion on additional supporting documents that affect the licensing process including regulatory guides, the standard review plan, interim staff guidance and other guidance for emergency planning.

7.1 Regulations for NPP Emergency Planning

Emergency Planning regulations are in 10 CFR 50, specifically 10 CFR 50.47 "Emergency Plans," and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities". Appendix E to 10 CFR 50 establishes the minimum requirements for emergency plans to attain an acceptable level of emergency preparedness. Appendix E addresses the emergency organization and duties of individuals; assessment actions to determine the magnitude of and continually assess the impact of, the release of radioactive materials; the activation of the emergency organization under each class of emergency; the notification procedures; emergency facilities and equipment; training, including for co-located licensees; recovery; and onsite protective actions during hostile action.

Appendix E also specifies that emergency action levels (EALs) are to be used as criteria for determining the need for taking emergency response actions, including, for example, notification of emergency response organizations, which depends upon the degree of degradation of plant safety during an emergency event.

Appendix E also describes the Emergency Response Data System (ERDS), which is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC that provides an automated transmission of a limited set of selected parameters. These parameters are specified for existing LWRs. No parameter set is specified for SMRs. A data set appropriate for each type of SMR would be needed.

7.2 Sub-tier Guidance Documents for NPP Emergency Planning

This section identifies and provides a discussion on the additional supporting documents including the NRC standard review plan, NRC regulatory guides, Environmental Protection Agency (EPA) Manual, NUREG documents, Interim Staff Guidance (ISG), and NRC policy (SECY) documents that provide detailed guidance on EP implementation. These are relevant to licensing a new reactor, including the near-term iPWRs, LMRs, and other SMRs, by the NRC.

7.2.1 NUREG-0800: Standard Review Plan

NUREG-0800 [3] establishes guidance for NRC staff for safety reviews of construction permit and operating license applications and amendments under 10 CFR Part 50 and for early site permit, design certification, combined license, standard design approval, and manufacturing license applications and amendments under 10 CFR Part 52. The Emergency Plan review is addressed in Chapter 13.3, "Emergency Planning" and addresses emergency planning zones, emergency action levels, emergency time estimates and emergency response facilities. Revision 3 of Chapter 13.3 was issued in March 2007. For reactor license applications, excluding design certifications, the NRC consults with the Federal Emergency Management Agency (FEMA) concerning offsite emergency planning and preparedness, since FEMA's Radiological Emergency Preparedness (REP) Program has lead responsibility for oversight of offsite emergency planning and preparedness. The final decision-making authority on emergency planning and preparedness adequacy is the NRC's. The SRP is not a substitute for NRC regulations and compliance with it is not required; however, applicants are required to identify differences between their facility design features, analytical techniques and procedures and the SRP acceptance criteria and provide an evaluation of how their proposed alternatives provide acceptable methods of compliance with NRC regulations.

Emergency action levels are required in 10 CFR 50.47 (b)(4) and Appendix E, Section IV.B and the applicant's emergency plan should include the emergency action level scheme described in Appendix 1 and Supplement 3 to NUREG-0654 [27]. New applications will likely follow the emergency action level scheme similar to that in Revision 4 of NEI 99-01, "Methodology for Development of Emergency Action Levels," [28], which was endorsed in Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," [29]. The majority of Revision 4 of NEI 99-01 may be applicable to any reactor design and should be used, but it may not be entirely applicable to SMRs, and the unique characteristics of the new reactor should be addressed in the emergency action level scheme, which should follow Regulatory Information Summary 2003-18, "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels," and its supplements.

While generally the EP guidance in this document applies to SMRs, a design specific review standard for iPWR is in development.

7.2.2 RG 1.101: Emergency Response Planning and Preparedness for Nuclear Power Reactors

Regulatory Guide, RG 1.101 endorses Revision 1 of NUREG-0654 to provide guidance to licensees and applicants on NRC acceptable methods for complying with regulations for emergency response plans and preparedness at nuclear power plants. NUREG-0654 provides specific evaluation criteria for determination of compliance with 10 CFR 50.47(b) standards and for FEMA review of offsite emergency plans and preparedness adequacy.

Regulatory Guide 1.101, Revision 3 reaffirmed the endorsement of NUREG-0654/FEMA-REP-1, Revision 1 and endorses the use of Revision 2 of NUMARC/NESP-007, "Methodology for the Development of Emergency Action Levels" [30] issued by the Nuclear Management and Resources Council (NUMARC) as an acceptable alternative.[1]

[1] NUMARC is now the Nuclear Energy Institute (NEI).

RG 1.101, Revision 5, issued in June 2005, provides guidance for co-located facilities for NRC regulation compliance concerning training-related content of emergency plans, including the conduct of emergency response planning activities and for interactions with offsite authorities in the years between full or partial participation of those offsite agencies in exercises. Co-located licensees are two different licensees whose licensed facilities are located either on the same site or on adjacent, contiguous sites, and that share most of the following emergency planning and siting elements: plume exposure and ingestion emergency planning zones; offsite governmental authorities; offsite emergency response organizations; public notification system; and/or emergency facilities (taken from 10 CFR 50 Appendix E, footnote to paragraph IV.F.2.c). This may apply to some SMRs if they are co-located with another nuclear power plant.

7.2.3 EPA-400-R-92-001: Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

EPA-400-R-92-001[31] provides a discussion of Protective Action Guides (PAGs) and their use in planning for protective actions for the safety of public health; PAGS for specific exposure pathways and timeframes; implementation guidance; and background information. The PAGs give dose levels at which initiation of protective actions are recommended. The PAGs for general use for response during the early phase of an incident are evacuation (or sheltering in certain cases) for a projected dose of 1-5 rem, where evacuation or sheltering should normally be initiated at 1 rem and administration of stable iodine at 25 rem with the approval of State medical officials. Later phases are also addressed in this manual.

EPA-400-R-92-001 notes that in NUREG-0396 [32], the size and shape of emergency planning zone (EPZs) are partially based on consideration of the numerical values of the PAGs. An additional basis is that the planning zone for evacuation and sheltering should be large enough to accommodate any urban and rural areas affected and involve the various off-site organizations needed for emergency response, and include the area in which acute health effects could occur. Experience that is gained through periodic emergency response training exercises is expected to provide an adequate basis for expanding the response to an actual incident to larger areas, if that is needed. It is not appropriate to use the maximum distance where a PAG might be exceeded as a basis for establishing the EPZ boundary for a facility.

Revised EPA Protective Action Guides (PAG), EPA-400-R-92-001, issued in 1992, provide that licensed facilities that can demonstrate that accident doses at the site boundary would not exceed the PAGs should not be required to have defined EPZs or comprehensive offsite emergency planning. The NRC has licensed several small reactors with an EPZ of 5 miles (for plume and 30 miles for ingestion), including Fort St. Vrain (842 MWt, HTGR), Big Rock Point (240 MWt, BWR) and La Crosse (165 MWt, BWR). With the SMR safety enhancements and the potential for reduced accident source terms and fission product releases, it may be appropriate for SMRs to develop similarly reduced EPZ sizes using a dose/distance approach.

7.2.4 NUREGs

NUREG-0396: Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants

NUREG-0396 [32] introduces the concept of generic EPZs as a basis for the planning of response actions. For serious accidents, predetermined protective actions would be taken if projected doses appeared to be at or above the applicable PAGs, based on information readily available in the reactor control room; that is, at predetermined emergency action levels. It recommends that EPZs be defined around each nuclear facility for both the "plume exposure pathway" and the "ingestion exposure pathway." The concept shows an EPZ-plume radius of 10 miles and an EPZ-ingestion of 50 miles. This same concept is shown in NUREG-0654/FEMA-REP-1 [27]. NUREG-0396 provides the technical basis for an EPZ, which is dose based; however, it does not provide guidance for actually establishing the EPZ.

NUREG-0396 identifies the items needed to scope the emergency planning efforts as the distance for which planning for the initiation of predetermined protective actions are warranted; the time dependent characteristics of potential releases and exposures; and the kinds of radioactive materials that can potentially be released to the environment. Judgment is to be used in determining the precise size and shape of the EPZs to take into consideration local conditions that may include demography, topography and land use, access routes, and local jurisdictional boundaries. For SMRs the time dependent characteristics of potential releases and subsequent exposures may be different than for current LWRs, particularly for LMRs. This is an area in which more study is needed.

NUREG-0654/FEMA-REP-1: Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

The NRC and FEMA jointly developed NUREG-0654/FEMA-REP-1 [27] to provide guidance to licensees and State and local governments in developing their plans to meet the 16 planning standards in 10 CFR 50.47(b). NUREG-0654 establishes specific evaluation criteria for each of the 16 planning standards that can be used by NRC and FEMA staff to evaluate whether the emergency plans meet the planning standards. The 16 planning standards are:

A. Assignment of Responsibility (Organization Control);
B. Onsite Emergency Organization;
C. Emergency Response Support and Resources;
D. Emergency Classification System;
E. Notification Methods and Procedures;
F. Emergency Communications;
G. Public Education and Information;
H. Emergency Facility and Equipment;
I. Accident Assessment;
J. Protective Response;
K. Radiological Exposure Control;
L. Medical and Public Health Support;

M. Recovery and Reentry Planning and Post-accident Operations;
N. Exercises and Drills;
O. Radiological Emergency Response Training; and
P. Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans.

EPZs must be defined for both the short term "plume exposure pathway" and the longer term "ingestion exposure pathway" and are defined as the areas for which planning is needed to assure that prompt and effective actions can be taken to protect the public in the event of an accident. For LWR power plants a radius of about 10 miles for the plume exposure pathway and about 50 miles for the ingestion exposure pathway was selected by the task force in NUREG-0396. In a NUREG-0654 footnote, it is stated that these radii are applicable to light water nuclear power plants, rated at 250 MWt or greater. Small water cooled power reactors (less than 250 MWt) and the Fort St. Vrain gas cooled reactor may use a plume exposure emergency planning zone of about 5 miles in radius and an ingestion pathway emergency planning zone of about 30 miles in radius. The alert and notification system will be scaled on a case-by-case basis. These are based on the lower potential hazard from these facilities, with lower radionuclide inventory and longer times to release significant amounts of activity for many accident scenarios. A similar argument may be able to be used for SMRs.

While licensees have primary responsibility for planning and implementing emergency measures within their site boundaries, the facility cannot do this alone and it is a necessary part of emergency planning to make advance arrangements with State and local organizations for special emergency assistance.

NUREG-0654, Section II.B, "Onsite Emergency Organization," provides guidance for meeting the organization requirements found in 10 CFR 50, Appendix E, Section IV.A. This guidance describes the onsite emergency organization and includes the staffing requirements found in NUREG-0654, Table B-1, "Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies" (see Table 7-1). Table 7-1 specifies a minimum of 10 on-shift responders in four functional areas and specifies seven on-shift responders who perform response duties that may be performed by shift personnel in addition to their other assigned functions. Firefighting and site access control must be staffed on a site-specific basis. The table specifies the number of augmenting responders within 30 and 60 minute timeframes. A detailed analysis is required to demonstrate that on-shift personnel that are also assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions that are specified in the emergency plan. On-shift staff must be able to cope with design-basis threats and design-basis accidents and implement emergency plans until the augmenting emergency response organization staff is mobilized. On-shift staff must be capable of taking emergency actions to safely shut down the reactor(s), mitigate accident consequences, notify augmented emergency response organization staff and offsite response organizations (within 15 minutes of a change in emergency classification level or issuance of a protective action recommendation), determine protective action recommendations for site personnel and the public, perform firefighting, and provide medical assistance if needed. These capabilities are primarily addressed in planning standards D. Emergency Classification System; E. Notification Methods and Procedures; I. Accident Assessment; and J. Protective Response.

Table 7-1 also notes that for each unaffected unit in operation, at least one shift foreman, one control room operator and one auxiliary operator are to be maintained, except in the case of units sharing a control room, where a shift foreman may be shared if all functions are covered. This will be particularly relevant to SMRs with multiple reactors and a shared control room.

For SMRs the categories of staff and their numbers for both on-shift and augmenting staff needed within what time frames will be key in determining staffing levels and what the appropriate augmenting time frames may be. With some of the passive safety features, augmenting in times longer than the 30 and 60 minutes specified in Table 7-1 may be feasible. This is dependent upon the SMR accident scenarios and the passive, as well as active, safety systems that are part of each SMR design or group of similar designs.

NUREG-0696: Functional Criteria for Emergency Response Facilities

NUREG-0696 [33] describes the systems and facilities that can be used by licensees of nuclear power plants to respond to emergency situations. The facilities described include the technical support center (TSC), onsite operational support center (OSC), near site emergency operations facility (EOF) and the emergency response functions of the control room. The safety parameter display system (SPDS) data system is described. NUREG-0696 guidance also describes the size and staffing of the TSC and the EOF and the time frame for that staffing (30 minutes for the TSC and 60 minutes for the EOF). These facilities and staffing are to support the control room in mitigating the consequences of accidents and support the licensee's capability to respond to abnormal plant conditions.

Although NUREG-0696 states that the walking time between the control room and the TSC shall not exceed 2 minutes, to facilitate face-to-face interaction, a number of nuclear power plants are licensed with longer distances, especially with current communication capabilities. This will be relevant for SMRs that have multiple reactors at a single site and may have a conceptual design with a single TSC or TSC that is shared between several reactors.

This guide will apply to SMRs. The staffing levels and the times to activate will need to be reviewed for SMRs considering their passive safety systems and the timelines of their accident scenarios. The facilities sizes will need to be reviewed based on the number of units.

NUREG-0737, Supplement 1: Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability

NUREG-0737 [34] was issued in January 1983 and provides additional clarification regarding emergency response capabilities. The supplement provides a summary and revision of the basic requirements for emergency response capabilities for various emergency response facilities, from a broad range of guidance documents. The guidance documents are not requirements, but are to be used as sources of guidance for licensees and NRC reviewers regarding acceptable means for meeting the basic requirements. NUREG-0737, Supplement 1, Table 2, "Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies," references NUREG-0654 as the source for this table (see Table 7-1).

Table 7-1. Minimum Staffing Requirements For NRC Licensees For Nuclear Power Plant Emergencies (See NUREG-0654 Evaluation Criteria B.5 [27]).

Major Functional Area	Major Tasks	Position Title or Expertise	On Shift*	Capability for Additions 30 min	60 min
Plant Operations and Assessment of Operational Aspects		Shift Supervisor (SRO)	1	- -	- -
		Shift Foreman (SRO)	1	- -	- -
		Control Room Operators	2	- -	- -
		Auxiliary Operators	2	- -	- -
Emergency Direction and Control (Emergency Coordinator)***		Shift Technical Advisor, Shift Supervisor or designated facility manager	1**	- -	- -
Notification/ Communication****	Notify licensee, State Local and Federal personnel & maintain communication		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment	Emergency Operations Facility (EOF) Director	Senior Manager	- -	- -	1
	Offsite Dose Assessment	Senior Health Physics (HP) Expertise		1	- -
	Offsite Surveys		- -	2	2
	Onsite (out-of-plant)		- -	1	1
	In-plant surveys	HP Technicians	1	1	1
	Chemistry/Radio-chemistry	Rad-Chem Technicians	1	- -	1
Plant System Engineering, Repair and Corrective Actions	Technical Support	Shift Technical Advisor	1	- -	- -
		Core/Thermal Hydraulics	- -	1	
		Electrical	- -	- -	1
		Mechanical	- -	- -	1
	Repair and Corrective Actions	Mechanical Maintenance/	1**	- -	1
		Rad Waste Operator			1
		Electrical Maintenance/	1**	1	1
		Instrument and Control (I&C) Technician	- -	1	- -

Table 7-1. Minimum Staffing Requirements For NRC Licensees For Nuclear Power Plant Emergencies (cont.)
(See NUREG-0654 Evaluation Criteria B.5 [27]).

Major Functional Area	Major Tasks	Position Title or Expertise	On Shift*	Capability for Additions 30 min	60 min
Protective Action (In-Plant)	Radiation Protection: a. Access Control b. HP Coverage for repair corrective actions, search and rescue first-aid & firefighting c. Personnel monitoring d. Dosimetry	HP Technicians	2**	2	2
Firefighting	- -	- -	Fire Brigade per Technical Specifications	Local Support	
Rescue Operations and First-Aid	- -	- -	2**	Local Support	
Site Access Control and Personal Accountability	Security, firefighting communications, personnel accountability	Security Personnel	All per Security plan		
Total			**10**	**11**	**15**

Notes:

* For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.

** May be provided by shift personnel assigned other functions.

*** Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.

**** May be performed by engineering aide to shift supervisor.

7.2.5 Other Guidance Documents

NSIR/DPR-ISG-01: Interim Staff Guidance, Emergency Planning for Nuclear Power Plants

Interim Staff Guidance, NSIR/DPR-ISG-01 [35], provides updated guidance for addressing EP requirements for nuclear power plants. This guidance is based on changes to EP regulations in 10 CFR 50.47, "Emergency Plans" and 10 CFR 50 Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," that were published November 23, 2011 in the Federal Register. This ISG provides an overview of emergency planning guidance documents. The issues addressed in this ISG include: On-Shift Staffing Analysis; Emergency Response Organization Augmentation at Alternative Facility; Licensee Coordination with Offsite Response Organizations; Protective Actions for Onsite Personnel; Challenging Drills and Exercises; Emergency Declaration Timeliness; EOF – Performance Based Approach; Backup Means for Alert and Notification Systems; and ORO Event Response Integration with Nuclear Power Plants. Each issue section includes background, discussion and guidance.

The on-shift staffing analysis addresses concerns regarding the concurrent assignment of responsibilities to on-shift emergency response organization personnel that can potentially overburden them and prevent the timely performance of their emergency plan functions. Licensees are to include a detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent them from performing their emergency plan assigned functions in a timely manner. The staffing analysis should include at least one event that results in the declaration of a General Emergency and radiological doses to the public that exceed the EPA protective action guides beyond the site boundary and that would require the promulgation of a protective action recommendation (PAR), even if no design-basis accident results in this emergency classification level and dose level. All of these issues will apply to SMRs and the On-Shift Staffing Analysis will be of particular interest.

This ISG has a section on the Emergency Operations Facility – Performance Based Approach. Nuclear power plant licensees have submitted several requests to the NRC to combine Emergency Operations Facilities (EOFs) for multiple plants within a state or in multiple states into a consolidated EOF. In some cases an EOF could serve units with more than one type of reactor technology, such as PWRs and BWRs or more than one design of the same reactor type. The EOF staff needs to be capable of understanding plant conditions for each type of reactor and for communicating useful information to offsite officials and media. The term "near-site" EOF is being deleted and replaced with functional capabilities; however, a location closer to the site is still needed for NRC and offsite agency staff if the EOF is more than 25 miles from the site. The EOF will have the capability to respond to simultaneous events occurring at more than one site. This will apply to SMRs and the issue of EOF staff needing to be capable of understanding plant conditions for each type of reactor will be of particular importance if the EOF for an SMR or multiple SMRs is a consolidated facility with different reactor technologies.

This ISG identifies the four risk-significant planning standards as classifying an emergency event (emergency action levels), notifying emergency responders and offsite officials of a declared emergency (including alert and notification systems), performing dose assessment and developing protective actions. The NRC considers the emergency response capabilities in these risk-significant planning standards to be critical for protecting public health and safety. These standards will apply to SMRs.

SECY-11-0152, Policy Issue (Information): Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors

SECY-11-0152 [36] discusses the NRC staff's intentions to develop a technology-neutral, dose-based, consequence-oriented emergency preparedness framework for SMR sites and take into account the various designs, modularity and collocation, and size of the emergency planning zone. The approach would be based on the concept of scaling commensurate with the accident source term, fission product release and dose characteristics for the SMR designs. The methodology for dose calculation is also being considered with NRC staff planning on working with stakeholders to develop general guidance on calculating the offsite dose. It is anticipated that industry will develop and implement a detailed calculation method that the NRC would review and approve.

As indicated in Section 7.2.3, revised EPA PAGs in EPA-400-R-92-001 [31], provides that licensed facilities that can demonstrate that accident doses at the site boundary would not exceed the PAGs should not be required to have defined EPZs or comprehensive offsite emergency planning. With the SMR safety enhancements and the potential for reduced accident source terms and fission product releases, it may be appropriate for SMRs to develop similarly reduced EPZ sizes using a dose/distance approach. An approach could be to have offsite emergency planning requirements scale with the SMR accident source term, fission product release and associated dose characteristics and be based on offsite dose and use the EPA PAGs to establish standard EPZ distances. An example of a scalable EPZ from SECY-11-0152 is shown in Table 7-2.

In this example of a scalable EPZ, if the expected offsite dose is greater than 1 rem off site, but less than 1 rem at 2 miles, then the requirements for the EPZ would be limited to the 2-mile zone. Specific EP requirements would consider both source term and event transient time. For example, an accident event leading to an offsite dose may be several hours, so the current requirement to notify offsite responsible State and local governmental agencies within 15 minutes of declaring an emergency may need to be evaluated based on the accident event transit time. Whether the source term is based on a single module or multiple modules or all modules at a site could depend on the accident scenario.

In addition SMRs have the potential to employ multiple reactors, as modules, on a single site and for that site to be collocated near industrial facilities or with large reactors or other types of SMRs. However, in response to the NRC's Regulatory Issue summary 2011-02 [37], no potential applicant indicated the intention for an SMR facility to be collocated in this manner. An issue will be emergency response organization staffing levels if SMR sites increase from two or four reactors to six or more reactors and if reactor modules have some common or shared systems on a single site.

Table 7-2. Example of a Scalable EPZ from SECY-11-0152, Table 1.

EPZ Category	Dose Limits	Plume Exposure EPZ	Ingestion Exposure EPZ	EP Plan Required	Offsite EP Plan
I	Projected dose at site boundary <1 rem	Site boundary	No; however, EPZ can expand based on event, if determined to be necessary	Yes	All hazards-license condition*
II	Dose at site boundary ≥1 rem, <1 rem at 2 miles	2 miles**	Yes; dosed-based distance, ad hoc basis***— Food and Drug Administration (FDA) food PAGs	Yes	Yes
III	Dose at 2 miles ≥1 rem, <1 rem at 5 miles	5 miles**	Yes; dosed-based distance, ad hoc basis*** —FDA food PAGs	Yes	Yes
IV	Dose at 5 miles ≥1 rem	10 miles**	Yes; per current regulations, ad hoc basis***	Yes	Yes

* The NRC would issue a license condition that will require the licensee to ensure that a certified offsite all-hazards plan exists (which provides the basic framework for responding to a wide variety of disasters).
** The staff will also consider the area needed to ensure an adequate planning basis for local response functions and the area in which acute health effects could occur.
*** Per NUREG-0396, actions that would provide dose savings for any such accident can be taken on an ad hoc basis using the same considerations that went into the initial action determinations.

7.3 Summary of Emergency Planning Requirements Applicability to Advanced SMRs

While some of the emergency planning regulations, regulatory guides and other guidance are fully applicable to SMRs, there are some aspects that warrant evaluation with respect to the differences between SMRs and light water power reactors. These can include, but are not limited to, size of the emergency planning zone, notification times, shared facilities, collocation with other SMRs and other nuclear power reactors, number of staffing positions, times for augmenting staffing and shared staffing.

Appendix E of 10 CFR 50 describes the Emergency Response Data System (ERDS), which is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC that provides an automated transmission of a limited set of selected parameters. A parameter set appropriate for each type of SMR would be needed.

NUREG-0654, Table B-1 (see Table 7-1) provides the specific guidance for staffing requirements for nuclear power plant emergencies. This table specifies a minimum of 10 on-shift responders in four functional areas and specifies 7 on-shift responders who perform response duties that may be performed by shift personnel in additional to their other assigned functions, with firefighting and site access control staffed on a site-specific basis. This table also provides for a shared shift foreman for a shared control room. This type of shared staff function will be particularly relevant for SMRs with multiple reactors and a shared control room or rooms. This table also specifies the number of augmenting responders within 30 and 60 minute timeframes. A number of current nuclear power plants have longer augmentation times than 30 and 60 minutes that have been approved on a case-by-case basis. For SMRs, with passive safety features, the time to augment the emergency staff will be relevant and depend upon the safety features and their impact in accident progression.

In NSIR/DPR-ISG-01, it states that there have been several requests to the NRC to combine EOFs for multiple plants within a state or in multiple states, where an EOF could serve multiple units or units with more than one type of reactor technology. This may apply to SMRs, and EOF staff will need to be capable of understanding plant conditions for each type of reactor technology, particularly if the EOF for an SMR is co-located with different reactor technologies.

The four risk-significant emergency planning standards are classifying an emergency event (emergency action levels), notifying emergency responders and offsite officials of a declared emergency (including alert and notification systems), performing dose assessment and developing protective actions. All of these standards apply to SMRs.

In SECY-11-0152, there is a discussion on an emergency preparedness framework for SMRs that includes an example of a scalable EPZ, based on the dose a distances from the site and utilizing the EPA PAGs. The NRC has licensed several small reactors with an EPZ of 5 miles for plume (and 30 miles for ingestion), including Fort St. Vrain HTGR (842 MWt), Big Rock Point BWR (240 MWt) and La Crosse BWR (165 MWt). With the SMR passive safety features and the potential for reduced accident source terms and fission product releases, it may be appropriate for SMRs to develop similarly reduced EPZ sizes using a dose/distance approach.

8. SUMMARY

The current wave of small modular reactor (SMR) designs all have the goal of reducing the cost of management and operations. By optimizing the system, the goal is to make these power plants safer, cheaper to operate and maintain, and more secure. In particular, the reduction in plant staffing can result in significant cost savings. The introduction of advanced reactor designs and increased use of advanced automation technologies in existing nuclear power plants will likely change the roles, responsibilities, composition, and size of the crews required to control plant operations. Similarly, certain security staffing requirements for traditional operational nuclear power plants may not be appropriate or necessary for SMRs due to the simpler, safer and more automated design characteristics of SMRs.

As a first step in a process to identify where regulatory requirements may be met with reduced staffing and therefore lower cost, this report identifies the regulatory requirements and associated guidance utilized in the licensing of existing reactors. The potential applicability of these regulations to advanced SMR designs is addressed taking into account the unique features of these types of reactors. Specific focus was placed on the potential for reduction in operating, security, and emergency response staffing. The following discussion presents some preliminary insights from this effort.

Operating Staff

The design features and concepts of operations for new generations of advanced reactors, as well as the introduction of new automated systems into existing plants, may require fewer licensed operators. Unless and until the NRC develops and establishes new regulations for MCR staffing for advanced reactors, applicants will have to submit exemption requests from the applicable regulations included primarily in 10 CFR 50.54(m). The exemptions may include variations in the prescribed number, composition, or qualifications of licensed operators. In SECY-02-0180, SECY-11-0098, and in NUREG-1791, the NRC has acknowledged that there may be a case for exemptions from MCR staffing regulations for advanced reactor designs, and that the regulations would not address all of the potential MCR configurations that may be proposed (e.g., more than two reactors controlled from the same MCR). The NRC staff has proposed a policy to address advanced reactor MCR staffing issue in both the near-term (i.e., no advanced SMR operating experience) and long term (i.e., subsequent to operational experience of licensed SMRs). This policy would involve evaluating requests for exemptions from 10CFR 54 (m) from both DC and CL applicants prior to the existence of SMR operating experience. Eventually, after operating experience of SMRs has been gained, the NRC would revise existing regulations and develop other regulations to provide specific control room staffing requirements for SMRs.

As outlined in SECY-11-0098, the NRC staff intends to pursue the short term strategy by revising the Standard Review Plan (NUREG-0800) and the two NUREG documents that deal directly with staff evaluation of control room staffing issues and related human engineering aspects, NUREG-0711 and NUREG-1791 (summarized in Section 3.2). The NRC intends to revise these staff guidance documents so that they identify and address to the extent possible differences in advanced reactor designs that might impact conduct-of-operations, operator performance and staffing levels.

Security

The NRC's physical security regulations for NPPs are generally technology neutral and therefore applicable for advanced SMRs. However, licensees for SMRs may wish to explore strategies to reduce cost and staffing while still achieving regulatory compliance. Each NPP is operated differently and the appropriate staffing level for the facility is determined considering: the number of personnel needed to ensure the protection of nuclear material and facilities; the number of personnel needed to maintain security programs to support the sustainability of a physical protection program, including the protective strategy; and the number of personnel needed to handle normal daily security operations. An overarching goal for reactor security is to establish and maintain a physical security program and infrastructure, to include a security organization capable of providing high assurance of the protection of nuclear material and facilities against adversary attack.

The NRC has stated expectations that advanced reactor designs will provide enhanced measures of safety and security. Additional research may be needed to assess the efficacy of any new security measures. In addition the NRC expects reactor designers to integrate security into the design and conduct a security assessment to evaluate the level of protection provided. The design approaches for SMR safety related systems potentially provide a larger measure of "intrinsic" security, although this would have to be demonstrated. In addition, depending on the particular SMR design, the plant footprint is likely to be smaller than that for a traditional LWR with a commensurate smaller number of targets and target sets and overall smaller footprint for security operations. This factor may in itself demonstrate cost savings for SMR security.

Emergency Response

While some of the emergency planning regulations, regulatory guides and other guidance are fully applicable to SMRs, there are some aspects that warrant evaluation with respect to the differences between SMRs and light water power reactors. These can include, but are not limited to, size of the emergency planning zone, notification times, shared facilities, collocation with other SMRs and other nuclear power reactors, number of staffing positions, and times for augmenting staffing and shared staffing. As an example, with the SMR passive safety features and the potential for reduced accident source terms and fission product releases, it may be appropriate for SMRs to develop reduced EPZ sizes using a dose/distance approach. There are precedents for this since the NRC has licensed several small reactors with an EPZ of 5 miles for plume (and 30 miles for ingestion).

NUREG-0654, Table B-1 (see Table 7-1) provides the specific guidance for staffing requirements for nuclear power plant emergencies. This table specifies a minimum of 10 on-shift responders in four functional areas and specifies 7 on-shift responders who perform response duties that may be performed by shift personnel in additional to their other assigned functions, with firefighting and site access control staffed on a site-specific basis. This type of shared staff function will be particularly relevant for SMRs with multiple reactors and shared control rooms. This table also specifies the number of augmenting responders within 30 and 60 minute timeframes. A number of current nuclear power plants have longer augmentation times than 30 and 60 minutes that have been approved on a case-by-case basis. For SMRs, with passive safety features, the time to augment the emergency staff will be relevant and depend upon the safety features and their impact in accident progression.

9. REFERENCES

1. USG (United States Government), 2008. Code of Federal Regulations, http://www.gpoaccess.gov/CFR/INDEX.HTML.
2. "Nuclear Power Plant Licensing Process," NUREG/BR-0298, Rev 2, U.S. Nuclear Regulatory Commission, Washington, DC, 2004.
3. "Standard Review Plan for the Review of Safety Analysis Reports Nuclear Power Plants, NUREG-0800, Draft Revision 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1996.
4. "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Regulatory Guide 1.70, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, 1978.
5. http://www.nrc.gov/about-nrc/regulatory/standards-dev.html.
6. Ankrum, et.al, "Codes and Standards and Other Guidance Cited in Regulatory Documents," NUREG/CR-5973, Rev. 3, Pacific Northwest Laboratory, Richland WA., August 1996.
7. "Clarification of TMI Action Plan Requirements," NUREG-0737, U.S. Nuclear Regulatory Commission, Washington, DC, November 1980.
8. Southern Nuclear Operating Company, "Vogtle Electric Generating Plant Units 3 and 4 Combined License Application: Part 2 - Final Safety Analysis Report," NRC ADAMS Number ML11180A100, June 24, 2011.
9. "TMI-Related Requirements for New Operating Licenses," NUREG-0694, U.S. Nuclear Regulatory Commission, Washington, DC, June 1980.
10. J. Persensky et al., "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," NUREG-1791, U.S. Nuclear Regulatory Commission, Washington, DC, July 2005.
11. J.M. O'Hara et al., "Human Factors Engineering Program Review Model," NUREG-0711, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2012.
12. B.P. Hallbert et al., "A Study of Control Room Staffing Levels for Advanced Reactors," NUREG.IA-0137, U.S. Nuclear Regulatory Commission, Washington, DC, November 2000.
13. C. Plott et al., "Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," NUREG/CR-6838, February 2004.
14. "Legal and Financial Policy Issues Associated with Licensing New Nuclear Power Plants," SECY-02-0180, U.S. Nuclear Regulatory Commission, Washington, DC, October 2002.
15. "Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs." SECY-10-0034, U.S. Nuclear Regulatory Commission, Washington, DC, March 2010.

16. “Operator Staffing for Small or Multi-Module Nuclear Power Plant Facilities,” SECY-11-0098, U.S. Nuclear Regulatory Commission, Washington, DC, July 2011.
17. “Recommendations for Enhancing Reactor Safety in the 21st Century” (ML111861807), issued July 13, 2011.
18. “NEI Position Paper: Control Room Staffing for Small Reactors,” Nuclear Energy Institute, Washington DC, September 2011 (NRC ADAMS number ML120690009).
19. “Gap Analysis Summary Report,” NP-RP-0612-023, NuScale Power, LLC, Corvallis Oregon, July 2012.
20. “General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant,” ANSI/ANS-54.1-1989, American Nuclear Society, April 1989.
21. Whitehead, D.W., C.S. Potter, III, and S.L. O’Connor. “Nuclear Power Plant Security Assessment Technical Manual,” SAND2007-5591, Sandia National Laboratories, Albuquerque, NM, 2007.
22. “Nuclear Power Plant Security Assessment Format and Content Guide,” Part 1 of 3, Information Systems Laboratories, Rockville MD, 2007.
23. Wyss, G.D., J.F. Clem, J.L. Darby, K. Dunphy-Guzman, J.P Hinton, K.W. Mitchiner, 2010. “Risk-Based Cost Benefit Analysis for Security Assessment Problems,” SAND2010-5095C, presented at the IEEE 44th Annual International Carnahan Conference on Security Technology, October 5-8, San Jose, CA.
24. http://www.nrc.gov/reading-rm/doc-collections/enforcement/security/index.html
25. FR (Federal Register), 2008. “Policy Statement on the Regulation of Advanced Reactors,” 73 Federal Register 60612.
26. “*Report to Congress: Advanced Reactor Licensing*,” U.S. Nuclear Regulatory Commission, Washington DC, 2012.
27. “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in support of Nuclear Power Plants,” NUREG-0654/FEMA-REP-1, U.S. Nuclear Regulatory Commission and Federal Emergency Management Agency, Washington, DC.
28. “Methodology for Development of Emergency Action Levels,” NEI 99-01, Nuclear Energy Institute, Washington DC, January 2003
29. “Emergency Response Planning and Preparedness for Nuclear Power Reactors,” Regulatory Guide, RG 1.101, U.S. Nuclear Regulatory Commission, Washington, DC, Revision 4, October 2003.
30. “Methodology for the Development of Emergency Action Levels,” NUMARC/NESP-007.
31. “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents,” EPA-400-R-90-001, Environmental Protection Agency, Washington DC, 1992.
32. “Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants,” NUREG-0396, U.S. Nuclear Regulatory Commission, Washington, DC, 1978.

33. "Functional Criteria for Emergency Response Facilities," NUREG-0696, U.S. Nuclear Regulatory Commission, Washington, DC, February 1981.

34. "Clarification of TMI Action Plan Requirements," Supplement 1, "Requirements for Emergency Response Capability," NUREG-0737, U.S. Nuclear Regulatory Commission, Washington, DC, January 1983.

35. "Emergency Planning for Nuclear Power Plants," Interim Staff Guidance, NSIR/DPR-ISG-01, U.S. Nuclear Regulatory Commission, Washington, DC, November 2011.

36. "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," SECY-11-0152, U.S. Nuclear Regulatory Commission, Washington, DC, October 28, 2011

37. NRC's Regulatory Issue Summary 2011-02, "Licensing Submittal Information and Design Development Activities for Small Modular Reactor Designs," dated February 2, 2011.

APPENDIX A

APPLICABILITY OF LICENSING EXISTING LICENSING REGULATIONS TO SMRS

Table A-1. Applicability of 10 CFR Part 50 Regulations to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	General Provisions		
50.1	Basis, purpose, and procedures applicable.	Applicable	Technology neutral.
50.2	Definitions.		
50.3	Interpretations.		
50.4	Written communications.		
50.5	Deliberate misconduct.		
50.7	Employee protection.		
50.8	Information collection requirements: OMB approval.		
50.9	Completeness and accuracy of information.		
	Requirement of License, Exceptions		
50.10	License required; limited work authorization.	Applicable	Technology neutral.
50.11	Exceptions and exemptions from licensing requirements.		
50.12	Specific exemptions.		
50.13	Attacks and destructive acts by enemies of the United States; and defense activities.		
	Classification and Description of Licenses		
50.20	Two classes of licenses.	Applicable	Technology neutral. Irrelevant to DC applicant.
50.21	Class 104 licenses; for medical therapy and research and development facilities.		
50.22	Class 103 licenses; for commercial and industrial facilities.		
50.23	Construction permits.		

Table A-1. Applicability of 10 CFR Part 50 Regulations to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Applications for Licenses, Certifications, and Regulatory Approvals; Form; Contents; Ineligibility of Certain Applicants		
50.30	Filing of applications for licenses; oath or affirmation.	Applicable	Technology neutral. Irrelevant to DC applicant.
50.31	Combining applications.		
50.32	Elimination of repetition.		
50.33	Contents of applications; general information.	Applicable	Technology neutral. Only (a) – (c) and (j) are relevant to a DC applicant.
50.34	Contents of applications; technical information.	Applicable (a) – (e)	(a) through (e) are technology neutral. (f) is titled *Additional TMI-related requirements*. Heavily focused on LWR concepts. For LMRs, lack of specific design features prohibits a judgment on much of this section. For iPWRs, would either require exemption from requirements for irrelevant SSCs – e.g., auxiliary feedwater system, SPDS and Containment Purge, or garner NRC agreement that functions are either irrelevant or alternately met.
50.34a	Design objectives for equipment to control releases of radioactive material in effluents— nuclear power reactors.	Applicable	Technology neutral
50.35	Issuance of construction permits.		
50.36	Technical specifications.		
50.36a	Technical specifications on effluents from nuclear power reactors.		
50.36b	Environmental conditions.	Applicable	Technology neutral. Irrelevant to DC applicant.
50.37	Agreement limiting access to Classified Information.	Applicable	Technology neutral
50.38	Ineligibility of certain applicants.		
50.39	Public inspection of applications.		

Table A-1. Applicability of 10 CFR Part 50 Regulations to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Standards for Licenses, Certifications, and Regulatory Approvals		
50.40	Common standards.	Applicable	Technology neutral
50.41	Additional standards for class 104 licenses.		
50.42	Additional standard for class 103 licenses.		
50.43	Additional standards and provisions affecting class 103 licenses and certifications for commercial power.		
50.44	Combustible gas control for nuclear power reactors.	Applicable	NuScale claims design specific features will warrant an exemption. Other reactor designs would likewise need to establish a design specific aspect that would negate the need for this requirement.
50.45	Standards for construction permits, operating licenses, and combined licenses.	Applicable	Technology neutral
50.46	Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.	Partially Applicable	Applicable for iPWRs. CFR 50.46 and Appendix K, which it invokes, are strongly focused on LWR technology. This requirement would be functionally applicable to SMR, but new requirements would have to be written.
50.46a	Acceptance criteria for reactor coolant system venting systems.	Applicable	Reactor designs must ensure that an accumulation of noncondensable gases could not interfere with post-accident circulation.
50.47	Emergency plans.	Applicable	Technology neutral.
50.48	Fire protection.	Not Applicable.	New reactors will fall under NFPA 805, which is technology neutral.
50.49	Environmental qualification of electric equipment important to safety for nuclear power plants.	Applicable	Technology Neutral

Table A-1. Applicability of 10 CFR Part 50 Regulations to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Issuance, Limitations, and Conditions of Licenses and Construction Permits		
50.50	Issuance of licenses and construction permits.	Applicable	Technology Neutral
50.51	Continuation of license.		
50.52	Combining licenses.		
50.53	Jurisdictional limitations.		
50.54	Conditions of licenses.	Applicable	Staffing requirements in (m) are based on numbers of reactor units and control rooms. Any exemption request would be evaluated by the NRC based on the guidance in NUREG-1791.
50.55	Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses.	Applicable	Technology neutral.
50.55a	Codes and standards.	Partially Applicable	Applicable to iPWRs. Many standards listed have been developed with a focus on LWR technology. All relevant Standards Development Organizations would have to evaluate the applicability of each of their codes to LMRs.
50.56	Conversion of construction permit to license; or amendment of license.	Applicable	Technology neutral
50.57	Issuance of operating license.		
50.58	Hearings and report of the Advisory Committee on Reactor Safeguards.		
50.59	Changes, tests and experiments.		
50.60	Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation.	Partially Applicable	Applicable to iPWRs. Many standards listed have been developed with a focus on LWR technology. All relevant Standards Development Organizations would have to evaluate the applicability of each of their codes to LMRs.
50.61	Fracture toughness requirements for protection against pressurized thermal shock events.		
50.61a	Alternate fracture toughness requirements for protection against pressurized thermal shock events.		

Table A-1. Applicability of 10 CFR Part 50 Regulations to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
50.62	Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light- water-cooled nuclear power plants.	Partly Applicable	Applicable for iPWRs. Heavily focused on LWR concepts. For LMRs, lack of specific design features prohibits a judgment on much of this section.
50.63	Loss of all alternating current power.	Applicable	Technology neutral
50.64	Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors.		
50.65	Requirements for monitoring the effectiveness of maintenance at nuclear power plants.		
50.66	Requirements for thermal annealing of the reactor pressure vessel.	Partly Applicable	Applicable for iPWRs. Heavily focused on LWR concepts. For LMR, lack of specific design features prohibits a judgment on much of this section.
50.67	Accident source term.	Applicable	Technology neutral
50.68	Criticality accident requirements.		
	Inspections, Records, Reports, Notifications		
50.69	Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors.	Applicable	Technology neutral
50.70	Inspections.		
50.71	Maintenance of records, making of reports.		
50.72	Immediate notification requirements for operating nuclear power reactors.		
50.73	License event report system.		
50.74	Notification of change in operator or senior operator status.		
50.75	Reporting and recordkeeping for decommissioning planning.	Applicable	Technology neutral
50.76	Licensee's change of status; financial qualifications.		
	US/IAEA Safeguards Agreement		
50.78	Facility information and verification.	Applicable	Technology neutral

Table A-1. Applicability of 10 CFR Part 50 Regulations to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Transfers of Licenses—Creditors' Rights—Surrender of Licenses		
50.80	Transfer of licenses.	Applicable	Technology neutral
50.81	Creditor regulations.		
50.82	Termination of license.		
50.83	Release of part of a power reactor facility or site for unrestricted use.		
	Amendment of License or Construction Permit at Request of Holder		
50.90	Application for amendment of license, construction permit, or early site permit.	Applicable	Technology neutral
50.91	Notice for public comment; State consultation.		
50.92	Issuance of amendment.		
	Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission		
50.100	Revocation, suspension, modification of licenses, permits, and approvals for cause.	Applicable	Technology neutral
50.101	Retaking possession of special nuclear material.		
50.102	Commission order for operation after revocation.		
50.103	Suspension and operation in war or national emergency.	Applicable	Technology neutral
	Backfitting		
50.109	Backfitting.	Applicable	Technology neutral

Table A-1. Applicability of 10 CFR Part 50 Regulations to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Enforcement		
50.110	Violations.	Applicable	Technology neutral
50.111	Criminal penalties.		
	Additional Standards for Licenses, Certifications, and Regulatory Approvals		
50.120	Training and qualification of nuclear power plant personnel.	Applicable	Technology neutral
50.150	Aircraft impact assessment.		
	Appendices		
Appendix A	General Design Criteria for Nuclear Power Plants	See Table A-4	All are applicable to iPWRs, some are not applicable to LMRs..
Appendix B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants	Applicable	Technology neutral
Appendix C	A Guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Construction Permits and Combined Licenses		
Appendix D	[Reserved]		
Appendix E	Emergency Planning and Preparedness for Production and Utilization Facilities	Applicable	Technology neutral
Appendix F	Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities	Not Applicable	
Appendix G	Fracture Toughness Requirements	Partly Applicable	See 50.61
Appendix H	Reactor Vessel Material Surveillance Program Requirements	Partly Applicable	See 50.61
Appendix I	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents	Applicable	The intent of this Appendix is technology neutral, despite its name.
Appendix J	Primary Reactor Containment Leakage Testing for Water-	Applicable	The intent of this Appendix is

Table A-1. Applicability of 10 CFR Part 50 Regulations to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Cooled Power Reactors		technology neutral, despite its name.
Appendix K	ECCS [Emergency Core Cooling system] Evaluation Models	Partly Applicable	Applicable for iPWRs. Heavily focused on LWR concepts. For LMRs, lack of specific design features prohibits a judgment on much of this section.
Appendix L	[Reserved]		
Appendix M	[Reserved]		
Appendix N	Standardization of Nuclear Power Plant Designs: Permits To Construct and Licenses To Operate Nuclear Power Reactors of Identical Design at Multiple Sites	Applicable	Technology neutral
Appendix O	[Reserved]		
Appendix P	[Reserved]		
Appendix Q	Pre-application Early Review of Site Suitability Issues	Applicable	Technology neutral
Appendix R	Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979	Not Applicable	New reactors will use NFPA 805, which is technology neutral.
Appendix S	Earthquake Engineering Criteria for Nuclear Power Plants	Applicable	Technology neutral

Table A-2. 10 CFR Part 51 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	General Provisions		
51.1	Scope.	Applicable	These general provisions are applicable regardless of reactor design or license type.
51.2	Subparts.		
51.3	Resolution of conflict.		
51.4	Definitions.		
51.5	Interpretations.		
51.6	Specific exemptions.		
	Subpart A—National Environmental Policy Act—Regulations Implementing Section 102(2)		
51.10	Purpose and scope of subpart; application of regulations of Council on Environmental Quality.	Applicable	These general provisions are applicable regardless of reactor design or license type.
51.11	Relationship to other subparts. [Reserved]		
51.12	Application of subpart to ongoing environmental work.		
51.13	Emergencies.		
51.14	Definitions.		
51.15	Time schedules.		
51.16	Proprietary information.		
51.17	Information collection requirements; OMB approval.		
	Preliminary Procedures		
	Classification of Licensing and Regulatory Actions		
51.20	Criteria for and identification of licensing and regulatory actions requiring environmental impact statements.	Applicable	Not applicable for a standard design certification. Applicable for a combined license.
51.21	Criteria for and identification of licensing and regulatory actions requiring environmental assessments.	Applicable	Technology neutral
51.22	Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review.	Applicable	Not applicable for a standard design certification. Applicable for a combined license.

Table A-2. 10 CFR Part 51 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
51.23	Temporary storage of spent fuel after cessation of reactor operation—generic determination of no significant environmental impact.	Applicable	This requirement is applicable regardless of reactor design or license type.
	Determinations to Prepare Environmental Impact Statements, Environmental Assessments or Findings of No Significant Impact, and Related Procedures		
51.25	Determination to prepare environmental impact statement or environmental assessment; eligibility for categorical exclusion.	Applicable	This requirement is applicable regardless of reactor design or license type.
51.26	Requirement to publish notice of intent and conduct scoping process.	Applicable	Not applicable for a standard design certification. Applicable for a combined license.
51.27	Notice of intent.		
	Scoping		
51.28	Scoping—participants.	Applicable	Not applicable for a standard design certification. Applicable for a combined license.
51.29	Scoping-environmental impact statement and supplement to environmental impact statement.		
	Environmental Assessment		
51.30	Environmental assessment.	Applicable	Only section 51.30(b) and (d) are applicable to a standard design certification.
51.31	Determinations based on environmental assessment.	Applicable	Only section 51.31(b) is applicable to a standard design certification.
	Finding of No Significant Impact		
51.32	Finding of no significant impact.	Applicable	These requirements are applicable regardless of reactor design or license type.
51.33	Draft finding of no significant impact; distribution.		
51.34	Preparation of finding of no significant impact.		
51.35	Requirement to publish finding of no significant impact; limitation on Commission action.		

Table A-2. 10 CFR Part 51 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Environmental Reports and Information—Requirements Applicable to Applicants and Petitioners for Rulemaking		
	General		
51.40	Consultation with NRC staff.	Applicable	These general provisions are applicable regardless of reactor design or license type.
51.41	Requirement to submit environmental information.		
	Environmental Reports—General Requirements		
51.45	Environmental report.	Applicable	This requirement is applicable regardless of reactor design or license type.
	Environmental Reports—Production and Utilization Facilities		
51.49	Environmental report—limited work authorization.	Applicable	Applicable only for a construction permit or combined license.
51.50	Environmental report-construction permit, early site permit, or combined license stage.	Applicable	Applicable only for a construction permit, early site permit, or combined license.
51.51	Uranium fuel cycle environmental data—Table S-3.		
51.52	Environmental effects of transportation of fuel and waste—Table S-4.		
51.53	Post-construction environmental reports.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
51.54	Environmental report—manufacturing license.	Applicable	Applicable only for a manufacturing license.
51.55	Environmental report-standard design certification.	Applicable	Applicable only for a standard design certification.
51.58	Environmental report-number of copies; distribution.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.

Table A-2. 10 CFR Part 51 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Environmental Reports—Materials Licenses		
51.60	Environmental report—materials licenses.	Potentially Applicable – see comment	These requirements are only applicable to material licenses. Facilities manufacturing modular reactor designs that are installed at a reactor site are subject to these requirements.
51.61	Environmental report—independent spent fuel storage installation (ISFSI) or monitored retrievable storage installation (MRS) license.		
51.62	Environmental report—land disposal of radioactive waste licensed under 10 CFR part 61.		
51.66	Environmental report—number of copies; distribution.		
51.67	Environmental information concerning geologic repositories.		
	Environmental Reports—Rulemaking		
51.68	Environmental report—rulemaking.		
	Environmental Impact Statements		
	Draft Environmental Impact Statements—General Requirements		
51.70	Draft environmental impact statement—general.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
51.71	Draft environmental impact statement—contents.		
51.72	Supplement to draft environmental impact statement.		
51.73	Request for comments on draft environmental impact statement.		
51.74	Distribution of draft environmental impact statement and supplement to draft environmental impact statement; news releases.		
	Draft Environmental Impact Statements—Production and Utilization Facilities		
51.75	Draft environmental impact statement—construction permit, early site permit, or combined license.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
51.76	Draft environmental impact statement—limited work authorization.		

Table A-2. 10 CFR Part 51 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
51.77	Distribution of draft environmental impact statement.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
	Draft Environmental Impact Statements—Materials Licenses		
51.80	Draft environmental impact statement—materials license.	Not Applicable	These requirements are only applicable to material licenses.
51.81	Distribution of draft environmental impact statement.		
	Draft Environmental Impact Statements—Rulemaking		
51.85	Draft environmental impact statement—rulemaking.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
51.86	Distribution of draft environmental impact statement.		
	Legislative Environmental Impact Statements—Proposals for Legislation		
51.88	Proposals for legislation.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
	Final Environmental Impact Statements—General Requirements		
51.90	Final environmental impact statement—general.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
51.91	Final environmental impact statement—contents.		
51.92	Supplement to the final environmental impact statement.		
51.93	Distribution of final environmental impact statement and supplement to final environmental impact statement; news releases.		

Table A-2. 10 CFR Part 51 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
51.94	Requirement to consider final environmental impact statement.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
	Final Environmental Impact Statements—Production and Utilization Facilities		
51.95	Post-construction environmental impact statements.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
	Final Environmental Impact Statements—Materials Licenses		
51.97	Final environmental impact statement—materials license.	Not Applicable	These requirements are only applicable to material licenses.
	Final Environmental Impact Statements—Rulemaking		
51.99	[Reserved]		
	NEPA Procedure and Administrative Action		
	General		
51.100	Timing of Commission action.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
51.101	Limitations on actions.		
51.102	Requirement to provide a record of decision; preparation.		
51.103	Record of decision—general.		
51.104	NRC proceeding using public hearings; consideration of environmental impact statement.		

Table A-2. 10 CFR Part 51 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Production and Utilization Facilities		
51.105	Public hearings in proceedings for issuance of construction permits or early site permits; limited work authorizations.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
51.105a	Public hearings in proceedings for issuance of manufacturing licenses.		
51.106	Public hearings in proceedings for issuance of operating licenses.		
51.107	Public hearings in proceedings for issuance of combined licenses; limited work authorizations.		
51.108	Public hearings on Commission findings that inspections, tests, analyses, and acceptance criteria of combined licenses are met		
	Materials Licenses		
51.109	Public hearings in proceedings for issuance of materials license with respect to a geologic repository.	Not Applicable	These requirements are only applicable to material licenses.
	Rulemaking		
51.110	[Reserved]		
	Public Notice of and Access to Environmental Documents		
51.116	Notice of intent.	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
51.117	Draft environmental impact statement—notice of availability.		
51.118	Final environmental impact statement—notice of availability.		
51.119	Publication of finding of no significant impact; distribution.	Applicable	
51.120	Availability of environmental documents for public inspection.		
51.121	Status of NEPA actions.		
51.122	List of interested organizations and groups.		
51.123	Charges for environmental documents; distribution to public; distribution to governmental agencies.		

Table A-2. 10 CFR Part 51 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Commenting		
51.124	Commission duty to comment.	Applicable	Requirement is for NRC review of environmental assessments performed by other federal agencies.
	Responsible Official		
51.125	Responsible official.	Applicable	
	Appendices		
Appendix A	Format for Presentation of Material in Environmental Impact Statements	Applicable	Not applicable for a standard design certification. Applicable for construction permit, early site permit, and combined license.
Appendix B	Environmental Effect of Renewing the Operating License of a Nuclear Power Plant	Applicable	Only pertains to the renewal of an operating license.

Table A-3. 10 CFR Part 52 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	General Provisions		
52.0	Scope; applicability of 10 CFR Chapter I provisions.	Applicable	These general provisions are applicable regardless of reactor design or license type.
52.1	Definitions.		
52.2	Interpretations.		
52.3	Written communications.		
52.4	Deliberate misconduct.		
52.5	Employee protection.		
52.6	Completeness and accuracy of information.		
52.7	Specific exemptions.		
52.8	Combining licenses; elimination of repetition.		
52.9	Jurisdictional limits.		
52.10	Attacks and destructive acts.		
52.11	Information collection requirements: OMB approval.		
	Subpart A—Early Site Permits		
52.12	Scope of subpart.	Not Applicable	Applicable only for applicants of an early site permit.
52.13	Relationship to other subparts.		
52.15	Filing of applications.		
52.16	Contents of applications; general information.		
52.17	Contents of applications; technical information.		
52.18	Standards for review of applications.		
52.21	Administrative review of applications; hearings.		
52.23	Referral to the Advisory Committee on Reactor Safeguards (ACRS).		
52.24	Issuance of early site permit.		
52.25	Extent of activities permitted.		
52.26	Duration of permit.		
52.27	Limited work authorization after issuance of early site permit.		

Table A-3. 10 CFR Part 52 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
52.28	Transfer of early site permit.	Not Applicable	Applicable only for applicants of an early site permit.
52.29	Application for renewal.		
52.31	Criteria for renewal.		
52.33	Duration of renewal.		
52.35	Use of site for other purposes.		
52.39	Finality of early site permit determinations.		
	Subpart B—Standard Design Certifications		
52.41	Scope of subpart.	Applicable	Subpart B governs the certification by rulemaking of the design of a nuclear power plant. Standard design certifications are included in the Appendices in 10 CFR Part 52. These regulations establish general requirements for design certifications that are independent of the reactor type.
52.43	Relationship to other subparts.		
52.45	Filing of applications.		
52.46	Contents of applications; general information.		
52.47	Contents of applications; technical information.		
52.48	Standards for review of applications.		
52.51	Administrative review of applications.		
52.53	Referral to the Advisory Committee on Reactor Safeguards (ACRS).		
52.54	Issuance of standard design certification.		
52.55	Duration of certification.		
52.57	Application for renewal.	Applicable	Applicable only for standard design certification renewal.
52.59	Criteria for renewal.		
52.61	Duration of renewal.		
52.63	Finality of standard design certifications.	Applicable	General requirements for design certifications that are independent of the reactor type.
	Subpart C—Combined Licenses		
52.71	Scope of subpart.	Applicable	Subpart C sets out the requirements and procedures applicable to issuance of combined licenses for nuclear

Table A-3. 10 CFR Part 52 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
			power facilities. These regulations establish general requirements for combined licenses that are independent of the reactor type.
52.73	Relationship to other subparts.	Applicable	Subpart C sets out the requirements and procedures applicable to issuance of combined licenses for nuclear power facilities. These regulations establish general requirements for combined licenses that are independent of the reactor type.
52.75	Filing of applications.		
52.77	Contents of applications; general information.		
52.79	Contents of applications; technical information in final safety analysis report.		
52.80	Contents of applications; additional technical information.		
52.81	Standards for review of applications.		
52.83	Finality of referenced NRC approvals; partial initial decision on site suitability.		
52.85	Administrative review of applications; hearings.		
52.87	Referral to the Advisory Committee on Reactor Safeguards (ACRS).		
52.89	Reserved.		
52.91	Authorization to conduct limited work authorization activities.		
52.93	Exemptions and variances.		
52.97	Issuance of combined licenses.		
52.98	Finality of combined licenses; information requests.		
52.99	Inspection during construction.		
52.103	Operation under a combined license.		
52.104	Duration of combined license.		
52.105	Transfer of combined license.		
52.107	Application for renewal.		
52.109	Continuation of combined license.		
52.110	Termination of license.		
	Subpart D [Reserved]		

Table A-3. 10 CFR Part 52 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
	Subpart E—Standard Design Approvals		
52.131	Scope of subpart.	Applicable	Subpart E governs the NRC staff review and approval of a final standard design. Standard designs are not subject to the rule making process utilized in Subpart B and are not incorporated into 10 CFR Part 52 as an appendix. Subpart E requirements establish the general requirements for applicants for a standard design approval that are independent of the reactor type. It is not expected, but is possible, that a reactor design would be licensed under this Subpart.
52.133	Relationship to other subparts.		
52.135	Filing of applications.		
52.136	Contents of applications; general information.		
52.137	Contents of applications; technical information.		
52.139	Standards for review of applications.		
52.141	Referral to the Advisory Committee on Reactor Safeguards (ACRS).		
52.143	Staff approval of design.		
52.145	Finality of standard design approvals; information requests.		
52.147	Duration of design approval.		
	Subpart F—Manufacturing Licenses		
52.151	Scope of subpart.	Potentially Applicable – see comment	Subpart F sets out the requirements and procedures applicable to issuance of a license authorizing manufacture of nuclear power reactors to be installed at sites not identified in the manufacturing license application. Facilities manufacturing modular reactor designs that are installed at a reactor site are subject to these requirements.
52.153	Relationship to other subparts.		
52.155	Filing of applications.		
52.156	Contents of applications; general information.		
52.157	Contents of applications; technical information in final safety analysis report.		
52.158	Contents of application; additional technical information.		
52.159	Standards for review of application.		
52.161	Reserved.		
52.163	Administrative review of applications; hearings.		
52.165	Referral to the Advisory Committee on Reactor Safeguards (ACRS).		
52.167	Issuance of manufacturing license.		
52.169	Reserved.		

Table A-3. 10 CFR Part 52 Applicability to iPWRs and LMRs.

Regulation	Regulation Title	SMR Applicability	Comments
52.171	Finality of manufacturing licenses; information requests.	Potentially Applicable – see comment	Subpart F sets out the requirements and procedures applicable to issuance of a license authorizing manufacture of nuclear power reactors to be installed at sites not identified in the manufacturing license application. Facilities manufacturing modular reactor designs that are installed at a reactor site are subject to these requirements.
52.173	Duration of manufacturing license.		
52.175	Transfer of manufacturing license.		
52.177	Application for renewal.		
52.179	Criteria for renewal.		
52.181	Duration of renewal.		
	Subpart G [Reserved]		
	Subpart H—Enforcement		
52.301	Violations.	Applicable	These requirements are independent of the reactor design and type of license.
52.303	Criminal penalties.		
	Appendices		
Appendix A	Design Certification Rule for the U.S. Advanced Boiling Water Reactor	Not Applicable	These appendices are specific to standard designs previously certified under Subpart C of 10 CFR Part 52.
Appendix B	Design Certification Rule for the System 80+ Design		
Appendix C	Design Certification Rule for the AP600 Design		
Appendix D	Design Certification Rule for the AP1000 Design		
Appendixes E Through M	[Reserved]		
Appendix N	Standardization of Nuclear Power Plant Designs: Combined Licenses to Construct and Operate Nuclear Power Reactors of Identical Design at Multiple Sites	Applicable	Applicable to applicants for a combined license but not applicable to applicants for a standard design certification.

Table A-4. 10 CFR Part 50 Appendix A Applicability to iPWRs and LMRs.			
General Design Criteria	**Title**	**SMR Applicability**	**Comments**
	I. Overall Requirements:		
1	Quality Standards and Records	Applicable	Technology Neutral
2	Design Bases for Protection Against Natural Phenomena		
3	Fire Protection		
4	Environmental and Dynamic Effects Design Bases		
5	Sharing of Structures, Systems, and Components		
	II. Protection by Multiple Fission Product Barriers:		
10	Reactor Design	Applicable	Technology Neutral
11	Reactor inherent Protection		
12	Suppression of Reactor Power Oscillations		
13	Instrumentation and Control		
14	Reactor Coolant Pressure Boundary		
15	Reactor Coolant System Design		
16	Containment Design		
17	Electric Power Systems		
18	Inspection and Testing of Electric Power Systems		
19	Control Room		
	III. Protection and Reactivity Control Systems:		
20	Protection System Functions	Applicable	Technology Neutral

Table A-4. 10 CFR Part 50 Appendix A Applicability to iPWRs and LMRs.			
General Design Criteria	**Title**	**SMR Applicability**	**Comments**
21	Protection System Reliability and Testability	Applicable	Technology Neutral
22	Protection System Independence		
23	Protection System Failure Modes		
24	Separation of Protection and Control Systems		
25	Protection System Requirements for Reactivity Control Malfunctions		
26	Reactivity Control System Redundancy and Capability		
27	Combined Reactivity Control Systems Capability		
28	Reactivity Limits		
29	Protection Against Anticipated Operational Occurrences		
	IV. Fluid Systems:		
30	Quality of Reactor Coolant Pressure Boundary	Applicable	Technology Neutral
31	Fracture Prevention of Reactor Coolant Pressure Boundary		
32	Inspection of Reactor Coolant Pressure Boundary		
33	Reactor Coolant Makeup	Applicable	Applicable to iPWRs, not applicable to LMRs since they do not have coolant makeup systems
34	Residual Heat Removal	Applicable	Technology Neutral
35	Emergency Core Cooling	Applicable	Applicable to iPWRs, not applicable to LMRs since they do not have ECCS
36	Inspection of Emergency Core Cooling System		
37	Testing of Emergency Core Cooling System		
38	Containment Heat Removal		

Table A-4. 10 CFR Part 50 Appendix A Applicability to iPWRs and LMRs.			
General Design Criteria	**Title**	**SMR Applicability**	**Comments**
39	Inspection of Containment Heat Removal System	Applicable	Applicable to iPWRs, not applicable to LMRs since they do not have ECCS
40	Testing of Containment Heat Removal System		
41	Containment Atmosphere Cleanup		
42	Inspection of Containment Atmosphere Cleanup Systems	Applicable	Applicable to iPWRs, not applicable to LMRs
43	Testing of Containment Atmosphere Cleanup Systems		
44	Cooling Water		
45	Inspection of Cooling Water System		
46	Testing of Cooling Water System		
	V. Reactor Containment:		
50	Containment Design Basis	Applicable	Technology Neutral
51	Fracture Prevention of Containment Pressure Boundary		
52	Capability for Containment Leakage Rate Testing		
53	Provisions for Containment Testing and Inspection		
54	Systems Penetrating Containment		
55	Reactor Coolant Pressure Boundary Penetrating Containment		
56	Primary Containment Isolation		
57	Closed Systems Isolation Valves		
	VI. Fuel and Radioactivity Control:		
60	Control of Releases of Radioactive Materials to the Environment	Applicable	Technology Neutral
61	Fuel Storage and Handling and Radioactivity Control		
62	Prevention of Criticality in Fuel Storage and Handling		
63	Monitoring Fuel and Waste Storage		

Table A-4. 10 CFR Part 50 Appendix A Applicability to iPWRs and LMRs.			
General Design Criteria	**Title**	**SMR Applicability**	**Comments**
64	Monitoring Radioactivity Releases	Applicable	Technology Neutral

APPENDIX B
APPLICABILITY OF EXISTING GUIDANCE DOCUMENTS TO SMRS

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
	CHAPTER 1			
	Introduction and General Description of Plant			
1.0	Introduction and Interfaces	03/2007	Applicable	Technology Neutral
	CHAPTER 2			
	Site Characteristics			
2.0	Site Characteristics and Site Parameters	03/2007	Applicable	Technology Neutral. Applicable to applicants for a combined license but not applicable to applicants for a standard design certification.
2.1.1, Rev. 3	Site Location and Description	03/2007		
2.1.2, Rev. 3	Exclusion Area Authority and Control	03/2007		
2.1.3, Rev. 3	Population Distribution	03/2007		
2.2.1-2.2.2, Rev. 3	Identification of Potential Hazards in Site Vicinity	03/2007		
2.2.3, Rev. 3	Evaluation of Potential Accidents	03/2007		
2.3.1, Rev. 3	Regional Climatology	03/2007		
2.3.2, Rev. 3	Local Meteorology	03/2007		
2.3.3, Rev. 3	Onsite Meteorological Measurements Programs	03/2007		
2.3.4, Rev. 3	Short-Term Atmospheric Dispersion Estimates for Accident Releases	03/2007		
2.3.5, Rev. 3	Long-Term Atmospheric Dispersion Estimates for Routine Releases	03/2007		
2.4.1, Rev. 3	Hydrologic Description	03/2007		
2.4.2, Rev. 4	Floods	03/2007		
2.4.3, Rev. 4	Probable Maximum Flood (PMF) on Streams and Rivers	03/2007		
2.4.4, Rev.3	Potential Dam Failures	03/2007		
2.4.5, Rev. 3	Probable Maximum Surge and Seiche Flooding	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
2.4.6, Rev. 3	Probable Maximum Tsunami Flooding	03/2007	Applicable	Technology Neutral. Applicable to applicants for a combined license but not applicable to applicants for a standard design certification.
2.4.7, Rev. 3	Ice Effects	03/2007		
2.4.8, Rev. 3	Cooling Water Canals and Reservoirs	03/2007		
2.4.9, Rev. 3	Channel Diversions	03/2007		
2.4.10, Rev. 3	Flooding Protection Requirements	03/2007		
2.4.11, Rev. 3	Low Water Considerations	03/2007		
2.4.12, Rev. 3	Groundwater	03/2007		
2.4.13, Rev. 3	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	03/2007		
2.4.14, Rev. 3	Technical Specifications and Emergency Operation Requirements	03/2007		
2.5.1, Rev. 4	Basic Geologic and Seismic Information	03/2007		
2.5.2, Rev. 4	Vibratory Ground Motion	03/2007		
2.5.3, Rev. 4	Surface Faulting	03/2007		
2.5.4, Rev. 3	Stability of Subsurface Materials and Foundations	03/2007		
2.5.5, Rev. 3	Stability of Slopes	03/2007		
	CHAPTER 3			
	Design of Structures, Components, Equipment, and Systems			
3.2.1, Rev. 2	Seismic Classification	03/2007	Applicable	Technology Neutral – underground siting can affect loads on SSCs.
3.2.2, Rev. 2	System Quality Group Classification	03/2007		
3.3.1, Rev. 3	Wind Loading	03/2007		
3.3.2, Rev. 3	Tornado Loads	03/2007		
3.4.1, Rev. 3	Internal Flood Protection for Onsite Equipment Failures	03/2007		
3.4.2, Rev. 3	Analysis Procedures	03/2007		
3.5.1.1, Rev. 3	Internally Generated Missiles (Outside Containment)	03/2007		
3.5.1.2, Rev. 3	Internally Generated Missiles (Inside Containment)	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
3.5.1.3, Rev. 3	Turbine Missiles	03/2007	Applicable	Technology Neutral
3.5.1.4, Rev. 3	Missiles Generated by Tornadoes and Extreme Winds	03/2007		
3.5.1.5, Rev. 4	Site Proximity Missiles (Except Aircraft)	03/2007		
3.5.1.6, Rev. 3	Aircraft Hazards	03/2007		
3.5.2, Rev. 3	Structures, Systems, and Components To Be Protected From Externally-Generated Missiles	03/2007		
3.5.3, Rev. 3	Barrier Design Procedures	03/2007		
3.6.1, Rev. 3	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	03/2007		
3.6.2, Rev 2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	03/2007		
3.6.3, Rev. 1	Leak-Before-Break Evaluation Procedures	03/2007	Generally Not Applicable	iPWRs and LMRs do not have primary piping at high pressure.
3.7.1, Rev. 3	Seismic Design Parameters	03/2007	Applicable	Technology Neutral
3.7.2, Rev. 3	Seismic System Analysis	03/2007		
3.7.3, Rev. 3	Seismic Subsystem Analysis	03/2007		
3.7.4, Rev. 2	Seismic Instrumentation	03/2007		
3.8.1, Rev. 2	Concrete Containment	03/2007	SMR specific	Different SMRs have different types of containments
3.8.2, Rev. 2	Steel Containment	03/2007		
3.8.3, Rev. 2	Concrete and Steel Internal Structures of Steel or Concrete Containments	03/2007		
3.8.4, Rev. 2	Other Seismic Category I Structures	03/2007	Applicable	Technology Neutral
3.8.5, Rev. 2	Foundations	03/2007		
3.9.1, Rev. 3	Special Topics for Mechanical Components	03/2007		
3.9.2, Rev. 3	Dynamic Testing and Analysis of Systems, Structures, and Components	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
3.9.3, Rev. 2	ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures	03/2007	Applicable	Technology Neutral
3.9.4, Rev. 3	Control Rod Drive Systems	03/2007		
3.9.5, Rev. 3	Reactor Pressure Vessel Internals	03/2007		
3.9.6, Rev. 3	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	03/2007		
3.9.7	Risk-Informed Inservice Testing of Pumps and Valves	08/1998		
3.9.8	Risk-Informed Inservice Inspection of Piping	09/2003		
3.10, Rev. 3	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	03/2007		
3.11, Rev. 3	Environmental Qualification of Mechanical and Electrical Equipment	03/2007		
3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports	03/2007		
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3	03/2007		
Branch Technical Position 3-1, Rev. 2	Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWRs	03/2007	Not Applicable	Applicable only to BWRs
Branch Technical Position 3-2, Rev. 2	Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary	03/2007		
Branch Technical Position 3-3, Rev. 3	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	03/2007	Applicable	Technology Neutral
Branch Technical Position 3-4, Rev. 2	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
	CHAPTER 4			
	Reactor			
4.2, Rev. 3	Fuel System Design	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
4.3, Rev. 3	Nuclear Design	03/2007		
4.4, Rev. 2	Thermal and Hydraulic Design	03/2007		
4.5.1, Rev. 3	Control Rod Drive Structural Materials	03/2007		
4.5.2, Rev. 3	Reactor Internal and Core Support Structure Materials	03/2007		
4.6, Rev. 2	Functional Design of Control Rod Drive System	03/2007		
Branch Technical Position 4-1, Rev. 3	Westinghouse Constant Axial Offset Control (CAOC)	03/2007	Not Applicable	Only applicable if SMR uses this operating control method
	CHAPTER 5			
	Reactor Coolant System and Connecting Systems	03/2007		
5.2.1.1, Rev. 3	Compliance With the Codes and Standards Rule, 10 CFR 50.55a	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
5.2.1.2, Rev. 3	Applicable Code Cases	03/2007		
5.2.2, Rev. 3	Overpressure Protection	03/2007		
5.2.3, Rev. 3	Reactor Coolant Pressure Boundary Materials	03/2007		
5.2.4, Rev. 2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	03/2007		
5.2.5, Rev. 2	Reactor Coolant Pressure Boundary Leakage Detection	03/2007		
5.3.1, Rev. 2	Reactor Vessel Materials	03/2007		
5.3.2, Rev. 2	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	03/2007		
5.3.3, Rev. 2	Reactor Vessel Integrity	03/2007		
5.4, Rev. 2	Reactor Coolant System Component and Subsystem Design	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
5.4.1.1, Rev. 2	Pump Flywheel Integrity (PWR)	03/2007	Applicable for some iPWRs	Some SMRs (e.g., NuScale) do not have reactor coolant pumps. Not applicable for LMRs.
5.4.2.1, Rev. 3	Steam Generator Materials	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
5.4.2.2, Rev. 2	Steam Generator Program	03/2007		
5.4.6, Rev. 4	Reactor Core Isolation Cooling System (BWR)	03/2007	Not Applicable	Applicable only to BWRs
5.4.7, Rev. 4	Residual Heat Removal (RHR) System	03/2007		
5.4.8, Rev. 3	Reactor Water Cleanup System (BWR)	03/2007		
5.4.11, Rev. 3	Pressurizer Relief Tank	03/2007	Applicable for some iPWRs	Some SMRs (e.g., NuScale) do not have a pressurizer relief tank. Not applicable for LMRs.
5.4.12, Rev. 1	Reactor Coolant System High Point Vents	03/2007	Applicable for iPWRs	Not applicable for LMRs
5.4.13	Isolation Condenser System (BWR)	03/2007	Not Applicable	Applicable only to BWRs
Branch Technical Position 5-1, Rev. 3	Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
Branch Technical Position 5-2, Rev. 3	Overpressure Protection of Pressurized-Water Reactors While Operating at Low Temperatures	03/2007	Applicable for iPWRs	Not applicable for LMRs
Branch Technical Position 5-3, Rev. 2	Fracture Toughness Requirements	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
Branch Technical Position 5-4, Rev. 4	Design Requirements of the Residual Heat Removal System	03/2007	Applicable for iPWRs	Applicable to system used for residual heat removal. Not applicable for LMRs.
	CHAPTER 6			
	Engineered Safety Features			
6.1.1, Rev. 2	Engineered Safety Features Materials	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
6.1.2, Rev. 3	Protective Coating Systems (Paints) - Organic Materials	03/2007	Applicable for some SMRs	NuScale does not use vessel insulation.
6.2.1, Rev. 3	Containment Functional Design	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design..

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
6.2.1.1.A, Rev. 3	PWR Dry Containments, Including Subatmospheric Containments	03/2007	Applicable for iPWRs	Applicable to system used for residual heat removal. Not applicable for LMRs.
6.2.1.1.B, Rev. 3	DRAFT Ice Condenser Containments	04/1996	Not Applicable	No SMRs are using ice condenser containments
6.2.1.1.C, Rev. 7	Pressure-Suppression Type BWR Containments	03/2007	Not Applicable	Applicable only to BWRs
6.2.1.2, Rev. 3	Subcompartment Analysis	03/2007	May be Applicable	Dependent upon SMR containment design.
6.2.1.3, Rev. 3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)	03/2007	Applicable for iPWRs	Not applicable for LMRs
6.2.1.4, Rev. 2	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
6.2.1.5, Rev. 3	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	03/2007	Applicable for some SMRs	A LOCA in NuScale does not result in core uncover.
6.2.2, Rev. 5	Containment Heat Removal Systems	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
6.2.3, Rev. 3	Secondary Containment Functional Design	03/2007	Applicable for some SMRs	NuScale does not have a secondary containment.
6.2.4, Rev. 3	Containment Isolation System	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
6.2.5, Rev. 3	Combustible Gas Control in Containment	03/2007	Applicable for iPWRs	Not applicable for LMRs
6.2.6, Rev. 3	Containment Leakage Testing	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
6.2.7, Rev. 1	Fracture Prevention of Containment Pressure Boundary	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design (e.g., close proximity of containment to vessel in NUScale may result in neutron embrittlement).
6.3, Rev. 3	Emergency Core Cooling System	03/2007	Applicable for iPWRs with modification	ECCS in iPWRs are generally passive systems. Not applicable for LMRs.

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
6.4, Rev. 3	Control Room Habitability System	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
6.5.1, Rev. 3	ESF Atmosphere Cleanup Systems	03/2007		
6.5.2, Rev. 4	Containment Spray as a Fission Product Cleanup System	03/2007	Applicable for some SMRs	NuScale does not have a containment spray system.
6.5.3, Rev. 3	Fission Product Control Systems and Structures	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
6.5.4, Rev. 4 DRAFT	Ice Condenser as a Fission Product Cleanup System	04/1996	Not Applicable	No SMRs are using ice condenser containments
6.5.5, Rev. 1	Pressure Suppression Pool as a Fission Product Cleanup System	03/2007	Not Applicable	Applicable only to BWRs
6.6, Rev. 2	Inservice Inspection and Testing of Class 2 and 3 Components	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
6.7, Rev. 3 DRAFT	Main Steam Isolation Valve Leakage Control System (BWR)	04/1996	Not Applicable	Applicable only to BWRs
Branch Technical Position 6-1	pH For Emergency Coolant Water for Pressurized Water Reactors	03/2007	Applicable for iPWRs	Not applicable for LMRs
Branch Technical Position 6-2, Rev. 3	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	03/2007	Applicable for iPWRs	Not applicable for LMRs
Branch Technical Position 6-3, Rev. 3	Determination of Bypass Leakage Paths in Dual Containment Plants	03/2007	Applicable for some iPWRs	A LOCA in NuScale does not result in core uncover. Not applicable for LMRs
Branch Technical Position 6-4, Rev. 3	Containment Purging During Normal Plant Operations	03/2007	Applicable for some SMRs	Not applicable to NuScale.
Branch Technical Position 6-5, Rev. 3	Currently the Responsibility of Reactor Systems Piping from the RWST (or BWST) and Containment Sump(s)to the Safety Injection Pumps	03/2007	Applicable for some iPWRs	NuScale does not have a RWST and most iPWRs do not have safety injection pumps. Not applicable for LMRs
	CHAPTER 7			
	Instrumentation and Controls			
7.0, Rev. 5	Instrumentation and Controls - Overview of Review Process	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
7.0-A, Rev. 5	Review Process for Digital Instrumentation and Control Systems	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
7.1, Rev. 5	Instrumentation and Controls - Introduction	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
7.1-T, Second Rev. 5	Table 7-1 Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety	03/2007		
Appendix 7.1-A, Second Rev. 5	Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety	03/2007		
Appendix 7.1-B, Rev. 5	Guidance for Evaluation of Conformance to IEEE Std. 279	03/2007		
Appendix 7.1-C, Rev 5	Guidance for Evaluation of Conformance to IEEE Std. 603	03/2007		
Appendix 7.1-D Second Issuance	Guidance for Evaluation of Conformance to IEEE Std. 7-4.3.2	03/2007		
7.2, Rev. 5	Reactor Trip System	03/2007		
7.3, Rev. 5	Engineered Safety Features Systems	03/2007		
7.4, Rev. 5	Safe Shutdown Systems	03/2007		
7.5, Rev. 5	Information Systems Important to Safety	03/2007		
7.6, Rev. 5	Interlock Systems Important to Safety	03/2007		
7.7, Rev. 5	Control Systems	03/2007		
7.8, Rev. 5	Diverse Instrumentation and Control Systems	03/2007		
7.9, Rev. 5	Data Communication Systems	03/2007		
Appendix 7-A, Rev. 5	General Agenda, Station Site Visits	03/2007		
Appendix 7-B, Rev. 5	Acronyms, Abbreviations, and Glossary	03/2007		
Branch Technical Position 7-1, Rev. 5	Guidance on Isolation of Low-Pressure Systems From the High-Pressure Reactor Coolant System	03/2007	Applicable for iPWRs	Not applicable for LMRs

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
Branch Technical Position 7-2, Rev. 5	Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines	03/2007	Applicable for some iPWRs	Applicable to iPWRs with accumulators. Not applicable for LMRs.
Branch Technical Position 7-3, Rev. 5	Guidance on Protection System Trip Point Changes for Operation With Reactor Coolant Pumps Out of Service	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
Branch Technical Position 7-4, Second Rev. 5	Guidance on Design Criteria for Auxiliary Feedwater Systems	03/2007	May be applicable for some iPWRs	Current iPWRs do not have auxiliary feedwater systems. Not applicable for LMRs.
Branch Technical Position 7-5, Rev. 5	Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	03/2007	Applicable for iPWRs	Not applicable for LMRs.
Branch Technical Position 7-6, Rev. 5	Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	03/2007	Applicable for some iPWRs	Not all iPWRs switch from injection to recirculation mode. Not applicable for LMRs.
Branch Technical Position 7-8, Rev. 5	Guidance for Application of Regulatory Guide 1.22	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
Branch Technical Position 7-9, Rev. 5	Guidance on Requirements for Reactor Protection System Anticipatory Trips	03/2007		
Branch Technical Position 7-10, Rev. 5	Guidance on Application of Regulatory Guide 1.97	03/2007		
Branch Technical Position 7-11, Rev. 5	Guidance on Application and Qualification of Isolation Devices	03/2007		
Branch Technical Position 7-12, Rev. 5	Guidance on Establishing and Maintaining Instrument Setpoints	03/2007		
Branch Technical Position 7-13, Rev. 5	Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
Branch Technical Position 7-14, Rev. 5	Guidance on Software Reviews for Digital Computer-Based Instrumentation and Controls Systems	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
Branch Technical Position 7-16 Withdrawn	Guidance on Level of Detail Required for Design Certification Applications Under 10 CFR Part 52	see ML0704 50253		
Branch Technical Position 7-17, Rev. 5	Guidance on Self-Test and Surveillance Test Provisions	03/2007		
Branch Technical Position 7-18, Rev. 5	Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems	03/2007		
Branch Technical Position 7-19, Rev. 5	Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems	03/2007		
Branch Technical Position 7-21, Rev. 5	Guidance on Digital Computer Real-Time Performance	03/2007		
	CHAPTER 8			
	Electric Power			
8.1, Rev. 3	Electric Power - Introduction	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
8.2, Rev. 4	Offsite Power System	03/2007		
8.3.1, Rev. 3	AC Power Systems (Onsite)	03/2007		
8.3.2, Rev. 3	DC Power Systems (Onsite)	03/2007		
8.4	Station Blackout	03/2007		
8-A, Rev. 1	General Agenda, Station Site Visits	03/2007		
Branch Technical Position 8-1, Rev. 3	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	03/2007	Applicable for some iPWRs	Applicable to iPWRs with accumulators. Not applicable for LMRs.
Branch Technical Position 8-2, Rev. 3	Use of Diesel-Generator Sets for Peaking	03/2007	May be applicable	Not evident that any SMR would use DGs for peaking.

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
Branch Technical Position 8-3, Rev. 3	Stability of Offsite Power Systems	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
Branch Technical Position 8-4, Rev. 3	Application of the Single Failure Criterion to Manually Controlled Electrically Operator Valves	03/2007		
Branch Technical Position 8-5, Rev. 3	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	03/2007		
Branch Technical Position 8-6, Rev. 3	Adequacy of Station Electric Distribution System Voltages	03/2007		
Branch Technical Position 8-7, Rev. 3	Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status	03/2007		
	CHAPTER 9			
	Auxiliary Systems			
9.1.1, Rev. 3	Criticality Safety of Fresh and Spent Fuel Storage and Handling	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
9.1.2, Rev. 4	New and Spent Fuel Storage	03/2007		
9.1.3, Rev. 2	Spent Fuel Pool Cooling and Cleanup System	03/2007		
9.1.4, Rev. 3	Light Load Handling System (Related to Refueling)	03/2007		
9.1.5, Rev. 1	Overhead Heavy Load Handling Systems	03/2007		
9.2.1, Rev. 5	Station Service Water System	03/2007		
9.2.2, Rev. 4	Reactor Auxiliary Cooling Water Systems	03/2007		
9.2.3 - Withdrawn	Demineralized Water Makeup System.	see ML063320108		
9.2.4, Rev. 3	Potable and Sanitary Water Systems	03/2007		
9.2.5, Rev. 3	Ultimate Heat Sink	03/2007		
9.2.6, Rev. 3	Condensate Storage Facilities	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
9.3.1, Rev. 2	Compressed Air System	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
9.3.2, Rev. 3	Process and Post-Accident Sampling Systems	03/2007		
9.3.3, Rev. 3	Equipment and Floor Drainage System	03/2007		
9.3.4, Rev. 3	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	03/2007	Applicable for iPWRs	Not applicable for LMRs.
9.3.5, Rev. 3	Standby Liquid Control System (BWR)	03/2007	Not applicable	Only applicable to BWRs.
9.4.1, Rev. 3	Control Room Area Ventilation System	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
9.4.2, Rev. 3	Spent Fuel Pool Area Ventilation System	03/2007		
9.4.3, Rev. 3	Auxiliary and Radwaste Area Ventilation System	03/2007		
9.4.4, Rev. 3	Turbine Area Ventilation System	03/2007		
9.4.5, Rev. 3	Engineered Safety Feature Ventilation System	03/2007		
9.5.1, Rev. 5	Fire Protection Program	03/2007		
9.5.2, Rev. 3	Communications Systems	03/2007		
9.5.3, Rev. 3	Lighting Systems	03/2007		
9.5.4, Rev. 3	Emergency Diesel Engine Fuel Oil Storage and Transfer System	03/2007		
9.5.5, Rev. 3	Emergency Diesel Engine Cooling Water System	03/2007		
9.5.6, Rev. 3	Emergency Diesel Engine Starting System	03/2007		
9.5.7, Rev. 3	Emergency Diesel Engine Lubrication System	03/2007		
9.5.8, Rev. 3	Emergency Diesel Engine Combustion Air Intake and Exhaust System	03/2007		
	CHAPTER 10			
	Steam and Power Conversion System			
10.2, Rev. 3	Turbine Generator	03/2007	Applicable	Technology neutral unless advanced power generation system is utilized.

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
10.2.3, Rev. 2	Turbine Rotor Integrity	03/2007	Applicable	Technology neutral unless advanced power generation system is utilized.
10.3, Rev. 4	Main Steam Supply System	03/2007	Applicable for iPWRs	May be applicable to LMRs utilizing conventional steam generators..
10.3.6, Rev. 3	Steam and Feedwater System Materials	03/2007		
10.4.1, Rev. 3	Main Condensers	03/2007		
10.4.2, Rev. 3	Main Condenser Evacuation System	03/2007		
10.4.3, Rev. 3	Turbine Gland Sealing System	03/2007		
10.4.4, Rev. 3	Turbine Bypass System	03/2007		
10.4.5, Rev. 3	Circulating Water System	03/2007		
10.4.6, Rev. 3	Condensate Cleanup System	03/2007		
10.4.7, Rev. 4	Condensate and Feedwater System	03/2007		
10.4.8, Rev. 3	Steam Generator Blowdown System	03/2007		
10.4.9, Rev. 3	Auxiliary Feedwater System (PWR)	03/2007	May be applicable for some iPWRs	Current iPWRs do not have auxiliary feedwater systems. Not applicable for LMRs.
Branch Technical Position, 10-1, Rev. 3	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants	03/2007		
Branch Technical Position, 10-2, Rev. 4	Design Guidelines for Avoiding Water Hammers in Steam Generators	03/2007	Applicable for iPWRs	May be applicable to LMRs utilizing conventional steam generators..
	CHAPTER 11			
	Radioactive Waste Management			
11.1, Rev. 3	Source Terms	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
11.2, Rev. 3	Liquid Waste Management System	03/2007		
11.3, Rev. 3	Gaseous Waste Management System	03/2007		
11.4, Rev. 3	Solid Waste Management System	03/2007		
11.5, Rev. 4	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
Branch Technical Position 11-3, Rev. 3	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
Branch Technical Position 11-5, Rev. 3	Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	03/2007		
Branch Technical Position 11-6	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	03/2007		
	CHAPTER 12			
	Radiation Protection			
12.1, Rev. 3	Assuring that Occupational Radiation Exposures Are As Low as is Reasonably Achievable	03/2007	Applicable	Technology Neutral.
12.2, Rev. 3	Radiation Sources	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
12.3-12.4, Rev. 3	Radiation Protection Design Features	03/2007		
12.5, Rev. 3	Operational Radiation Protection Program	03/2007	Applicable	Technology Neutral.
	CHAPTER 13			
	Conduct of Operations			
13.1.1, Rev. 5	Management and Technical Support Organization	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
13.1.2-13.1.3, Rev. 6	Operating Organization	03/2007		
13.2.1, Rev. 3	Reactor Operator Requalification Program; Reactor Operator Training	03/2007		
13.2.2, Rev. 3	Non-Licensed Plant Staff Training	03/2007		
13.3, Rev. 3	Emergency Planning	03/2007		
13.4, Rev. 3	Operational Programs	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
13.5.1.1	Administrative Procedures - General	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
13.5.1.2 DRAFT	Administrative Procedures - Initial Test Program (Content subsumed into SRP Section 14.2)	03/2007		
13.5.2.1, Rev. 1	Operating and Emergency Operating Procedures	03/2007		
13.5.2.2 DRAFT	Maintenance and Other Operating Procedures (Content subsumed into SRP Section 17.5)			
13.6	Physical Security	03/2007		
13.6.1	Physical Security - Combined License	03/2007		
13.6.2	Physical Security - Design Certification	03/2007		
13.6.3	Physical Security - Early Site Permit	03/2007		
	CHAPTER 14			
	Initial Test Program and ITAAC-Design Certification			
14.2, Rev. 3	Initial Plant Test Program - Design Certification and New License Applicants	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
14.2.1	Generic Guidelines for Extended Power Uprate Testing Programs	08/2006		
14.3	Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.1	[Reserved]			
14.3.2	Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.3	Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.4	Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
14.3.5	Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
14.3.6	Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.7	Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.8	Radiation Protection Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.9	Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.10	Initial Test Program and D-RAP - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.11	Containment Systems and Severe Accidents - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
14.3.12	Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria	03/2007		
	CHAPTER 15			
	Accident Analysis			
15.0, Rev. 3	Introduction - Transient and Accident Analyses	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
15.0.1	Radiological Consequence Analyses Using Alternate Source Terms	07/2000		
15.0.2	Review of Transient and Accident Analysis Methods	01/2006		
15.0.3	Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors	03/2007	Applicable for iPWRs	Not applicable for LMRs.

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
15.1.1 - 15.1.4, Rev. 2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	03/2007	Applicable for iPWRs	May be applicable to LMRs utilizing conventional steam generators..
15.1.5, Rev. 3	Steam System Piping Failures Inside and Outside of Containment (PWR)	03/2007	Applicable for iPWRs	Not applicable for LMRs.
15.1.5.A, Rev. 2	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	07/1981		
15.2.1-15.2.5, Rev. 2	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
15.2.6, Rev. 2	Loss of Non-Emergency AC Power to the Station Auxiliaries	03/2007		
15.2.7, Rev. 2	Loss of Normal Feedwater Flow	03/2007		
15.2.8, Rev. 2	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	03/2007		
15.3.1-15.3.2, Rev. 2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	03/2007		
15.3.3-15.3.4, Rev. 3	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	03/2007		
15.4.1, Rev. 3	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	03/2007		
15.4.2, Rev. 3	Uncontrolled Control Rod Assembly Withdrawal at Power	03/2007		
15.4.3, Rev. 3	Control Rod Misoperation (System Malfunction or Operator Error)	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
15.4.4-15.4.5, Rev. 2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	03/2007	Not Applicable	Only applicable to BWRs.
15.4.6, Rev. 2	Inadvertent Decrease in Boron Concentration in the Reactor Coolant (PWR)	03/2007	Applicable for iPWRs	Not applicable for LMRs.
15.4.7, Rev. 2	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
15.4.8, Rev. 3	Spectrum of Rod Ejection Accidents (PWR)	03/2007	Applicable for iPWRs	Not applicable for LMRs.
15.4.8.A, Rev. 2	Radiological Consequences of a Control Rod Ejection Accident (PWR)	07/1981		
15.4.9, Rev. 3	Spectrum of Rod Drop Accidents (BWR)	03/2007	Not Applicable	Only applicable to BWRs.
15.4.9.A, Rev. 2	Radiological Consequences of Control Rod Drop Accident (BWR)	07/1981		
15.5.1-15.5.2, Rev. 2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	03/2007	Applicable for iPWRs	Not applicable for LMRs.
15.6.1, Rev. 2	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	03/2007	Applicable for iPWRs	Not applicable for LMRs.
15.6.2, Rev. 2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	07/1981	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
15.6.3, Rev. 2	Radiological Consequences of Steam Generator Tube Failure (PWR)	07/1981	Applicable for iPWRs	Not applicable for LMRs.
15.6.4, Rev. 2	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	07/1981	Not Applicable	Only applicable to BWRs.

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
15.6.5, Rev. 3	Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	03/2007	Applicable for iPWRs	Not applicable for LMRs.
15.6.5.A, Rev. 2	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	07/1981		
15.6.5.B, Rev. 2	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Leakage From Engineered Safety Feature Components Outside Containment	07/1981		
15.6.5.D, Rev. 2	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR)	07/1981	Not Applicable	Only applicable to BWRs.
15.7.3, Rev. 2	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (content of this section has been relocated to BTP 11-6)	07/1981	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
15.7.4,	Rev. 2 Radiological Consequences of Fuel Handling Accidents	07/1981		
15.7.5, Rev. 2	Spent Fuel Cask Drop Accidents	07/1981		
15.8, Rev. 2	Anticipated Transients Without Scram	03/2007		
15.9	Boiling Water Reactor Stability	03/2007	Not Applicable	Only applicable to BWRs.
	CHAPTER 16			
	Technical Specifications			
16.0, Rev. 2	Technical Specifications	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
16.1, Rev. 1	Risk-Informed Decision Making: Technical Specifications	03/2007		

Table B-1. Applicability of LWRStandard Review Plan in NUREG-0800 [3] to iPWRs and LMRs.

Section/Revision	Title	Date	Applicability	Comments
	CHAPTER 17			
	Quality Assurance			
17.1, Rev. 2	Quality Assurance During the Design and Construction Phases	07/1981	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
17.2, Rev. 2	Quality Assurance During the Operations Phase	07/1981		
17.3	Quality Assurance Program Description	07/1981		
17.4	Reliability Assurance Program (RAP)	03/2007		
17.5	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	03/2007		
17.6	Maintenance Rule	03/2007		
	CHAPTER 18			
	Human Factors Engineering			
18.0, Rev. 2	Human Factors Engineering	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
	CHAPTER 19			
	Severe Accidents			
19.0, Rev. 2	Probabilistic Risk Assessment and Severe Accident Evaluation		Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
19.1, Rev. 2	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed			
19.2	Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance			

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)	--	11/1970	Applicable for iPWRs	Not applicable for LMRs.
1.2	Thermal Shock to Reactor Pressure Vessels (Safety Guide 2)	W	07/1991	Not Applicable	(**Withdrawn** -- See 56 FR 36175, 07/31/1991)
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors	2	06/1974	Not Applicable	Only applicable to BWRs
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors	2	06/1974	Applicable for iPWRs	Not applicable for LMRs.
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5)	--	03/1971	Not Applicable	Only applicable to BWRs
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6)	--	03/1971	Applicable	Applicable for all SMRs but application will be dependent upon SMR design..

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.7	Control of Combustible Gas Concentrations in Containment	3	03/2007	Applicable for some iPWRs.	Not an issue for NuScale. Not applicable for LMRs.
1.8	Qualification and Training of Personnel for Nuclear Power Plants	3	05/2000	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.9	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	4	03/2007		
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures	W	07/1981	Not Applicable	(**Withdrawn** -- See 46 FR 37579, 07/21/1981)
1.11	Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11)	1	03/2010	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.12	Nuclear Power Plant Instrumentation for Earthquakes	2	03/1997		
1.13	Spent Fuel Storage Facility Design Basis	2	03/2007		
1.14	Reactor Coolant Pump Flywheel Integrity	1	08/1975		
1.15	Testing of Reinforcing Bars for Category I Concrete Structures	W	07/1981	Not Applicable	(**Withdrawn** -- See 46 FR 37579, 07/21/1981)
1.16	Reporting of Operating Information -- Appendix A Technical Specifications (for Comment)	W	08/2009	Not Applicable	(**Withdrawn** -- See 74 FR 40244, 08/11/2009)

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.17	Protection of Nuclear Power Plants Against Industrial Sabotage	W	07/1991	Not Applicable	(**Withdrawn** -- See 56 FR 30777, 07/05/1991)
1.18	Structural Acceptance Test for Concrete Primary Reactor	W	07/1981	Not Applicable	(**Withdrawn** -- See 46 FR 37579, 07/21/1981)
1.19	Nondestructive Examination of Primary Containment Liner Welds	W	07/1981	Not Applicable	(**Withdrawn** -- See 46 FR 37579, 07/21/1981)
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	3	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants	2	06/2009		
1.22	Periodic Testing of Protection System Actuation Functions (Safety Guide 22)	--	02/1972		
1.23	Meteorological Monitoring Programs for Nuclear Power Plants (Safety Guide 23)	1	03/2007		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)	--	03/1972	Applicable for iPWRs	Not applicable for LMRs.
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)	--	03/1972	Applicable for iPWRs	May be applicable for LMRs but review will be dependent upon design.
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	4	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.27	Ultimate Heat Sink for Nuclear Power Plants (for Comment)	2	01/1976		
1.28	Quality Assurance Program Criteria (Design and Construction)	4	06/2010		
1.29	Seismic Design Classification	4	03/2007		
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety	--	08/1972		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.					
Guide Number	**Title**	**Rev.**	**Publish Date**	**SMR Applicability**	**Comments**
	Guide 30)				
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	3	04/1978	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.32	Criteria for Power Systems for Nuclear Power Plants	3	03/2004		
1.33	Quality Assurance Program Requirements (Operation)	2	02/1978		
1.34	Control of Electroslag Weld Properties	1	03/2011		
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments	3	07/1990	SMR Specific	Dependent upon SMR containment design.
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	--	07/1990		
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	--	02/1973	Applicable to some SMRs	Some SMRs do not have thermal insulation around the vessel (e.g., NuScale).
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	1	03/2007	Applicable for iPWRs	Not applicable for LMRs.
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	W	09/2010	Not Applicable	(**Withdrawn** -- See 75 FR 54921, 09/09/2010)

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	W	11/2010	Not Applicable	(**Withdrawn** -- See 75 FR 70044, 11/16/2010)
1.40	Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants	1	02/2010	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments	--	03/1973		
1.42	Interim Licensing policy on as low as Practicable for Gaseous Radiodine Releases from Light-Water-Cooled Nuclear Power Reactors.	W	03/1976	Not Applicable	(**Withdrawn** -- See 41 FR 11891, 03/22/1976)
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	1	03/2011	SMR Specific	Not clear what metals will be utilized in SMRs
1.44	Control of the Use of Sensitized Stainless Steel	1	03/2011		
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage	1	05/2008	Applicable	Applicable for all SMRs but application will be dependent upon SMR design – more applicable to iPWRs.
1.46	Protection Against Pipe Whip Inside Containment	W	09/1985	Not Applicable	(**Withdrawn** -- See 50 FR 9732, 03/11/1985)
1.47	Bypassed and Inoperable Status Indication for Nuclear Power	1	02/2010	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
	Plant Safety Systems				
1.48	Design Limit and Loading Combinations for Seismic Category I Fluid System Components	W	03/1985	Not Applicable	(**Withdrawn** -- See 50 FR 9732, 03/11/1985)
1.49	Power Levels of Nuclear Power Plants	W	07/2007	Not Applicable	(**Withdrawn** -- See 72 FR 36737, 07/05/2007)
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	1	03/2011		
1.51	Inservice Inspection of ASME code Class 2 and 3 Nuclear Power Plants components	W	07/1975	Not Applicable	(**Withdrawn** -- See 40 FR 30540, 07/21/1975)
1.52	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants	4	09/2012	Applicable for some iPWRs	Not all iPWRs have an atmospheric cleanup system. Not applicable for LMRs.
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	2	11/2003	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants	2	10/2010	SMR Specific	Not clear if protective coatings will be utilized in SMRs
1.55	Concrete Placement in Category I Structures	W	07/1981	Not Applicable	(**Withdrawn** -- See 46 FR 37579, 07/21/1981)

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.56	Maintenance of Water Purity in Boiling Water Reactors (for Comment)	W	02/2010	Not Applicable	(**Withdrawn** -- See 75 FR 7526, 02/19/2010)
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	1	03/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.58	Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel	W	07/1991	Not Applicable	(**Withdrawn** -- See 56 FR 36175, 07/31/1991)
1.59	Design Basis Floods for Nuclear Power Plants	2	08/1977	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	1	12/1973		
1.61	Damping Values for Seismic Design of Nuclear Power Plants	1	03/2007		
		--	10/1973		
1.62	Manual Initiation of Protective Actions	1	06/2010		
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	3	02/1987		
1.64	Quality Assurance Requirements for the Design of Nuclear Power plants	W	07/1991	Not Applicable	(**Withdrawn** -- See 56 FR 36175, 07/31/1991)
1.65	Materials and Inspections for	1	04/2010	Applicable	Applicable for all SMRs but application will be

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
	Reactor Vessel Closure Studs				dependent upon SMR design.
1.66	Nondestructive Examination of Tubular Products	W	10/1977	Not Applicable	(**Withdrawn** -- See 42 FR 54478, 10/6/1977)
1.67	Installation of Overpressure Protection Devices	W	04/1983	Not Applicable	(**Withdrawn** -- See 48 FR 19101, 04/27/1983)
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	3	03/2007	Applicable for iPWRs	Not applicable for LMRs.
1.68.1	Initial Test Program of Condensate and Feedwater Systems for Light-Water Reactors	2	09/2012		
1.68.2	Initial Startup Test Program To Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	2	04/2010		
1.68.3	Preoperational Testing of Instrument and Control Air Systems	1	09/2012	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants Concrete Radiation Shields for Nuclear Power Plants	1	05/2009		
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)	3	11/1978	Applicable for iPWRs	Generally applicable for LMRs.

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.71	Welder Qualification for Areas of Limited Accessibility	1	03/2007	Applicable	Technology Neutral
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin	2	11/1978		
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	--	01/1974	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.74	Quality Assurance Terms and Definitions	W	09/1989	Not Applicable	(**Withdrawn** -- See 54 FR 38919, 09/21/1989)
1.75	Physical Independence of Electric Systems	3	02/2005	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	1	03/2007		
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	--	05/1974	Applicable for iPWRs	Not applicable for LMRs.
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	1	12/2001	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	1	09/1975	Applicable for iPWRs	Not applicable for LMRs.
1.80	Preoperational Testing of	W	05/1982	Not Applicable	(**Withdrawn** -- See 47 FR 19258, 05/4/1982)

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
	Instrument Air Systems				
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	1	01/1975	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	4	03/2012	Applicable for iPWRs	Not applicable for LMRs.
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tube	W	11/2009	Not Applicable	(**Withdrawn** -- See 74 FR 58324, 11/12/2009)
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	35	10/2010	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.85	Materials code case Acceptability - - ASME Section III, Division 1	W	06/2003	Not Applicable	(**Withdrawn** 06/01/2003)
1.86	Termination of Operating Licenses for Nuclear Reactors	--	06/1974	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors	--	06/1975		
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Record	W	07/1991	Not Applicable	(**Withdrawn** -- See 56 FR 36175, 07/31/1991)
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	1	06/1984	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	2	11/2012	SMR specific	Different SMR designs utilize different types of containment.
1.91	Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants	1	02/1978	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	3	09/2012		
1.93	Availability of Electric Power Sources	1	03/2012		
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	W	09/2010	Not Applicable	(**Withdrawn --** See 75 FR 54921, 09/09/2010)
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	W	01/2002	Not Applicable	(**Withdrawn** 01/01/2002)
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	1	06/1976	Not Applicable	Only applicable to BWRs.
1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	4	06/2006	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (for Comment)	--	03/1976	Not Applicable	Only applicable to BWRs.
1.99	Radiation Embrittlement of Reactor Vessel Materials	2	05/1988	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants	3	09/2009		
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors	5	06/2005		
1.102	Flood Protection for Nuclear Power Plants	1	09/1976		
1.103	Post-tensioned Prestressing Systems for Concrete Reactor and Containment	W	07/1981	Not Applicable	(**Withdrawn** -- See 46 FR 37579, 07/21/1981)
1.104	Overhead Crane Handling Systems for Nuclear Power Plants	W	07/1981	Not Applicable	(**Withdrawn** -- See 44 FR 49321, 07/27/1981)
1.105	Setpoints for Safety-Related Instrumentation	3	12/1999	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	2	2/1012		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	2	06/2011	SMR specific	Different SMR designs utilize different types of containment.
1.108	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants	W	08/1993	Not Applicable	(**Withdrawn** -- See 58 FR 41813, 08/5/1993)
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	1	10/1977	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (for Comment)	--	03/1976		
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	1	07/1977	Applicable for iPWRs	Not applicable for LMRs.
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	1	03/2007		
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the	1	04/1977		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
	Purpose of Implementing Appendix I				
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit	3	10/2008	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.115	Protection Against Low-Trajectory Turbine Missiles	2	01/2012		
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	W	09/2010	Not Applicable	(**Withdrawn** -- See 75 FR 54921, 09/09/2010)
1.117	Tornado Design Classification	1	04/1978	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.118	Periodic Testing of Electric Power and Protection Systems	3	04/1995		
1.119	Surveillance Program for New Fuel Assembly Designs	W	06/1977	Not Applicable	(**Withdrawn** -- See 42 FR 33387, 06/30/1977)
1.120	Fire Protection Guidelines for Nuclear Power Plants	W	08/2001	Not Applicable	(**Withdrawn** 08/1/2001)
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (for Comment)	--	08/1976	Applicable for iPWRs	Not applicable for LMRs.
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1	02/1978	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.123	Quality Assurance Requirements for Control of Procurement of Items and Service for Nuclear Power Plants	W	07/1991	Not Applicable	(**Withdrawn** -- See 56 FR 36175, 07/31/1991)
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports	2	02/2007	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	2	03/2009		
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	2	03/2010		
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	1	03/1978		
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	2	02/2007		
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	2	02/2007		
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component	2	03/2007		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
	Supports				
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (for Comment)	W	04/2009	Not Applicable	(**Withdrawn** -- See 74 FR 18000, 04/20/2009)
1.132	Site Investigations for Foundations of Nuclear Power Plants	2	10/2003	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors	1	05/1981	Applicable for iPWRs	Not applicable for LMRs.
1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants	3	03/1998	Applicable	Technology Neutral
1.135	Normal Water Level and Discharge at Nuclear Power Plants (for Comment)	W	08/2009	Not Applicable	(**Withdrawn** -- See 74 FR 39349, 08/06/2009)
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	3	03/2007	SMR Specific	Applicable to SMRs with concrete containments.
1.137	Fuel-Oil Systems for Standby Diesel Generators	1	10/1979	Applicable	Technology Neutral
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	2	12/2003		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.139	Guidance for Residual Heat Removal (for Comment)	W	06/2008	Not Applicable	(**Withdrawn** -- See 73 FR 32750, 06/10/2008)
1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants	2	06/2001	Applicable for iPWRs	Possibly applicable for LMRs.
1.141	Containment Isolation Provisions for Fluid Systems	1	07/2010	Applicable	Applicable
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)	2	11/2001	Applicable	Applicable
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	2	11/2001	Applicable for iPWRs	Possibly applicable for LMRs.
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants	W	07/1991	Not Applicable	(**Withdrawn** -- See 56 FR 36175, 07/31/1991)
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	1	02/1983	Applicable	Technology Neutral
1.146	Qualification of Quality	W	07/1991	Not Applicable	(**Withdrawn** -- See 56 FR 36175, 07/31/1991)

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.					
Guide Number	**Title**	**Rev.**	**Publish Date**	**SMR Applicability**	**Comments**
	Assurance Program Audit Personnel for Nuclear Power Plants				
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	16	10/2010	Applicable	Technology Neutral
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants	W	01/2010	Not Applicable	(**Withdrawn** -- See 75 FR 2894, 01/19/2010)
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations	4	04/2011	Applicable	Technology Neutral
1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations	W	02/2008	Not Applicable	(**Withdrawn**-- See 73 FR 7766, 02/11/2008)
1.151	Instrument Sensing Lines	1	07/2010	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants	3	07/2011		
1.153	Criteria for Safety Systems (12/85)	1	06/1996		
1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors	W	01/2011	Not Applicable	(**Withdrawn** -- See 76 FR 2726, 01/14/2011)

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.155	Station Blackout	--	08/1988	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.156	Qualification of Connection Assemblies for Nuclear Power Plants	1	07/2011		
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance	--	05/1989	Not Applicable	iPWR ECCS are passive systems. This guidance is for active ECCS response.
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants	--	02/1989	Applicable Applicable	Technology Neutral Technology Neutral
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	2	10/2011		
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	3	05/2012		
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb	--	06/1995		
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels	--	02/1996		
1.163	Performance-Based Containment Leak-Test Program	--	09/1995		
1.164	(Not yet issued)			Not Applicable	
1.165	Identification and	W	04/2010	Not Applicable	(**Withdrawn** -- See 75 FR 22868, 04/30/2010)

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.					
Guide Number	**Title**	**Rev.**	**Publish Date**	**SMR Applicability**	**Comments**
	Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion				
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post earthquake Actions	--	03/1997	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event	--	03/1997		
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	02/2004		
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	09/1997		
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	09/1997		
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	09/1997		
1.172	Software Requirements Specifications for Digital Computer Software Used in	--	09/1997		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
	Safety Systems of Nuclear Power Plants				
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	09/1997	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	2	05/2011	Applicable	Technology Neutral
1.175	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing	--	08/1998		
1.176	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance	W	02/2008	Not Applicable	(**Withdrawn**-- See 73 FR 7766, 02/11/2008)
1.177	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications	1	05/2011	Applicable	Technology Neutral
1.178	An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping	1	09/2003		
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors	1	06/2011		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	1	10/2003	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)	--	09/1999	Applicable	Technology Neutral
1.182	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants	--	05/2000	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	--	07/2000		
1.184	Decommissioning of Nuclear Power Reactors	--	07/2000	Applicable	Technology Neutral
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	--	07/2000		
1.186	Guidance and Examples for Identifying 10 CFR 50.2 Design Bases	--	12/2000		
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments	--	11/2000		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.188	Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses	1	09/2005	Applicable	Technology Neutral
1.189	Fire Protection for Nuclear Power Plants	2	10/2009	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence	--	03/2001		
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown	--	05/2001		
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code	--	06/2003		
1.193	ASME Code Cases Not Approved for Use	3	10/2010		
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	--	06/2003		
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors	--	05/2003		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors	1	01/2007	Applicable	Technology Neutral
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors	--	05/2003		
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	--	11/2003		
1.199	Anchoring Components and Structural Supports in Concrete	--	11/2003		
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	2	03/2009		
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	1	05/2006	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	--	02/2005	Applicable	Technology Neutral
1.203	Transient and Accident Analysis Methods	--	12/2005	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
1.204	Guidelines for Lightning Protection of Nuclear Power Plants	--	11/2005	Applicable	Technology Neutral
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	1	12/2009		
1.206	Combined License Applications for Nuclear Power Plants (LWR Edition)	--	06/2007	Applicable to iPWRs	Generally applicable to LMRs.
1.207	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors	--	03/2007	Applicable to iPWRs	Not applicable to LMRs.
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	--	03/2007	Applicable	Technology Neutral
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	--	03/2007		
1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	--	06/2008		
1.211	Qualification of Safety-Related Cables and Field Splices for	--	04/2009		

Table B-2. Applicability of Regulatory Guides to iPWRs and LMRs.

Guide Number	Title	Rev.	Publish Date	SMR Applicability	Comments
	Nuclear Power Plants				
1.212	Sizing of Large Lead-Acid Storage Batteries	--	11/2008	Applicable	Applicable for all SMRs but application will be dependent upon SMR design.
1.213	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	--	05/2009		
1.215	"Guidance for ITAAC Closure Under 10 CFR Part 52."	1	05/2012		
1.216	Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure	--	08/2010		
1.217	Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts	--	08/2011		
1.218	Condition Monitoring Techniques for Electric Cables Used in Nuclear Power Plants	--	04/2012		
1.219	Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors	--	12/2011		
1.221	Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants	--	10/2011		

APPENDIX C
CODES AND STANDARDS IDENTIFIED IN THE SRP

A comprehensive list of standards cited in the SRP and other regulatory documents (CFR, Regulatory Guides, NRC Bulletins, Information Notices, Circulars, Generic Letters, and Policy Statements) was developed in NUREG/CR-5973 [6]. There are over 4000 citations of standards in NRC documents; over 300 are cited in the NRC Standard Review Plan (NUREG-0800) [3]. Table B-1 provides the list of standards identified in the SRP.

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American Concrete Institute	ACI 349	"Code Requirements for Nuclear Safety Related Concrete Structures"
American Institute of Steel Construction	ANSI/AISC N690	"Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities"
American National Standard	ANSI N658 (ANS 51.7)	"Single Failure Criteria for PWR Fluid Systems"
American National Standard	ANSI/ANS-56.8-1994	Containment System Leakage Testing Requirements
American National Standards Institute	ANSI B 16.11	"Forged Steel Fittings, Socket-Welding and Threaded"
American National Standards Institute	ANSI B 16.25	"Butt Welding Ends - Pipe, Valves, Flanges, and Fittings"
American National Standards Institute	ANSI B 16.34	"Steel Valves with Flanged and Butt Welding Ends"
American National Standards Institute	ANSI B 16.5	"Steel Pipe Flanges and Flanged Fittings"
American National Standards Institute	ANSI B 16.9	"Wrought Steel Butt Welding Fittings"
American National Standards Institute	ANSI B.16.41	"Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants"
American National Standards Institute	ANSI B96.1	"Specification for Welded Aluminum-Alloy Field-Erected Storage Tanks"
American National Standards Institute	ANSI N14.6-1993	"Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More."
American National Standards Institute	ANSI N18.2	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants
American National Standards Institute	ANSI N18.7-1976	"Administrative Controls and Quality Assurance for the Operation Phase of Nuclear Power Plants"
American National Standards Institute	ANSI N323A-1997	"American National Standard Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments."
American National Standards Institute	ANSI N41.6-1972	Functional Qualification Requirements for Actuators for Power Operated Valve Assemblies for Nuclear Power Plants"
American National Standards Institute	ANSI N42.17A-1989	"Performance Specifications for Health Physics Instrumentation—Portable Instrumentation for Use in Normal Environmental Conditions."
American National Standards Institute	ANSI N42.18-2004	"Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents"

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American National Standards Institute	ANSI N42.20-2003	"Performance Criteria for Active Personnel Radiation Monitors."
American National Standards Institute	ANSI N42.28-2002	"American National Standard for Calibration of Germanium Detectors for In Situ Gamma Ray Measurements."
American National Standards Institute	ANSI N45 N551.4	"Functional Qualification of Motor Drives for Safety Related Code Class 2 and 3 Pumps for Nuclear Power Plants"
American National Standards Institute	ANSI N45.2	"Quality Assurance Program Requirements for Nuclear Power Plants," Washington, DC.
American National Standards Institute	ANSI S2.32-1982 (R2004)	"Methods for the Experimental Determination of Mechanical Mobility, Part 2: Measurements Using Single-Point Translational Excitation."
American National Standards Institute	ANSI Standard N45.2.1-1973	"Cleaning of Fluid Systems and Associated Components"
American National Standards Institute	ANSI Std C84.1-1989	"American National Standard for Electric Power Systems and Equipment — Voltage Ratings (60 Hz)."
American National Standards Institute	ANSI/AISC N690-1994, including Supplement 2 (2004)	"Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities"
American National Standards Institute	ANSI/ANS 56.5-1979 (reaffirmed 1987)	"PWR and BWR Containment Spray System Design Criteria"
American National Standards Institute	ANSI/ASME B16.34	"Valves - Flanged, Threaded, and Welding End"
American National Standards Institute	ANSI/ASME B31.1	"Power Piping"
American National Standards Institute	ANSI/HPS N13.11-2001	"Personnel Dosimetry Performance—Criteria for Testing."
American National Standards Institute	ANSI/HPS N13.1-1999	"Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities."
American National Standards Institute	ANSI/HPS N13.14-1994	"Internal Dosimetry Programs for Tritium Exposure—Minimum Requirements."
American National Standards Institute	ANSI/HPS N13.30-1996	"Performance Criteria for Radiobioassay"
American National Standards Institute	ANSI/HPS N13.42-1997	"Internal Dosimetry for Mixed Fission Activation Products."
American National Standards Institute	ANSI/HPS N13.6	"Practice for Occupational Radiation Exposure Records Systems."
American National Standards Institute	ANSI/ISA-S7.3-1976	"Quality Standard for Instrument Air."
American National Standards Institute	ANSI/ISO/IEC 17025	"General Requirements for the Competence of Testing and Calibration Laboratories," Washington, DC.
American National Standards Institute	ANSI/NCSL Std. Z540-1-1994	"Calibration Laboratories and Measuring and Test Equipment - General Requirements."
American National Standards Institute	N210-76	"Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, Design"
American National Standards Institute/American Nuclear Society	ANS 57.2/ANSI N210-1976	"Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations"

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American National Standards Institute/American Nuclear Society	ANSI/ANS 3.1-1993 R99	"Selection, Qualification, and Training of Personnel for Nuclear Power Plants."
American National Standards Institute/American Nuclear Society	ANSI/ANS 3.2 1982	"Standard for Administrative Controls for Nuclear Power Plants"
American National Standards Institute/American Nuclear Society	ANSI/ANS 5.4	"Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel"
American National Standards Institute/American Nuclear Society	ANSI/ANS 51.1	"Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants-Replaces ANSI N18.2"
American National Standards Institute/American Nuclear Society	ANSI/ANS 52.1	"Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants"
American National Standards Institute/American Nuclear Society	ANSI/ANS 57.1-1992	"Design Requirements for Light Water Reactor Fuel Handling Systems"
American National Standards Institute/American Nuclear Society	ANSI/ANS 57.2-1983	"Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants"
American National Standards Institute/American Nuclear Society	ANSI/ANS 57.3-1983	"Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants"
American National Standards Institute/American Nuclear Society	ANSI/ANS 58.2-1988	"Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture"
American National Standards Institute/American Nuclear Society	ANSI/ANS Standard 18.1	"Source Term Specification"
American National Standards Institute/American Nuclear Society	ANSI/ANS Std. 4.5-1980	"Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors."
American National Standards Institute/American Nuclear Society	ANSI/ANS-18.1-1999	"American National Standard Radioactive Source Term for Normal Operation of Light Water Reactors"
American National Standards Institute/American Nuclear Society	ANSI/ANS-40.37-1993	"American National Standard For Mobile Low-Level Radioactive Waste Processing Systems." Proposed 2007 draft for public comments

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American National Standards Institute/American Nuclear Society	ANSI/ANS-51.1-1983	"Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," (replaced ANSI N18.2-1974; reaffirmed 1988; withdrawn 1998).
American National Standards Institute/American Nuclear Society	ANSI/ANS-52.1-1983	"Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," (replaced ANS Trial Use Standard N212-1974; reaffirmed 1988; withdrawn 1998).
American National Standards Institute/American Nuclear Society	ANSI/ANS-55.4-1993	(1999), "Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants."
American National Standards Institute/American Nuclear Society	ANSI/ANS-55.6-1993	"Liquid Radioactive Waste Processing System for Light Water Reactor Plants"
American National Standards Institute/American Nuclear Society	ANSI/ANS-57.3-1983	"Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants"
American National Standards Institute/American Nuclear Society	ANSI/ANS-59.51-1997	"Fuel Oil Systems for Safety-Related Emergency Diesel Generators."
American National Standards Institute/American Nuclear Society	ANSI/ANS-59.52-1998	"Lubricating Oil Systems for Safety-Related Diesel Generators."
American National Standards Institute/American Nuclear Society	ANSI/ANS-HPSSC-6.8.1-1981	"Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors"
American National Standards Institute/American Society of Mechanical Engineers	ANSI/ASME B31.1	"Power Piping"
American National Standards Institute/American Society of Mechanical Engineers	ANSI/ASME N278.1-1975	"Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard"
American National Standards Institute/American Society of Mechanical Engineers	ANSI/ASME N551.1	"Standard for Qualification of ASME Code Class 2 & 3 Pump Assemblies for Safety Systems Service, General Requirements"

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American National Standards Institute/American Society of Mechanical Engineers	ANSI/ASME N551.2	"Standard for Qualification of ASME Code Class 2 & 3 Pumps for Safety Systems Service"
American National Standards Institute/American Society of Mechanical Engineers	ANSI/ASME NQA-1-1983	"Quality Assurance Program Requirements for Nuclear Facilities."
American National Standards Institute/American Society of Mechanical Engineers	ANSI/ASME NQA-1a-1983 Addenda	"Addenda to ANSI/ASME NQA-1-1983, Quality Assurance Program Requirements for Nuclear Facilities."
American National Standards Institute/Institute of Electrical and Electronics Engineers	ANSI IEEE 309-1991	"Test Procedure for Geiger-Mueller Counters."
American Nuclear Society	ANS 3.2	"Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."
American Nuclear Society	ANS 5.1	"Decay Heat Power for Light Water Reactors"
American Nuclear Society	ANS Trial Use Standard N212	"Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants"
American Nuclear Society	ANS-58.23-200X	"Standard on Methodology for Fire PRA"
American Nuclear Society	ANSI/ANS-2.8-1992	“Determining Design Basis Flooding at Power Reactor Sites.”
American Nuclear Society	ANSI/ANS-3.11-2005	"Determining Meteorological Information at Nuclear Facilities"
American Nuclear Society	ANSI/ANS-58.21-2003	"External-Events PRA Methodology." La Grange Park, IL.
American Petroleum Institute	API Standard 620, Sixth Edition	Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks"
American Petroleum Institute	API Standard 650, Sixth Edition, Revision 1	"Welded Steel Tanks for Oil Storage"
American Society for Testing and Materials	ASTM A-262 1970	"Detecting Susceptibility to Intergranular Attack in Stainless Steels"; Practice A "Oxalic Acid Etch Test for Classification of Etch Structures of Stainless Steels"; Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack.”
American Society for Testing and Materials	ASTM A-708	"Standard Recommended Practices for Detection of Susceptibility to Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel."

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American Society for Testing and Materials	ASTM A-708-1974,	"Detection of Susceptibility to Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel"
American Society for Testing and Materials	ASTM D3911-03	"Standard Test Method for Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions."
American Society for Testing and Materials	ASTM D5144-00	"Standard Guide for Use of Protective Coating Standards in Nuclear "Structural Welding Code"
American Society for Testing and Materials	ASTM Designation: D 1689-95	"Standard Guide for Developing Conceptual Site Models for Contaminated Sites"
American Society for Testing and Materials	ASTM Designation: D 5447-04	"Standard Guide for Application of a Ground-Water Flow Model to a Site-Specific Problem"
American Society for Testing and Materials	ASTM Designation: D 5490-93	"Standard Guide for Comparing Ground-Water Flow Model Simulations to Site-Specific Information"
American Society for Testing and Materials	ASTM Designation: D 5609-94	"Standard Guide for Defining Boundary Conditions in Ground-Water Flow Modeling
American Society for Testing and Materials	ASTM Designation: D 5610-94	"Standard Guide for Defining Initial Conditions in Ground-Water Flow Modeling"
American Society for Testing and Materials	ASTM Designation: D 5611-94	"Standard Guide for Conducting a Sensitivity Analysis of a Ground-Water Flow Model Application"
American Society for Testing and Materials	ASTM Designation: D 5880-95.	"Standard Guide for Subsurface Flow and Transport Modeling"
American Society for Testing and Materials	ASTM E-1820-05a	"Standard Test Method for Measurement of Fracture Toughness"
American Society for Testing and Materials	ASTM E-185-1982	"Surveillance Tests on Structural Materials in Nuclear Reactors" Annual Book of ASTM Standards, Part 30,
American Society for Testing and Materials	ASTM E-185-82	"Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels."
American Society for Testing and Materials	ASTM E-399-05	"Standard Test Method for Linear-Elastic Plan-Strain Fracture Toughness KIC of Metallic Materials"
American Society for Testing and Materials	ASTM, A-262-1970	"Detecting Susceptibility to Intergranular Attack in Stainless Steels"; Practice A "Oxalic Acid Etch Test for Classification of Etch Structures of Stainless Steels"; Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack."
American Society for Testing and Materials	Standard C776-89	"Standard Specification for Sintered Uranium Dioxide Pellets"
American Society for Testing Materials	ASTM E-208-95a	"Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American Society of Civil Engineers	ASCE 4-98	"Seismic Analysis of Safety-Related Nuclear Structures and Commentary," (Section 3.5.3.2 for embedded walls and Sections 3.5.3.1 through 3.5.3.3 for earth retaining walls).
American Society of Civil Engineers	ASCE Report	"Seismic Response of Buried Pipes and Structural Components"
American Society of Civil Engineers	ASCE Standard No. 7-05; ASCE/SEI 7-05	"Minimum Design Loads for Buildings and Other Structures,"
American Society of Civil Engineers	SEI/ASCE 37	"Design Loads on Structures During Construction"
American Society of Civil Engineers	Technical Engineering and Design Guide	"Bearing Capacity of Soils"
American Society of Civil Engineers	Technical Engineering and Design Guide	"Settlement Analysis"
American Society of Civil Engineers		"Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures"
American Society of Civil Engineers		"Suggested Specification for Structures of Aluminum Alloys 6061-T6 and 6067-T6"
American Society of Civil Engineers/Structural Engineering Institute	ASCE/SEI 7-05	"Minimum Design Loads for Buildings and Other Structures"
American Society of Heating, Refrigerating, and Air-Conditioning Engineers		"2005 ASHRAE Handbook — Fundamentals"
American Society of Mechanical Engineers	ANSI/ASME Standard NQA -1	"Quality Assurance Program Requirements for Nuclear Facility Applications"
American Society of Mechanical Engineers	ASME AG-1-1997	"Code on Nuclear Air and Gas treatment"
American Society of Mechanical Engineers	ASME B30.2-2005	"Overhead and Gantry Cranes - Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist."
American Society of Mechanical Engineers	ASME B30.9-2003	"Slings"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	"Code Cases: Nuclear Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	"Section III, Division 1, Valves - Flanged, Threaded, and Welding End Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	"Section III, Nuclear Power Plant Components," and "Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components"

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	"Section VIII, Division 1, Pressure Vessels"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	"Sections III, V, and XI"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section II, "Materials Specifications"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section II, "Materials," Parts A, B, and C; Section III, "Rules for Construction of Nuclear Facility Components"; and Section IX, "Welding and Brazing Qualifications"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section II, "Materials," Parts A, B, and C; Section III, "Rules for Construction of Nuclear Plant Components," Division 1, including Appendix I and Division 2; and Section IX, "Welding and Brazing Qualifications"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section II, "Materials," Parts A, B, C, and D; and Section III, "Rules for Construction of Nuclear Plant Components," Division 1, including Appendix I
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section II, “Materials,” Tables 2A, 2B and 4; Section III, “Rules for Construction of Nuclear Facility Components,” Division 1; and Section IX, “Welding and Brazing Qualifications."
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section II, Parts A, B, and C, Section III, subsections NB, NC, and ND, Article D-1000, and Appendix I, and Section IX
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Code for Concrete Reactor Vessels and Containments," Division 2, "Code for Concrete Reactor Vessels and Containments
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Nuclear Power Plant Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Nuclear Power Plant Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Nuclear Power Plant Components,” and Section VIII, Division 1, “Pressure Vessels"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Nuclear Power Plant Components." Article NB-7000, "Protection against Overpressure."
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Rules for Construction of Nuclear Facility Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Rules for Construction of Nuclear Facility Components," Article NCA-2000, "Classification of Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Rules for Construction of Nuclear Facility Components," Subsection NCA, "General Requirements for Division 1 and Division 2," and Division 1, Subsection NB, "Class 1 Components"

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Rules for Construction of Nuclear Facility Components."
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Rules for Construction of Nuclear Power Plant Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, "Rules for Construction of Nuclear Power Plant Components, Division 1, Appendix N, "Dynamic Analysis Methods"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Appendix G, "Protection Against Nonductile Failure"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Article NB-7000, "Overpressure Protection"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Article NB-7511.1, “Spring-Loaded Valves"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division 1, "Nuclear Power Plant Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division 1, Appendix O, "Rules for the Design of Safety Valve Installations"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division 1, Subsection NE, "Class MC Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division 1, Subsection NE, "Class MC Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division 1, Summer 1977 Addenda, Subsection NC, "Class 2 Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division 2, "Code for Concrete Reactor Vessels and Containments"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division 2, "Code for Concrete Reactor Vessels and Containments"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division 2, Subsection CC, "Code for Concrete Reactor Vessels and Containments"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, Division I, "Nuclear Power Plant Components" 3.9.1-10 Revision 3
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section III, including Appendix G, "Protection Against Nonductile Failure"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section IX, "Welding and Brazing Qualifications."

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Concrete Components of Light-Water Cooled Power Plants"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water Cooled Plants"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section XI, "Rules for Inservice Inspection of Nuclear Power Plants Components."
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section XI, "Rules of Inservice Inspection of Nuclear Power Plant Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section XI, Division 1, Appendix G, " Fracture Toughness Criteria for Protection Against Failure"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Section XI, Subsection IWV, "Inservice Testing of Valves in Light-Water Reactor Power Plants"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Sections II, "Materials," III, "Rules for Construction of Nuclear Facility Components," V, "Nondestructive Examination," IX, "Welding and Brazing Qualifications," XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Sections II, III, and XI
American Society of Mechanical Engineers	ASME Boiler and Pressure Vessel Code	Sections II, III, V, IX, and XI
American Society of Mechanical Engineers	ASME Code AG-1	"Code for Nuclear Air and Gas Treatment," 1991 (including the AG-1a-92 Addenda thereto).
American Society of Mechanical Engineers	ASME N509-1989	"Nuclear Power Plant Air Cleaning Units and Components"
American Society of Mechanical Engineers	ASME N510-1989	"Testing of Nuclear Air Cleaning Systems"
American Society of Mechanical Engineers	ASME NOG1-2004	"Rules for Construction of Overhead and Gantry Cranes."

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
American Society of Mechanical Engineers	ASME OM-S/G-2000	"Standards and Guides For Operation of Nuclear Power Plants" Part 3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems," including the addenda and Part 7, "Requirements for Thermal Expansion Test
American Society of Mechanical Engineers	ASME RA-S-2002	"Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications." ASME: New York, NY.
American Society of Mechanical Engineers	ASME RA-Sa-2003 (Addendum A to RA-S-2002)	"Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications." ASME: New York, NY.
American Society of Mechanical Engineers	ASME RA-Sb-2005 (addendum B to RA-S-2002)	"Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications." . ASME: New York, NY.
American Society of Mechanical Engineers	ASME Std. NQA-1-1994	"Quality Assurance Requirements for Nuclear Facility Applications"
American Society of Mechanical Engineers	ASTM A-370-05	"Standard Test Methods and Definitions for Mechanical Testing of Steel Products," Annual Book of ASTM Standards, Parts 1, 2, 3, 4, or 31
American Society of Mechanical Engineers	Boiler and Pressure Vessel Code	Section III, "Rules for Construction of Nuclear Power Plant Components"
American Society of Mechanical Engineers	Boiler and Pressure Vessel Code	Section III, "Rules for Construction on Nuclear Power Plant Components," and XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"
American Society of Mechanical Engineers	BTP 3-1-3 Revision 2	
American Society of Mechanical Engineers	NQA-2. ASME	"Quality Assurance Requirements for Nuclear Facility Applications," Washington, DC.
American Society of Mechanical Engineers		"ASME Code for Operation and Maintenance of Nuclear Power Plants"
American Society of Mechanical Engineers		NQA-1-1994 Edition, "Quality Assurance Requirements for Nuclear Facility Applications," Revision and Consolidation of ASME NQA-1-1989 and ASME NQA-2-1989 Editions
American Water Works Association	AWWA D100	"AWWA Standard for Steel Tanks-Standpipes, Reservoirs, and Elevated Tanks for Water Storage"
American Welding Society	AWS D1.1-1981	"Structural Welding Code"
Committee on the Safety of Nuclear Installations	NEA/CSNI/R(93)14	"Separate Effects Test Matrix for Thermal-Hydraulic Code Validation"
Committee on the Safety of Nuclear Installations	NEA/CSNI/R(96)17	"Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients"
Diesel Engine Manufacturers Association		(DEMA) Standard 1974.

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
IEC Publication	IEC Std. 60880-2	"Software for Computers Important to Safety for Nuclear Power Plants - Part 2: Software Aspects of Defense Against Common Cause Failures, Use of Software Tools and of Pre-Developed Software"
Institute of Electrical and Electronics Engineers	ANSI/IEEE Std 344-1987	"IEEE Recommended Practice for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE 344-1987	"IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, Appendix D, "Test Duration and Number of Cycles"
Institute of Electrical and Electronics Engineers	IEEE 344-1987	"Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Standard 308-2001	"IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."
Institute of Electrical and Electronics Engineers	IEEE Standard 344-1987	"IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, Appendix D, "Test Duration and Number of Cycles"
Institute of Electrical and Electronics Engineers	IEEE Standard 603-1991	"IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Standard 765-1983	"IEEE Standard for Preferred Power Supply (PPS) for Nuclear Power Generating Stations." (2002 is latest revision)
Institute of Electrical and Electronics Engineers	IEEE Std 100-2000	"The Authoritative Dictionary of IEEE Standards Terms 7th Edition."
Institute of Electrical and Electronics Engineers	IEEE Std 1012-1998	"IEEE Standard for Software Verification and Validation."
Institute of Electrical and Electronics Engineers	IEEE Std 1042-1987	"IEEE Guide to Software Configuration Management."
Institute of Electrical and Electronics Engineers	IEEE Std 1058.1-1991	"IEEE Standard for Software Project Management Plans."
Institute of Electrical and Electronics Engineers	IEEE Std 1061-1998	"IEEE Standard for a Software Quality Metrics Methodology."
Institute of Electrical and Electronics Engineers	IEEE Std 12207.0-1996	"IEEE/EIA Standard - Industry Implementation of International", Standard ISO/IEC 12207:1995 (ISO/IEC 12207), "Standard for Information Technology - Software Life Cycle Processes."
Institute of Electrical and Electronics Engineers	IEEE Std 1228-1994	"IEEE Standard for Software Safety Plans."
Institute of Electrical and Electronics Engineers	IEEE Std 1540-2001	"IEEE Standard for Life Cycle Processes – Risk Management."

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
Institute of Electrical and Electronics Engineers	IEEE Std 279-1971	"Criteria for Protection Systems for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 317-1983	"IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generation Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 323-1974	"IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 334-1971	"IEEE Trial-Use Guide for Type Tests of Continuous-Duty Class 1 Motors Installed Inside the Containment of Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 344-1971	"Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 379-2000	"IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems"
Institute of Electrical and Electronics Engineers	IEEE Std 382-1972	"IEEE Trial-Use Guide for Type Test of Class 1 Electric Valve Operators for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 383-1974	"IEEE Standard for Type Test of Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 450-2002	"Recommended Practice for Maintenance, Testing and Replacement of Vented Lead-Acid Batteries for Stationary Applications."
Institute of Electrical and Electronics Engineers	IEEE Std 484-2002	"Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications."
Institute of Electrical and Electronics Engineers	IEEE Std 535-1986	"IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 572-1985	"IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 603-1980	"IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 603-1991	"IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 603-1998	"IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 7-4.3.2-2003	"IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std 828-1990	"IEEE Standard for Software Configuration Management Plans."

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
Institute of Electrical and Electronics Engineers	IEEE Std 835 -1994	"Standard Power Cable Ampacity Tables."
Institute of Electrical and Electronics Engineers	IEEE Std C37.90.1-2002	"IEEE Standard for Surge Withstand Capability (SWC) Tests for Relays and Relay Systems Associated with Electric Power Apparatus."
Institute of Electrical and Electronics Engineers	IEEE Std C62.23-1995 (R 2001)	"IEEE Application Guide for Surge Protection of Electric Generating Plants."
Institute of Electrical and Electronics Engineers	IEEE Std C62.36-2000	"IEEE Standard Test Methods for Surge Protectors Used in Low-Voltage Data, Communications, and Signaling Circuits."
Institute of Electrical and Electronics Engineers	IEEE Std C62.41.1-2002	"IEEE Guide on the Surge Environment in Low-Voltage (1000V and Less) AC Power Circuits."
Institute of Electrical and Electronics Engineers	IEEE Std C62.41.2-2002	"IEEE Recommended Practice on Characterization of Surges in Low-Voltage (1000 V and Less) AC Power Circuits."
Institute of Electrical and Electronics Engineers	IEEE Std C62.45-2002	"IEEE Recommended Practice on Surge Testing for Equipment Connected to Low-Voltage (1000 V and Less) AC Power Circuits."
Institute of Electrical and Electronics Engineers	IEEE Std. 1008-1987	"IEEE Standard for Software Unit Testing."
Institute of Electrical and Electronics Engineers	IEEE Std. 1012-1998	"IEEE Standard for Software Verification and Validation."
Institute of Electrical and Electronics Engineers	IEEE Std. 1028-1988	"IEEE Standard for Software Reviews and Audits"
Institute of Electrical and Electronics Engineers	IEEE Std. 1028-1997	"IEEE Standard for Software Reviews."
Institute of Electrical and Electronics Engineers	IEEE Std. 1042-1987	"IEEE Guide to Software Configuration Management"
Institute of Electrical and Electronics Engineers	IEEE Std. 1050-1996	"IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std. 1074-1995	"IEEE Standard for Developing Software Life Cycle Processes"
Institute of Electrical and Electronics Engineers	IEEE Std. 1100-1999	"IEEE Recommended Practice for Powering and Grounding Electronic Equipment" (IEEE Emerald Book).
Institute of Electrical and Electronics Engineers	IEEE Std. 1184-2006	"Guide for Batteries for Uninterruptible Power Supply Systems."
Institute of Electrical and Electronics Engineers	IEEE Std. 1290-1996	"IEEE Guide for Motor Operated Valve (MOV) Motor Application, Protection, Control, and Testing in Nuclear Power Generating Stations"

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
Institute of Electrical and Electronics Engineers	IEEE Std. 1375-1998	"Guide for the Protection of Stationary Battery Systems."
Institute of Electrical and Electronics Engineers	IEEE Std. 141-1993	"Recommended Practice for Electric Power Distribution for Industrial Plants," (Red Book).
Institute of Electrical and Electronics Engineers	IEEE Std. 142-1991	"IEEE Recommended Practice for Grounding of Industrial and Commercial Power Systems"
Institute of Electrical and Electronics Engineers	IEEE Std. 242-2001	"Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems," (Buff Book).
Institute of Electrical and Electronics Engineers	IEEE Std. 279-1971	"Criteria for Protection Systems for Nuclear Power Generating Stations."
Institute of Electrical and Electronics Engineers	IEEE Std. 323-2000	"IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."
Institute of Electrical and Electronics Engineers	IEEE Std. 338-1987	"Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems."
Institute of Electrical and Electronics Engineers	IEEE Std. 367-1996	"IEEE Recommended Practice for Determining the Electric Power Station Ground Potential Rise and Induced Voltage from a Power Fault."
Institute of Electrical and Electronics Engineers	IEEE Std. 379-2000	"Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems"
Institute of Electrical and Electronics Engineers	IEEE Std. 384-1992	"IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits"
Institute of Electrical and Electronics Engineers	IEEE Std. 387-1995	"Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations."
Institute of Electrical and Electronics Engineers	IEEE Std. 399-1997	"Recommended Practice for Power Systems Analysis," (Brown Book).
Institute of Electrical and Electronics Engineers	IEEE Std. 473-1985	"IEEE Recommended Practice for an Electromagnetic Site Survey (10 kHz to 10 GHz)"
Institute of Electrical and Electronics Engineers	IEEE Std. 485-1987	"Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications."
Institute of Electrical and Electronics Engineers	IEEE Std. 485-1997	"Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications."
Institute of Electrical and Electronics Engineers	IEEE Std. 487-2000	"IEEE Recommended Practice for the Protection of Wire-Line Communication Facilities Serving Electric Supply Locations."
Institute of Electrical and Electronics Engineers	IEEE Std. 497-2002	"IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
Institute of Electrical and Electronics Engineers	IEEE Std. 498-1990	"IEEE Standard Requirements for the Calibration and Control of Measuring and Test Equipment Used in Nuclear Facilities."
Institute of Electrical and Electronics Engineers	IEEE Std. 518-1982	"IEEE Guide for the Installation of Electrical Equipment to Minimize Noise Inputs to Controllers from External Sources"
Institute of Electrical and Electronics Engineers	IEEE Std. 603-1991	"IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std. 610.12-1990	"IEEE Standard Glossary of Software Engineering Terminology."
Institute of Electrical and Electronics Engineers	IEEE Std. 665-1987	"IEEE Guide for Generating Station Grounding"
Institute of Electrical and Electronics Engineers	IEEE Std. 665-1995 (reaffirmed 2001)	"IEEE Guide for Generating Station Grounding"
Institute of Electrical and Electronics Engineers	IEEE Std. 666-1991	"IEEE Design Guide for Electrical Power Service Systems for Generating Stations"
Institute of Electrical and Electronics Engineers	IEEE Std. 7-4.3.2-2003	"IEEE Standard for Digital Computers in Safety Systems of Nuclear Power Generating Stations."
Institute of Electrical and Electronics Engineers	IEEE Std. 741-1997	"IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations."
Institute of Electrical and Electronics Engineers	IEEE Std. 80-2000	"IEEE Guide for Safety in AC Substation Grounding."
Institute of Electrical and Electronics Engineers	IEEE Std. 81.2-1991	"IEEE Guide for Measurement of Impedance and Safety Characteristics of Large, Extended or Interconnected Grounding Systems"
Institute of Electrical and Electronics Engineers	IEEE Std. 81-1983	"IEEE Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Ground System."
Institute of Electrical and Electronics Engineers	IEEE Std. 828-1990	"IEEE Standard for Software Configuration Management Plans."
Institute of Electrical and Electronics Engineers	IEEE Std. 829-1983	"IEEE Standard for Software Test Documentation."
Institute of Electrical and Electronics Engineers	IEEE Std. 830-1993	"IEEE Recommended Practice for Software Requirements Specifications."
Institute of Electrical and Electronics Engineers	IEEE Std. 934-1987	"Requirements for Replacement Parts for Class 1E Equipment in Nuclear Power Generating Stations."
Institute of Electrical and Electronics Engineers	IEEE Std. 946-2004	"Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations."

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
Institute of Electrical and Electronics Engineers	IEEE Std. C37.101-1993	"IEEE Guide for Generator Ground Protection."
Institute of Electrical and Electronics Engineers	IEEE Std. C37.106 - 2003	"Guide for Abnormal Frequency Protection for Power Generating Plants."
Institute of Electrical and Electronics Engineers	IEEE Std. C37.1-1994	"IEEE Standard Definition, Specification, and Analysis of Systems Used for Supervisory Control, Data Acquisition, and Automatic Control."
Institute of Electrical and Electronics Engineers	IEEE Std. C57.13.3-1983	"IEEE Guide for the Grounding of Instrument Transformer Secondary Circuits and Cases" (reaffirmed 1990).
Institute of Electrical and Electronics Engineers	IEEE Std. C62.23-1995	"IEEE Application Guide for Surge Protection of Electric Generating Plants"
Institute of Electrical and Electronics Engineers	IEEE Std. C62.41.1-2002	"IEEE Guide on the Surge Environment in Low-Voltage (1000 V and Less) AC Power Circuits."
Institute of Electrical and Electronics Engineers	IEEE Std. C62.41.2-2002	"IEEE Recommended Practice on Characterization of Surges in Low-Voltage (1000 V and Less) AC Power Circuits."
Institute of Electrical and Electronics Engineers	IEEE Std. C62.41-1991	"IEEE Recommended Practice on Surge Voltages in Low-Voltage AC Power Circuits"
Institute of Electrical and Electronics Engineers	IEEE Std. C62.45-1992	"IEEE Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Circuits"
Institute of Electrical and Electronics Engineers	IEEE Std. C62.45-2002	"IEEE Recommended Practice on Surge Testing for Equipment Connected to Low-Voltage (1000 V and Less) AC Power Circuits"
Institute of Electrical and Electronics Engineers	IEEE Std. C62.92.1-2000	"IEEE Guide for the Application of Neutral Grounding in Electrical Utility Systems, Part I - Introduction."
Institute of Electrical and Electronics Engineers	IEEE Std. C62.92.2-1989	"IEEE Guide for the Application of Neutral Grounding in Electrical Utility Systems, Part II - Grounding of Synchronous Generator Systems" (reaffirmed 2001).
Institute of Electrical and Electronics Engineers	IEEE Std. C62.92.3-1993	"IEEE Guide for the Application of Neutral Grounding in Electrical Utility Systems, Part III - Generator Auxiliary Systems" (reaffirmed 2000).
Institute of Electrical and Electronics Engineers	IEEE/EIA Std. 12207.0-1996	"Industry Implementation of International Standard ISO/IEC 12207: 1995 (ISO/IEC 12207) Standard for Information Technology Software Life Cycle Processes"
Institute of Electrical and Electronics Engineers	Std 338-1987	"Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems."
Institute of Electrical and Electronics Engineers/American Nuclear Society	IEEE/ANS 7-4.3.2-1982	"American Nuclear Society and IEEE Standard Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations."
International Electrotechnical Committee	IEC 60880-2-2002	"Software for Computers in the Safety Systems of Nuclear Power Stations - Part 2, Software Aspects of Defense Against Common Cause Failures, Use of Software Tools and of Pre-developed Software."

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
International Electrotechnical Committee	IEC 61000-3-2	"Electromagnetic Compatibility (EMC) - Part 3-2: Limits - Limits for Harmonic Current Emissions," International Electrotechnical Commission, 2001.
International Electrotechnical Committee	IEC 61000-3-4	"Electromagnetic Compatibility (EMC) - Part 3-4: Limits - Limitation of Emission of Harmonic Currents in Low-Voltage Power Supply Systems for Equipment with Rated Current Greater than 16 A," International Electrotechnical Commission, 1998.
International Electrotechnical Committee	IEC 61000-4-1	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 1: Overview of Immunity Tests," 1992.
International Electrotechnical Committee	IEC 61000-4-10	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 10: Damped Oscillatory Magnetic Field Immunity Test"
International Electrotechnical Committee	IEC 61000-4-11	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 11: Voltage Dips, Short Interruptions, and Voltage Variations Immunity Test"
International Electrotechnical Committee	IEC 61000-4-12	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 12: Oscillatory Waves Immunity Tests"
International Electrotechnical Committee	IEC 61000-4-13	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 13: Immunity to Harmonics and Interharmonics"
International Electrotechnical Committee	IEC 61000-4-16	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 16: Test for Immunity to Conducted, Common Mode Disturbances in the Frequency Range 0 Hz to 150 kHz"
International Electrotechnical Committee	IEC 61000-4-2	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 2: Electrostatic Discharge Immunity Test,"
International Electrotechnical Committee	IEC 61000-4-3	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 3: Radiated, Radio-Frequency, Electromagnetic Field Immunity Test,"
International Electrotechnical Committee	IEC 61000-4-4	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 4: Electrical Fast Transient/Burst Immunity Test,"
International Electrotechnical Committee	IEC 61000-4-5	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 5: Surge Immunity Test"
International Electrotechnical Committee	IEC 61000-4-6	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 6: Immunity to Conducted Disturbances, Induced by Radio-Frequency Fields"
International Electrotechnical Committee	IEC 61000-4-7	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 7: General Guide on Harmonics and Interharmonics Measurements and Instrumentation, for Power Supply Systems and Equipment Connected Thereto"
International Electrotechnical Committee	IEC 61000-4-8	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 8: Power Frequency Magnetic Field Immunity Test"

Table C-1. Codes and Standards Identified in the Standard Review Plan.

SDO	Number	Title
International Electrotechnical Committee	IEC 61000-4-9	"Electromagnetic Compatibility (EMC) - Part 4: Testing and Measurement Techniques, Section 9: Pulse Magnetic Field Immunity Test"
International Electrotechnical Committee	IEC 61000-6-4	"Electromagnetic Compatibility (EMC) - Part 6: Generic Standards, Section 4: Emission Standard for Industrial Environments"
International Organizations for Standardization	ISO 7626-5:1994	"Vibration and shock - Experimental determination of mechanical mobility - Part 5: Measurements using impact excitation with an exciter which is not attached to the structure."
International Society of Automation	ISA-S67.02-1980	"Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants"
International Society of Automation	ISA-S67.04-1994	"Setpoints for Nuclear Safety-Related Instrumentation."
International Society of Automation	ISA-S67.04-1994, Part I	"Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants."
International Society of Automation	ISA-S67.04-1994, Part II	"Methodology for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."
Manufacturers Standardization Society	MSS-SP-25	"Marking for Valves, Fittings, Flanges, and Unions"
National Fire Protection Agency	NFPA 804	"Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants."
National Fire Protection Agency	NFPA 805	"Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."
Structural Engineering Institute/American Society of Civil Engineers	SEI/ASCE 37-02	"Design Loads on Structures During Construction"
U.S. Environmental Protection Agency	EPA/600/R-94/038d	Quality Assurance Handbook for Air Pollution Quality Assurance Handbook for Air Pollution
U.S. Environmental Protection Agency	EPA-454/R-99-005	Meteorological Monitoring Guidance for Regulatory Modeling Applications
U.S. Environmental Protection Agency	EPA-600/4-76-030a	Weber, A. H. "Atmospheric Dispersion Parameters in Gaussian Plume Modeling,"
U.S. Environmental Protection Agency	PB 94-205804 EPA 402-R-94-012	"A Technical Guide to Ground-Water Model Selection at Sites Contaminated with Radioactive Substances
U.S. State Department	SD-STD-02.01	"Certification Standard, Test Method for Vehicle Crash Testing of Perimeter Barriers and Gates," Washington, DC. Unclassified.
Underwriters Laboratories (UL)	UL 752	"Standard for Bullet Resisting Equipment." Unclassified.
Underwriters' Laboratories, Inc.		"Building Materials List"

APPENDIX D-
SECURITY TOPICAL AREAS ADDRESSED IN 10CFR73

Table D-1. Applicability of Physical Security Requirements IN 10CFR 73 to SMRs.

Security Topic Area	Security Regulations	Existing LWR	Future SMR
Performance Evaluation Program Organization	10 CFR Part 73, Appendix B, Section VI, C.3.(a) through (l); 10 CFR 73.55(b)(6)	X	X
Establishment of Security	10 CFR 11.11; 10 CFR 73.55(d); 10 CFR Part 73, Appendix C, Section II, B.3.(a)	X	X
Qualification for Employment in Security 1. Training and Facility Personnel 2. Security Personnel training 3. Local Law Enforcement Liaison 4. Security Personnel Equipment 5. Work-Hour Controls	 10 CFR 73.55(c)(4); 10 CFR 73.55(d)(3); 10 CFR 50.120; 10 CFR Part 73, Appendix B, Section VI 10 CFR Part 73, Appendix B, Section VI 10 CFR 73.55(k)(9); 10 CFR Part 73, Appendix C, Section II 10 CFR Part 73, Appendix B, Section VI, G.2 10 CFR Part 26, Subpart I	X	X
Physical Barriers 1. Vehicle Barrier System 2. Channeling Barrier System 3. Protected Area Barrier 4. Vital Area Barrier 5. Delay Barriers	 10 CFR 73.55(e)(10) 10 CFR 73.55(e)(6) 10 CFR 73.55(e)(8) 10 CFR 73.55(e)(9) 10 CFR 73.55(e)(3)(C)(ii) and (iii)	X	X
Target Sets	10 CFR 73.55(f)	X	X
Security Posts and Structures	10 CFR Part 73, Appendix C, Section II, B.3.c(v)(3)	X	X
Access Requirements 1. Access Authorization and Fitness for Duty 2. Insider Mitigation Program 3. Picture Badge Systems 4. Searches a. VBS Access Control Point b. Packages and Materials c. PA Vehicle Search d. PA Personnel Search 5. PA Access Controls a. Escort and Visitor Requirements	 10 CFR 73.56, 10 CFR Part 26 10 CFR 73.55(b)(9) 10 CFR 73.55(g)(6)(ii) 10 CFR 73.55(h)(2) 10 CFR 73.55(h)(3) 10 CFR 73.55(h)(3) 10 CFR 73.55(h)(3) 10 CFR 73.55(g); 10 CFR 73.56 10 CFR 73.55(g)(7)	X	X

Table D-1. Applicability of Physical Security Requirements IN 10CFR 73 to SMRs.

Security Topic Area	Security Regulations	Existing LWR	Future SMR
6. VA Access Controls 7. Waterborne Threat	10 CFR 73.55(g)(4) 10 CFR 73.55(e)(10)(ii)		
Surveillance, Observation, Monitoring 1. Illumination 2. Surveillance 3. Intrusion Detection Equipment 4. CAS and SAS Operation 5. Security Patrols a. OCA b. PA and VA	 10 CFR 73.55(i)(6) 10 CFR 73.55(i)(5) 10 CFR 73.55(i)(1)(2)(3) 10 CFR 73.55(i)(4) 10 CFR 73.55(i)(5) Same as above Same as above	X	X
Communications 1. Notifications 2. System Descriptions	 10 CFR 73.55(j); 10 CFR 73.55(k)(8)(iii) Same as above	X	X
Review, Evaluation, and Audit of the Security Program	10 CFR 73.55(m)	X	X
Response Requirements	10 CFR 73.55(k)	X	X
Special Situations Affecting Security 1. Refueling/Major Maintenance 2. Construction and Maintenance	 10 CFR 73.55(b)(3)(i) Same as above	X	X
Testing and Maintenance 1. Intrusion Detection & Access Control Devices 2. Search Equipment 3. Communications Equipment 4. Security Personnel Equipment 5. Firearms 6. Active Vehicle Barrier System	 10 CFR 73.55(n) Same as above Same as above Same as above Same as above; 10 CFR Part 73, Appendix B, Section VI, G.3 10 CFR 73.55(e)(10)(i)(B)	X	X
Compensatory Measures 1. PA Physical Barriers 2. VA Barriers 3. Perimeter Intrusion Detection Alarm System 4. PA Lighting 5. VA Portal Alarms 6. Closed Circuit Television/Nonfixed	 10 CFR 73.55(o); 10 CFR Part 73, Appendix G, reportable safeguard events, Sections I and II Same as above Same as above Same as above Same as above Same as above	X	X
Camera System 7. Playback/Recorded Video System 8. Security Computer System 9. PA Device	 Same as above Same as above Same as above	X	X

Table D-1. Applicability of Physical Security Requirements IN 10CFR 73 to SMRs.

Security Topic Area	Security Regulations	Existing LWR	Future SMR
10. Vehicle Barrier System 11. Channeling Barrier System 12. Other Security Equipment	Same as above Same as above Same as above		
Records 1. Access Authorization Records 2. Suitability, Physical, and Psychological Qualification Records for Security Personnel 3. PA and VA Access Control Records a. PA Visitor Access Records b. VA Access Transactions Records c. Vitalization and Devitalization Records d. Vital Area Access List Review 4. Security Plans and Procedures 5. Security Patrols, Inspections, and Tests 6. Maintenance 7. CAS/SAS Alarm Annunciation and Security Responses 8. LLEA Liaison 9. Record of Audits and Reviews 10. Records of Security-Related Keys 11. Security Training and Qualification Records 12. Firearms Testing and Maintenance Records 13. Engineering Analysis for the Vehicle Barrier System	10 CFR 73.70, 10 CFR 73.55(q) 10 CFR 73.55(g)(4)(i), 10 CFR 73.56(o) 10 CFR Part 73, Appendix B, Section VIB 10 CFR 73.70(a) and (b) 10 CFR 73.70(c) 10 CFR 73.70(d) 10 CFR 73.55(q) 10 CFR 73.56(j) 10 CFR 73.55(q)(2) 10 CFR 73.70(e) 10 CFR 73.70(e) 10 CFR 73.70(f); 10 CFR 73.55(q)(2) 10 CFR 73.55(k); 10 CFR Part 73, Appendix C, Section II, B.3.d 10 CFR 73.55(q)(4) 10 CFR 73.70(e) and (h) 10 CFR Part 73, Appendix B; 10 CFR 73:70 10 CFR 73.70(e), 10 CFR Part 73, Appendix B, Section VI, G.3 10 CFR 73.55(e)(2); 10 CFR 73.70; 10 CFR 73.55(c)(7) and (8); 10 CFR 73.21, 10 CFR 73 22	**X**	**X**
Digital Systems Security	10 CFR 73.54; 10 CFR 73.55(b)(8)	**X**	**X**
Temporary Suspension of Security Measures 1. Suspension of Security Measures 2. Suspension of Security Measures During Severe Weather or Other Hazardous Conditions 3. Notification	10 CFR 73.55(p); 10 CFR 50.54; 10 CFR 73.71; 10 CFR 50.72 10 CFR 73.55(p) 10 CFR 73.55(p), 10 CFR 73.71, 10 CFR 50.72	**X**	**X**

DISTRIBUTION

External Distribution

1 Robert N. Hill
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, IL 60439
bobhill@anl.gov (electronic copy)

1 Jim Kinsey
Idaho National Laboratory
PO Box 1625
jim.kinsey@inl.gov (electronic copy)

1 Brian Robinson
U.S. Department of Energy
NE-74
1000 Independence Avenue SW
Washington, DC 20585
brian.robinson@nuclear.energy.gov (electronic copy)

1 Tanju Sofu
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, IL 60439
tsofu@anl.gov (electronic copy)

Internal Distribution

1	MS0421	Sabina Jordan	0249 (electronic copy)
1	MS0721	Peter B. Davies	6200 (electronic copy)
1	MS0736	Evaristo Bonano	6220 (electronic copy)
1	MS0748	Jeffrey LaChance	6231 (electronic copy)
1	MS0748	Timothy Wheeler	6231 (electronic copy)
1	MS0748	Cathy Ottinger Farnum	6231 (electronic copy)
1	MS0757	Felicia Durán	6612 (electronic copy)
1	MS0757	Gregory D. Wyss	6612 (electronic copy)
1	MS1136	Bobby Middleton	6221 (electronic copy)
1	MS1136	Gary Rochau	6221 (electronic copy)
1	MS1234	Gregory Baum	6812 (electronic copy)
1	MS1361	Mark K. Snell	6833 (electronic copy)
1	MS1377	Marie Y. Arrieta	6813 (electronic copy)
1	MS0899	Technical Library	9536 (electronic copy)

Sandia National Laboratories