Evaluation of the Applicability of 10 CFR Part 52 Content of Application Regulatory Requirements to non-Light Water Reactors

Review and justifications for regulations found to be generically not applicable for non-light water reactors

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1 INTRODUCTION

Section 50.1, “General provisions,” to Title 10 of the *Code of Federal Regulations* (10 CFR), states, in part, “The regulations in this part are promulgated by the Nuclear Regulatory Commission pursuant to the Atomic Energy Act of 1954 (AEA), as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242), to provide for the licensing of production and utilization facilities.” Further, the U.S. Nuclear Regulatory Commission (NRC) mission states the following:

> The NRC licenses and regulates the Nation's civilian use of radioactive materials to provide reasonable assurance of adequate protection of public health and safety and to promote the common defense and security and to protect the environment.

The NRC generally implements the relevant portions of the Atomic Energy Act (AEA), as codified in U.S. Code (USC), through regulations, although the NRC retains the authority to establish the level of protection that it considers adequate and reasonable.

Regulations from the CFR for nuclear power plants generally have two characteristics: (1) they contain assumptions about the facility, and (2) they evoke that adequate protection is assured, in part, through compliance.

The delineation between regulations that apply and those that do not is nested in the former – the assumptions in each regulation. Many regulations in 10 CFR Part 50 and 10 CFR Part 52 were tailored to large light water reactors (LWRs) and make assumptions about the technology in the language of each requirement. Many of the assumptions that these regulations make can be applied to other reactor technologies, besides large LWRs. However, some regulations make assumptions that are specific to either large reactors or reactors that are water-cooled. These regulations do not apply, that is their underlying purpose does not apply and they are not technically relevant, to reactors that are not large LWRs. Identification of these regulations should facilitate more streamlined non-light water reactor (non-LWR) applications and more efficient NRC reviews by focusing the application and the NRC staff’s review thereof on the information that is directly relevant to the NRC’s safety findings.

For the regulations that do apply, the premise is that if compliance is demonstrated, the intent of the regulation is met, and adequate protection is assured. However, because some regulations are overly prescriptive or are technology-specific, this logic does not hold for all regulations for all reactors. If a non-LWR meets the underlying intent of applicable regulations without prescriptive compliance, the process to demonstrate this conclusion should be streamlined and consistent between Part 50 and Part 52. However, the question of how to optimize exemptions or otherwise streamline the documentation of meeting the underlying intent of applicable regulations is not the topic of this paper.

This document presents an evaluation of the applicability of 10 CFR Part 52 content of application regulatory requirements to non-LWRs. Specifically, the content of application requirements in 10 CFR 52.79 and associated Part 50 references have been considered. Thus, it is not intended to be an exhaustive review of all regulatory requirements.

2 APPLICABILITY OF REGULATIONS FOR NON-LIGHT WATER REACTORS

2.1 Disposition of applicability of regulations for contents of the FSAR

Applicants for a combined license are required by 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report,” to submit a final safety analysis report (FSAR). Section 52.79 to 10 CFR contains 47 specific requirements for information that must be submitted. Most of these requirements apply to non-LWR designs, and compliance with them should be discussed in the FSAR. The remainder of the 10 CFR 52.79 requirements do not apply to non-LWRs and are discussed in this paper, with justifications. The regulations that do not apply make assumptions about design features that are not present in non-LWRs. Since many 10 CFR 52.79 requirements point back to 10 CFR Part 50, the relevant section from 10 CFR Part 50 is discussed, where appropriate.

Table 2-1 summarizes the disposition of the 10 CFR 52.79 requirements with their generic applicability to non-LWRs. We recognize that the applicability of some regulations to non-LWRs may depend on the specifics of a reactor design. In such cases, the development and use of performance-based “entry conditions” may be an acceptable solution.

Table 2-1: Applicability of 10 CFR 52.79 to non-light water reactors

Section	Short description	Applicable
52.79(a)(1)	Site envelope and boundary	Yes
52.79(a)(2)	Design and analysis of SSCs	Yes
52.79(a)(3)	Radioactive materials produced in operation	Yes
52.79(a)(4)	Principal design criteria	Partial
52.79(a)(5)	Transient analysis	Partial
52.79(a)(6)	Fire protection	Partial
52.79(a)(7)	Pressurized thermal shock	No
52.79(a)(8)	Combustible gas control	Yes
52.79(a)(9)	Station blackout	No
52.79(a)(10)	Environmental qualification of electric equipment	Yes
52.79(a)(11)	Codes and standards	No
52.79(a)(12)	Primary containment leakage rate testing program	No
52.79(a)(13)	Reactor vessel material surveillance program	No
52.79(a)(14)	Operator training program	Yes
52.79(a)(15)	Maintenance rule	Yes
52.79(a)(16)	Effluent monitoring and sampling	No
52.79(a)(17)	Three Mile Island requirements	No
52.79(a)(18)	Risk-informed treatment of SSCs	Yes
52.79(a)(19)	Earthquake criteria	Yes
52.79(a)(20)	Unresolved and generic safety issues	Yes
52.79(a)(21)	Emergency planning	Yes
52.79(a)(22)	Emergency planning with state and local governments	Yes

Section	Short description	Applicable
52.79(a)(23)	Reserved	-
52.79(a)(24)	Prototype operational conditions	Yes
52.79(a)(25)	Quality Assurance Program – design	Yes
52.79(a)(26)	Organizational structure for operations	Yes
52.79(a)(27)	Quality Assurance Program – operation	Yes
52.79(a)(28)	Preoperational testing and initial operations	Yes
52.79(a)(29)	Operational plans	Yes
52.79(a)(30)	Technical Specifications	Yes
52.79(a)(31)	Multi-unit sites	Yes
52.79(a)(32)	Technical qualifications of the applicant	Yes
52.79(a)(33)	Training Program description	Yes
52.79(a)(34)	Operator requalification	Yes
52.79(a)(35)	Physical security plans	Yes
52.79(a)(36)	Safeguards and other security plans	Yes
52.79(a)(37)	Incorporation of operational insights	Yes
52.79(a)(38)	Severe accidents	No
52.79(a)(39)	Radiation Protection Program description	Yes
52.79(a)(40)	Fire Protection Program description	Yes
52.79(a)(41)	Standard Review Plan evaluation	No
52.79(a)(42)	Anticipated transients without scram	No
52.79(a)(43)	Criticality accidents	Yes
52.79(a)(44)	Fitness-for-Duty Program description	Yes
52.79(a)(45)	Minimization of contamination	Yes
52.79(a)(46)	Probabilistic risk assessment summary	Yes
52.79(a)(47)	Aircraft impact assessment	Yes

2.2 Format of this Paper

The regulations identified in Table 2-1 as not applicable, i.e., the response in the Applicable column is “No,” or “Partial,” are discussed in this paper in the context of generic non-applicability to non-LWRs. Specific regulations discussed in this paper are shown in Table 2-2. An additional column is added to indicate the NRC staff position on the generic non-applicability to non-LWRs of each regulation.

Table 2-2: Regulations generically not applicable to non-light water reactors

Section	Short description	Non-applicability description	NRC staff position[1]
52.79(a)(4)	Principal design criteria	General design criteria	In agreement
52.79(a)(5)	Transient analysis	ECCS and RCS vents	In agreement
52.79(a)(6)	Fire protection	Fire protection General Design Criterion	In agreement

52.79(a)(7)	Pressurized thermal shock	Pressurized thermal shock	In agreement
52.79(a)(9)	Station blackout	Station blackout	In agreement
52.79(a)(11)	Codes and standards	Codes and standards	In agreement
52.79(a)(12)	Primary containment leakage rate testing program	Primary containment leakage rate testing program	In agreement
52.79(a)(13)	Reactor vessel material surveillance program	Reactor vessel material surveillance program	In agreement
52.79(a)(16)	Effluent monitoring and sampling	Effluent monitoring and sampling	Partial agreement
52.79(a)(17)	Three Mile Island requirements	Three Mile Island requirements	In agreement
52.79(a)(38)	Severe accidents	Severe accidents	In agreement
52.79(a)(41)	Standard Review Plan evaluation	Standard Review Plan evaluation	In agreement
52.79(a)(42)	Anticipated transients without scram	Anticipated transients without scram	In agreement

[1] Per the NRC staff white paper, "Non-light water review strategy." Sep. 2019.

This paper discusses each of the regulations in Table 2-2. Each non-applicability discussion has the same format and includes the following items:

- Purpose
- Technical justification
- Regulatory justification

The purpose section describes the portion(s) of the regulation that does not apply generically to non-LWRs. Strikeout text is used to denote this non-applicability. Certain regulations do not apply in full, whereas others only partially do not apply. The technical justification section describes the technical reason for why the particular regulation does not apply and suggests entry criteria where useful in identifying non-LWR design characteristics. The regulatory justification provides information such as regulatory precedent that shows that the intent of the regulation was for light water reactors.

2.3 Legal Justification

Additionally, the following legal justification shows legal compliance with the relevant portions of the AEA, as codified in U.S. Code (USC). The regulations identified as not applicable in Table 2-2, above, do not preclude compliance with applicable law or the Commission's required statutory findings. AEA Section 182 ("License Applications"), 42 USC 2232, requires that applicants provide such technical information, including the specific characteristics of the facility, as the Commission may, by rule or regulation, deem necessary to enable it to find that operation of the facility will be in accordance with the common defense and security and provide adequate protection of the public health and safety. AEA Section 185.b, 42 USC 2235(b), "Construction Permits and Operating Licenses," requires, in pertinent part, that an application contain sufficient information to support the issuance of a combined license. Given that the requirements identified in the corresponding regulations listed in Table 2-2 either do not directly apply to non-LWRs or are specific to the characteristics of, or risk of events in, LWRs, compliance with those identified as not applicable is not necessary to support the NRC's required statutory findings under the AEA for non-LWRs.

3 GENERAL DESIGN CRITERIA – 10 CFR 52.79(A)(4)

3.1 Purpose

This section explains why 10 CFR 52.79(a)(4) does not fully apply to non-LWRs. Specifically, the part of the requirement that does not apply is 10 CFR 52.79(a)(4)(i), which points to Appendix A, "General design criteria for nuclear power plants," to Part 50 of 10 CFR. The specific non-applicability is shown in the below quoted text, using strike out:

> ~~(10)~~The principal design criteria for the facility. ~~Appendix A to part 50 of this chapter, "General Design Criteria for Nuclear Power Plants," establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;~~
>
> (ii) The design bases and the relation of the design bases to the principal design criteria;
>
> (iii) Information relative to materials of construction, arrangement, and dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety.

3.2 Technical Justification

The portion of 10 CFR 52.79(a)(4) that does not apply is only with regard to the reference to 10 CFR Part 50, Appendix A. The remainder of 10 CFR 52.79(a)(4) applies and should be covered in the FSAR. Specifically, the FSAR should propose principal design criteria, discuss the design bases and the relation of the design bases to the principal design criteria, and provide information relative to material of construction arrangement, and dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety.

3.3 Regulatory Justification

Principal design criteria are required for each facility licensed under 10 CFR Part 52. Since 10 CFR 52.79 was developed after LWRs were already operating, many of the requirements for contents of the safety analysis report were informed by the operating LWR fleet. Specifically, Appendix A to 10 CFR Part 50 sets forth the "minimum requirements" for water-cooled reactor principal design criteria, referred to as the General Design Criteria (GDC), and notes that the GDC may provide guidance in establishing the PDC for other types of nuclear power units.

These points are reinforced by Regulatory Guide (RG) 1.232, "Guidance for developing principal design criteria for non-light-water reactors," Revision 0, which notes:

> A key part of the regulatory requirements is in the general design criteria (GDC) in Appendix A to 10 CFR Part 50. These high-level GDC requirements support the design of the current nuclear power plants and are addressed in 10 CFR 50.34, "Contents of Applications; Technical Information." Because the current GDC are based on LWR technology, the NRC developed the non-LWR design criteria, included as appendices to this RG, to provide guidance for developing PDC for non-LWR technology.

Although non-LWRs licensed under 10 CFR Part 50 and 10 CFR Part 52 must propose principal design criteria, they do not need to comply with the GDC described in 10 CFR Part 50, Appendix A. As stated by the NRC staff in RG 1.232:

> Together, these requirements recognize that different requirements may need to be adapted for non-LWR designs and that the GDC in 10 CFR 50 Appendix A are not regulatory requirements for non LWR designs but provide guidance in establishing the PDC for non-LWR designs.

RG 1.232 is a guidance document and therefore imposes no regulatory requirements itself. Thus, while the RG offers design criteria that might be useful for non-LWR technologies, its use is voluntary. This fact is reflected in the "Purpose of Regulatory Guides" section of RG 1.232:

> The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required.

Therefore, Appendix A to 10 CFR Part 50 does not strictly apply to non-LWRs or otherwise impose binding legal requirements on non-LWR combined license applicants. NRC staff also recently acknowledged that this regulation is based on LWR technology, and that the principal design criteria provided by the non-LWR designer or applicant establish the necessary design, fabrication, construction, testing, and performance of safety-significant structures, systems, and components (SSCs).[1] Instead of utilizing Appendix A to 10 CFR Part 50, principal design criteria should be proposed in the FSAR, as per 10 CFR 52.79(a)(4).

4 ECCS AND RCS VENTS – 10 CFR 52.79(A)(5)

4.1 Purpose

This section explains why 10 CFR 52.79(a)(5) does not apply, in part, to non-light water reactors. The parts of the regulation that do not apply are the analysis and evaluation of emergency core cooling system (ECCS) cooling performance and the need for high-point vents following postulated loss of coolant accidents, to be performed in accordance with the requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50.46a, "Acceptance criteria for reactor coolant system venting systems." The specific non-applicability is shown in the below-quoted text, using strike out:

> An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. ~~Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter;~~

4.2 Technical Justification

4.2.1 Background

10 CFR 52.79(a)(5) requires an analysis of the ECCS' cooling performance following postulated loss of coolant accidents, as well as an evaluation of whether high point vents are required. Since 10 CFR 52.79 was developed with LWRs in mind, it explicitly references systems that are incorporated by LWRs, such as the ECCS. The ECCS is used by LWRs to provide core decay heat removal capability in the event of a failure of the reactor coolant system. The successful operation of the decay heat removal capability of the ECCS is important for ensuring that LWRs do not exceed fuel safety limits following a loss of coolant accident.

4.2.2 ECCS Cooling Performance

Non-LWRs do not possess an ECCS. A loss of coolant accident is defined in 10 CFR 50.46(c)(1) in the following way:

> Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

A LOCA is a specific major accident for LWRs and the ECCS is a specific LWR system. LOCAs might not be possible or not be technically relevant for non-LWRs; subsequently, ECCS likely does not exist in non-LWRs or has no equivalent. The remainder of the applicable regulation in 10 CFR 52.79(a)(5) already requires the safety analysis during normal operations and transient conditions, which is broad enough to scope the relevant challenging accidents, and should be documented in the FSAR. The specific requirement in 10 CFR 52.79(a)(5) relates to an LWR specific accident and system. Therefore, the requirement to analyze the performance of an ECCS following a loss of coolant accident is not technically relevant for non-LWRs and those accidents that are technically relevant are already scoped by the applicable portion of 10 CFR 52.79(a)(5).

4.2.3 Need for high-point vents following postulated loss of coolant accidents

High point vents are required for LWRs for the reactor coolant system, the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases could cause the loss of function of these systems. 10 CFR 52.79(a)(5) requires an analysis to determine whether, following a LOCA, high point vents are needed as part of the system design to eliminate the challenge posed by a possible accumulation of these noncondensible gases, in order to ensure that adequate core cooling can be maintained.

As discussed in Section 4.2.2, the requirement to evaluate LOCAs is specific to LWRs. LOCAs might not be possible or not be technically relevant for non-LWRs, non-LWRs might not have a reactor coolant system or a reactor vessel head, and non-LWRs might not need to protect against the accumulation of noncondensible gases because the noncondensible gasses or the systems that need to be protected might not exist. The remainder of the applicable regulation in 10 CFR 52.79(a)(5) already requires a safety analysis during normal operations and transient conditions, which is broad enough to scope the relevant challenging accidents, and should be documented in the FSAR. Therefore, the requirement to analyze the need for high-point vents following postulated LOCAs is not technically relevant for non-LWRs and those accidents that are technically relevant are already scoped by the applicable portion of 10 CFR 52.79(a)(5).

4.3 Regulatory Justification

The relevant portion of 10 CFR 52.79(a)(5) for this non-applicability is the following text, “Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter.” Section 50.46 and Section 50.46a to 10 CFR are further discussed in this section.

4.3.1 10 CFR 50.46

10 CFR 50.46 specifically applies to boiling and pressurized water reactors, as stated in 10 CFR 50.46(a)(1)(i):

> Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section.

Non-LWRs are not boiling or pressurized water reactors. The NRC staff has recently acknowledged that this regulation does not apply to non-LWRs, stating, in part:

> However, the second sentence of each citation states that an analysis and evaluation of the ECCS cooling performance shall be provided in accordance with 10 CFR 50.46, which is not applicable to non-LWRs.[1]

Therefore, the design features that are assumed to be present by this regulation do not exist in non-LWRs, and this regulation is not applicable.

4.3.2 10 CFR 50.46a

10 CFR 50.46a has the following language in terms of applicability of the regulation:

> Each nuclear power reactor must be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. High point vents are not required for the tubes in U-tube steam generators.

The need for this regulation was established in 1981 as an amendment to 10 CFR 50.44 because this requirement is more closely related to the regulation for ECCS in 10 CFR 50.46. The need for high point venting was specifically recognized as important to safety in reactor systems that can use natural circulation of the reactor coolant. It was also recognized as important in reactor systems that utilize an ECCS, in order to assure that any noncondensibles could be vented before they cause operational concerns with the ECCS pumps. As such, this requirement strives to mitigate the core cooling function and does not apply to “aggravation” that will be inevitably placed on the containment.[1]

Specifically, the original requirement in 10 CFR 50.44 and the subsequent requirement in 10 CFR 50.46a seek to assure the safety of plants that rely on natural circulation and ECCS for cooling, as stated by the NRC:

> This process is regarded as an important safety feature in accident sequences that credit natural circulation of the reactor coolant system. In other sequences, the pockets of noncondensible gases may interfere with pump operation. The high point vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus, prevents further accident progression.[2]

The assumption that non-LWRs have an ECCS, a reactor coolant system, a reactor vessel head, or systems required to maintain adequate core cooling, considering the accumulation of noncondensible gases, is too prescriptive and not technically relevant. The NRC staff also recently acknowledged that this regulation is based on LWR technology and is likely to not apply to non-LWRs.[1] Therefore, because these design features do not exist in non-LWRs, this regulation is not applicable.

5 FIRE PROTECTION GENERAL DESIGN CRITERION – 10 CFR 52.79(A)(6)

5.1 Purpose

This section explains why 10 CFR 52.79(a)(6) does not apply, in part, to non-LWRs. Specifically, 10 CFR 52.79(a)(6) does not apply to non-LWRs insofar as it requires compliance with GDC 3 of Appendix A to 10 CFR Part 50, with regard to the fire protection plan. The specific non-applicability is shown in the below quoted text, using strike out:

> A description and analysis of the fire protection design features for the reactor necessary to comply with ~~10 CFR part 50, appendix A, GDC 3, and~~ § 50.48 of this chapter.

5.2 Technical Justification

The portion of 10 CFR 52.79(a)(6) that does not apply is only with regard to the reference to 10 CFR Part 50, Appendix A, GDC 3. The remainder of 10 CFR 52.79(a)(6) applies to non-LWRs and should be covered in the FSAR.

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[1] 68 FR 54129, September 16, 2003
[2] 68 FR 54133, September 16, 2003

5.3 Regulatory Justification

As discussed in Section 3.3, the GDC do not apply to non-LWRs because they are based on water-cooled reactor technology. Therefore, GDC 3 of Appendix A to 10 CFR does not apply to non-LWRs as the entire set of the GDC do not apply to non-LWRs.

6 PRESSURIZED THERMAL SHOCK – 10 CFR 52.79(A)(7)

6.1 Purpose

This section explains why 10 CFR 52.79(a)(7) does not apply to non-LWRs. Specifically, no portion of 10 CFR 52.79(a)(7) applies to non-LWRs, as shown using strike out in the below quoted text:

> ~~A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for the reactor vessel beltline materials as defined in 10 CFR 50.60 and 50.61(b)(1) and (b)(2) of this chapter;~~

6.2 Technical Justification

Pressurized thermal shock is an event or transient in pressurized water reactors causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. This kind of event challenges the integrity of the reactor pressure vessel, especially in vessels that have experienced embrittlement from exposure to a high neutron fluence [2]. Because the reactor pressure vessel is a vital component in LWR designs, its integrity needs to be ensured.

Non-LWRs are neither LWRs, generally, nor pressurized water reactors, specifically. Pressurized thermal shock is an event historically identified particular to pressurized water reactors. Since non-LWRs will analyze and identify important events in different ways than historical events specific to LWRs, maintaining applicability of an event specifically important to a pressurized water reactor introduces inconsistencies for non-LWR regulation. Further, 10 CFR 52.79(a) already requires a thorough safety analysis of any design that is the subject of a combined license application.

6.3 Regulatory Justification

6.3.1 10 CFR 50.60

The requirements of 10 CFR 52.79(a)(7) point to 10 CFR 50.60, “Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation,” and 10 CFR 50.61, “Fracture toughness requirements for protection against pressurized thermal shock events.”

Specifically, 10 CFR 50.60(a) states the applicability of the regulation as follows:

> Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.

Non-LWRs do not have the design features assumed by 10 CFR 50.60, specifically as they are not light-water nuclear power reactors. The NRC staff has recently acknowledged that this regulation does not apply to non-LWRs.[1] Therefore, the requirements of 10 CFR 50.60 do not apply.

6.3.2 10 CFR 50.61

Additionally, 10 CFR 50.61(b)(1) describes the applicability of this regulation as follows, in part:

> For each pressurized water nuclear power reactor for which an operating license has been issued under this part or a combined license issued under Part 52 of this chapter, other than a nuclear power reactor facility for which the certification required under § 50.82(a)(1) has been submitted, the licensee shall have projected values of RT_{PTS} or RT_{MAX-X}, accepted by the NRC, for each reactor vessel beltline material…

Because non-LWRs are not pressurized water nuclear power reactors, the design features assumed by 10 CFR 50.61 do not exist. The NRC staff has recently acknowledged that this regulation does not apply to non-LWRs.[1] Therefore, the requirements of 10 CFR 50.61 do not apply.

7 STATION BLACKOUT – 10 CFR 52.79(A)(9)

7.1 Purpose

This section explains why 10 CFR 52.79(a)(9) does not apply to non-LWRs. Specifically, no portion of 10 CFR 52.79(a)(9) applies to non-LWRs, as shown using strike out in the below quoted text:

> ~~The coping analyses, and any design features necessary to address station blackout, as described in § 50.63 of this chapter;~~

7.2 Technical Justification

10 CFR 50.63, “Loss of all alternating current power,” requires LWRs to submit analyses, plans, procedures, and other information related to the ability of the plant to cope and recover from a station blackout. Currently operating LWRs in the U.S. are large and subsequently produce large amounts of decay heat following reactor shutdown. Because of this large amount of decay heat, these LWRs have many safety-related systems, especially active core cooling systems, that rely on alternating current (AC) electrical power to operate. The AC electrical power must be available after reactor shutdown to keep these safety-related systems operational in order to ensure that LWR safety limits are maintained.

Non-LWRs generally are designed such that they do not rely on the use of offsite AC power or emergency onsite power to shut down the reactor, ensure that the core is cooled, or ensure appropriate containment integrity is maintained in the event of an indefinite duration station blackout. Therefore, station blackout is not challenging for most non-LWRs because the reactor has been designed with this LWR experience in mind. Other regulations in 10 CFR 52.79(a) already require a thorough safety analysis. If a station blackout event were relevant to the design, it would be scoped under other portions of 10 CFR 52.79(a). Therefore, station blackout is not technically relevant for non-LWRs and 10 CFR 50.63 is not applicable.

7.3 Regulatory Justification

The requirements of 10 CFR 52.79(a)(9) point to 10 CFR 50.63. Specifically, 10 CFR 50.63(a)(1) outlines the applicability of this requirement, as follows:

> Each light-water-cooled nuclear power plant licensed to operate under this part, each light-water-cooled nuclear power plant licensed under subpart C of 10 CFR part 52 after the Commission makes the finding under § 52.103(g) of this chapter, and each design for a light-water-cooled nuclear power plant approved under a standard design approval, standard design certification, and manufacturing license under part 52 of this chapter must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2.

The purpose of this rule was to ensure that LWRs can withstand a total loss of AC electric power (i.e., station blackout) for a specified duration and can maintain reactor core cooling during that period. The assumption in this regulation is that AC electric power is needed for both essential and nonessential service and is provided by offsite power. These systems are assumed to provide power for various safety functions, including decay heat removal and containment heat removal.[3]

Non-LWRs are not light-water-cooled nuclear power plants and generally do not require AC electric power to maintain reactor core cooling. The NRC staff has recently acknowledged that this regulation does not apply to non-LWRs.[1] Additionally, other regulations in 10 CFR 52.79(a) already require a thorough safety analysis. If a station blackout event were relevant to the design, it would be scoped under other portions of 10 CFR 52.79(a). Therefore, the design features that are assumed to be present by this regulation do not exist in non-LWRs, and this regulation is not applicable.

8 CODES AND STANDARDS – 10 CFR 52.79(A)(11)

8.1 Purpose

This section explains why 10 CFR 52.79(a)(11) does not apply to non-LWRs. Specifically, effectively no portion of 10 CFR 52.79(a)(11) applies to non-LWRs, as shown using strike out in the below quoted text:

> ~~A description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter.~~

8.2 Technical Justification

10 CFR 50.55a, “Codes and standards,” requires compliance with certain codes and standards from the American Society of Mechanical Engineers (ASME) and the Institute of Electrical and Electronics Engineers (IEEE). Specifically, 10 CFR 50.55a(a) incorporates codes and standards from ASME and IEEE by reference, and the remainder of Section 50.55a specifies when the incorporated codes and standards must be followed. In the September 2020 NRC Staff Draft White Paper, “*Analysis of Applicability of NRC Regulations for Non-Light Water Reactors*,” NRC staff acknowledge that 10 CFR 50.55a(a) “does not itself impose requirements,” and thus the basis for calling it applicable to non-LWRs is unclear. Further,

[3] 53 FR 23203, June 21, 1988

continuing to guide individual applicants to assess the list of standards is a missed opportunity for regulatory efficiency.

Sections 50.55a(b)-(g) of 10 CFR require that certain components and systems meet specific ASME codes. These sections refer to (b) the ASME Boiler and Pressure Vessel (ASME BPV) Code and the ASME Operation and Maintenance (ASME OM) Code, (c)-(e) Quality Groups, (f) preservice and inservice testing, and (g) preservice and inservice inspection. Each of these requirements are specifically for boiling and pressurized water-cooled nuclear reactors. Examples of differences in technologies between LWRs (i.e., boiling and pressurized water-cooled nuclear reactors) and non-LWRs include operating temperatures and operating pressures. Most non-LWRs are designed to operate at significantly different conditions than those for which these code requirements were developed to address, including temperature, materials, pressure, and other characteristics. Further, certain ASME Codes that are incorporated by reference by the NRC specifically state that they are not applicable to non-LWRs. In contrast, non-LWRs might be able to use other portions of ASME Codes, which are not for pressure vessels, or other industry vessel standards altogether, such as ASTM standards. Therefore, these portions of 10 CFR 50.55a(b) through (g) are not technically relevant to non-light water reactors.

10 CFR 50.55a(h)(2) is dated and thus not applicable to non-LWR applications. Conversely, 10 CFR 50.55a(h)(3) does apply for safety related instrumentation and control in non-LWRs.

10 CFR 50.55a(z) provides for alternatives to 10 CFR 50.55a and is therefore outside of the scope of this document.

8.3 Regulatory Justification

Section 50.55a of 10 CFR is broken up into several paragraphs, which are further grouped for purposes of discussion in this document and shown in Table 8-1. There are five groups discussed: (1) LWR-specific, (2) Outdated, (3) Reserved, (4) Quality, and (5) Safety-related instrumentation and control (I&C). 10 CFR 50.55a(z) is ungrouped because this paragraph provides for alternatives to 10 CFR 50.55a and is therefore outside of the scope of this document. This discussion focuses on the first and second group, LWR-specific and Outdated, respectively.

Table 8-1: 10 CFR 50.55a groups

10 CFR 50.55a paragraph	Group
(a) Documents approved for incorporation by reference	
(1) American Society of Mechanical Engineers (ASME)	LWR-specific
(i) ASME Boiler and Pressure Vessel Code, Section III	LWR-specific
(ii) ASME Boiler and Pressure Vessel Code, Section XI	LWR-specific
(iii) ASME Code Cases: Nuclear Components	LWR-specific
(iv) ASME Operation and Maintenance Code	LWR-specific
(v) ASME Quality Assurance Requirements	Quality
(2) Institute of Electrical and Electronics Engineers (IEEE)	
(i) IEEE standard 279-1968	Outdated
(ii) IEEE standard 279-1971	Outdated
(iii) IEEE standard 603-1991	Safety-related I&C

(iv) IEEE standard 603-1991, correction sheet	Safety-related I&C
(3) U.S. Nuclear Regulatory Commission (NRC) Public Document Room	
(i) NRC Regulatory Guide 1.84, Revision 36	LWR-specific
(ii) NRC Regulatory Guide 1.147, Revision 17	LWR-specific
(iii) NRC Regulatory Guide 1.192, Revision 1	LWR-specific
(b) Use and conditions on the use of standards	
(1) Conditions on ASME BPV Code Section III	LWR-specific
(2) Conditions on ASME BPV Code, Section XI	LWR-specific
(3) Conditions on ASME OM Code	LWR-specific
(4) Conditions on Design, Fabrication, and Materials Code Cases	LWR-specific
(5) Conditions on inservice inspection Code Cases	LWR-specific
(6) Conditions on ASME OM Code Cases	LWR-specific
(c) Reactor coolant pressure boundary	LWR-specific
(d) Quality Group B components	LWR-specific
(e) Quality Group C components	LWR-specific
(f) Preservice and inservice testing requirements	LWR-specific
(g) Preservice and inservice inspection requirements	LWR-specific
(h) Protection and safety systems	
(1) Reserved	Reserved
(2) Protection systems	Outdated
(3) Safety systems	Safety-related I&C
(i)-(y) Reserved	Reserved
(z) Alternatives to codes and standards requirements	None

8.3.1 LWR-specific 10 CFR 50.55a requirements

The paragraphs discussed under this section are 10 CFR 50.55a(a)(1)(i)-(a)(1)(iv), 10 CFR 50.55a(a)(3), 10 CFR 50.55a(b), 10 CFR 50.55a(c), 10 CFR 50.55a(d), 10 CFR 50.55a(e), 10 CFR 50.55a(f), and 10 CFR 50.55a(g).

Section 50.55a(a)(1)(i)-(a)(1)(iv) of 10 CFR refer to the following items, respectively:

- ASME Boiler and Pressure Vessel Code, Section III
- ASME Boiler and Pressure Vessel Code, Section XI
- ASME Code Cases: Nuclear Components—(A) ASME BPV Code Case N-513-3 Mandatory Appendix I; ASME BPV Code Case N-513-3, “Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1,” Mandatory Appendix I, “Relations for F m, F b, and F for Through-Wall Flaws”
- ASME Operation and Maintenance Code

10 CFR 50.55a(a)(3) refers to the following NRC RGs and respective restrictions:

- RG 1.84, Revision 36, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," dated August 2014, with the requirements in paragraph (b)(4) of this section.
- RG 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," dated August 2014, which lists ASME Code Cases that the NRC has approved in accordance with the requirements in paragraph (b)(5) of this section.
- RG 1.192, Revision 1, "Operation and Maintenance Code Case Acceptability, ASME OM Code," dated August 2014, which lists ASME Code Cases that the NRC has approved in accordance with the requirements in paragraph (b)(6) of this section.

10 CFR 50.55a(b) provides requirements on the use and conditions of the codes and RGs listed in 10 CFR 50.55a(a)(1)(i)-(a)(1)(iv) and 10 CFR 50.55a(a)(3), respectively. Therefore, 10 CFR 50.55a(b) is not discussed separately.

10 CFR 50.55a(c), 10 CFR 50.55a(d), 10 CFR 50.55a(e), 10 CFR 50.55a(f), and 10 CFR 50.55a(g) provide requirements on the use and conditions of the ASME BPV Code. Therefore, 10 CFR 50.55a(c), 10 CFR 50.55a(d), 10 CFR 50.55a(e), 10 CFR 50.55a(f), and 10 CFR 50.55a(g) are not discussed separately from the ASME BPV Code discussion in Section 8.3.1.1.

8.3.1.1 ASME BPV Code

The NRC has incorporated by reference parts of the ASME BPV Code Section III in its regulations. Specifically, the only portions of the ASME BPV Code Section III that are incorporated by reference are located in Division 1. The ASME Codes generally apply only to boiling and pressurized water-cooled reactors, as stated during the time the rule was created:[4]

> It has been generally recognized that, for boiling and pressurized water-cooled reactors, pressure vessels, piping, pumps, and valves which are part of the reactor coolant pressure boundary should, as a minimum, be designed, fabricated, inspected, and tested in accordance with the requirements of the applicable American Society of Mechanical Engineers (ASME) codes in effect at the time the equipment is purchased…

Further, the NRC underscored the importance of important to safety components in water-cooled reactor designs:[5]

> The Commission considers that a significant improvement in the level of quality in design, fabrication, and testing of systems and components important to safety of water-cooled reactors will be afforded by compliance with the requirements of more recent versions of the codes than those specified in the amendments, or portions thereof, and encourages such compliance whenever practicable, regardless of the date of purchase of equipment or the provisions of these amendments.

[4] 36 FR 11423, June 12, 1971
[5] 36 FR 11424, June 12, 1971

The use of the ASME BPV Code is required by the regulations for LWRs. Further, many non-LWRs will operate at temperatures and with materials that might not be covered by those ASME Codes that are incorporated by reference into 10 CFR 50.55a. Because non-LWRs are not water-cooled designs, 10 CFR 50.55a(a)(1)(i)-(iii), 10 CFR 50.55a(a)(3)(i)-(ii), 10 CFR 50.55a(b)(1)-(2), 10 CFR 50.55a(b)(4)-(5), and 10 CFR 50.55a(c)-(g) do not apply. The NRC staff has recently acknowledged that these regulations do not apply to non-LWRs.[1, 3] Therefore, because these design features do not exist in non-LWRs, this regulation is not applicable.

8.3.1.2 ASME OM Code

The NRC has incorporated by reference parts of the ASME OM Code. As stated in RG 1.192, Revision 2, the ASME OM Code was developed in the context of rules for the IST and inservice examination of pumps, valves, and dynamic restraints:

> In 1990, the ASME published the initial edition of the OM Code that provides rules for IST and inservice examination of pumps, valves, and dynamic restraints (snubbers). The OM Code was developed and is maintained by the ASME Committee on Operation and Maintenance of Nuclear Power Plants. The OM Code was developed in response to the ASME Board on Nuclear Codes and Standards directive that transferred responsibility for development and maintenance of rules for the IST and inservice examination of pumps, valves, and dynamic restraints (snubbers) from the ASME Section XI Subcommittee on Nuclear Inservice Inspection to the ASME OM Committee. The ASME intended the OM Code to replace Section XI rules for IST and inservice examination of pumps, valves, and dynamic restraints (snubbers), and the Section XI rules for IST and inservice examination of these components that had been incorporated by reference into NRC regulations have been deleted from Section XI.

The ASME OM Code was incorporated by reference in NRC regulations in September 1999, with the following clarification from the NRC:[6]

> These provisions provide updated rules for the construction of components of light water-cooled nuclear power plants, and for the inservice inspection and inservice testing of those components

Further the ASME OM Code is specific to LWRs and LWR components. As such, 10 CFR 50.55a(a)(1)(iv), 10 CFR 50.55a(a)(3)(iii), 10 CFR 50.55a(b)(3), and 10 CFR 50.55a(b)(6) do not apply to non-LWRs. The NRC staff has recently acknowledged that these regulations do not apply to non-LWRs.[1, 3] Therefore, because these design features do not exist in non-LWRs, this regulation is not applicable.

8.3.2 Outdated 10 CFR 50.55a requirements

The paragraphs discussed under this section are 10 CFR 50.55a(a)(2)(i)-(a)(2)(ii) and 10 CFR 50.55a(h)(2).

Sections 50.55a(a)(2)(i) and(a)(2)(ii) of 10 CFR both have the following reference, "referenced in paragraph (h)(2) of this section." 10 CFR 50.55a(h)(2) states as follows:

> *Protection systems.* For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements in IEEE Std

[6] 64 FR 51370, September 22, 1999

> 279-1968, “Proposed IEEE Criteria for Nuclear Power Plant Protection Systems,” or the requirements in IEEE Std 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations,” or the requirements in IEEE Std 603-1991, “Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603-1991 and the correction sheet dated January 30, 1995.

Since future non-LWR applications will be filed after 1999, neither IEEE 279-1968, IEEE 279-1971, nor IEEE 603-1991 will apply to such applications. Therefore, the requirements of 10 CFR 50.55a(a)(2)(i)-(a)(2)(ii) and 10 CFR 50.55a(h)(2) are dated and do not apply to the already-filed non-LWR application or to any future non-LWR application that could be filed. Therefore, 10 CFR 50.55a(h)(3) will apply to non-LWRs going forward. [3]

9 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM – 10 CFR 52.79(A)(12)

9.1 Purpose

This section explains why 10 CFR 52.79(a)(12) does not apply to non-LWRs. Specifically, 10 CFR 52.79(a)(12), in its entirety, does not apply to non-LWRs, as shown using strike out in the below quoted text:

> ~~A description of the primary containment leakage rate testing program, and its implementation, necessary to ensure that the containment meets the requirements of Appendix J to 10 CFR 50;~~

9.2 Technical Justification

10 CFR 52.79(a)(12) requires compliance with Appendix J to 10 CFR 50. Appendix J, "Primary reactor containment leakage testing for water-cooled power reactors,” to 10 CFR Part 50 was developed as a technology-specific appendix for water-cooled reactors. In this appendix are descriptions for necessary tests to verify that the primary containment or related systems do not exceed allowable leakage rates as specified in the design’s technical specifications. In addition to confirming that leakage rates are below set limits, monitoring leakage for a water-cooled containment provides confidence that the containment structure is maintained during its service life.

This regulation makes the assumption that the design has a primary reactor containment and that its leakage has a significance to safety. In contrast, these design features are likely not to be present in non-LWRs or not have an impact on safety. As identified in SECY-18-0096, and approved in the corresponding staff requirements memorandum, non-LWRs may rely upon “functional containment” without a pressure retaining containment structure. Other regulations in 10 CFR 52.79(a) already require a thorough safety analysis and an analysis of the radioactive materials produced during operation. If leakage of any sort were relevant to the design, it would be scoped under other portions of 10 CFR 52.79(a). Because of drastic differences in reactor design between LWR and non-LWR designs, there is no such equivalent system in non-LWRs for which 10 CFR 52.79(a)(12) was written. Therefore, preoperational and verification testing as described in Appendix J does not apply to non-LWRs.

9.3 Regulatory Justification

The relevant portion of 10 CFR 52.79(a)(12) for this non-applicability is the following text, "A description of the primary containment leakage rate testing program, and its implementation, necessary to ensure that the containment meets the requirements of Appendix J to 10 CFR 50." Appendix to J to 10 CFR Part 50 expressly applies only to water-cooled reactors:

> One of the conditions of all operating licenses under this part and combined licenses under part 52 of this chapter for water-cooled power reactors as specified in § 50.54(o) is that primary reactor containments shall meet the containment leakage test requirements set forth in this appendix.

Non-LWRs are not water-cooled nuclear power reactors. Additionally, the NRC staff has recently acknowledged that this regulation does not apply to non-LWRs.[1] Therefore, the design features that are assumed to be present by this regulation do not exist in non-LWRs, and this regulation is not applicable.

10 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM – 10 CFR 52.79(A)(13)

10.1 Purpose

This section explains why 10 CFR 52.79(a)(13) does not apply to non-LWRs. Specifically, 10 CFR 52.79(a)(13), in its entirety, does not apply to non-LWRs, as shown using strike out in the below quoted text:

> ~~A description of the reactor vessel material surveillance program required by Appendix H to 10 CFR Part 50 and its implementation;~~

10.2 Technical Justification

Commercial LWR designs utilize reactor pressure vessels that are important components to the safety of the reactors. Reactor vessels are typically pressure vessels for LWRs because of the pressurized nature of the coolant, among other considerations. Since the reactor pressure vessel is the main boundary for the reactor coolant boundary and is intended to operate for over 40 years without replacement, the structural integrity is of importance from a safety perspective.

Specifically, the structural integrity is determined through fracture mechanics evaluations that include the measurements or estimates of the fracture toughness of the material resulting from exposure to neutron irradiation and the thermal environment. During operation, neutrons escaping from the reactor core impact the reactor pressure vessel beltline materials, causing embrittlement to those materials. The main factors affecting steel embrittlement include the following [4]:

- Type of steel and its composition and microstructure
- Exposure temperature
- Neutron environment
- Stress state

- Combined embrittlement effects

To ensure structural integrity, if a reactor pressure vessel using a ferritic material exceeds a neutron fluence of 10^{17} n/cm^2, for neutron energies greater than 1 MeV, in the beltline region of the reactor pressure vessel, a material surveillance program is required by Appendix H to 10 CFR Part 50. Specifically, it is important for LWRs to maintain the reactor vessel integrity in order for the fission product barrier to be effective, as well as the maintenance of the reactor coolant.[7]

This regulation makes the assumption that the design has a reactor pressure vessel, the material of which must be monitored to assure the safe operation of the facility. However, non-LWRs might not use pressure vessels for many reasons, including not having a pressurized system. Further, this regulation discusses concerns with carbon steels, which are common in LWR applications but might not be used in non-LWR designs. Other regulations in 10 CFR 52.79(a) already require a thorough safety analysis, which would encompass degradation and potential leakage or structural support concerns. Because of drastic differences in reactor design between LWR and non-LWR designs, there is no such equivalent system in non-LWRs for which 10 CFR 52.79(a)(13) was written. Therefore, reactor vessel material surveillance program as described in Appendix H does not apply to non-LWRs.

10.3 Regulatory Justification

The relevant portion of 10 CFR 52.79(a)(13) for this non-applicability is the following text, "A description of the reactor vessel material surveillance program required by Appendix H to 10 CFR Part 50 and its implementation." Appendix H to 10 CFR Part 50 expressly applies only to light water nuclear power reactors:

> The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment.

Non-LWRs are not light water nuclear power reactors. Additionally, the NRC staff has recently acknowledged that this regulation does not apply to non-LWRs.[1] Therefore, the design features that are assumed to be present by this regulation do not exist in non-LWRs, and this regulation is not applicable.

11 EFFLUENT MONITORING AND SAMPLING – 10 CFR 52.79(A)(16)

11.1 Purpose

This section explains why 10 CFR 52.79(a)(16) does not apply to non-light water reactors. Specifically, 10 CFR 52.79(a)(16) does not apply to non-light water reactors, in part, as shown using strike out in the below quoted text:

> (i) The information with respect to the design of equipment to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, as described in § 50.34a(d) of this chapter;

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[7] 60 FR 65456, December 19, 1995

> ~~(ii) A description of the process and effluent monitoring and sampling program by Appendix I to 10 CFR part 50 and its implementation.~~

11.2 Technical Justification

The control of radioactive materials in liquid and gaseous effluents is required by 10 CFR 52.79 to ensure that occupational dose limits and dose limits to members of the public are maintained to levels as low as reasonably achievable (ALARA) and within design objectives. In order to meet the requirements of this section, an applicant must provide design objectives for limiting effluents and means for keeping radioactive materials in effluents ALARA. In addition, information describing equipment and procedures to control radioactive material in effluents as well as maintenance descriptions for radioactive waste systems must be provided.

The regulations in 10 CFR 52.79(a)(16)(ii) and 10 CFR 50.34a(a) point to Appendix I, "Numerical guides for design objectives and limiting conditions for operation to meet the criterion 'As low as is reasonably achievable' for radioactive materials in light-water-cooled nuclear power plants," to 10 CFR Part 50.

Appendix I to 10 CFR Part 50 is technology-specific and was written for the use of LWRs. Applying numerical guides developed for one technology, LWRs, to a different technology, non-LWRs, is not technologically relevant as non-LWRs might have entirely different effluent compositions and pathways. Further, 10 CFR Part 20, "Standards for the protection against radiation," provides numerical guides for many different isotopes that are technology-agnostic and are applicable to non-LWRs. Therefore, Appendix I to 10 CFR Part 50 is not technologically relevant for non-LWRs and does not apply.

11.3 Regulatory Justification

The requirements of Appendix I to 10 CFR Part 50 state the applicability of the regulation in the title of the appendix and also in Section II, as follows:

> Guides on design objectives for light-water-cooled nuclear power reactors licensed under 10 CFR part 50 or part 52 of this chapter.

Because any potential effluents, and their subsequent limits are dependent on the reactor technology, it's important to highlight that this appendix is for LWRs. For non-LWRs, any potential effluents might vary significantly. Therefore, since non-LWRs are not LWRs, 10 CFR 50, Appendix I does not apply. Furthermore, non-LWR applicants must comply with other portions of 10 CFR 50.34a(d), namely 10 CFR 50.34a(b)(2), and 10 CFR Part 20, which address the control of gaseous and liquid effluents during normal operations and related ALARA considerations.

12 THREE MILE ISLAND REQUIREMENTS – 10 CFR 52.79(A)(17)

12.1 Purpose

This section explains why 10 CFR 52.79(a)(17) does not apply to non-LWRs. Specifically, 10 CFR 52.79(a)(17), in its entirety, does not apply to non-LWRs, as shown using strike out in the below quoted text:

> ~~The information with respect to compliance with technically relevant positions of the Three Mile Island requirements in § 50.34(f) of this chapter, with the exception of § 50.34(f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v).~~

12.2 Technical Justification

10 CFR 50.34(f) was created for LWRs following the Three Mile Island (TMI) Unit 2 accident. The TMI accident occurred on March 28, 1979 and involved a partial meltdown of the Unit 2 reactor core. The accident involved a small radioactive release, but had no detectable health effects on plant workers or the public. The majority of regulatory changes that occurred following the TMI accident were in the areas of emergency response planning, reactor operator training, human factors engineering, and radiation protection.

The TMI accident involved a failure on the secondary side of the plant, specifically of the main feedwater pumps failing to send water to the steam generators. Because heat was not able to be removed from the reactor, the turbine automatically tripped and subsequently, the reactor tripped. To control the increasing pressure on the primary side, the pilot operated relief valve opened, which was located on top of the pressurizer. However, this valve failed to close and became stuck open. Control room instrumentation falsely indicated that the valve was closed, meaning plant personnel were unaware that steam (i.e., cooling water) was being lost through the pressurizer valve. These events were that of a small break LOCA.

This regulation was written specifically to ensure that LWRs in the U.S. were retrofitted to ensure that the probability of an accident such as the TMI accident was reduced or eliminated. Non-LWRs do not use water for cooling, and do not contain a feedwater system, pressurizer, or pilot-operated relief valves on a pressurizer. As noted in Section 4.2.2, a LOCA is a specific major accident for LWRs. Thus, LOCAs might not be possible and likely are not technically relevant for most non-LWR designs. Since the TMI accident was a small break LOCA, there is no discernible analogue in non-LWR designs.

Further, a major issue during the TMI accident was that the instrumentation did not accurately indicate to the control room staff the ongoing conditions in the plant. However, many non-LWRs do not plan to have reactor operators or might not have credited operator actions, such as the operators at TMI. Instead, the plants may function fully automatically. Although human factors are taken into consideration for usability of the monitoring functions of non-LWR designs, they are not a critical safety consideration as they were during the TMI accident. In conclusion, the highly automated nature of non-LWRs obviates the human errors that occurred during the TMI accident.

12.3 Regulatory Justification

The requirements of 10 CFR 52.79(a)(17) point to 10 CFR 50.34(f). Specifically, 10 CFR 50.34(f) states the applicability of the regulation as follows:

> In addition to the requirements of paragraph (a) of this section, each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982, shall meet the requirements in paragraphs (f)(1) through (3) of this section. This regulation applies to the pending applications by Duke Power Company (Perkins Nuclear Station, Units 1, 2, and 3) … In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under part 52 of this chapter shall

> demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section, except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

Because non-LWRs are not light water nuclear power reactors, are not expected to be challenged by LOCAs, and operate under different control systems from the LWRs of the 1970s and earlier, the design features assumed by 10 CFR 50.34(f) do not exist. The NRC staff has recently acknowledged that this regulation does not apply to non-LWRs.[1] Therefore, the requirements of 10 CFR 50.34(f) do not apply to non-LWRs.

13 SEVERE ACCIDENTS – 10 CFR 52.79(A)(38)

13.1 Purpose

This section explains why 10 CFR 52.79(a)(38) does not apply to non-LWRs. Specifically, no portion of 10 CFR 52.79(a)(38) applies to non-LWRs, as shown using strike out in the below quoted text:

> ~~For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.~~

13.2 Technical Justification

The NRC historically has defined an LWR severe accident as an accident involving multiple failures of equipment or function, whose likelihood is generally lower than design-basis accidents, but where consequences may be higher.[8] Thus, by definition, severe accidents are postulated events whose probability of occurrence is so low that they are excluded from the spectrum of design-basis accidents postulated by NRC regulations. Moreover, they involve multiple failures that may result in changes to the reactor core configuration and significant radionuclide releases from the damaged core. For LWRs licensed under Part 52, 10 CFR 52.79(a)(38) makes this clear insofar as it refers to severe accidents as involving "challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass."

It is recognized that the requirement to consider severe accident mitigation design alternatives (SAMDAs) is a NEPA-based requirement derived from a judicial decision[9] and implemented through certain regulations in 10 CFR Part 51, "Environmental protection regulations for domestic licensing and related regulatory functions." However, the NRC's concept of credible severe accident progression sequences (and related SAMDAs) does not appear to apply to other technologies, outside of LWRs.

The example severe accidents (i.e., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass) do not have an equivalent in non-LWRs and do not necessitate further evaluation. Therefore, the requirements of 10 CFR 52.79(a)(38) do not apply and no further information should be included in the FSAR.

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[8] NUREG-1437, Vol. 1 at 5-1 (1996); NUREG-1437, Rev. 1 at 1-27 (2013)
[9] See *Limerick Ecology Action v. NRC*, 869 F.2d 719 (3rd Cir. 1989).

13.3 Regulatory Justification

The requirements of 10 CFR 52.79(a)(38) state the applicability of the regulation as follows:

> For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.

The NRC staff has recently acknowledged that this regulation does not apply to non-LWRs and could be included in the non-LWR design probabilistic risk assessment (PRA), as stated, in part:

> Furthermore, 10 CFR 52.47(a)(23), 10 CFR 52.79(a)(38), 10 CFR 52.137(a)(23), and 10 CFR 52.157(f)(23) require that applications for LWR designs include a description and analysis of design features for the prevention and mitigation of severe accidents and their consequences. The risk consideration associated with these BDBEs [beyond design basis events], as well as other low frequency event sequences, will be considered as part of the required PRA.

Because non-LWRs are not an LWR design, the design features assumed by 10 CFR 52.79(a)(38) do not exist. Furthermore, other regulations that are applicable to non-LWRs require applicants to address the adequacy of SSCs for the prevention and mitigation of the consequences of accidents, including BDBEs. Therefore, the requirements of 10 CFR 52.79(a)(38) do not apply.

14 STANDARD REVIEW PLAN EVALUATION – 10 CFR 52.79(A)(41)

14.1 Purpose

This section explains why 10 CFR 52.79(a)(41) does not apply to non-light water reactors. Specifically, 10 CFR 52.79(a)(41), in its entirety, does not apply to non-light water reactors, as shown using strike out in the below quoted text:

> ~~For applications for light-water-cooled nuclear power plant combined licenses, an evaluation of the facility against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement~~

14.2 Technical Justification

10 CFR 52.79(a)(41) specifically requires applicants for an LWR combined license to perform an evaluation against the light water SRP, otherwise known as NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.” Because NUREG-0800 was written specifically for LWRs, it makes assumptions that the design that would follow this guidance would be an LWR. Non-LWRs do not have the design features assumed by NUREG-0800.

Although there are several standard review plans issued as NUREGs, none of those standard review plans apply to the review of a non-LWR license application. Therefore the requirements of 10 CFR 52.79(a)(41) are not technically relevant for non-LWRs.

14.3 Regulatory Justification

10 CFR 52.79(a)(41) describes its applicability as follows:

> For applications for light-water-cooled nuclear power plant combined licenses, an evaluation of the facility against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application…

Because non-LWRs are not light-water-cooled nuclear power plants, the design features assumed by 10 CFR 52.79(a)(41) do not exist. The NRC staff has recently acknowledged that this regulation does not apply to non-LWRs.[1] Therefore, the requirements of 10 CFR 52.79(a)(41) do not apply.

15 ANTICIPATED TRANSIENTS WITHOUT SCRAM – 10 CFR 52.79(A)(42)

15.1 Purpose

This section explains why 10 CFR 52.79(a)(42) does not apply to non-LWRs. Specifically, 10 CFR 52.79(a)(42), in its entirety, does not apply to non-LWRs, as shown using strike out in the below quoted text:

> ~~Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram (ATWS) events in § 50.62 of this chapter.~~

15.2 Technical Justification

Anticipated transient without scram is defined in 10 CFR 50.62, "Requirements for the reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," in the following way:

> …an anticipated operational occurrence as defined in appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of appendix A of this part.

Appendix A, "General design criteria for nuclear power plants," to 10 CFR Part 50 defines anticipated operational occurrences as follows:

> …those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

Concern related to the consequences of an ATWS, the uncertainty around the reliability of the reactor protection system, and some precursor events in the industry (primarily an event at Browns Ferry 3) led to the ATWS rule in June 1984. The rule included required improvements in the design and operation of LWRs to, "reduce the likelihood of failure of the reactor protection system to shut down the reactor (scram) following anticipated transients and to mitigate the consequences of anticipated transients without scram (ATWS) event.”[10]

The ATWS rule was developed for LWRs and has specific equipment requirements for various types of LWRs. These requirements include the following items:

- For pressurized water reactors, equipment to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip
- For boiling water reactors, a standby liquid control system with the capability of injecting borated water into the reactor pressure vessel at a minimum flow rate
- For boiling water reactors, an alternate rod injection system that is independent from the existing reactor trip system

Non-LWRs generally do not rely on auxiliary or emergency feedwater systems and do not use borated water for reactivity control. Because of the substantial differences in design of non-LWRs and LWRs, the equipment requirements do not have technical relevance.

The ATWS rule was developed in response to concern about ATWS events for LWRs. The requirements of 10 CFR 50.62 are specific to LWRs and do not apply directly to non-LWRs. Other regulations in 10 CFR 52.79(a) already require a thorough safety analysis. If ATWS events were relevant to the design, it would be scoped under other portions of 10 CFR 52.79(a). Therefore, the ATWS requirements of 10 CFR 52.79(a)(44) and 10 CFR 50.62 are not technically relevant for non-light water reactors.

15.3 Regulatory Justification

The relevant portion of 10 CFR 52.79(a)(42) for this non-applicability is the following text:

> Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram (ATWS) events in § 50.62 of this chapter

10 CFR 50.62 specifically applies to commercial light water cooled nuclear reactors, as stated in 10 CFR 50.62(a):

> The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted.

Non-LWRs are not light-water-cooled nuclear power plants. Additionally, the NRC staff has recently acknowledged that this regulation does not apply to non-LWRs.[1] Other regulations in 10 CFR 52.79(a) already require a thorough safety analysis. If ATWS events were relevant to the design, it would be scoped under other portions of 10 CFR 52.79(a). Therefore, the design features that are assumed to be present by this regulation do not exist in non-LWRs and this regulation is not applicable.

[10] 49 FR 26036, June, 1984

16 REFERENCES

[1] Nuclear Regulatory Commission Staff, “Non-light water review strategy.” Sep. 2019.

[2] Christopher Boyd, “Pressurized Thermal Shock, PTS.” Nuclear Regulatory Commission, 2008.

[3] Nuclear Regulatory Commission Staff, "Draft White Paper Analysis of Applicability of NRC Regulations for Non-Light Water Reactors." Sep. 2020.

[4] L. E. Steele, “Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels,” IAEA, Vienna, Austria, Technical Report 163, 1975.