NUSCALE™

Power for all humankind

NuScale Standard Plant
Design Certification Application

Exemptions

PART 7

Revision 4
January 2020

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

1. 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) Reactor Coolant System Venting

1.1 Introduction and Request

1.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from the requirements of 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) that specify high point vents for the reactor coolant system (RCS) and reactor pressure vessel (RPV) head. The underlying purpose of the requirements is to prevent the accumulation of noncondensible gases that may inhibit core cooling during natural circulation. The NuScale design supports core cooling without reliance on high point venting of the RCS and RPV head. The accumulation of noncondensible gases in other systems required to maintain adequate core cooling would not cause a loss of function of those systems, and therefore, the underlying intent of the requirements are met without high point vents.

1.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:
>
> . . .
>
> (4) 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 emergency core cooling system (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;
>
> . . .
>
> (8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f). . .

10 CFR 50.46a states the following:

> 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. Acceptable venting systems must meet the following criteria:
>
> (a) The high point vents must be remotely operated from the control room.
>
> (b) The design of the vents and associated controls, instruments and power sources must conform to appendix A and appendix B of this part.
>
> (c) The vent system must be designed to ensure that:
>
> (1) The vents will perform their safety functions; and
>
> (2) There would not be inadvertent or irreversible actuation of a vent.

10 CFR 50.34(f) states, in part:

> . . .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, . . .
>
> . . .
>
> (2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.
>
> . . .
>
> (vi) Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)

1.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the requirements contained in 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) specifying high point vents for the RCS and RPV head.

1.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale design, as reflected in the Final Safety Analysis Report (FSAR), will not conform with 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). Conformance with 10 CFR 50.34(f)(2)(vi) is discussed in FSAR Table 1.9-5, "NuScale Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)."

1.2 Justification for Exemption

The underlying purpose of 10 CFR 50.46a, requiring high point vents for the RCS and RPV, is to preclude an accumulation of noncondensible gases that may inhibit core cooling. As stated in 68 FR 54123:

> This requirement permitted venting of noncondensible gases that may interfere with the natural circulation pattern in the reactor coolant system. 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.

Similarly, as stated in NUREG-0737 for TMI Item II.B.1, the purpose of 10 CFR 50.34(f)(2)(vi) is to prevent the accumulation of noncondensible gases that may inhibit core cooling during natural circulation.

The NuScale design supports natural circulation core cooling without reliance on the RCS and RPV high point venting specified by 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). In the NuScale design, natural circulation is not inhibited by the accumulation of noncondensible gases and core cooling is not dependent on pump operation. Therefore, the underlying purpose of the requirements is met without high point vents.

1.2.1 Technical Basis

The NuScale design includes an RCS that is integral to the RPV; the core, steam generator, and pressurizer are contained in the vessel. Therefore, the high point of the RCS flow loop and pressurizer is the high point of the RPV. The accumulation of noncondensible gases in the RCS and pressurizer steam space is minimized during normal operation via the high point degasification line.

As described in FSAR Section 5.4.4, the ECCS includes three reactor vent valves (RVVs) located on the top of the RPV that discharge to the CNV upon ECCS actuation, thereby venting any noncondensible gases accumulated in the pressurizer space. Thus, during ECCS operation the ability of the ECCS to maintain adequate core cooling is not impeded. The RCS does not include separate post-accident high point vent capability. During decay heat removal system (DHRS) cooling events, accumulation of noncondensable gases in the pressurizer will not impact the ability of the DHRS to maintain core cooling because the pressurizer volume is not in the DHRS cooling flow path. Accumulation of noncondensable gas in the RPV during DHRS operation does not affect the RPV level because the liquid phase is incompressible. Liquid circulation in the RPV during DHRS operation is therefore not significantly affected by accumulation of noncondensable gas in the RPV. Noncondensible gas accumulation within the secondary system was calculated and considered in the DHRS performance analysis, summarized in FSAR Section 5.4.3, and determined not to impede DHRS operation.

There are no other systems necessary to maintain adequate core cooling that require high point venting. As described in FSAR Section 6.2, accumulated noncondensible gases vented to the containment vessel during ECCS operation will not challenge adequate core cooling. Therefore, the underlying purpose of the 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) requirements to preclude the accumulation of noncondensible gases that may prevent adequate core cooling is met without reliance on the features required by this rule.

1.3 Regulatory Basis

1.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR Part 50 regulations are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption will not impact the consequences of any design basis event and will not create new accident precursors. The NuScale design does not rely on high point vents in the RCS and RPV to accomplish safety functions and any noncondensible gases in the systems prior to, during, and following an accident will not accumulate in the NuScale Power Module in such a way that inhibits the operation of ECCS. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that are necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The design of the RCS, RPV, and other systems required to maintain adequate core cooling precludes the accumulation of noncondensible gases that may inhibit the core cooling during natural circulation.

1.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the requirements of 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi).

2. 10 CFR 50.44 Combustible Gas Control

2.1 Introduction and Request

2.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from 10 CFR 50.44(c)(2) combustible gas control. The NuScale Power Plant design does not require controls to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by a hydrogen combustion event. The NuScale containment vessel (CNV) structural integrity, safe shutdown, and accident mitigating functions are maintained during design basis and beyond design basis events, without inerting the CNV atmosphere or limiting hydrogen concentrations within the CNV. The NuScale Power Plant design meets the underlying purpose of the rule, to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by a hydrogen combustion event, without the controls required by 10 CFR 50.44(c)(2).

2.1.2 Regulatory Requirements

10 CFR 52.47(a)(12) requires a design certification applicant to include:

> An analysis and description of the equipment and systems for combustible gas control as required by 10 CFR 50.44.

10 CFR 50.44(c)(2) states:

> Combustible gas control. All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

2.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the combustible gas control requirements of 10 CFR 50.44(c)(2).

2.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design as reflected in the Final Safety Analysis Report (FSAR) will not conform with 10 CFR 50.44(c)(2).

2.2 Justification for Exemption

The underlying purpose of 10 CFR 50.44 is to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by a hydrogen combustion event. As discussed in NuScale TR-0716-50424, "Combustible Gas Control," the control requirements of 10 CFR 50.44(c)(2) are not necessary to ensure the structural integrity of the

NuScale CNV, safe shutdown functions, and accident mitigation features, during design basis or beyond design basis events. The NuScale CNV design, as described in FSAR Section 6.2.5, does not require an inerted atmosphere or design features to reduce hydrogen concentrations in order to meet the underlying purpose of the rule.

The NuScale request for exemption from 10 CFR 50.44(c)(2) is consistent with the requirements for currently operating nuclear power plants in that hydrogen combustion is not a failure threat to CNV integrity. The requirements of 10 CFR 50.44(b), applicable to currently licensed reactor designs (as of October 16, 2003), require the control of combustible gases only for boiling water reactors with Mark I, II, or III type containments and pressurized water reactors with ice condenser containments. With respect to other containment designs, the Nuclear Regulatory Commission (NRC) determined that:

> …combustion was not a failure threat for large, dry containments and that there was no basis for requiring combustible gas control measures, such as inerting or igniters, in these plants… Severe accident management strategies should account for a threat to containment integrity from a combustion event late in a core meltdown accident sequence.

See attachment 2 of SECY-00-0198, "Feasibility Study for a Risk-Informed Alternative to 10 CFR 50.44, 'Standards for Combustible Gas Control System in Lightwater-cooled Power Reactors.'" The 10 CFR 50.44 rulemaking documents (68 FR 54123) state: "hydrogen combustion is not a significant threat to the integrity of large, dry containments or subatmospheric containments," and therefore, the rule does not require hydrogen controls for these containment designs. The rulemaking documents go on to state: "The NRC believes that accumulation of combustible gases beyond 24 hours can be managed by licensee implementation of the severe accident management guidelines (SAMGs) or other ad hoc actions because of the long period of time available to take such action."

Similar to large, dry containment designs, hydrogen combustion is not a significant threat to NuScale CNV structural integrity and it is not necessary to maintain an inert containment atmosphere or limit hydrogen concentrations in the CNV. For NuScale, compliance with 10 CFR 50.44(c)(2) would result in undue hardship that is significantly in excess of those incurred by others similarly situated (i.e., large, dry or subatmospheric containment designs which do not require hydrogen controls because containment structural integrity is not challenged).

2.2.1 Technical Basis

The underlying purpose of 10 CFR 50.44, to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by a hydrogen combustion event, is met by the NuScale CNV design as described in FSAR Section 6.2.5 and NuScale Technical Report TR-0716-50424. The NuScale CNV can withstand postulated worst-case hydrogen ignition during the first 72 hours of a design basis or beyond design basis event. The NuScale Power Plant design is able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. The NuScale design meets the underlying purpose of the rule without the hydrogen control capabilities required by 10 CFR 50.44(c)(2).

2.3 Regulatory Basis

2.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 are found in 10 CFR 50.12 and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). Application of this requirement to the NuScale Power Plant design is not necessary to achieve the underlying purpose of the rule, which is to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by a hydrogen combustion event. This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any DBE, and does not create new accident precursors. Therefore, the exemption does not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of 10 CFR 50.44, to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by a hydrogen combustion event, is met by the NuScale CNV design as described in FSAR Section 6.2.5 and NuScale TR-0716-50424.

Special circumstances are present (10 CFR 50.12(a)(2)(iii)) in that compliance would result in undue hardship or other costs that are significantly in excess of those incurred by others similarly situated. Similar to large, dry containment designs, hydrogen combustion is not a significant threat to NuScale CNV structural integrity and it is not necessary to maintain an inert containment atmosphere or limit hydrogen concentrations in the CNV. For NuScale, compliance with 10 CFR 50.44(c)(2) would result in undue hardship that is significantly in excess of those incurred by others similarly situated (i.e., large, dry or subatmospheric containment designs, which do not require hydrogen controls because containment structural integrity is not challenged).

2.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the requirements of 10 CFR 50.44(c)(2).

3. 10 CFR 50.62(c)(1) Reduction of Risk from Anticipated Transients Without Scram

3.1 Introduction and Request

3.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from the portion of 10 CFR 50.62(c)(1) requiring diverse equipment to initiate a turbine trip, under conditions indicative of an anticipated transient without scram (ATWS). The NuScale Power Plant design does not include an auxiliary or emergency feedwater system, and therefore, the portion of the rule requiring diverse and automatic auxiliary feedwater system (AFWS) initiation is not applicable. The underlying purpose of 10 CFR 50.62 is to reduce the risk associated with ATWS events. The NuScale Power Plant is designed to reduce the risk of an ATWS event via redundancy, diversity, and independence within the NuScale module protection system (MPS). The MPS design reduces the probability of a failure to scram. When combined with the NuScale Power Plant response to ATWS events, the MPS design results in an ATWS contribution to core damage frequency lower than the safety goal identified in 10 CFR 50.62 rulemaking documents. Therefore, the underlying purpose of the rule is met, without the diverse turbine trip capabilities specified in 10 CFR 50.62(c)(1).

3.1.2 Regulatory Requirements

10 CFR 52.47(a)(15) requires a design certification applicant to include:

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

10 CFR 50.62(c)(1) states:

> Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

3.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the portion of 10 CFR 50.62(c)(1) requiring equipment diverse and independent from the reactor trip system (RTS) to automatically initiate a turbine trip under conditions indicative of an ATWS. The portion of 10 CFR 50.62(c)(1) requiring diverse AFWS initiation is not applicable to the NuScale design, and therefore not within the scope of this exemption request.

3.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale design, as reflected in the Final Safety Analysis Report (FSAR), will not conform with 10 CFR 50.62(c)(1). The NuScale design will be exempt from the portion of 10 CFR 50.62(c)(1) requiring diverse turbine trip capability. The portion of 50.62(c)(1) that requires diverse equipment to initiate AFWS is not applicable to the NuScale design.

3.2 Justification for Exemption

As discussed in SECY-83-293, dated July 19, 1983, 10 CFR 50.62 was promulgated to reduce the risk of an ATWS. The underlying purpose of 10 CFR 50.62 is to reduce the risk from common-cause failures in the reactor protection system leading to a failure to scram (see detailed discussions within SECY-83-293 and NUREG-1780, September 2003). The value-impact calculations discussed in SECY-83-293 were derived from the fundamental constraints of existing plant designs, which limited the options for design enhancements that were considered at the time the rule was promulgated. Digital instrumentation and control designs, similar to those utilized in the NuScale design, were not considered. As discussed further below, the NuScale design process integrated risk reduction for ATWS events during design activities, unconstrained by an existing RTS design. To meet the underlying purpose of the rule, as discussed in SECY-83-293, NuScale designed the MPS to limit the risk from common-cause failures leading to a failure to scram.

The underlying purpose of the specific design features required by 10 CFR 50.62(c)(1) is to reduce the risk associated with ATWS events (see SECY-83-293 and NUREG-1780). The NuScale design does not rely on diverse and independent turbine trip functionality to mitigate the consequences of ATWS events. Additionally, the NuScale Power Plant design does not include an AFWS (or rely on the decay heat removal system to mitigate the consequences of ATWS events); therefore, the portion of 50.62(c)(1) that requires diverse capability to initiate AFWS is not applicable to the NuScale design. The NuScale design does not rely on the design-specific functionality specified in 50.62(c)(1) to meet the underlying purpose of the rule, i.e., to reduce risk associated with ATWS events.

The NuScale Power Plant design represents material circumstances not considered when 10 CFR 50.62 was adopted. The rulemaking process evaluated the nuclear plant design features required by 10 CFR 50.62(c) via design-specific value-impact calculations for the nuclear plants under consideration at the time the rule was drafted (see SECY-83-293, and NUREG-1780). The design requirements within 10 CFR 50.62(c)(1) were delineated for large pressurized water reactors based on the risk reduction it offered for ATWS events for the specific designs evaluated, the specific plant response capabilities of those designs, and the cost of implementing the various options for those designs. 10 CFR 50.62(c)(1) establishes requirements to incorporate additional safety features independent from the "existing reactor trip system," i.e., established designs and operating plants at that time.

The design-specific evaluations discussed in SECY-83-293 did not consider the design options available to future designs. The plant designs that were considered differ significantly from the NuScale design, and the NuScale plant response to an ATWS event differs significantly from the plant responses that were considered during the 50.62 rulemaking. Because of the MPS diversity and passive safety features incorporated into the NuScale Power Plant design, which results in risk reduction from ATWS events, it would be in the public interest to grant an exemption from the specific design features required by 10 CFR 50.62(c)(1).

There are several conflicting regulatory documents describing reduction of risk associated with ATWS events, creating a situation where application of the regulation in the particular circumstances conflicts with other rules or requirements of the Commission. SECY 90-016 dated January 1990, proposed that all evolutionary reactor designs should include a diverse scram system. The Commission approved the NRC Staff's position in the staff requirements memorandum to SECY 90-016, dated June 1990; however, the Commission required that the NRC Staff accept an alternative approach when applicants can demonstrate that the consequences of ATWS events are acceptable. This policy is not reflected in regulation. 10 CFR 50.62 does not contain requirements applicable to NuScale for diverse scram capabilities, and the regulatory requirements of 10 CFR 50.62 applicable to NuScale (i.e., 50.62(c)(1)) do not require demonstration that the consequences of ATWS events are acceptable. Furthermore, the policy statement of SECY 90-016 (incorporated into NUREG-0800, Standard Review Plan Section 15.8, Revision 2) requiring a diverse scram system, includes diverse instrumentation and controls (I&C) equipment from sensor output to final actuation device. This conflicts with the NuScale Design Specific Review Standard Chapter 7 Appendix C that discusses I&C design simplicity. This further conflicts with the Commission's Policy Statement on the Regulation of Advanced Reactors (73 FR 60612) which recognizes a preference for "simplified, inherent, passive, or other innovative means to accomplish safety functions," including "highly reliable and less complex shutdown" systems and "inherent safety, reliability, redundancy, diversity, and independence in safety systems." Applying the policy language of SECY 90-016 and Standard Review Plan Section 15.8 to the NuScale design would add additional systems and complexity with no additional safety benefit over the MPS design. As supported within NuScale Design Specific Review Standard Chapter 7 Appendix C and the Policy Statement on the Regulation of Advanced Reactors, the simplicity of the NuScale design affords public benefits by avoiding adverse consequences that arise from added complexity within the design.

3.2.1 Technical Basis

As defined in 10 CFR 50.62(b), an ATWS is an anticipated operational occurrence followed by failure of the reactor trip portion of the protection system. For the NuScale Power Plant, the protection system is the MPS. The purpose of 10 CFR 50.62 is to reduce the risk from common-cause failures in the MPS leading to a failure to scram. The NuScale design incorporates diversity within the MPS, thereby reducing the risk from common-cause failures leading to a failure to scram. The safety goal described in SECY 83-293 is that "the estimated core melt frequency due to ATWS events should probably be no more than about 1E-5 per year." The probabilistic risk assessment model used to evaluate the NuScale design is discussed in FSAR Chapter 19. As described in FSAR Section 19.1.9, the NuScale ATWS contribution to single module core damage frequency is significantly less than the target of 1.0E-5 per reactor year provided in SECY 83-293. The NuScale Power Plant plant response to an ATWS event does not rely on diverse turbine trip functionality to reduce the risks associated with an ATWS. A diverse system to trip the turbine is not required to meet the underlying purpose of the rule, and diverse actuation of AFWS is not applicable to the NuScale Power Plant design, which does not include an AFWS.

<u>MPS Diversity</u>

The NuScale design incorporates diversity within the MPS to reduce the risk associated with ATWS events. As described in FSAR Chapter 7, the NuScale design meets the underlying purpose of 10 CFR 50.62 with a robust RTS that has internal diversity, reducing the probability of a failure to scram.

As discussed in FSAR Sections 7.1 and 7.2, the MPS utilizes the highly integrated protection system (HIPS) platform. NuScale Topical Report TR-1015-18653 describes integration of fundamental I&C design principles into the HIPS design. The HIPS platform encompasses the principles of independence, redundancy, predictability and repeatability, and diversity and defense-in-depth. The highly integrated protection system was developed specifically to provide a simple and reliable solution for nuclear power plant I&C applications. These key design concepts of the HIPS platform contribute to simplicity in both the functionality of the NuScale MPS and in its implementation.

Internal diversity within the MPS is consistent with the underlying purpose of the diverse scram requirements of 10 CRF 50.62(c)(2), although 10 CRF 50.62(c)(2) is not applicable to NuScale, and consistent with the intent of SECY-90-016. The NuScale design was not constrained by an existing RTS design, i.e., one without internal diversity (see discussion above concerning material circumstances that were not considered when the rule was adopted) that resulted in the additional diverse scram system requirements specified in 50.62(c)(2). The diversity within MPS provides a simpler solution than described in 50.62(c)(2), i.e., risk reduction from common cause failures without the addition of a separate system from sensor output to interruption of power to the control rods. The NuScale MPS design, by significantly reducing the probability of a failure to scram, meets the underlying purpose of the rule, i.e., to reduce the risk from common-cause failures leading to a failure to scram.

ATWS Response

As described in FSAR Section 15.8, the NuScale Power Plant design does not rely on diverse turbine trip functionality during ATWS events to reduce the risks associated with an ATWS. To provide insights on the NuScale plant response to postulated ATWS events, the spectrum of event sequences were modeled to include ATWS as discussed in FSAR Section 19.2.2. The NuScale plant response to an ATWS event, with NuScale design features such as passive cooling and low power density core, protects against fuel damage, limiting the risk of an ATWS. Without the design attributes required by 10 CFR 50.62(c)(1) and without relying on diverse and independent decay heat removal system actuation, the NuScale ATWS contribution to single module core damage frequency is significantly less than the target of 1.0E-5 per reactor year provided in SECY 83-293 (FSAR Section 19.1.9). Therefore, diverse system to trip the turbine is not required to meet the underlying purpose of the rule, and diverse actuation of AFWS is not applicable to the NuScale Power Plant design.

3.3 Regulatory Basis

3.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR Part 50 regulations are found in 10 CFR 50.12 and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under

10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). The NuScale Power Plant design incorporates diversity within the MPS, reducing the risk from common-cause failures leading to a failure to scram. The NuScale design does not rely on diverse turbine trip functionality to reduce the risks associated with an ATWS. Therefore, exemption from the provisions of 10 CFR 50.62(c)(1) requiring diverse turbine trip capabilities will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that are necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(i)) in that application of the regulation in the particular circumstances conflicts with other rules or requirements of the Commission. Commission regulations, policy, and guidance describing reduction of risk associated with ATWS events conflict. The staff requirements memorandum to SECY 90-016 is not supported by 10 CFR 50.62. The policy statement of SECY 90-016, which is incorporated into Standard Review Plan Section 15.8, conflicts with the NuScale Design Specific Review Standard Chapter 7, Appendix C and the Policy Statement on the Regulation of Advanced Reactors.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The NuScale Power Plant design does not rely on diverse turbine trip functionality to reduce the risks associated with ATWS. The NuScale design incorporates diversity within the MPS that sufficiently reduces the risk of common-cause failures leading to a failure to scram. The provisions of 10 CFR 50.62(c)(1) requiring diverse turbine trip capabilities are therefore not required for NuScale to meet the underlying purpose of the rule.

Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that other material circumstances are present which were not considered when the regulation was adopted. 10 CFR 50.62 establishes requirements to incorporate additional safety features for "existing reactor trip system[s]," i.e., designs that were established at the time of the issuance of the rule. The nuclear plant design features that formed the basis of 10 CFR 50.62(c)(1) were evaluated via design-specific value-impact calculations for the nuclear plant designs under review at the time the rule was drafted, as documented in SECY-83-293 and NUREG-1780. These designs do not reflect the NuScale design. The NuScale design includes enhanced safety features that sufficiently reduce the risk from ATWS events and also maintains a simpler I&C configuration than the separate equipment considered at the time of the adoption of 10 CFR 50.62. Therefore, it is in the public interest to grant an exemption from the diverse turbine trip feature required by 10 CFR 50.62(c)(1).

3.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the portion of 10 CFR 50.62(c)(1) requiring diverse equipment to initiate a turbine trip under conditions indicative of an ATWS. The portion of 10 CFR 50.62(c)(1) requiring diverse AFWS initiation is not applicable to the NuScale design.

4. 10 CFR 50, Appendix A, Electric Power Systems GDCs

4.1 Introduction and Request

4.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from General Design Criterion (GDC) 17 because there are no safety-related functions in the NuScale design that rely on electrical power. The underlying purpose of GDC 17 is to ensure sufficient electric power is available to accomplish plant safety-related functions. The design of the NuScale Power Plant provides passive safety systems and features to accomplish plant safety-related functions without reliance on electrical power. Therefore, NuScale meets the underlying purpose of the rule without the need for electric power systems specified in GDC 17.

Exemptions from the GDC 18 requirements for inspection and testing of electric power systems and the electric power provisions of GDCs 34, 35, 38, 41, and 44 are requested to address conforming changes. The underlying purpose, to ensure sufficient electric power is available to accomplish the safety functions of the respective systems, is met without reliance on electric power.

4.1.2 Regulatory Requirements

10 CFR 52.47(a) states in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information: . . .
>
> (3) The design of the facility including:
>
> (i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled 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…

The introduction to 10 CFR 50, Appendix A states in part:

> Also there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

10 CFR 50, Appendix A, GDC 17 states:

> Criterion 17 - Electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.
>
> The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.
>
> Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.
>
> Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

10 CFR 50, Appendix A, GDC 18 states:

> Criterion 18 - Inspection and testing of electric power systems. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

10 CFR 50, Appendix A, GDC 34 states:

> Criterion 34 - Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and

other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

10 CFR 50, Appendix A, GDC 35 states:

Criterion 35 - Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

10 CFR 50, Appendix A, GDC 38 states:

Criterion 38 - Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

10 CFR 50, Appendix A, GDC 41 states:

Criterion 41 - Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that

> for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

10 CFR 50, Appendix A, GDC 44 states:

> Criterion 44 - Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

4.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from GDC 17, pertaining to electric power systems, and GDC 18, pertaining to inspection and testing of those electric power systems.

In addition, NuScale requests an exemption from the related provisions of GDC 34, 35, 38, 41, and 44, which describe capabilities for specific systems with respect to electric power. For each of these GDC, exemption is sought from the phrase "for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available)." Exemption from this provision of GDC 34, 35, 38, 41, and 44 is consistent with the proposed exemptions from GDC 17 and GDC 18. GDC 33, which includes the same provision, is subject to a separate exemption request (see DCA Part 7, Section 5).

4.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the Final Safety Analysis Report (FSAR), will not conform with the requirements of GDC 17, GDC 18, or the power provisions of GDC 34, 35, 38, 41, and 44. GDC 17 and 18 will not be principal design criteria for the NuScale design.

The NuScale Power Plant design will conform to principal design criteria instead of GDC 34, 35, 38, 41, and 44. The NuScale principal design criteria 33, 34, 35, 38, 41, and 44 eliminate the power provisions within specific system requirements of each applicable GDC, but are otherwise identical. The NuScale design bases will include the modified design criterion, as stated in FSAR Section 3.1. In each case the phrase "…for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available)…" has been eliminated from the principal design criteria. For example, the proposed NuScale principal design criterion 34 states in part:

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that the system safety-related function can be accomplished, assuming a single failure.

The Principal Design Criteria for the NuScale Design are stated in FSAR Section 3.1.

4.2 Justification for Exemption

As stated in GDC 17, the purpose of the onsite and offsite power systems is for each to:

> provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

As discussed in FSAR Chapter 8, the NuScale Power Plant systems and features relied upon for safe shutdown, core cooling, containment isolation and integrity, and reactor coolant pressure boundary (RCPB) integrity are not dependent on electrical power, whether alternating current (AC) or direct current (DC), to perform their safety-related functions. Therefore, the underlying purpose of the rule is met without reliance on electric power.

GDC 34, 35, 38, 41, and 44 specify systems to perform certain plant safety functions, including electric power system requirements for those functions. The safety-related functions addressed by these GDC are accomplished by the NuScale design via passive systems or inherent design characteristics, without reliance on electric power. Therefore, the GDC provisions for independence and redundancy of power systems do not apply because the safety-related aspects of the functions specified are performed without electric power availability.

GDC 18 identifies inspection and testing requirements for electric power systems important to safety. As described in FSAR Chapter 8, the inspection and testing criterion of GDC 18 is not applicable to, and is not part of the design basis for, NuScale Power Plant electric power systems.

The underlying purpose of GDC 17 (as well as the testing and inspection provisions of GDC 18 and the power provisions of GDC 34, 35, 38, 41, and 44) is to ensure power is available to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled, and containment integrity and other vital functions are maintained in the event of postulated accidents. The NuScale Power Plant design does not rely on power to accomplish these safety-related functions, and therefore, the underlying purpose of the rule is met without the power systems required by GDC 34, 35, 38, 41, and 44.

4.2.1 Technical Basis

GDC 17

As described in FSAR Chapter 8, the NuScale Power Module design does not rely on electrical power to achieve and maintain safe shutdown, to provide core cooling, to ensure containment vessel isolation and integrity, or to ensure RCPB integrity during and following a design basis event. The safety-related systems actuate without electrical power and their continued operation relies on mechanisms based on fundamental physical and thermodynamic principles that do not require electrical power (e.g., gravity; natural circulation; convective, radiative, and conductive heat transfer; condensation; and evaporation). Therefore, electrical power is not required to actuate or operate systems or components that perform safety-related functions in the event of postulated accidents.

The NuScale Power Plant safety-related functions are achieved and maintained with no reliance on electrical power; therefore, neither the AC power systems nor the DC power systems are required to be safety-related. The nonsafety-related onsite AC power systems are described in Section 8.3.1. The nonsafety-related DC power systems are described in Section 8.3.2. The NuScale Power Plant is designed with offsite (with or without connections to an external transmission grid) and onsite power systems. In the event of a loss of the AC electrical power supply, a DC electrical power supply system is available to perform certain functions. Qualified isolation devices, described in FSAR 7.1.2.2, provide electrical isolation between NuScale power systems and safety-related equipment.

In current operating nuclear plant designs, safety-related systems require electric power to function; offsite and onsite power systems provide redundant means of supplying electric power to perform those plant safety-related functions. The layers of defense in depth reflected by the GDC are maintained in the NuScale Power Plant design, but without reliance on electric power to perform plant safety-related functions. Further, NuScale addresses defense-in-depth considerations (as shown below) through multiple layers of defense incorporated into plant safety systems, as well as the design and operation of AC and DC power systems. NuScale created multiple independent and redundant layers of defense to ensure that no single layer, no matter how robust, is exclusively relied upon (consistent with NUREG/KM-0009 dated April 2016). The NuScale Power Plant design ensures that an appropriate level of independence, redundancy, and testability is provided to instill a high level of confidence that defense-in-depth considerations have been fully addressed. The NuScale Power Plant design provides:

- no reliance on electrical power (AC or DC) for systems relied upon for safe shutdown, core cooling, containment isolation and integrity, and RCPB integrity.
- redundant passive safety system components.
- multiple onsite AC power sources independent of offsite power availability (see FSAR Section 8.3).
- redundant DC electrical power sources that are independent of AC power (see FSAR Section 8.3).
- sufficient capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout (see FSAR Section 8.4).
- neither onsite nor offsite power is required for the mitigation of Chapter 15 events.
- AC and DC power systems are classified as not risk-significant (see FSAR Section 3.2 Classification of Structures, Systems, and Components and FSAR Sections 19.1 and 19.2 for probabilistic risk assessments).

FSAR Chapter 15 demonstrates that, with no electrical power available, (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The NuScale Power Plant design does not require electric power to meet the underlying purpose of GDC 17, which is to ensure plant safety-related function reliability and availability. The underlying purpose of GDC 17 is met without reliance on electric power.

GDC 18

As discussed in FSAR Chapter 8, the NuScale Power Plant electric power supply systems do not contain any safety-related or risk-significant structures, systems, and components that are required to meet GDC 18. NuScale topical report TR-0815-16497, “Safety Classification of Passive Nuclear Power Plant Electrical Systems” and FSAR Chapter 8 describe the classification of electrical structures, systems, and components, and the applicability of GDC 18, consistent with NRC guidance. In describing the technical rationale for compliance with GDC 18, NUREG-0800 discusses the scope of GDC 18 within Standard Review Plan Sections 8.3.1, Revision 4, and 8.3.2, Revision 4, in terms of Class 1E systems. For example, Standard Review Plan Section 8.3.1 states "the AC power system should provide the capability to perform integral testing of Class 1E systems on a periodic basis." Also, Standard Review Plan Section 8.3.1 states: "In ensuring that the proposed periodic onsite testing capabilities of the onsite ac power system satisfies the requirements of GDC 18 … verify that the design has the built-in capability to permit integral testing of Class 1E systems…" As discussed in FSAR Chapter 8, NuScale AC and DC power systems are nonsafety-related and non-Class 1E. Consistent with NRC guidance and NRC-endorsed industry guidance documents, the inspection and testing criterion of GDC 18 is not applicable to NuScale Power Plant power systems.

As described in FSAR Sections 8.2 and 8.3, the inspection and testing criterion of GDC 18 is not applicable to NuScale AC power systems. The NuScale AC power systems are designed to permit periodic inspection and testing to assess the operability and functionality of the systems and the condition of their components for operational, commercial, and plant investment protection purposes.

As described in FSAR Section 8.3.2, although the highly reliable DC power system and the normal DC power system are not safety-related or risk-significant, the highly reliable DC power system performs functions that warrant the application of surveillance, testing, and quality assurance provisions typically applied to Class 1E DC power systems using a graded approach. These provisions ensure that the highly reliable DC power system is designed to permit appropriate periodic inspection and testing.

GDC 34, 35, 38, 41, and 44

GDCs 34, 35, 38, 41, and 44 identify plant safety-related functions, including electric power system requirements for those functions. The safety-related functions addressed by these GDC are accomplished within the NuScale design via passive systems without reliance on power. The NuScale DSRS Section 8.3.1 states that "GDC 33, 34, 35, 38, 41, and 44 are not applicable to passive designs having the capability to automatically establish and maintain safe-shutdown conditions after design-basis events (DBEs) for 72 hours, without operator action, following a loss of both offsite and onsite ac power sources." As discussed in the NuScale FSAR within the applicable system descriptions, the safety-related functions addressed by these GDC (except GDC 33, which is subject to a separate exemption request) are provided in the NuScale design, with suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities, without reliance on electric power. Therefore, the GDC provisions for independence and redundancy of power systems do not apply because the safety-related functions are performed without electric power availability.

4.3 Regulatory Basis

4.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption will not impact the consequences of any design basis event or create new accident precursors. The NuScale Power Plant design does not rely on electric power systems to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that is necessary to maintain the secure status of the plant. The exemption request does not impact the security power system. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of GDC 17 (as well as the testing and inspection provisions of GDC 18 and the power provisions of GDC 34, 35, 38, 41, and 44) is to ensure sufficient electric power capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The NuScale design does not rely on electric power systems to accomplish safety-related functions or meet these acceptance criteria, and therefore the underlying purpose of the rule is met without the need for power.

Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that other material circumstances are present, which were not considered when the regulation was adopted. GDC 17 was written to encompass nuclear power plant designs currently in operation, which rely on electric power to perform safety-related functions. In the designs considered at the time of the rulemaking, loss of electrical power would cause the loss of safety-related functions; therefore, redundant power system requirements were incorporated into GDC 17. The underlying purpose of GDC 17, which is to ensure sufficient electric power to accomplish safety-related functions, did not consider the potential for passive designs that do not rely on electric power to perform safety-related functions. It is in the public interest to grant the exemption based on the passive design of the NuScale Power Plant, which provides safety system functions without reliance on electric power.

4.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from GDC 17, the testing and inspection provisions of GDC 18, and the power provisions of GDC 34, 35, 38, 41, and 44.

5. 10 CFR 50, Appendix A, GDC 33 Reactor Coolant Makeup

5.1 Introduction and Request

5.1.1 Summary

NuScale Power, LLC (NuScale), requests an exemption from General Design Criterion (GDC) 33, which requires a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary (RCPB), because the NuScale Power Plant design does not require makeup to protect against breaks in the RCPB. The reactor pressure vessel and containment vessel (CNV) design, in conjunction with the passive design and operation of the emergency core cooling system, ensure that the core is not uncovered and adequate core cooling is maintained in the event of a break in the RCPB. Therefore, NuScale meets the underlying purpose of the rule without relying on a reactor coolant makeup system as specified in GDC 33.

5.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:
>
> . . .
>
> (3) The design of the facility including:
>
> (i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled 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. . . .

The introduction to 10 CFR 50, Appendix A states, in part:

> Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

10 CFR 50, Appendix A, GDC 33 states:

> Criterion 33 - Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

5.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the requirements of GDC 33, which requires a system to provide reactor coolant makeup for protection against small breaks in the RCPB.

5.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the Final Safety Analysis Report (FSAR), will not conform with GDC 33. GDC 33 will not be a principal design criterion for the NuScale design.

5.2 Justification for Exemption

As stated in GDC 33, the purpose of the requirement is to provide “protection against small breaks in the reactor coolant pressure boundary.” The safety function of the system is “to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary.”

The NuScale reactor pressure vessel and CNV design retain sufficient reactor coolant system (RCS) inventory that, in conjunction with safety actuation setpoints to isolate the chemical and volume control system (CVCS) from the RCS and operation of the emergency core cooling system, adequate core cooling is maintained and the minimum critical heat flux ratio, the relevant specified acceptable fuel design limit, is greater than the safety limit in the event of a small break in the RCPB. Therefore, the design meets the underlying purpose of GDC 33.

5.2.1 Technical Basis

The NuScale Power Plant design includes a CVCS that maintains RCS inventory during normal operation as described in FSAR Section 9.3.4. However, CVCS makeup is not relied on to protect against exceeding minimum critical heat flux ratio during design basis events because reactor coolant inventory is preserved within the NuScale Power Module by isolating connected systems from the RCS and from containment at safety setpoints.

The reactor pressure vessel and CNV design retain sufficient inventory, in conjunction with the safety actuation setpoints to isolate CVCS from the RCS and operation of the emergency core cool cooling system, to ensure specified fuel design limits are not exceeded and core cooling is maintained with a design basis leak or break in the RCPB, as described in FSAR Section 6.3. Therefore, the purpose of GDC 33 is met without without relying on a reactor coolant makeup system.

5.3 Regulatory Basis

5.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts". The exemption requirements for 10 CFR 50 regulations are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption will not affect the performance or reliability of power operations, will not impact the consequences of any design basis event, and will not create new accident precursors. The NuScale Power Plant design incorporates specific design provisions ensuring adequate reactor coolant inventory so that RCPB leaks and small breaks do not result in loss of core cooling and specific acceptable fuel design limits are not exceeded. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that are necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of GDC 33 is to provide "protection against small breaks in the reactor coolant pressure boundary." The NuScale Power Plant design incorporates specific design provisions that ensure adequate reactor coolant inventory for protection against small breaks in the RCPB, thus meeting the underlying purpose of GDC 33.

5.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption from the requirements of GDC 33 for the NuScale design certification.

6. 10 CFR 50.54(m), Control Room Staffing

6.1 Introduction and Request

6.1.1 Summary

10 CFR 50.54(m) specifies minimum licensed operator staffing requirements and responsibilities as a license condition on operating licenses. These requirements do not address a design with more than three units on a site or more than two units operated from a single control room. Further, licensee decisions regarding licensed operator staffing, including the number, composition, and qualifications of licensed personnel are more appropriately based on features unique to the design rather than the existing large, light water reactor-based staffing levels. Because 10 CFR 50.54(m) is applicable only to a licensee, an exemption from the regulations is not appropriate for a design certification applicant. Therefore, NuScale Power, LLC is requesting that design-specific staffing requirements be incorporated in the design certification to be applicable to licensees referencing the NuScale Power Plant design certification, in lieu of the current requirements of 10 CFR 50.54(m).

6.1.2 Regulatory Requirements

The introduction to 10 CFR 50.54 states:

> The following paragraphs of this section, with the exception of paragraphs (r) and (gg), and the applicable requirements of 10 CFR 50.55a, are conditions in every nuclear power reactor operating license issued under this part. The following paragraphs with the exception of paragraph (r), (s), and (u) of this section are conditions in every combined license issued under part 52 of this chapter, provided, however, that paragraphs (i) introductory text, (i)(1), (j), (k), (l), (m), (n), (w), (x), (y), (z), and (hh) of this section are only applicable after the Commission makes the finding under § 52.103(g) of this chapter.

10 CFR 50.54(m) states:

> (m) (1) A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.
>
> (2) Notwithstanding any other provisions of this section, by January 1, 1984, licensees of nuclear power units shall meet the following requirements:
>
> (i) Each licensee shall meet the minimum licensed operator staffing requirements in the following table:

Minimum Requirements[1] Per Shift for On-Site Staffing of Nuclear Power Units by Operators and Senior Operators Licensed Under 10 CFR Part 55

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
One	Senior Operator	2	2	2	2	2
	Operator	2	3	3	4	4
Two	Senior Operator		2	3	[3]3	3
	Operator		3	4	[3]5	5
Three	Senior Operator				3	4
	Operator				5	6

[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.

(ii) Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit. If a single senior operator does not hold a senior operator license on all fueled units at the site, then the licensee must have at the site two or more senior operators, who in combination are licensed as senior operators on all fueled units.

(iii) When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.

(iv) Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

6.1.3 Requested Action

NuScale Power, LLC requests that minimum licensed operator staffing requirements specific to the NuScale Power Plant design be adopted as requirements applicable to licensees referencing the NuScale Power Plant design certification, in lieu of the requirements stated in 10 CFR 50.54(m). NuScale proposes the following provisions be included in Section V, Applicable Regulations, of the NuScale Power Plant design certification rule.

V. Applicable Regulations

* * *

C. A licensee referencing this appendix is exempt from portions of the following regulations:

1. Paragraph (m) of 10 CFR 50.54—Conditions of licenses—codified as of [date of NuScale Power Plant design certification]. In place, the following requirements shall be conditions of such licenses:

a. A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

b. Licensees shall meet the following requirements:

(1) Each licensee shall meet the minimum licensed operator staffing requirements in the following table:

Minimum Requirements[1] Per Shift for On-Site Staffing of NuScale Power Plants by Operators and Senior Operators Licensed Under 10 CFR Part 55

Number of nuclear power units operating[2]	Position	One to twelve units One control room
None	Senior Operator	1
	Operator	2
One to twelve	Senior Operator	3
	Operator	3

[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 MODE 1, 2, or 3 as defined by the unit's technical specifications.

(2) Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit.

(3) When a nuclear power unit is in MODE 1, 2, or 3, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator a licensed operator or senior operator shall be present at the controls at all times. In addition to the senior operator and licensed operator or senior operator present at the controls, a licensed operator or

senior licensed operator shall be in the control room envelope at all times.

(4) Each licensee shall have present, during alteration or movement of the core of a nuclear power unit (including fuel loading, fuel transfer, or movement of a module that contains fuel), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

6.1.4 Effect on NuScale Regulatory Conformance

As a result of the approval and adoption of the proposed requirements in the NuScale Power Plant design certification, licensees referencing the NuScale Power Plant design certification shall comply with the NuScale-specific staffing regulations in lieu of the current staffing rules in 10 CFR 50.54(m).

As required by 10 CFR 52.47(a)(8), the NuScale Final Safety Analysis Report (FSAR) includes the information necessary to demonstrate compliance with 10 CFR 50.34(f)(2)(iii), considering the staffing requirements requested herein.

6.2 Justification for Rulemaking

6.2.1 Technical Basis

The staffing regulation proposed by NuScale Power, LLC is consistent with the approach detailed in SECY-11-0098, July 2011. SECY-11-0098 proposes an approach to resolving staffing issues for small or multi-module reactor facilities based on the guidance of NUREG-0800, Chapter 18, Revision 2; NUREG-0711, Revision 2; and NUREG-1791, July 2005. The NuScale-proposed staffing requirements are consistent with NUREG-0800, Chapter 18; NUREG-0711, Revision 3; NUREG-1791; and NUREG/CR-6838, February 2004 (endorsed in NUREG-0711) for the review criteria of plant staffing levels that require an exemption from 10 CFR 50.54.

Chapter 18 of the NuScale FSAR addresses the review elements of NUREG-1791. The concept of operations is described in FSAR Section 18.7.2.2. FSAR Section 18.2 describes the relevant operating experience considered to identify and address staffing-related lessons learned. The functional requirements analysis and function allocation is presented in FSAR Section 18.3. The specific tasks that are needed to accomplish functions and their staffing implications are identified in FSAR Section 18.4. FSAR Section 18.5 contains the proposed staffing plan for safe and reliable operation, including job definitions of the licensed operators. Additional data and analyses are addressed by FSAR Section 18.6 (important human actions), FSAR Section 18.7 (human-system interface design), FSAR Section 18.8 (procedure development), and FSAR Section 18.9 (training program development). The design implementation element addressed in FSAR Section 18.11 verifies the implemented design accurately reflects the verified and validated design. Finally, FSAR Section 18.12 addresses the human performance monitoring program.

Based upon the above, the proposed staffing requirements ensure safe operation of the plant under all relevant operational conditions.

6.3 Regulatory Basis

The requested action is authorized by law. This action is not inconsistent with the Atomic Energy Act of 1954, as amended.

The requested action does not present an undue risk to the public health and safety. This action does not affect the performance or reliability of power operations, does not impact the consequences of any design basis event, and does not create new accident precursors. The proposed staffing requirements are based upon the advanced design features and human factors engineering analysis using the methodology provided in NUREG-0711 and NUREG-1791, and validated to demonstrate that the required staff can perform all required functions. Therefore, the requested action does not present an undue risk to the public health and safety.

The requested action is consistent with the common defense and security. The requested action does not affect the design, function, or operation of structures or plant equipment necessary to maintain the secure status of the plant. The proposed action has no impact on plant security or safeguards procedures. Therefore, the requested action is consistent with common defense and security.

Application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the staffing requirements of 10 CFR 50.54(m)(2) is achieved with design-specific staffing requirements based upon advanced design features of the NuScale Power Plant design and a human factors engineering analysis using the methodology provided in NUREG-0711 and NUREG-1791.

Other material circumstances are present, which were not considered when the regulation was adopted. The existing regulation incorporates implicit assumptions and specifies requirements that are not appropriate to apply to the NuScale Power Plant design. Examples include, but are not limited to, a maximum of three units and three control rooms, no more than two units per control room, and at least one senior operator in the control room for each unit in operation.

6.4 Conclusion

On the basis of the information presented, NuScale Power, LLC requests that the Nuclear Regulatory Commission adopt the proposed requirements as a license condition for licensees referencing the NuScale Power Plant design certification, in lieu of the current license condition of 10 CFR 50.54(m).

7. 10 CFR 52, App. A, GDC 52 Containment Leakage Rate Testing

7.1 Introduction and Request

7.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from General Design Criterion (GDC) 52, which requires that the containment be designed so that periodic integrated leak rate testing (ILRT) can be conducted at containment design pressure. The underlying purpose of GDC 52 is to provide design capability for testing that assures that leakage integrity of containment is maintained and that containment vessel (CNV) leakage does not exceed allowable leakage rate values. The NuScale Power Plant CNV design meets the underlying purpose of the requirement by providing CNV design specifications, design capability for inspection and examination, and design capability for testing that together provide assurance that leakage integrity of containment is maintained and that CNV leakage does not exceed allowable leakage rate values. NuScale is requesting an exemption from GDC 52 because ILRT is not required to meet the underlying purpose of the rule.

7.1.2 Regulatory Requirements

10 CFR 52.47(a) states in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information: . . .
>
> (3) The design of the facility including:
>
> (i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled 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…

The introduction to 10 CFR 50, Appendix A states in part:

> Also there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

10 CFR 50, Appendix A, GDC 52 states:

> Criterion 52 - Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

10 CFR 50.54(o), Conditions of Licenses, states:

> Primary reactor containments for water cooled power reactors ... shall be subject to the requirements set forth in appendix J to this part.

Appendix J to 10 CFR 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," states, in part:

> 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. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for these tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases; and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.

7.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50 Appendix A, GDC 52, capability for containment leakage rate testing at design pressure.

10 CFR 50 Appendix J specifies Type A testing directly related to GDC 52. While Appendix J is not applicable to a design certification applicant, NuScale requests that, with approval of the GDC 52 exemption, the NuScale Power Plant design certification rule include exemption from the requirements of 10 CFR 50 Appendix J Type A tests for plants referencing the NuScale design.

7.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale design, as reflected in the FSAR, will not conform with 10 CFR 50 Appendix A, GDC 52. GDC 52 will not be a principal design criterion for the NuScale design.

As a result of the approval and adoption of the proposed Appendix J exemption in the NuScale design certification rule, plants referencing the NuScale design shall be exempt from the requirements of 10 CFR 50 Appendix J Type A tests.

7.2 Justification for Exemption

The underlying purpose of GDC 52 is to ensure that the containment is designed to enable testing to provide assurance that the leakage integrity of containment is maintained during its service life and to provide assurance of the performance of the overall containment system as a barrier to fission product releases. 10 CFR 50, Appendix J identifies containment leakage rate inspection and testing requirements for licensees, including periodic ILRT (Type A tests) as described in GDC 52, as well as local leak rate tests (LLRTs) for equipment penetrations and valves that represent potential containment leakage pathways (Type B and C tests). Appendix J identifies the purpose of containment ILRT as:

> …to assure that leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases…

The NuScale Power Plant CNV design allows testing and inspection, other than as anticipated by GDC 52, which meets the underlying purpose of the rule to assure CNV leakage integrity. The alternate NuScale CNV design, testing, and inspection requirements provide equivalent CNV leakage integrity assurance, and thus meets the underlying purpose of the rule.

7.2.1 Technical Basis

<u>Underlying purpose of the rule.</u> The NuScale containment provides design capability for testing and inspection which assures that the leakage integrity of containment is maintained and that CNV leakage does not exceed allowable leakage rate values, thereby meeting the underlying purpose of the rule. NuScale's containment design allows for comprehensive inspection and examination to provide assurance that no unknown leakage pathways exist. NuScale's containment design ensures that containment flanges maintain contact pressure at accident temperature conditions, concurrent with peak accident pressure. Because no unknown leakage pathways exist, and because penetration and CIV designs support accurate LLRT results, and because containment flange bolting preload verifications ensure the containment flanges are installed per design, quantification of overall containment leakage can be accomplished using Type B and C tests.

As described in FSAR Section 6.2, the NuScale Power Plant CNV is an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class MC component (inclusive of all access and inspection openings, penetrations for emergency core cooling system trip/reset valves, and openings for electrical penetration assemblies). As permitted by ASME NCA-2134(c), the complete CNV is constructed and stamped as an ASME Class 1 vessel in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. All penetrations that are potential leakage pathways are either ASME Class 1 flanged joints capable of Type B testing or ASME Class 1 welded nozzles with isolation valves capable of Type C testing. Because the potential vessel leakage pathways are testable containment penetrations, total CNV leakage can be quantified via 10 CFR 50, Appendix J, Type B and C tests, thus assuring that CNV leakage does not exceed allowable leakage rate values.

Comprehensive inservice inspection ensures that no new leakage pathways develop over the life of the containment system.

NuScale containment leakage integrity assurance is described in NuScale's Technical Report TR-1116-51962. The primary elements discussed in TR-1116-51962 include:

- factory inspection and testing, including hydrostatic testing with zero leakage, to assure initial containment leakage integrity per ASME Section III, Class 1 pressure vessel requirements (i.e., assures that no unknown leakage pathways exist)
- Preservice design pressure leakage testing that loads CNV bolted flange connections to containment design pressure and confirms no visible leakage under these conditions.
- preservice and periodic Type B and C testing of penetrations to assure that overall containment leakage does not exceed allowable leakage rate values (i.e., quantifies overall containment leak rates)
- Containment flange bolting preloading verifications that confirm the flange bolting is preloaded to the design value that maintains flange contact pressure at accident temperature conditions, concurrent with peak accident pressure.
- ASME Section III, Class 1, design, construction, inspection, examination, and testing, and ASME Section XI (inservice testing and inspection, repair and replacement, scheduled examinations, non-destructive examination (NDE) methods, and flaw size characterization, including post-maintenance inspection, examination, and testing for CNV repairs or modifications) to assure continued leakage integrity (i.e., assures that no unknown leak pathways develop over time)
- Type B and C testing, inspections, and administrative controls (e.g., configuration management and procedural requirements for system restoration) to assure leakage integrity associated with activity-based failure mechanisms (i.e., assures that CNV penetrations and CIVs remain within allowable leakage rate values after system and component modifications or maintenance)

The alternate NuScale CNV design, testing, examination, and inspection requirements provide assurance that the leakage integrity of containment is maintained during its service life to support performance of the overall containment system as a barrier to fission product releases, and thereby meet the underlying purpose of the rule.

<u>Circumstances not considered when the regulation was adopted.</u> Other material circumstances are present which were not considered when the regulation was adopted. The requirements of GDC 52 and the test criteria described in Appendix J were established for containment designs different from the NuScale CNV design. A typical containment (i.e., typical of the current operating fleet of nuclear power plants) is a large, permanent, welded steel plate structure, with multiple levels, sub-compartments, and internal structures. The steel containment structures create potential leak pathways, and CNV inspections are not able to access all relevant portions of the typical containment. Unknown leakage pathways may develop over time via degradation or damage of the steel liner and could go undetected due to limited inspection access availability. For this type of containment design, Type A tests serve to identify any unknown leakage pathways within the massive structure.

As described in FSAR Section 6.2, the NuScale CNV is a pressure vessel fabricated from a combination of low alloy steel and austenitic stainless steel that houses, supports, and protects the reactor pressure vessel from external hazards and provides a barrier to the release of fission products. All low alloy steel surfaces are clad on both sides of the vessel to minimize corrosion and degradation due to the environment. The NuScale Power Plant CNV design is significantly different than a typical containment structure as it is designed, constructed, inspected, and tested in accordance with the specifications of ASME Code, Section III, Class 1 pressure vessels (requirements for the material, design, fabrication, examination, and testing), and ASME Code, Section XI (requirements for inservice testing and inspection, repair and replacement, scheduled examinations, testing, inspections, NDE methods, and flaw size characterization). Based on the high pressure and safety functions of the CNV, preservice and inservice requirements for the CNV meet ASME Class 1 criteria, similar to the reactor pressure vessel. The CNV design allows for visual inspection of the entire inner and outer surfaces; therefore, developing a leak through the metal pressure boundary is implausible. Welds between the CNV and the containment isolation valves are ASME Class 1 and will be inspected with a volumetric and surface exam each test interval. The CNV is designed to accommodate comprehensive inspections of welds, including volumetric and surface inspections. All welds are accessible, and there are no areas that cannot be inspected.

The NuScale CNV is a small ASME pressure vessel, and is more comparable to typical reactor pressure vessels in design and dimensions than typical containment structures. GDC 52 and Appendix J Type A testing criteria does not align with typical reactor pressure vessel leakage testing requirements. As discussed in TR-1116-51962, application of Type A testing requirements for the NuScale CNV would likely yield inaccurate leakage results due to the limited effectiveness of Type A acceptance criteria when applied to the NuScale design. NuScale instead applies leakage integrity testing, examination, and inspection that more appropriately aligns with typical pressure vessel requirements. In this way, NuScale provides containment leakage integrity assurance that was not considered when the regulation was adopted.

<u>Benefit to public health and safety.</u> Exemption from GDC 52 and Appendix J Type A tests would result in benefit to the public health and safety due to the inherent safety features of the NuScale containment which is integral to the NPM. NuScale has achieved an improvement in safety over existing plants through simplicity of design and passive safety systems. The integral design of the NPM eliminates most of the reactor piping found on typical reactor vessels, thereby reducing the possibility of a pipe rupture that would result in a loss-of-coolant-accident (LOCA). The NPM is designed to reduce the consequences of design basis LOCAs by using redundant, simplified, passive safety-related systems that eliminate the need for emergency core cooling system (ECCS) pumps, accumulators, and water storage tanks found on conventional PWRs. The availability of passive safety systems for decay heat removal and emergency core cooling, as well as other features of the NuScale design, eliminates the need for external power under accident conditions. The result is a design with a core damage frequency that is lower than the current light water reactor fleet. As described in TR-1116-51962, these inherent safety features of the NuScale containment design present unique challenges to performing integrated leak rate testing at containment design pressure, thereby limiting the ability of the NuScale design to conform with Appendix J Type A testing acceptance criteria and limiting the effectiveness of Type A tests for the NuScale design. The simple, safe NuScale design provides benefit to public health and safety, due to safety features inherent in the design.

Exemption from GDC 52 and Appendix J Type A tests would also result in benefit to the public health and safety by maintaining occupational radiation doses as low as reasonably achievable (ALARA). As described in TR-1116-51962, the NuScale containment design presents unique challenges to performing integrated leak rate testing at containment design pressure. Accessibility constraints and the permanent or temporary installation of a large quantity of additional CNV instrumentation for Type A testing would expose occupational radiation workers to unnecessary radiation doses to support testing, maintenance, and calibration. This unnecessary exposure would be required to perform Type A tests, while Type A tests would provide questionable value to confirming leakage integrity of the NuScale containment. Instead, the NuScale Power Plant design utilizes Type B and C tests to quantify containment leakage.

Industry experience and NRC studies have documented the relative effectiveness of Type A, B, and C tests, concluding that most ILRT failures (Type A) result from leakage that is detectable by LLRTs (Type B and C), (e.g., see Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01, Revision 3). Documented cases where ILRTs have identified leakage not detected during LLRTs (e.g., NRC Bulletin 78-09, "BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures") involve identification of post-maintenance issues. However, the periodicity requirements of Type A tests (10 to 15 years) are such that post-maintenance leakage assurance is not the purpose of these tests. NRC and industry studies have shown that containment leakage pathways primarily originate within equipment penetrations and valves which are effectively detected via LLRT, e.g., most ILRT failures result from leakage that is detectable by Type B and C testing (see NUREG-1273, dated April, 1988, NUREG-1493, dated July, 1995, NUREG-1777, dated August, 2003, Electric Power Research Institute (EPRI) TR-104285, dated August, 1994).

The NuScale CNV is designed to allow testing and inspection that provides reasonable assurance that the allowable containment leakage is not exceeded during its service life, and therefore assures the performance of the overall containment system as a barrier to fission product releases. Performance of Type A tests would expose occupational radiation workers to unnecessary radiation doses without a commensurate safety benefit.

Undue hardship. Compliance with GDC 52 and Appendix J Type A tests would result in undue hardships that are significantly in excess of those contemplated when the regulation was adopted. As discussed above, GDC 52 and Appendix J requirements were written to address the containment structure designs currently in operation. As described in TR-1116-51962, the NuScale containment design presents unique challenges to performing integrated leak rate testing at containment design pressure. Type A testing requirements were not written to address the unique challenges of the NuScale containment design, and the regulation does not provide for testing acceptance criteria or alternatives which adequately address the temperature and pressure variations expected during Type A testing within the NuScale containment. The temperature and pressure impacts on Type A testing and associated acceptance criteria for the NuScale design increase the likelihood of inaccurate results, false test failures, and multiple testing iteration requirements. The relative hardship of such challenges is increased considering the rated power of the NuScale small modular reactor compared to the LLWR designs contemplated by the regulation. Therefore, application to the NuScale design represents undue hardships that are significantly in excess of those contemplated when the regulation was adopted.

Continued leakage integrity is assured by the advanced design of the NuScale containment system. The containment system is a small, high pressure, ASME Section III, Class 1 vessel. Comprehensive preservice and inservice inspections and tests, applying ASME Class 1 criteria, ensure continued system leakage integrity. All surface areas and welds are accessible for inspection. All penetration pathways will be tested to Type B or C criteria at accident or design pressures. These features ensure that continued leakage integrity of the containment system is maintained. Therefore, the NuScale Power Plant CNV is designed to allow alternative testing and inspection that provide equivalent assurance that the allowable containment leakage is not exceeded during its service life, and therefore assures the performance of the overall containment system as a barrier to fission product releases.

7.3 Regulatory Basis

7.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 regulations are found in 10 CFR 50.12 and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption will not affect the performance or reliability of power operations, will not impact the consequences of any design basis event, and will not create new accident precursors. The NuScale Power Plant CNV design, inspection, testing provides assurance that no unknown leakage pathways exist, and CNV leakage is quantified via Type B and C tests, which provides assurance that CNV leakage does not exceed the allowable leakage rate. The NuScale CNV design, inspection, and testing provide assurance of the overall containment system as a barrier to fission product releases to reduce the risk from reactor accidents.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that are necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The NuScale Power Plant CNV is designed to support inspection, testing, and monitoring that verify that no unknown CNV leakage pathways exist or develop, and Appendix J Type B and C tests, which quantify containment leakage. The NuScale CNV design, inspection, and testing assure that CNV leakage rates do not exceed design assumptions and assure the performance of the overall containment system as a barrier to fission product releases. NuScale is requesting an exemption from GDC 52 because integrated leakage rate testing, as prescribed for Appendix J Type A tests, is not required to meet the underlying purpose of the rule.

Special circumstances are present (10 CFR 50.12(a)(2)(iii)) in that compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted. The prescriptive Appendix J Type A testing criteria, and the temperature and pressure impacts on acceptance criteria for the NuScale design, limits the effectiveness of Type A testing and increases the likelihood of inaccurate results, false test failures, and multiple testing iteration requirements.

Special circumstances are present (10 CFR 50.12(a)(2)(iv)) in that the exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption. As discussed above, NuScale's design, testing, inspection, and examination provide leakage integrity assurance consistent with the underlying purpose of the rule; therefore, there is no decrease in safety. Additionally, the inherent safety features of the NuScale design, as well as the ability to maintain occupational radiation doses as low as reasonably achievable by avoiding unnecessary tests, results in benefits to public health and safety.

Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that other material circumstances are present, which were not considered when the regulation was adopted. The requirements of GDC 52 and the Type A test, described in Appendix J, were established for containment designs different from the NuScale CNV design. The NuScale CNV ASME Class 1 pressure vessel design, with comprehensive ASME preservice and inservice inspections and testing requirements, represents material circumstances not considered when the regulation was adopted. It is in the public interest to grant an exemption to GDC 52 due to the containment leakage assurance afforded by the NuScale CNV design, comprehensive ASME preservice and inservice inspections and testing, and periodic 10 CFR 50 Appendix J LLRTs, which together assure the performance of CNV leakage integrity.

7.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the requirements of GDC 52. Additionally, NuScale requests that the NuScale design certification rule include an exemption from 10 CFR 50 Appendix J Type A tests for licensees referencing the NuScale design.

8. 10 CFR 50, Appendix A, GDC 40 Testing of Containment Heat Removal System

8.1 Introduction and Request

8.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from General Design Criterion (GDC) 40, which requires design provisions for periodic pressure and functional testing of the containment heat removal system. Periodic pressure and functional testing is not necessary to satisfy the underlying purpose of ensuring the operability and performance of the containment heat removal function for the NuScale Power Plant design. Operability and performance of the passive components comprising the containment heat removal function for the NuScale design is ensured through inspections in accordance with GDC 39. Containment leakage integrity and emergency core cooling system (ECCS) operability is addressed by other GDC. Therefore, the NuScale design meets the underlying purpose of the rule without designing for periodic pressure and functional testing of the containment heat removal function.

8.1.2 Regulatory Requirements

10 CFR 52.47(a) states in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:
>
> (3) The design of the facility including:
>
> (i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled 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;

The introduction to 10 CFR 50, Appendix A states in part:

> Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

10 CFR 50, Appendix A, GDC 40 states:

> Criterion 40-Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

8.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from GDC 40, periodic pressure and functional testing of the containment heat removal system.

8.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the FSAR, will not conform with GDC 40. GDC 40 will not be a principal design criterion for the NuScale design.

8.2 Justification for Exemption

The underlying purpose of GDC 39 and GDC 40 is to ensure the operability and performance of the containment heat removal function. GDC 39 and 40 address a traditional design that includes an active containment heat removal system, where a combination of inspection and testing is needed to ensure integrity, operability, and performance of the components and system. The NuScale Power Plant containment heat removal function does not have any components that require periodic pressure or functional testing to ensure operability and performance of the containment heat removal function. The operability and performance of the containment heat removal function is ensured through periodic inspections, pursuant to GDC 39. Therefore, the underlying purpose of GDC 40 is met without designing for periodic pressure and functional testing of the containment heat removal function.

8.2.1 Technical Basis

In the NuScale Power Plant design, containment heat removal is an inherent characteristic ensured by the materials and physical configuration of the containment vessel (CNV) partially immersed in the reactor pool, which functions as an ultimate heat sink. The configuration provides the ability to remove containment heat. The containment heat removal function is ensured without reliance on electrical power, valve actuation, cooling water flow, or other active system or component operations. Further design details of this function are described in FSAR Section 6.2.2.

Designing the containment heat removal function for periodic pressure and functional testing, as specified by GDC 40, is not necessary because the containment heat removal function is an inherent characteristic of the NuScale CNV design. Periodic inspection of the containment heat removal surfaces, as addressed by GDC 39, will assess for surface fouling or degradation that could potentially impede heat transfer from the CNV. The inspections ensure the operability and performance of the function. Further details of these inspections and the conformance with GDC 39 is provided in FSAR Section 6.2.2.

Structural and leakage integrity of the CNV, as related to providing an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment pursuant to GDC 16 and maintaining effective CNV heat transfer during a loss-of-coolant accident, is addressed by GDC 50, 51, and 53. Testing and inspection for CNV integrity with respect to GDC 50, 51, and 53 is addressed in FSAR Section 6.2.1.6 (GDC 52 is subject to a separate exemption request; see DCA Part 7, Section 7). Testing and inspection of the ECCS is addressed by GDC 36 and 37, as discussed in FSAR Section 6.3.

8.3 Regulatory Basis

8.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 are found in 10 CFR 50.12 and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is consistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption will not affect the performance or reliability of power operations, will not impact the consequences of any design basis event, and will not create new accident precursors. The containment heat removal function does not have any active components that require periodic pressure or functional testing to ensure operability and performance of the containment heat removal function. The operability and performance of the containment heat removal function is ensured through periodic inspections, pursuant to GDC 39. Accordingly, designing for periodic pressure and functional testing of components, as specified in GDC 40, is not necessary to ensure the containment heat removal function. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of GDC 40 is to ensure the operability and performance of the containment heat removal function. The containment heat removal function is an inherent characteristic of the design provided without reliance on electrical power, valve actuation, cooling water flow, or other active system or component operations. The operability and performance of the passive containment heat removal function is ensured by periodic inspections of the CNV, pursuant to GDC 39, thereby satisfying the underlying purpose of GDC 40.

8.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption from the requirements of GDC 40 for the NuScale design certification.

9. 10 CFR 50, Appendix A, GDC 55, 56, and 57 Containment Isolation

9.1 Introduction and Request

9.1.1 Summary

NuScale Power, LLC (NuScale) requests exemptions from General Design Criteria (GDC) 55, 56, and 57, which specify containment isolation provisions for piping system lines penetrating primary containment, as applied to several containment penetrations in the NuScale Power Plant design. These GDC generally require an isolation barrier inside containment and one outside containment for the purpose of ensuring reliable containment isolation. With respect to GDC 55 and 56, the NuScale Power Plant design meets the purpose of the requirements through redundant containment isolation valves (CIVs) outside containment, with appropriate design provisions to ensure adequate isolation reliability. With respect to GDC 57, the NuScale decay heat removal system (DHRS) meets the underlying purpose of the rule through a closed system inside containment and a closed system outside containment to provide redundant containment barriers.

9.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:
>
> . . .
>
> (3) The design of the facility including:
>
> (i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled 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. . . .

The introduction to 10 CFR 50, Appendix A states, in part:

> Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

10 CFR 50, Appendix A, GDC 55 states:

> Criterion 55 - Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:
>
> (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
>
> (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
>
> (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
>
> (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.
>
> Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.
>
> Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

10 CFR 50, Appendix A, GDC 56 states:

> Criterion 56 - Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:
>
> (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
>
> (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

10 CFR 50, Appendix A, GDC 57 states:

Criterion 57 - Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

9.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50, Appendix A, GDC 55 for the lines with penetrations CNV6, CNV7, CNV13, and CNV14 to allow the placement of two CIVs outside containment rather than locating one of the CIVs inside containment as specified in GDC 55.

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50, Appendix A, GDC 56 for the lines with penetrations CNV5, CNV10, CNV11, and CNV12 to allow the placement of two CIVs outside containment rather than locating one of the CIVs inside containment as specified in GDC 56.

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50, Appendix A, GDC 57 for the lines with penetrations CNV3, CNV4, CNV22, and CNV23 to allow the use of a closed system outside containment rather than providing a CIV as specified in GDC 57.

9.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the FSAR, will not conform with GDC 55, 56, and 57 for the specified lines that penetrate containment; alternate containment isolation provisions are provided for the specified lines.

Other lines subject to the requirements of GDC 57 are in compliance with the applicable provisions. FSAR Section 6.2.4 provides additional discussion on the compliance of these lines.

9.2 Justification for Exemption

GDC 16 requires that a reactor containment and associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. GDC 54 requires, in part, that piping systems penetrating primary reactor containment be provided with isolation capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating the piping systems. GDC 55, 56, and 57 prescribe specific containment isolation provisions for lines that penetrate containment that are part of the reactor coolant pressure boundary (RCPB), connected directly to the containment atmosphere, or closed inside containment, respectively. The underlying purpose of GDC 55, 56, and 57 is to provide reasonable assurance of reliable containment isolation capability, reflecting the importance to safety of isolating each piping system to support the function of containment as an essentially leak-tight barrier against the uncontrolled release of radioactivity. To achieve that purpose, GDC 55, 56, and 57 generally require redundant isolation barriers (a CIV or a closed system), physically separated by the primary containment boundary.

For lines subject to GDC 55 and 56, the NuScale design includes redundant CIVs outside containment, with appropriate design provisions to ensure reliable isolation, even in the event of a single, active failure in the containment isolation provisions. For the DHRS lines subject to GDC 57, the proposed isolation provisions are redundant closed system isolation barriers. Thus, the one CIV outside containment normally required by GDC 57 is replaced by a closed system, which functions as a redundant passive containment barrier in place of the active CIV. This barrier design provides redundant essentially leak-tight barriers between the containment interior and the environs, without relying on an active component to perform the isolation function. Therefore, the containment isolation provisions specified by GDC 55, 56, and 57 are not necessary to achieve the underlying purpose of the rule.

9.2.1 Technical Basis

GDC 55 and GDC 56 Lines

As detailed below, the NuScale Power Plant lines penetrating containment that are part of the RCPB or connected directly to the containment atmosphere include two primary system containment isolation valves (PSCIVs) arranged in series outside containment. As discussed in FSAR Section 6.2.4, each set of two PSCIVs shares a single valve body. The single-body precludes pipe break between the two valves. The valve body is welded directly to the vessel nozzle safe-end outside the CNV. Welding directly to the CNV nozzle safe-end precludes pipe break or leakage between the CNV and the inboard PSCIV. The bolted connection and valve stem packing that forms part of the pressure boundary of the valve includes double seals and a means to detect, measure, and terminate leakage past the seals. The PSCIVs are remotely actuated by an automatic instrumentation and control signal or operator action, and fail closed on a loss of power. Each valve in a pair has a separate instrumentation and control division to provide independence and redundancy.

The reactor coolant system (RCS) injection (CNV6), pressurizer spray supply (CNV7), and RCS discharge (CNV13) lines that penetrate containment are part of the RCPB that penetrate containment, and are thus subject to GDC 55. Each of the three RCS lines has two PSCIVs outside containment. The PSCIVs are Seismic Category 1, American Society of Mechanical Engineers (ASME) Code Class 1 components and constructed in accordance with ASME Boiler and Pressure Vessel Code (BPVC), Section III, Subsection NB. The piping and valves are designed to preclude a breach of containment integrity in conformance with Standard Review Plan (SRP) Section 3.6.2. Therefore, although the RCS lines are not part of an engineered safety feature (ESF) system or required for safe shutdown, the isolation provisions for the RCS lines otherwise meet the intent of the provisions defined by NuScale DSRS Section 6.2.4, acceptance criterion 4.

The reactor pressure vessel high point degasification line (CNV14) is part of the RCPB that penetrates containment, and is thus subject to GDC 55. The CNV14 line has two PSCIVs outside containment. The PSCIVs are Seismic Category 1, ASME Code Class 1 components and constructed in accordance with ASME BPVC Section III, Subsection NB. The piping and valves are designed to preclude a breach of containment integrity in conformance with SRP 3.6.2. Therefore, although the reactor pressure vessel high point degasification line is not part of an ESF system or required for safe shutdown, the isolation provisions for the line otherwise meet the intent of the acceptable alternate provisions defined by NuScale DSRS 6.2.4, acceptance criterion 4.

The containment evacuation line (CNV10) penetrates primary reactor containment and connects directly to containment atmosphere, and is thus subject to GDC 56. The CNV10 line has two PSCIVs outside containment. The PSCIVs are Seismic Category 1, ASME Code Class 2 components, but are constructed and stamped as Class 1 in accordance with ASME BPVC, Section III, Subsection NB. The piping and valves are designed to preclude a breach of integrity in conformance with SRP 3.6.2. Therefore, although the containment evacuation line is not part of an ESF system or required for safe shutdown, the isolation provisions for the line otherwise meet the intent of the provisions defined by NuScale DSRS 6.2.4, acceptance criterion 4.

The containment flood and drain line (CNV11) penetrates primary reactor containment and connects directly to the containment atmosphere, and is thus subject to GDC 56. The CNV11 line has two PSCIVs outside containment. The PSCIVs are Seismic Category 1, ASME Code Class 2 components, but constructed and stamped as Class 1 in accordance with ASME BPVC, Section III, Subsection NB. The piping and valves are designed to preclude a breach of integrity in conformance with SRP 3.6.2. Therefore, although the containment flooding and drain line is not part of an ESF system or required for safe shutdown, the isolation provisions for the line otherwise meet the intent of the provisions defined by NuScale DSRS 6.2.4, acceptance criterion 4.

The control rod drive system (CRDS) supply (CNV12) and return (CNV5) lines penetrate primary reactor containment and are neither part of the RCPB nor connected directly to containment atmosphere. However, these lines inside containment are not credited as GDC 57 barriers and are conservatively treated as if the lines connect directly to containment atmosphere. Thus, these lines are considered subject to GDC 56. Each of the two CRDS lines has two PSCIVs outside containment. The PSCIVs are Seismic Category 1, ASME Code Class 2 components, but constructed and stamped as Class 1 in accordance with ASME BPVC Section III, Subsection NB. The piping and valves are designed to preclude a breach of integrity in conformance with SRP 3.6.2. Therefore, although the CRDS lines are not part of an ESF system or required for safe shutdown, the isolation provisions for the lines otherwise meet the intent of the provisions defined by NuScale DSRS 6.2.4, acceptance criterion 4.

GDC 57 Lines

The decay heat removal system (DHRS) lines (CNV3, CNV4, CNV 22, and CNV 23) penetrate primary reactor containment and are neither part of the RCPB nor connected directly to the containment atmosphere, and are thus subject to GDC 57. As described in FSAR Section 5.4.3, following containment isolation operation (which isolates the main steam and feedwater lines outboard of DHRS), the DHRS is a closed system inside and outside containment. The external DHRS loop does not include containment isolation valves. The two containment isolation barriers are provided by the essentially leaktight closed-loop DHRS outside containment and connecting piping, and the closed-loop steam generator (SG) system within the reactor pressure vessel and connecting piping. The DHRS is a Seismic Category I, ASME Code, Section III, Class 2 design and has a design temperature and design pressure equal to that of the reactor pressure vessel.

Inside containment the SG system and connected piping comprise a closed system consistent with NuScale DSRS 6.2.4, acceptance criterion 15. The feedwater lines supplying each steam generator form part of the closed system inside containment. Relief valves are provided in the feedwater header of each steam generator. These relief valves provide protection against thermally induced overpressure as required by Regulatory Guide 1.141 and American Nuclear Society Standard 56.2 during conditions where the steam generators are water solid between the feedwater and main steam isolation valves. The steam generators are only water solid for limited periods during portions of the startup and shutdown processes. If an inadvertent containment isolation occurred during such a period, then the SG system and DHRS would be water solid. Residual heat would provide heat to the SG and thermally induced overpressure of the solid water lines would be possible. The relief valves discharge from the SG system to the inside of containment. The relief valves are qualified for this condition and are designed to re-seat following the overpressure transient.

Outside containment, the DHRS is a closed system outside containment that functions as the second containment barrier. Consistent with the intent of GDC 57, the outside closed system barrier is redundant to the inside barrier and is physically separated from the inside barrier by the containment vessel. Rather than having a CIV, the closed system outside containment functions as a passive barrier requiring no action to isolate containment. The closed system outside containment is protected from missiles, designed to Seismic Category I and Quality Group B standards, and has a design temperature and design pressure at least equal to that for the containment; it has a design temperature and design pressure equal to that of the reactor pressure vessel. The design of the closed system outside containment precludes a breach of piping integrity in conformance with SRP Section 3.6.2. No single failure will cause a failure of both isolation barriers.

The use of closed systems inside and outside containment as an alternative isolation provision for the NuScale DHRS is not addressed by the GDC or by other regulatory guidance. However, although not directly applicable to a GDC 57, the DHRS outside containment otherwise meets the criteria for a closed system outside containment described in NuScale DSRS 6.2.4, acceptance criterion 5. Further, NuScale DSRS 6.2.4 acceptance criterion 1 recognizes closed systems both inside and outside containment as acceptable alternate containment isolation barriers for instrument lines, while acceptance criterion 6 accepts other types of sealed-closed barriers in place of isolation valves. Therefore, the alternative isolation provisions used for the DHRS penetrations meet the underlying purpose of GDC 57 by ensuring reliable containment isolation through the use of redundant isolation barriers.

9.3 Regulatory Basis

9.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). The containment isolation features included in the NuScale design include alternate containment isolation provisions that meets the underlying purpose of GDC 55, 56, and 57. This exemption will not affect the performance or reliability of power operations, will not impact the consequences of any design basis event, and will not create new accident precursors. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that is necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of GDC 55, 56 and 57--to ensure reliable containment isolation by providing two barriers per containment penetration--is accomplished with the alternative provisions for containment isolation included in the NuScale design.

Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that there are other material circumstances not considered when GDC 57 was adopted. GDC 57 did not anticipate lines penetrating containment that are closed systems both inside and outside containment. Such a design fulfills the purpose of the containment isolation provisions of GDC 57 by providing redundant, passive barriers between the environment inside containment and the environment outside containment.

9.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the provisions of GDC 55, 56, and 57 discussed in this section.

10. 10 CFR 50, Appendix K, Emergency Core Cooling System Evaluation Model

10.1 Introduction and Request

10.1.1 Summary

NuScale Power, LLC requests an exemption from the requirements of 10 CFR 50 Appendix K regarding required and acceptable features of emergency core cooling system (ECCS) evaluation models used to calculate cooling performance following postulated loss-of-coolant accidents (LOCAs). Many of the phenomena that are the subject of 10 CFR 50 Appendix K requirements are not encountered in design basis LOCAs for the NuScale Power Module (NPM) and are not relevant to the NuScale LOCA evaluation model. The underlying purpose of the requirements is to ensure that the consequences of postulated LOCAs from a spectrum of pipe break sizes and locations are conservatively calculated by the LOCA evaluation model. Designing the NPM so that certain phenomena are not encountered during a postulated LOCA is an acceptable alternative and meets the underlying purpose of the requirements.

10.1.2 Regulatory Requirements

10 CFR 52.47(a)(4) requires in part:

> ...Analysis and evaluation of emergency core cooling system (ECCS) cooling performance… following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46…of this chapter.

10 CFR 50.46(a)(1)(ii) states:

> Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of [10 CFR 50] appendix K ECCS Evaluation Models.

10 CFR 50 Appendix K states in part:

> I.A.4 The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards—"Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors." Approved by Subcommittee ANS-5, ANS Standards Committee, October 1971).
>
> I.A.5 The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L.C., "Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, page 7, May 1962).
>
> I.B Each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and

outside of the cladding, both as functions of time. To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The degree of swelling and rupture shall be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation....

I.C.1.b For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F.J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans American Society of Mechanical Engineers, 87, No. 1, February, 1965).

I.C.5.a Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer co-efficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

I.C.7.a The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

10.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale is requesting an exemption from the requirements of 10 CFR 50 Appendix K identified in Section 10.1.2.

10.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale design, as reflected in the Final Safety Analysis Report, does not conform with the requirements of 10 CFR 50 Appendix K identified in Section 10.1.2.

10.2 Justification for Exemption

The underlying purpose of 10 CFR 50 Appendix K is to ensure that the consequences of postulated LOCAs from a spectrum of pipe break sizes and locations are conservatively calculated by the LOCA evaluation model. The Appendix K requirements are based on the event progression and phenomena encountered during a LOCA for a traditional light water reactor design. Rather than modeling the elements of a traditional LOCA, the NuScale LOCA evaluation model reflects the phenomena and processes that are postulated to occur in the NPM during accident conditions. Certain elements of the Appendix K requirements are not modeled because they are precluded by the design of the NPM.

Designing the NPM so that traditional design basis event phenomena are not encountered during a postulated LOCA is an acceptable alternative to the Appendix K modeling requirements and meets the underlying purpose of the requirements.

10.2.1 Technical Basis

The NuScale Topical Report TR-0516-49422 describes the evaluation model for the analysis of design basis LOCAs in the NPM. The topical report includes a description and sample calculations of NPM loss-of-coolant accident scenarios and an assessment of the relative importance of phenomena and processes that may occur in the NPM during accident conditions. The topical report provides the technical basis to demonstrate that the LOCA evaluation model conservatively calculates the consequences of postulated LOCAs and the underlying purpose of 10 CFR 50 Appendix K is met.

Exemption from the Appendix K requirements identified in Section 10.1.2 is technically justified as follows:

Fission Product Decay (I.A.4): Appendix K specifies that heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in Proposed American Nuclear Society Standard, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," dated 1971. The NuScale LOCA evaluation model uses an implementation of Draft ANS-5.1/N18.5, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors," dated 1973 with a 20 percent uncertainty added in place of the 1971 ANS standard. A bounding form of the 1973 ANS standard is an acceptable alternative to the 1971 ANS standard.

Metal-Water Reaction Rate (I.A.5): Appendix K specifies that the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation. Calculated cladding temperatures for design basis LOCAs are well below the level where cladding oxidation occurs on a time scale of a LOCA event for the NPM. Maintaining core coverage and avoiding critical heat flux during postulated LOCAs precludes the occurrence of significant transient cladding oxidation. Therefore, the required features of I.A.5 are excluded from the NuScale LOCA evaluation model.

Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters (I.B): Appendix K specifies requirements for predicting cladding swelling and rupture during a LOCA. Calculated cladding temperatures for design basis LOCAs in the NPM are well below the threshold for cladding swelling and rupture. Peak cladding temperatures in the NPM occur at steady state normal operation. Because swelling and rupture do not occur during normal operation, they will not occur in the NPM during a LOCA event. Therefore, this requirement is not relevant for the NuScale LOCA evaluation model and the required features are excluded from the model.

Discharge Model (I.C.1.b): Appendix K specifies that the break flow rate during blowdown shall be calculated by use of the Moody model for all times after the discharging fluid has been calculated to be two-phase in composition. For the NPM, single-phase flow through the break may recur after the transition to two-phase flow. Following this transition, the NuScale LOCA evaluation methodology utilizes the single-phase critical flow model instead of the Moody model. Use of this model is conservative for single-phase break flow, and is an acceptable alternative to the Appendix K requirement.

Post-CHF Heat Transfer Correlations (I.C.5.a): Appendix K requires validation of heat transfer correlations for post-CHF regimes of transition and film boiling. Critical heat flux does not occur in the NPM for postulated LOCAs, so heat transfer beyond CHF is not encountered. This requirement is not relevant to the NuScale LOCA evaluation model and is therefore excluded from the model.

Core Flow Distribution During Blowdown (I.C.7.a): Appendix K requires that flow calculations during blowdown shall take into account cross flow between core regions. In the NuScale LOCA evaluation model, the core is represented by three non-interacting channels. The hot channel represents the hot fuel assembly, the average channel represents the remainder of the fuel assemblies, and a third channel represents core bypass flow. The assumption of no cross flow between the core regions results in a conservative flow distribution. Therefore, cross flow is excluded from the NuScale LOCA evaluation model.

10.3 Regulatory Basis

10.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 regulations are found in 10 CFR 50.12 and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). Some processes and phenomena specified in Appendix K are not encountered in the NPM during a postulated LOCA. Not modeling these processes and phenomena has no adverse impact on the ability of the LOCA evaluation model to conservatively calculate the consequences of a postulated LOCA, will not impact the consequences of any design basis event, and will not create new accident precursors. The LOCA evaluation model conservatively models the processes and phenomena experienced by the NPM during a postulated LOCA. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that is necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of 10 CFR 50 Appendix K is to ensure that the consequences of postulated LOCAs are conservatively calculated by the LOCA evaluation model. This purpose is met by the NuScale LOCA evaluation model without inclusion of specific model features prescribed by the Appendix K requirements identified in Section 10.1.2 because the NuScale design precludes the underlying phenomena during a postulated LOCA, or because the alternative model features conservatively account for the processes and phenomena experienced by the NPM during a postulated LOCA.

Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that other material circumstances are present, which were not considered when the regulation was adopted. The requirements of Appendix K reflect the phenomena encountered during blowdown, reflood, and refill of a reactor vessel as a result of a large break in primary system piping. These phenomena are not encountered by the NPM due to the integrated design that eliminates large primary system piping and core uncovery. It is in the public interest to grant an exemption to Appendix K due to the demonstrated performance of the NPM in reducing the consequences of a postulated LOCA compared to light water reactors that were considered when Appendix K was adopted.

10.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the requirements of 10 CFR 50, Appendix K, as discussed above.

11. 10 CFR 50.34(f)(2)(xx) Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

11.1 Introduction and Request

11.1.1 Summary

NuScale Power, LLC, (NuScale) requests an exemption from the portions of 10 CFR 50.34(f)(2)(xx) related to pressurizer level indicators. 10 CFR 50.34(f)(2)(xx) specifies power provisions for pressurizer relief valves, block valves, and level indicators. The NuScale Power Plant design does not include pressurizer relief valves or pressurizer block valves; therefore, portions of the rule applicable to such valves are not technically relevant. The underlying purpose of the requirement is to enable natural circulation cooling in a loss of offsite power condition. The NuScale Power Plant design does not rely on pressurizer level indication to achieve and maintain natural circulation in a loss of electrical power condition, and therefore meets the underlying purpose of the rule.

11.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:
>
> . . .
>
> (8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island [TMI] requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v);

10 CFR 50.34(f) states, in part:

> (f) Additional TMI-related requirements.
>
> . . .
>
> 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).
>
> . . .
>
> (xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B)

motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)

11.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the portions of 10 CFR 50.34(f)(2)(xx) requiring power from vital buses and emergency power sources for pressurizer level indication.

11.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the FSAR, will not conform with the provisions of 10 CFR 50.34(f)(2)(xx) related to pressurizer level indicators, as discussed in FSAR Table 1.9-5, "Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)." Additionally, the NuScale Power Plant design does not include pressurizer relief valves or pressurizer block valves; therefore, portions of the rule applicable to such valves are not technically relevant.

11.2 Justification for Exemption

Pursuant to NUREG-0578, July 1979, Recommendation 2.1.1, referenced in NUREG-0737, November 1980, TMI Item II.G.1, the basis of the requirement is as follows:

> In some designs, loss of pressurizer heaters due to a loss of offsite power requires the use of the high-pressure emergency core cooling system to maintain reactor pressure and volume control for natural circulation cooling. Similarly, in some designs the inability to close the power-operated relief valve upon loss of offsite power could result in additional challenges to the high-pressure emergency core cooling system. Finally, proper functioning of the pressurizer level instrumentation is necessary to maintain satisfactory pressure control for natural circulation cooling using the pressurizer heaters.

Thus, the underlying purpose of the requirement is to enable and maintain natural circulation cooling in a loss of offsite power condition. In the NuScale Power Plant design, following the loss of electrical power, the passive decay heat removal system (DHRS) is able to achieve and maintain natural circulation cooling of the reactor coolant system without electrical power. The pressurizer level instrumentation is not necessary to maintain natural circulation cooling, thus achieving the underlying purpose of the rule.

11.2.1 Technical Basis

As stated in FSAR Section 5.4.5, level indicators support pressurizer level and pressure controls during normal operation; however, this indication is not relied upon to establish or maintain natural circulation cooling during transient conditions. The DHRS removes decay heat and brings the reactor coolant system to safe shutdown conditions without reliance on pressurizer level indication. The DHRS design, as discussed in FSAR Section 5.4.3, is a passive system and does not require electrical power to actuate or operate. Therefore, the NuScale reactor is designed to achieve and maintain natural circulation conditions in the event of loss of electrical power, and the underlying purpose of the rule is met without the features specified in the requirement.

11.3 Regulatory Basis

11.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption will not affect the performance or reliability of power operations, will not impact the consequences of any design basis event, and will not create new accident precursors. The NuScale design does not rely on pressurizer level indication to achieve or maintain natural circulation cooling upon the loss of electrical power. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that is necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the 10 CFR 50.34(f)(2)(xx) requirement, to achieve and maintain natural circulation with the loss of electrical power, is accomplished by passive design features that do not require electrical power to operate or actuate. Specifically, natural circulation cooling is achieved and maintained without reliance on pressurizer level indication.

11.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the portions of 10 CFR 50.34(f)(2)(xx) related to pressurizer level indicators.

12. 10 CFR 50.34(f)(2)(xiii), Pressurizer Heater Power Supplies

12.1 Introduction and Request

12.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from 10 CFR 50.34(f)(2)(xiii), which requires power supplies for pressurizer heaters and associated motive and control interfaces to establish and maintain natural circulation in hot shutdown conditions. The underlying purpose of the requirement is to enable natural circulation cooling in a loss of offsite power condition. The NuScale design does not rely on pressurizer heaters to achieve and maintain natural circulation in a loss of electrical power condition, and therefore meets the underlying purpose of the rule.

12.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:
>
> . . .
>
> (8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island [TMI] requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v);

10 CFR 50.34(f) states, in part:

> (f) Additional TMI-related requirements.
>
> . . .
>
> 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).
>
> . . .
>
> (xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's [pressurized water reactors] only) (II.E.3.1)

12.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the requirements of 10 CFR 50.34(f)(2)(xiii).

12.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale design, as reflected in the FSAR, does not conform with 10 CFR 50.34(f)(2)(xiii), as discussed in FSAR Table 1.9-5, "Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)."

12.2 Justification for Exemption

Pursuant to NUREG-0578, July 1979, Recommendation 2.1.1, referenced in NUREG-0737, November 1980, TMI Item II.E.3.1, the basis of the requirement is as follows:

> In some designs, loss of pressurizer heaters due to a loss of offsite power requires the use of the high-pressure emergency core cooling system to maintain reactor pressure and volume control for natural circulation cooling. Similarly, in some designs the inability to close the power-operated relief valve upon loss of offsite power could result in additional challenges to the high-pressure emergency core cooling system. Finally, proper functioning of the pressurizer level instrumentation is necessary to maintain satisfactory pressure control for natural circulation cooling using the pressurizer heaters.

Thus, the underlying purpose of the requirement is to enable natural circulation cooling in a loss of offsite power condition. In the NuScale design, following the loss of electrical power, the passive decay heat removal system (DHRS) is able to achieve and maintain natural circulation cooling of the reactor coolant system without electrical power. The pressurizer heater is not necessary to maintain natural circulation cooling, thus achieving the underlying purpose of the rule.

12.2.1 Technical Basis

As stated in FSAR Section 5.4.5, the pressurizer heaters help maintain the pressurizer pressure during normal operation, however the heaters are not relied upon to establish or maintain natural circulation cooling during transient conditions. The DHRS removes decay heat and brings the reactor coolant system to safe shutdown conditions without reliance on pressurizer heaters. The DHRS design, as discussed in FSAR Section 5.4.3, is a passive system and does not require electrical power to actuate or operate. Therefore, the NuScale Power Module is designed to achieve and maintain natural circulation conditions in the event of loss of electrical power, and the underlying purpose of the rule is met without the features specified in the requirement.

12.3 Regulatory Basis

12.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR Part 50 regulations are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any design basis event, and does not create new accident precursors. The NuScale design does not rely on pressurizer heaters to achieve or maintain natural circulation cooling upon the loss of electrical power. Therefore, the exemption does not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that is necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the 10 CFR 50.34(f)(2)(xiii) requirement, to maintain natural circulation cooling to achieve and maintain hot shutdown conditions with the loss of electrical power, is accomplished by passive design features that do not require electrical power to operate or actuate. Additionally, natural circulation cooling is achieved and maintained without reliance on the features required by this rule.

12.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the portions of 10 CFR 50.34(f)(2)(xiii) related to power supplies for pressurizer heaters.

13. 10 CFR 50.34(f)(2)(xiv)(E) Containment Evacuation System Isolation

13.1 Introduction and Request

13.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the containment evacuation system (CES). The rule requires automatic containment isolation on a high radiation signal of systems that provide a path to the environs from containment. The underlying purpose of the rule is to limit radiological releases by ensuring containment isolation for such systems during events where reliance on a high containment pressure isolation signal may not be sufficient. The NuScale Power Plant design meets the purpose of the rule by ensuring reliable and dependable isolation of CES upon any event involving radiological consequences inside of the CNV. Therefore, alternate means to prevent radiological release from the CES to the environs are provided, such that automatic CES isolation on high radiation signal is not required to meet the underlying purpose of the rule.

13.1.2 Regulatory Requirements

10 CFR 52.47(a) states in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:...
>
> (8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f)...

10 CFR 50.34(f) states, in part:

> (f) 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…

10 CFR 50.34(f)(2)(xiv) requires containment isolation systems which, in part:

> (E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

13.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES.

13.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale CES design as reflected in the Final Safety Analysis Report (FSAR) will not conform with 10 CFR 50.34(f)(2)(xiv)(E).

13.2 Justification for Exemption

Isolation of the CES upon a high radiation signal is not required to ensure containment isolation for systems that provide direct paths to the environs. The underlying purpose of 10 CFR 50.34(f)(2)(xiv)(E) is to limit radiological releases by ensuring containment isolation for systems that provide paths to the environs where reliance on a high containment pressure isolation signal may not be sufficient. The discussions related to 10 CFR 50.34(f)(2)(xiv)(E) within NUREG-0578, dated October 1979; NUREG-0660, dated May 1980; and NUREG-0737, dated November 1980, concern containment purge and vent system isolation, which provide a path to the environs. NRC identified a need to initiate isolation of these systems for core damage scenarios when the event includes release into the containment but may not initiate containment isolation (i.e., containment pressure does not reach the isolation setpoint for small loss of coolant flowrate into containment). See NUREG-0578 Section 2.1.4, NUREG-0660, Task II.E.4.2, and NUREG-0737, Section II.E.4.2.

The NuScale design satisfies the intent of the 10 CFR 50.34(f)(2)(xiv) requirement for dependable containment isolation logic by using two automatic containment isolation signals: high containment pressure and low low pressurizer level. These containment isolation signals, along with other emergency safety feature actuation system logic, are shown to meet all design bases event acceptance criteria with no credit for nonsafety-related systems. Further, all analyzed in-containment events leading to degraded core or core damage conditions, result in both a low low pressurizer signal and, a high containment pressure signal. Any event leading to core damage or degradation, will result in a low low pressurizer level signal. Therefore, events that could result in a challenge to core integrity result in generation of dependable containment isolation signals.

13.2.1 Technical Basis

The NuScale design differs from that of a traditional large light water reactor design of a TMI era vintage because reactor core uncovery, and resulting core damage, cannot occur without reaching a low low pressurizer containment isolation setpoint. An event similar to the Three Mile Island Unit 2 accident is precluded by the NuScale plant design. The pressurizer is located well above the level of the reactor core and not connected to the reactor vessel by piping. Any decrease in reactor vessel inventory to the level of the reactor core would also result in complete emptying of the pressurizer and generation of a low low level containment isolation signal.

Analyses provided in Section 19.2 demonstrate it is impossible for the NuScale design to have a severe accident condition, with resultant core damage, without generation of a low low pressurizer level signal prior to onset of core damage. Additionally, all of the severe accidents generate a high containment pressure signal prior to onset of core damage, except for those involving a CVCS LOCA outside of containment.

Table 19.2-2 summarizes the status of mitigating systems for each of the core damage simulations. As shown in Tables 19.2-3 through 19.2-9, which give timelines for severe accident events, all events result in a containment isolation signal, either as a containment high pressure signal or low low pressurizer signal, well in advance of any core damage. In almost all cases there is more than one containment isolation signal present prior to core damage, providing diverse protection.

For design basis events, all events meet their thermal hydraulic and radiological release acceptance criteria without isolating CES on a high radiation signal, and none of the analyses credit nonsafety-related radiological instrumentation or holdup (such as the reactor building or gaseous radioactive waste system). None of these events result in degraded or damaged core conditions.

As discussed in FSAR Section 15.6, none of the analyzed DBE loss of RCS inventory events results in any fuel failure. Therefore, high radiation conditions inside containment, that would actuate a high radiation containment signal, would not occur for any of these events. Additionally, containment isolation due to generation of a low low pressurizer level signal occurs for all analyzed events, except for certain steam generator tube failure scenarios. However, as previously discussed, these scenarios do not involve any fuel failure and do not result in high radiation conditions inside of the containment.

Therefore, the underlying purpose of 10 CFR 50.34(f)(2)(xiv)(E) to ensure containment isolation for systems that provide paths to the environs is accomplished without reliance on features required by the rule.

13.3 Regulatory Basis

13.3.1 Criteria of 10 CFR 50.12

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 regulations are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The Nuclear Regulatory Commission has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption will not impact the consequences of any design basis event and will not create new accident precursors. The NuScale Power Plant design does not rely on a high radiation signal for CES containment isolation. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. Automatic isolation of the CES on a high radiation signal is not required to meet the underlying purpose of 10 CFR 50.34(f)(2)(xiv)(E) because alternate means to preclude a path to the environs are provided in the NuScale plant design. The CES isolates upon a low low pressurizer level signal, which occurs for any event leading to core damage or degradation. Additionally, the CES isolates upon a high containment pressure signal for any in-containment event leading to core damage or degradation. Therefore, the underlying purpose of the rule to ensure containment isolation for systems that provide paths to the environs is accomplished without reliance on the features required by the rule.

13.4 Conclusion

On the basis of the information presented above, NuScale requests that the NRC grant an exemption from 10 CFR 50.34(f)(2)(xiv)(E) for the NuScale design certification.

14. 10 CFR 50.46, Fuel Rod Cladding Material

14.1 Introduction and Request

14.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from the requirement of 10 CFR 50.46 regarding the use of zircaloy or ZIRLO as a fuel rod cladding material. The NuScale Power Plant fuel design uses AREVA's M5® alloy for the fuel rod cladding material. The underlying purpose of the requirement, which implies zircaloy or ZIRLO is to be used as a fuel rod cladding material, is to ensure that nuclear power facilities have adequately demonstrated the cooling performance of their emergency core cooling system (ECCS). The use of M5® cladding material is an acceptable alternative and, when evaluated pursuant to 10 CFR 50.46, meets the underlying purpose of the requirements.

14.1.2 Regulatory Requirements

10 CFR 52.47(a)(4) requires a design certification application to include, in part:

> ...Analysis and evaluation of emergency core cooling system (ECCS) cooling performance … following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 … of this chapter;

10 CFR 50.46(a)(1)(i) states in part:

> 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...

14.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale is requesting an exemption from the requirement of 10 CFR 50.46 regarding the use of zircaloy or ZIRLO as a fuel rod cladding material. An exemption is required because 10 CFR 50.46 does not anticipate the use of fuel rods with cladding materials other than zircaloy or ZIRLO.

10 CFR Part 50 Appendix K, Paragraph I.A.5, which also implicitly assumes that zircaloy or ZIRLO is to be used as the fuel rod cladding material, is subject to a separate exemption request (see DCA Part 7, Section 10).

14.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design as reflected in the Final Safety Analysis Report, will not conform with the fuel rod cladding materials implied in 10 CFR 50.46. The NuScale Power Plant design will include M5® fuel rod cladding material, but otherwise will meet the requirements of 10 CFR 50.46.

14.2 Justification for Exemption

The underlying purpose of 10 CFR 50.46 is to ensure that nuclear power facilities have adequately demonstrated the cooling performance of their ECCS by meeting defined acceptance criteria with an acceptable evaluation model. Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, BAW-10227P-A, Revision 1 (ML15162B047) demonstrates that the use of the acceptance criteria in 10 CFR 50.46 is acceptable to demonstrate adequate ECCS performance for reactors using M5® cladding material. Applicability of AREVA Fuel Methodology for the NuScale Design, TR-0116-20825-P-A, Revision 1, demonstrates the applicability of BAW-10227P-A, Revision 1 to the NuScale Power Plant fuel design. Therefore, the underlying purpose of 10 CFR 50.46 is met.

10 CFR 50 Appendix K, Paragraph I.A.5 requires that the Baker-Just equation be used in the ECCS evaluation model to determine the rate of energy release, hydrogen generation, and cladding oxidation. The Baker-Just equation specified in 10 CFR 50 Appendix K assumes the use of materials that have different chemical compositions than AREVA's M5® alloy. BAW-10227P-A, Revision 1 demonstrates that the Baker-Just equation is conservative relative to the oxidation performance of M5® cladding material; however, NuScale has separately requested an exemption from 10 CFR 50 Appendix K Paragraph I.A.5 because it is not applicable to NuScale's ECCS evaluation model. See DCA Part 7, Section 10.

14.2.1 Technical Basis

BAW-10227P-A, Revision 1, Section 4.2, demonstrates that the effectiveness of the ECCS will not be adversely affected by the use of M5® fuel rod cladding compared to zircaloy fuel rod cladding, and that ECCS acceptance criteria in 10 CFR 50.46 are appropriate for reactors using M5® fuel rod cladding to demonstrate acceptable ECCS performance. Therefore, NuScale's use of M5® fuel rod cladding material meets the underlying purpose of 10 CFR 50.46.

14.3 Regulatory Basis

14.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR 50 are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law [10 CFR 50.12(a)(1)]. This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The Nuclear Regulatory Commission has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety [10 CFR 50.12(a)(1)]. This exemption does not impact the consequences of any design basis event and does not create new accident precursors. The M5® fuel rod cladding has been evaluated in BAW-10227P-A, Revision 1, to confirm that operation with this cladding material does not increase the probability of occurrence or the consequences of an accident. The evaluation also concluded that no new or different type of accident will be created that could pose a risk to public health and safety. Therefore, the exemption does not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security [10 CFR 50.12(a)(1)]. The exemption does not affect the design, function, or operation of structures or plant equipment necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. The special nuclear material in this fuel product is handled and controlled in accordance with approved procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present [10 CFR 50.12(a)(2)(ii)] in that application of the regulation in the particular circumstances does not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of 10 CFR 50.46, which implies that Zircaloy or ZIRLO is to be used as the fuel rod cladding material, is to ensure that nuclear power facilities have adequately demonstrated the cooling performance of their emergency core cooling systems. This purpose is met by application of the acceptance criteria in 10 CFR 50.46 with the M5® fuel cladding material.

14.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption from 10 CFR 50.46 for the NuScale design certification to allow the use of M5® fuel rod cladding material.

15. 10 CFR 50, Appendix A, Criterion 27, Combined Reactivity Control Systems Capability

15.1 Introduction and Request

15.1.1 Summary

NuScale Power, LLC, (NuScale) requests an exemption from General Design Criterion (GDC) 27 to depart from its historical implementation as a requirement to achieve and maintain subcriticality following a postulated accident, using only safety-related equipment and with margin for stuck rods. Under postulated accident conditions, the NuScale Power Plant design protects public health and safety by reliably controlling reactivity using control rods to assure, with an assumed worst rod stuck out (WRSO), reactor core cooling is maintained. The NuScale Power Plant also relies on insertion of control rods to achieve and maintain reactor subcriticality to assure a safe, stable shutdown condition in the long term following a postulated accident. However, under the conditions that (1) the highest worth control rod assembly is assumed to not insert, (2) the chemical and volume control system (CVCS) is unavailable, and (3) during a small window of operational conditions when boron concentration is low and decay heat is low, a return to low power could occur following an initial period of subcriticality. During such a return to power, core cooling is maintained because the resultant power level is limited and the associated heat generated is within the capacity of the passive heat removal systems. Accordingly, the post-accident shutdown function is of reduced safety significance in the NuScale design.

Therefore, NuScale requests an exemption from GDC 27 to implement a design-specific Principal Design Criterion (PDC) 27. NuScale’s reactivity control design basis complies with the express requirements and original intent of GDC 27; however, this exemption request and PDC 27 respond to the shutdown requirement of GDC 27 derived from its historical implementation. NuScale's PDC 27 maintains the existing protection function capability of GDC 27, while clarifying that the shutdown function can be met with all control rods inserted, provided the specified acceptable fuel design limits (SAFDLs) for critical heat flux (CHF) would not be exceeded even if a return to power with an assumed stuck rod occurred. As a result of this exemption, the NuScale Power Plant design conforms to a principal design criterion for sufficient post-accident shutdown capability and reliability, rather than of GDC 27 as historically implemented.

15.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility

as a whole, and must include the following information:

. . .

(3) The design of the facility including:

(i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled 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. . .

The introduction to 10 CFR 50, Appendix A states, in part:

[T]here may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

10 CFR 50, Appendix A, GDC 27 states:

Criterion 27-Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

15.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from GDC 27 to the extent it has been implemented to require demonstration of long-term shutdown under post-accident conditions with an assumed worst rod stuck out.

15.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the Final Safety Analysis Report (FSAR), conforms to a principal design criterion requiring sufficient reactivity control capability and reliability to assure (1) core coolability under postulated accident conditions, with appropriate margin for stuck rods, and (2) post-accident shutdown with all control rods inserted, provided certain conditions are met. The NuScale Power Plant design bases relate to the modified design criterion, as stated in FSAR Section 3.1.

The first paragraph of NuScale PDC 27 requires the reactivity control systems to have a combined capability of reliably controlling reactivity changes to assure the capability to cool the core is maintained under postulated accident conditions, with appropriate margin for stuck rods. This provision mirrors GDC 27, except that the phrase "in conjunction with poison addition by the emergency core cooling system" has been deleted. The NuScale emergency core cooling system (ECCS) does not add poison. While GDC 27 does not require ECCS poison addition, this change within PDC 27 is made for clarity. NuScale implements the first paragraph of PDC 27 by demonstrating the capability of the reactivity control systems to perform their protection function; that is, to ensure that the capability to cool the core is maintained under accident condition, without regard to subcriticality.

The second paragraph of NuScale PDC 27 requires the control rods to have the capability of holding the reactor core subcritical under cold conditions, following a postulated accident. An equivalent provision is not explicit in GDC 27; rather, the additional provision states and modifies the implied requirement, derived from historical implementation of GDC 27, for the reactivity control systems to perform a shutdown function following a postulated accident. This provision does not require margin for stuck rod if certain conditions on safety are met, as explained below.

The shutdown provision of NuScale PDC 27 requires that, in order to demonstrate shutdown capability with all control rods inserted, a return to power (assuming a stuck rod) must not cause exceedance of the specified acceptable fuel design limits (SAFDLs) for critical heat flux (CHF). Meeting the specified acceptable fuel design limit (SAFDL) for minimum critical heat flux ratio (MCHFR) ensures adequate heat transfer from the fuel cladding to the reactor coolant and is a key parameter that is used to ensure that the fuel is protected in the NuScale design. MCHFR is the primary SAFDL used as the fuel-related acceptance criteria for Chapter 15 analyses. The other fuel-related SAFDLs used in Chapter 15, such as Fuel Centerline Melt and Fuel Enthalpy, are used primarily for LOCA and Rod Ejection. Using the MCHFR SAFDL is appropriate for overcooling return to power scenarios because it ensures no additional fuel failures following a postulated accident, and thereby prevents further radiological release. These return to power scenarios are limited to very low power levels where fuel centerline melt and fuel enthalpy are not limiting. Other fuel-related SAFDLs such as cladding strain or fuel rod internal pressure are analyzed throughout the operating cycle for normal operation and abnormal operating occurrences, but are not analyzed for infrequent events or accidents, and are not analyzed in Chapter 15.

NuScale PDC 27 is not intended to expand the applicability of GDC 27 beyond its current scope, but only to clarify the shutdown criterion for design basis events within the scope of GDC 27. Shutdown capability after reactivity accidents, specifically rod ejection, is not addressed by GDC 27 or GDC 28. Rather than addressing reactivity insertion for reactivity control or shutdown, GDC 28 is intended to restrict the amount of positive reactivity that can be inserted from reactivity accidents, including rod ejection, and thus limit the consequences of such events. Accordingly, GDC 28 remains the relevant criterion for evaluating reactivity control system design with respect to postulated reactivity accidents, as reflected in the review guidance of NUREG-0800.

15.2 Justification for Exemption

History and Intent of GDC 27

The stated requirement of GDC 27 is to "assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained." The regulatory history of the adoption of the General Design Criteria indicates that GDC 27 (and comparable provisions of GDC 26) was concerned with the protection function of the reactivity control systems that is the capability to protect the core under transient conditions. This protection function is distinct from the shutdown function also performed by the reactivity control systems. While the draft criteria explicitly addressed subcriticality and shutdown, the adopted GDC 27 only explicitly addresses fuel protection. Therefore, the original underlying intent of GDC 27 is consistent with its plain language—to assure core cooling under accident conditions, with reliability and margin requirements appropriate for the radiological hazards associated with that function.

While the underlying intent of GDC 27 was originally limited to fuel protection, subsequent implementation reflects a design and licensing basis for which "beyond the short term, all PWRs remain subcritical indefinitely, and the NRC has not licensed a power reactor that did not remain subcritical beyond the short term following an accident through the use of safety-related structures, systems and components (SSC)." (Letter from F. Akstulewicz, NRC, to T. Bergman, NuScale Power, Sept. 8, 2016, "Response to NuScale Gap Analysis Summary Report for Reactor Systems Reactivity Control Systems, Addressing Gap 11, General Design Criterion 27 (PROJ 0769).") This historical implementation is consistent with implementation of the related GDCs on decay heat removal (GDC 34) and emergency core cooling (GDC 35), for which the design basis heat removal capability has been tied to a shutdown reactor.

Based on the precedent, NRC Staff concluded that their "current view is that GDC 27 requires that the reactor be reliably controlled and that the reactor achieve and maintain a safe, stable condition, including subcriticality beyond the short term, using only safety-related equipment following a postulated accident with margin for stuck rods." Therefore, Staff stated if under postulated accident conditions, "the NuScale design does not ensure that the reactor would remain subcritical beyond the short term [using only safety-related equipment with margin for stuck rods], it would appear that NuScale would need to request an exemption from GDC 27, with respect to re-criticality."

NuScale Power Plant Design Basis and Safety with Respect to Intent of GDC 27

The NuScale Power Module (NPM) design basis assures fuel protection under postulated accident conditions assuming WRSO and under conservative safety analysis methodology, but relies on all control rods inserted to assure long-term subcriticality under normal and accident conditions. This NuScale reactivity control design basis reflects the importance to safety of the protection and shutdown functions considering unique design characteristics and the safety and risk profile of the overall plant design. The NuScale design relies on fully passive safety systems, which will reduce the reactor coolant system temperature to low temperatures.

The NuScale reactivity control capability is sufficiently safe because, under postulated accident conditions, the reactivity control systems assure the capability to cool the core is maintained, thereby assuring applicable dose criteria are met. However, following a postulated accident, maintaining subcriticality is not necessary to support safety-related system functions (e.g., core heat removal). Even under a return to power the passive NuScale safety systems would continue to adequately remove reactor heat to ensure fuel design limits are not further challenged. Therefore, a postulated return to power would not present a radiological risk for the NuScale design.

The NuScale reactivity control shutdown capability is sufficiently reliable because the probability for a return to power is insignificant. The safety-related control rods, alone, can hold the reactor subcritical to a conservative minimum reactor coolant system temperature. The probability for a stuck control rod and a failure on demand of boron addition by the CVCS system is less than 1E-5 per reactor year. A bounding probability for a return to power is calculated to be less than 1E-6 per reactor year, accounting for the reliability of reactivity control systems and the likelihood that the reactor is in a state that can subsequently lead to a return to power.

Therefore, NuScale's design basis meets the original underlying intent of GDC 27-that is to assure core cooling and thus radiological safety under postulated accident conditions. Pursuant to the historical implementation of GDC 27, the NuScale design basis further assures a sufficiently reliable reactivity control shutdown function, in a manner consistent with the importance to safety of that function in the NuScale design.

As a result of this exemption, NuScale's PDC 27 will reflect the NuScale design basis for the reactivity control protection and shutdown functions. PDC 27 will explicitly allow the reliance on all control rods inserted to assure shutdown following a postulated accident, provided certain conditions are met. Due to the unique design features of the NuScale design, under limiting analytical assumptions this approach is necessary for the design to exclusively rely on passive shutdown and decay heat removal systems, rather than a more complex system (such as boron addition).

The NRC, through the Policy Statement on the Regulation of Advanced Reactors, has recognized the potential safety benefits of simplified, passive safety features. NRC "expects that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions." Specifically, NRC "encouraged" the "use of inherent or passive means" to provide "highly reliable and less complex shutdown and decay heat removal" functions. NRC stated that such an attribute "could assist in establishing the acceptability or licensability" of the design.

The NuScale Power Plant design relies only on safety related control rods to ensure subcriticality and passive heat removal systems to remove decay heat. No operator actions are required to perform either function. In the unlikely event of a return to power, the reactor inherently limits the power to a level that can be safely removed by the passive, safety-related heat removal systems, and active boron injection is available as an additional method of controlling reactivity. Accordingly, the exemption is consistent with NRC's recognition of enhanced safety afforded by simplified and passive safety systems to provide shutdown and decay heat removal. Chapter 15 and 19 of the NuScale FSAR demonstrate enhanced margins of safety for the design, in addition to the simplified, inherent, and passive means for reactivity control and decay heat removal.

15.2.1 Technical Basis

The reactivity control capability of the NuScale design meets the underlying intent of GDC 27 by (1) complying with the explicit fuel protection requirement (ensure capability to cool the core), and (2) fulfilling the intent of the precedential shutdown requirement by providing sufficiently safe and sufficiently reliable shutdown capability to provide reasonable assurance of adequate protection of public health and safety.

Reactivity Control Fuel Protection Capability

Under the NuScale design basis, during normal operation sufficient negative reactivity is maintained (instantaneous shutdown margin) to ensure that the capability to cool the core is maintained under accident conditions, by rapid control rod insertion with an assumed WRSO. Chapter 15 of the NuScale FSAR provides a safety analysis to demonstrate that the NuScale SSCs perform their design basis functions under analyzed events using conservative methodology and assumptions. The safety analysis demonstrates the protection function of NuScale's PDC 27 (equivalent to the original requirement of GDC 27) is met by providing sufficient reactivity control such that core cooling is maintained under accident conditions.

Maintaining a shutdown condition in the long term is not necessary to meet the protection function requirement, as long as the capability to cool the core is maintained. Using conservative and bounding assumptions, including WRSO, NuScale's Chapter 15 safety analysis of a postulated return to power conditions demonstrates that the passive heat removal safety systems adequately remove reactor heat to ensure the CHF SAFDL is not exceeded by the return to power. Therefore, the capability to cool the core is maintained.

Description of Reactivity Control Shutdown (Long Term) Capability

Subsequent to a reactor trip, control rods with all control rods inserted provide sufficient negative reactivity to shut down the reactor and maintain it in a shutdown condition indefinitely. In the event of a control rod that failed to insert, a potential return to power can occur subsequent to a reactor trip only under all of the following conditions:

- the trip occurred late in the fuel cycle,
- the highest worth control rod is assumed stuck out,
- decay heat levels are very low, and
- a nonsafety-related means for boration is not available.

A return to power will be prevented early in cycle for the equilibrium core design due to a favorable moderator temperature coefficient. A return to power is also prevented late in cycle when high decay heat levels prevail; with sufficient decay heat present, negative reactivity from voiding will prevent a return to power. With the highest worth control rod assembly stuck out and the chemical and volume control system unavailable, shutdown conditions are maintained for more than 30 days after a design basis event, except for a small window of initial conditions when boron concentration is low and decay heat is low. For a normal fuel cycle, high decay heat levels at the end of cycle will prevent a return to power for 30 days after a reactor trip.

While a return to power does not challenge core cooling, in the event of a control rod that failed to insert, soluble boron can be added to the reactor coolant system using the nonsafety-related CVCS, if needed. Soluble boron can also be added to the containment using the nonsafety-related containment flood and drain system (CFDS), where it would subsequently enter the RCS through natural circulation when the ECCS reactor vent valves (RVVs) and reactor recirculation valves (RRVs) are opened.

Basis for Shutdown Capability that Is Sufficiently Safe

The safety-related control rods, alone, are capable of maintaining the core subcritical under cold conditions, without margin for stuck rods. This shutdown capability for the NuScale design is sufficiently safe because of the reduced radiological safety significance of shutdown in the NuScale design, which ensures fuel is protected and heat is removed through passive means, even in the unlikely event of a return to low power. In the event of a return to power, fuel is protected through inherent physical processes that control reactivity, limit power, and cool the core.

The inherent means for limiting the power is dependent on the heat removal system used. For design basis events that rely on heat removal using natural circulation flow through the RVVs and RRVs, the heat produced from a return to power with a nominal value for shutdown margin will be limited to less than 100 kW (0.06 percent of rated power) by negative reactivity feedback from moderator density. The low core heat level increases moderator temperature and generates voiding in the core, which, in combination with the elevated RCS temperature due to the heatup of the reactor pool from the residual heat of multiple reactors being shut down upon a loss of all AC power, provides negative feedback to keep core power very low. If decay heat exceeds 100 kW, the reactor will be maintained with $k_{eff}<1$ even with a worst rod stuck out because of negative density reactivity. Therefore, the heat produced after a return to power with natural circulation flow through the RVVs and RRVs is bounded by maximum decay heat. Consequently, the maximum decay heat curve, with the reactor at $k_{eff}<1$, is used to evaluate maintaining fuel integrity and ECCS performance. Maintaining fuel integrity is addressed in FSAR Sections 15.6.5 and 15.6.6 and ECCS performance is addressed in FSAR Section 6.3.

For design basis events that rely on heat removal using the DHRS, heat produced from a return to power will be limited by negative moderator temperature feedback. A return to power for a generic cooldown transient with DHRS is evaluated in Chapter 15 of the FSAR to demonstrate that fuel cladding integrity is maintained (i.e. the CHF SAFDL is not exceeded). The time to a return to power and the peak power level attained in the evaluation is based on conservative assumptions for the purpose of demonstrating fuel protection and is not an indication of realistic shutdown capability. Without AC power available, all events that rely on DHRS for heat removal will transition after 24 hours to heat removal using natural circulation flow through the RVVs and RRVs, as addressed above. With a nominal value for shutdown margin, a return to power does not occur for at least 24 hours with the worst control rod stuck out.

Basis for Shutdown Capability that Is Sufficiently Reliable

The shutdown capability for the NuScale design is sufficiently reliable because the probability for a return to power is insignificant. A bounding probability for a return to power is calculated to be less than 1E-6 per reactor year. Calculation of the bounding probability takes into account the reliability of reactivity control systems and the likelihood that the reactor is in a state that can subsequently lead to a return to power. Using a trip frequency of one trip per reactor year, the contributions to a bounding probability for a return to power are as follows:

- The probability that one out of 16 control rods fails to insert is 2E-4 per demand (1.3E-5 per demand per control rod x 16). This probability is bounding as only some of the control rods with a higher rod worth would lead to a return to power when stuck out. Further, industry experience with stuck control rods involved control rods that did not fully insert rather than stuck in the fully withdrawn position. The contribution of external initiating events to the reliability of control rod insertion is negligible because control rods are designed to insert under conditions during and after external initiating events.

- The probability of a CVCS failure to insert soluble boron is 8E-3 per demand. This probability is a bounding probability for a failure to insert soluble boron as it does not take into account alternative means for increasing soluble boron using the CFDS. Using the CFDS, soluble boron can be added to the containment and subsequently to the RCS through natural circulation when the reactor vent valves (RVVs) and reactor recirculation valves (RRVs) are opened. The contribution of external initiating events to the unavailability of these systems is small because of the low probability of external initiating events.
- The probability that the reactor is in a state which could result in a return to power with a WRSO (based on time in cycle and time at power) is 4E-2 to 1E-1 per year. This probability is bounding as it assumes that a return to power can occur shortly after a restart at any time during the fuel cycle, if limited decay heat is produced due to limited time at power. In reality, a return to power will be prevented early in cycle for the equilibrium core design due to a favorable moderator temperature coefficient. A return to power is also prevented late in cycle when high decay heat levels prevail; with sufficient decay heat present, negative reactivity from voiding will prevent a return to power.

Therefore, a bounding probability for a return to power is calculated to be between 7E-8 to 2E-7 per reactor year.

The NRC and the nuclear industry have not established a goal for acceptable shutdown reliability. However, in the following instances, NRC has considered whether the probability for an inadvertent return to power is acceptable, and whether the risk from an anticipated transient without scram (ATWS) is acceptable:

- Generic Safety Issue (GSI) 22, Inadvertent Boron Dilution Events. The probability for an inadvertent return to power due to boron dilution during a shutdown or refueling was calculated to have a probability 2E-4 per reactor year. A 3E-5 per reactor year probability was calculated for offsite radiological release of gap activity from any leak already present in the fuel. Given the low probability and low consequences from such an event, GSI-22 was closed with no new requirements and without implementing potential plant changes.
- NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States. The potential for core damage due to rapid boron dilution may occur with a frequency on the order of 1E-5 per reactor year (this assessment was performed using conservative system parameters without which core damage would be avoided). No further regulatory requirements were developed to reduce the potential for rapid boron dilution.
- GSI-185, Control of Recriticality Following Small-Break LOCAs in PWRs. The probability for core damage due to inadvertent boron dilution during a small-break LOCA transient

was calculated to be 3E-8 for B&W designs. Due to the low risk associated with this issue GSI-185 was closed with no changes to existing regulations or guidance.

- 10 CFR 50.62 ATWS. The intent of the ATWS rule is to reduce the CDF contribution from ATWS to less than 1E-5 per reactor year (U.S. NRC, SECY 83-293, "Amendments To 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," July 19, 1983). A failure to scram after an anticipated transient, which was calculated to occur between 1E-4 to 1E-5 per reactor year for operating LWRs, is considered a beyond design basis event. The potential for ATWS events are, however, evaluated in order to ensure that the ATWS CDF contribution is acceptably low.

The bounding probability for a return to low power for the NuScale design is significantly smaller than what was considered for GSI-22 and NUREG-1449. Furthermore, the bounding probability for a return to low power for the NuScale design is significantly smaller than the threshold of 1E-4 that has been proposed for event classification of beyond design basis events of previous advanced designs, and the 1E-5 per reactor year goal for ATWS CDF contribution.

Based on these precedent examples, the CDF of 1E-4 per reactor year constitutes a sufficiently small probability of return to power to provide reasonable assurance of shutdown. This probability aligns with NRC's safety goal of 1E-4 total CDF, although no fuel damage and thus no radiological consequences are predicted due to a return to low power in the NuScale reactor.

The bounding probability for a return to low power for the NuScale design is significantly smaller than 1E-4 per reactor year. Therefore, the NuScale shutdown capability is sufficiently reliable to provide reasonable assurance of adequate protection of public health and safety.

15.2.2 Risk Assessment

As discussed above, a bounding probability for a return to power is calculated to be less than 1E-6 per reactor year. Further, there is no fuel damage or offsite release predicted to occur as a result of a postulated return to power in the NuScale design. Accordingly, the requested exemption does not result in an increase in CDF or risk.

Therefore, design or operational requirements to further reduce the low probability or reduce the consequences of a return to power are not warranted for the NuScale design basis. A further implication of the low probability of return to power is that nonsafety-related systems do not need to be designed for external events in order to ensure a sufficiently low probability for a return to power.

15.3 Regulatory Basis

15.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR Part 50 regulations are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption will not impact the consequences of any design basis event and will not create new accident precursors. The NuScale plant incorporates reactivity control provisions to assure the capability to cool the core is maintained under postulated accident conditions, and to reliably and safely shutdown the reactor. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that is necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of GDC 27, as originally promulgated, was to limit the consequences of postulated accidents by assuring the capability to cool the core, with appropriate margin for stuck rods. As a result of this exemption, NuScale's design basis will meet the original underlying purpose of GDC 27. Subsequent implementation of GDC 27 has been interpreted to require long-term shutdown following the postulated accident, with the same stuck rod assumption. The underlying purpose of the shutdown requirement is to ensure a safe, stable reactor in the long-term following an accident. Application of the stuck rod assumption as part of the NuScale design basis for shutdown is not necessary to meet this purpose, because a stuck rod leading to return to power is unlikely, and, if it did occur, the return to power would be inherently limited and adequate core cooling would be maintained by passive safety systems.

Special circumstances are present (10 CFR 50.12(a)(2)(iv)) in that the exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption. Application of GDC 27, as interpreted by the NRC, would require NuScale to incorporate additional features (such as a safety-related boron injection system), increasing complexity of the shutdown systems. The NRC's Policy Statement on the Regulation of Advanced Reactors recognized simplified, passive safety features, including highly reliable and less complex shutdown and decay heat removal systems as a benefit to the public health and safety. NuScale's shutdown function is highly reliable, as demonstrated by the probability of a return to power. More importantly, a return to power would not result in fuel damage, and thus this exemption would not result in an increase in CDF or risk. Therefore, there is no identified decrease in safety as a result of this exemption.

15.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification to allow the design to address a principal design criterion that differs from GDC 27. PDC 27 maintains the existing protection function capability of GDC 27, while clarifying that the shutdown function can be met with all control rods in, provided the SAFDLs for CHF would not be exceeded even if a return to power occurred.

16. 10 CFR 50.34(f)(2)(viii) Post-Accident Sampling

16.1 Introduction and Request

16.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from 10 CFR 50.34(f)(2)(viii), requiring capability for post-accident sampling of the reactor coolant system and containment. The rule requires the capability to obtain and analyze samples without exceeding prescribed radiation dose limits to any individual. The underlying purpose of the rule is to ensure the capability to assess the presence and extent of core damage. The NuScale Power Plant design meets the underlying purpose of the rule by ensuring the capability to assess the presence and extent of core damage during an accident by other means, which benefits public health and safety by avoiding unnecessary operator dose, preventing the spread of contamination, and reducing the potential for radioactive leaks and spills.

16.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information
>
> . . .
>
> (8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v);

10 CFR 50.34(f)(2) states, in part;

> (f) Additional TMI-related requirements.
>
> . . .
>
> (2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operation licensing stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.
>
> . . .
>
> (viii) Provide the capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials[11] without radiation exposures to any individual

> exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

10 CFR 50.34, Footnote 11, states;

> The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

16.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the post-accident sampling requirements of 10 CFR 50.34(f)(2)(viii).

16.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the Final Safety Analysis Report (FSAR), will not conform with the provisions of 10 CFR 50.34(f)(2)(viii). The NuScale design will retain capabilities allowing a licensee to sample the reactor coolant system and containment.

16.2 Justification for Exemption

16.2.1 Purpose and History of Requirement

The underlying purpose of 10 CFR 50.34(f)(2)(viii) is to ensure the capability for plant operators to assess the presence and extent of core damage following an accident. As stated in NUREG-0578, Section 2.1.8:

> The NRC staff and the ACRS have for some years emphasized the need for special features and instruments to aid in accident diagnosis and control. Although some degree of capability of this type was available at TMI-2, and exists on other plants, the TMI-2 experience shows that more is needed. The Offices of Standards Development and Nuclear Reactor Regulation have agreed to expedite revision of Regulatory Guide 1.97, which deals with this subject area… In the meantime, the following provisions are recommended for early implementation on all plants to provide a uniform, minimum capability in this area. Recommendations: a. Improved Post-Accident Sampling Capability Review and upgrade the capability to obtain samples from the reactor coolant system and containment atmosphere under high radioactivity conditions. Provide the capability for chemical and spectrum analysis of high-level samples on site.

Thus, improved post-accident sampling capability was specified as an interim measure to assist in accident diagnosis and control, while the NRC developed revised requirements for instrumentation capabilities that would address a similar need. While operating plants had some sampling capability, the Task Force determined additional capabilities were required, and that TMI-2 accident conditions challenged the ability to perform sampling. As stated in NUREG-0578, Section 2.1.8.a:

> Chemical and radiological analysis of reactor coolant liquid and gas samples can provide substantial information regarding core damage and coolant characteristics. . . .
>
> Timely information from reactor coolant and containment air samples can be important to reactor operators for their assessment of system conditions and can influence subsequent actions to maintain the facility in a safe condition. Following an accident, significant amounts of fission products may be present in the reactor coolant and containment air, creating abnormally high radiation levels throughout the facility. These high radiation levels may delay the obtaining of information from samples because people taking and analyzing the samples would be exposed to high levels of radiation. . . .
>
> Prompt acquisition and spectrum analysis of reactor coolant samples within several hours after the initial scram would have indicated that significant core damage had occurred; perhaps with such information, earlier remedial actions could have been taken. Similarly, analysis of an early containment air sample would have indicated the presence of hydrogen, significant core damage, and the possibility of a hydrogen explosion in the containment.

In sum, the Task Force found that analysis of samples could provide substantial and important information to operators, which could assist in managing the accident and in informing emergency response efforts. The Task Force determined that effective radiation protection measures were necessary to ensure that such sampling capability could be effectively used when needed, such that operators could take timely actions to manage the event.

In the years since the TMI accident, significant improvements and a considerable amount of knowledge and industry experience have been realized in the areas of understanding risks associated with plant operations and developing better strategies for managing severe accident response. Insights about plant risks and alternate severe accident assessment tools have reduced the necessity of these post-accident sampling requirements. In certain instances, the use of a post-accident sampling system (PASS) can degrade the plant emergency response by diverting resources to non-essential activities and create a radiation release pathway.

In 1993, during its review of licensing issues pertaining to evolutionary and advanced light water reactors (ALWRs), the NRC staff evaluated requirements for PASS specified in 10 CFR 50.34(f)(2)(viii) in developing SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-water Reactor Designs." In the SRM for SECY-93-087 (ML003708056), the Commission approved the staff's positions, with modification, relaxing some of the requirements for post-accident sampling as implemented under item II.B.3 of NUREG-0737.

In the late 1990s, Combustion Engineering Owners Group (CEOG), Westinghouse Owners Group (WOG), and the BWR Owners Group (BWROG) submitted topical reports for NRC staff's review to eliminate PASS requirements. As stated by the CEOG in CE NPSD-1157, Revision 1 (ML003699802):

> [I]ncreased knowledge of accident phenomenology and the considerable amount of operating experience that have been gained in the years since NUREG-0737 was issued have led to a better understanding of degraded core behavior and the role that a PASS would play in various accident scenarios. This better understanding supports the conclusion that PASS does not play a significant role in controlling the plant emergency management response to severe accidents. In certain instances, use of PASS can even degrade the plant emergency response by diverting limited resources to non-essential activities and/or creating a radiation release pathway into the auxiliary building. It has also been determined that the role of the PASS in emergency planning is minimal and primarily confirmatory.

The staff concluded that the topical reports provided adequate basis to eliminate the PASS as a required system for post-accident sampling. As discussed in the safety evaluation for CE NPSD-1157, Revision 1 (ML003734524), the NRC based its decision "on the acceptability of the proposal to eliminate PASS on the benefit that the information obtained from PASS would provide in accident management and emergency response. If this information was considered to be necessary, and therefore, planned to be obtained shortly after a severe accident, then a PASS would be prudent to ensure that samples could be taken promptly and exposure minimized. However … the information is not considered to be beneficial for accident management or emergency response. Therefore, there is considered to be sufficient time to establish an alternate sampling capability if samples were considered to be beneficial in the longer term."

As addressed below, the information that could be obtained from post-accident sampling is not necessary for accident management and emergency response, because the NuScale design provides for sufficient information through other means.

16.2.2 Technical Basis

The underlying purpose of 10 CFR 50.34(f)(2)(viii) is to ensure the capability for plant operators to assess the presence and extent of core damage following an accident. In the NuScale design, this capability is provided by radiation monitors under the bioshield and by core exit thermocouples. The NuScale design is capable of classifying a fuel damage event at the alert level threshold utilizing the radiation monitors under the bioshield and the core exit thermocouples.

The NuScale design philosophy for major accident scenarios, including core damage events, is to isolate the NuScale power module (NPM) to preserve the primary coolant inventory and contain the potential post-accident source term. The process of taking a sample from the primary coolant or containment would necessarily require the transfer of potentially radioactive post-accident material from inside containment to the outside of containment. In lieu of such a process, the NuScale design relies upon other means to indicate the presence of core damage, namely radiation monitors under the bioshield and core exit thermocouples. It is preferable to maintain the primary coolant inventory and potential accident source term radioactive materials inside the NPM if the necessary information can be obtained otherwise. This design philosophy will result in a lower potential for facility contamination and personnel radiation exposure.

Primary Coolant Dissolved Gases (Including Hydrogen):

Based on the design of NuScale reactor module being insusceptible to an accumulation of non-condensable gases interfering with post-accident natural circulation, there is little benefit to requiring grab samples of post-accident reactor coolant for dissolved gas analysis for the sake of ensuring post-accident natural circulation. Therefore, the post-accident sampling of reactor coolant for dissolved gases is not required in the NuScale design.

Containment Hydrogen and Oxygen:

The NuScale design has features that support the capability to continuously monitor hydrogen and oxygen concentrations in the containment atmosphere using process sampling system (PSS) in-line monitors during accident conditions. This exemption does not impact the ability of a licensee to establish combustible gas monitoring following a design basis or beyond design basis event as described in TR-0716-50424, if needed.

Primary Coolant Chlorides:

A high concentration of chlorides in the reactor coolant can cause stress corrosion cracking of stainless steel components in contact with the coolant. The purpose of sampling the post-accident reactor coolant for chlorides is to ensure that chloride-induced SCC of stainless steel components will not occur in the long term. Chlorides can be introduced into the RCS by the incoming water from external source containing chloride. The NuScale design does not employ automatic safety injection or coolant makeup, so the chloride concentration will remain unchanged. Furthermore, the NuScale design does not utilize large quantities of chlorinated cable insulation inside containment; therefore, any containment water being circulated back into the reactor core during ECCS recirculation in accident scenario will not have high chloride concentration. Therefore, post-accident reactor coolant sampling for chloride is not necessary the NuScale plant design.

Primary Coolant Boron Concentration:

The purpose of sampling the reactor coolant for boron is to ensure that there is adequate shutdown margin in the RCS to enable safe shutdown to be achieved. The capability to ascertain the RCS boron concentration is an important long term issue when water, other than the original reactor coolant inventory, will be used to refill the reactor vessel or to flood the containment. Because there is no automatic coolant makeup or safety injection capabilities, the total boron concentration in the primary coolant will remain unchanged.

Therefore, post-accident boron samples are not necessary for the NuScale design.

Primary Coolant and Containment Radionuclide Concentration:

The purpose of sampling the post-accident reactor coolant for radionuclide content is to verify that the integrity of the fuel rod cladding has not been breached during an accident. The capability to measure reactor coolant radionuclides also supports the Emergency Action Level (EAL) classification in the Site Emergency Plan.

The NuScale design utilizes radiation monitors under the bioshield and core exit thermocouples to assess core damage. Therefore, post-accident sampling for radionuclide content is not necessary for the NuScale design

16.3 Regulatory Basis

16.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR Part 50 regulations are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any DBE, and does not create new accident precursors. Therefore, the exemption does not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). This exemption does not affect the design, function, or operation of structures or plant equipment necessary to maintain the secure status of the plant. The exemption has no impact on plant security or safeguards procedures. Therefore, the exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)). The underlying purpose of 10 CFR 50.34(f)(2)(viii) is to ensure the capability for plant operators to assess the presence and extent of core damage following an accident. The NuScale design provides for core damage assessment through the use of core exit thermocouples and radiation monitors under the bioshield, as described in FSAR Section 9.3.2.

Special circumstances are present (10 CFR 50.12(a)(2)(iv)) in that the requested exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption. In lieu of obtaining post-accident samples, operators will rely on other means to assess the presence and extent of core damage following an accident. By maintaining containment isolation as the preferred accident response, the spread of potentially highly radioactive material to systems outside of the NuScale power module will be prevented, avoiding unnecessary operator dose, preventing the spread of contamination to systems outside of the NuScale power module, and reducing the potential for leaks and spills that could result in additional dose to the public. Consistent with past determinations by licensees and NRC, there is negligible safety benefit to maintaining post-accident sampling capabilities required by

10 CFR 50.34(f)(2)(viii) where alternative instrumentation can provide the necessary information to inform operators for accident management and emergency response. Therefore, there is no identified decrease in safety as a result of this exemption.

16.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the requirements of 10 CFR 50.34(f)(2)(viii).

17. 10 CFR 50, Appendix A, Criterion 19, Control Room

17.1 Introduction and Request

17.1.1 Summary

NuScale Power, LLC, (NuScale) requests an exemption from General Design Criterion (GDC) 19 to depart from the portion of the rule requiring equipment outside the control room with a potential capability for subsequent cold shutdown of the reactor when the control room is evacuated.

The underlying intent of the remote shutdown portion of GDC 19 is to provide means for operators to place and maintain the reactor in a safe condition in the event of a control room evacuation. NuScale is requesting an exemption to clarify the GDC requirements as they apply to the NuScale design and substitute the term "safe shutdown" for "cold shutdown." Cold shutdown (i.e., the reactor coolant system at atmospheric pressure and < 200°F following a reactor cooldown) is not necessary to ensure a long-term safe condition following control room evacuation for NuScale's passive advanced light water reactor design. Cold shutdown is not a defined Mode under NuScale's Technical Specification definitions.

In the event of a main control room (MCR) evacuation, all reactors are tripped and decay heat removal and containment isolation are initiated prior to operators evacuating the MCR. These actions result in passive cooling that achieves and maintains safe shutdown (i.e., Mode 3 where $k_{eff} < 0.99$ and all NPM reactor coolant temperatures are < 420°F). Operators can also place the reactors in safe shutdown from outside the MCR in the module protection system (MPS) equipment rooms within the reactor building.

Following shutdown and initiation of passive cooling, the NuScale design does not rely on operator action outside of the MCR to maintain a safe stable shutdown condition, and control room signals are isolated preventing unintended signals from impacting unit conditions. In addition, no instrumentation or controls are necessary outside the MCR to maintain the NuScale Power Modules (NPMs) in a safe shutdown condition.

The design includes a remote shutdown station (RSS) to monitor plant conditions; however, there are no displays, alarms, or controls in the RSS necessary to achieve or maintain safe shutdown of the NPMs. If the MCR is evacuated, the RSS serves as a central location for the operators to monitor the modules in a safe shutdown condition with DHRS in service for each module. Additionally, the RSS provides defense-in-depth capability to monitor the plant remotely from the MCR and control balance of plant equipment to support asset protection and long-term plant recovery in events where the MCR becomes uninhabitable.

Therefore, NuScale requests an exemption from GDC 19 to implement a design-specific Principal Design Criterion (PDC) 19 that meets the underlying purpose of GDC 19's requirement for means to maintain the reactor in a safe condition in the event of a control room evacuation. Additional changes to PDC 19 are incorporated to improve clarity of the design criterion. As a result of this exemption, the NuScale Power Plant design conforms to a principal design criterion for safe shutdown capability in the event of MCR evacuation rather than a requirement for cold shutdown.

17.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:
>
> . . .
>
> (3) The design of the facility including:
>
> (i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), 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. . .

The introduction to 10 CFR 50, Appendix A states, in part:

> [T]here may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

10 CFR 50, Appendix A, GDC 19 states:

> *Criterion 19 - Control room*. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
>
> Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997,

applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

17.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from GDC 19, which requires equipment outside the control room providing a potential capability for cold shutdown of the reactor through the use of suitable procedures.

17.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the Final Safety Analysis Report (FSAR), conforms to a Principal Design Criterion (PDC) requiring the design capability for safe shutdown from equipment outside the control room, in lieu of the requirements for "design capability for prompt hot shutdown" and "potential capability for subsequent cold shutdown" as specified in GDC 19. PDC 19 also clarifies the requirements for control room radiation protection consistent with the current rule. NuScale's Principal Design Criterion is stated in FSAR Section 3.1, and reflected below:

Criterion 19 - Control room.

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.

Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided with a design capability for safe shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe shutdown condition.

17.2 Justification for Exemption

The principal requirement of GDC 19 is to provide a control room from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents, while providing adequate radiation protection for personnel. A secondary requirement is that facilities have equipment at appropriate locations outside the control room " (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures."

The underlying intent of the remote shutdown portion of GDC 19 is to provide means for operators to maintain the reactor in a safe condition in the event of a control room evacuation. Proposed GDC 11, later finalized as GDC 19, would have required it to "be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause" (32 Federal Register 10216, emphasis added). During the public comment period, industry was concerned that the requirements of GDC 11 could be interpreted to require a second control room (SECY-R 143, Jan. 28, 1971). In response, NRC clarified the requirements in final GDC 19 to spell out separately the "design capability" for prompt hot shutdown and "potential capability for subsequent cold shutdown" in the longer term. There is no indication in the rulemaking record that the NRC intended to alter the intent of proposed GDC 11 with respect to maintaining a "safe condition." NRC guidance is consistent with the interpretation that the "cold shutdown" language was not directed at reactor coolant system temperature, but rather a safe stable shutdown condition. For example, in summarizing the remote shutdown capability requirements in Information Notice 91-53, NRC stated that conditions that could preclude control room accessibility "warrant the use of a remote shutdown system to achieve safe shutdown of the plant."

Therefore, NuScale concludes that the cold shutdown provision was not intended to be stricter than the originally proposed "safe condition," but rather to allow facilities to rely on "potential capabilities" outside the control room for operators to establish a long term safe condition. Accordingly, the ability to maintain the reactor in a safe shutdown condition in the event of control room evacuation satisfies the underlying purpose of GDC 19. As discussed below, NuScale's proposed PDC 19, replacing "cold shutdown" with "safe shutdown" of the reactor, meets the underlying purpose of the design criterion for NuScale's passive advanced light water reactor design.

17.2.1 Technical Basis

The underlying intent of the remote shutdown portion of GDC 19 is to provide means for operators to maintain the reactor in a safe condition in the event of a control room evacuation. For NuScale's passive advanced light water reactor design, the establishment of PDC 19 to require remote "safe shutdown" capability instead of "cold shutdown" is supported and consistent with NRC guidance, such as SECY-94-084, Policy and Technical issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, which applies to passive residual heat removal systems and RG 1.189, Fire Protection for Nuclear Power Plants regarding fire in the main control room.

As stated in Design-Specific Review Standard for NuScale SMR Design (DSRS) Section 7.0, DSRS Chapter 7 Acceptance Criteria and Review Process Item 3, "Level of Review Applied to I&C Systems":

> *iii. Safe shutdown systems function to achieve and maintain a safe shutdown condition of the plant. The safe shutdown systems include I&C systems used to maintain the reactor core in a subcritical condition and provide adequate core cooling to achieve and maintain both hot and cold shutdown conditions, as defined in SECY 95-132 "Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems in Passive Plant Designs (SECY 94-084)."*

> The term safe shutdown is footnoted with:

The NRC considers a "safe stable shutdown condition" for advanced passive LWRs to be a condition by which all plant conditions are stable and within regulatory limits and the reactor coolant system pressure is stabilized and reactor coolant temperature is less than or equal to 215 degrees Celsius (C) (420 degrees Fahrenheit (F)).

As stated in NuScale Power DCA Part 4, Generic Technical Specifications, passive core cooling provided by the Decay Heat Removal System, Emergency Core Cooling System, or an appropriate water level in the containment can be used to remove core decay heat. The use of PASSIVE COOLING systems allows extended operation with no operator action required in MODE 3 once initiated. (i.e., Safe Shutdown where $k_{eff} < 0.99$ and all NPM reactor coolant temperatures are < 420°F).

The NuScale design addresses GDC 19's intent with respect to control room evacuation in two ways. First, the NuScale main control room (MCR) is designed specifically with the ability to place and maintain the reactors in safe shutdown in the event of an MCR evacuation event. Prior to evacuation of the MCR, operators trip the reactors, initiate decay heat removal and initiate containment isolation. These actions result in passive cooling that achieves safe shutdown of the reactors. Operators can also achieve safe shutdown of the reactors from outside the MCR in the module protection system (MPS) equipment rooms within the reactor building. Following shutdown and initiation of passive cooling, the NuScale design does not rely on operator action, instrumentation, or controls outside of the MCR to maintain safe shutdown, and control room signals are isolated preventing unintended signals from impacting unit conditions. The design includes a remote shutdown station (RSS) to monitor conditions; however, there are no displays, alarms, or controls in the RSS necessary to achieve or maintain safe shutdown of the reactors, as there is no manual control of safety-related equipment allowed from the RSS.

In the MCR, each NuScale Power Module (NPM) is provided two redundant, safety-related, hard-wired switches in separate and independent divisions in order to trip the reactor and establish the conditions required to passively achieve safe shutdown. NuScale credits operation of these switches consistent with the guidance in RG 1.189, Section 5.4.4, Control Room Fires. Additionally, the operation of the MCR switches described above is verified by ITAAC 02.05.01 and ITAAC 02.05.13. In accordance with GDC 26 and NuScale PDC 27, the NPM control rods, alone, can hold the reactor subcritical to a conservative minimum reactor coolant system temperature.

Alternate means outside of the MCR are provided to shut down each reactor and establish the conditions required to passively achieve safe shutdown. For each NPM, two separate MPS equipment rooms are provided. Within each MPS equipment room, safety-related components can be physically manipulated to trip its respective reactor and passively achieve safe shutdown.

Following shutdown from the MCR shutdown switches or MPS equipment rooms, the RSS only provides indication to monitor unit conditions; control room signals are isolated preventing unintended signals from impacting the unit conditions. There is no manual control of safety-related equipment allowed from the RSS. The module control system (MCS) equipment located in the RSS is similar to the MCS equipment located in the MCR, but configured in a way that allows monitoring of plant systems. Although no subsequent operator actions are required, the RSS has several video display units which can be used to monitor plant conditions. The video display units are comparable to those provided in the control room and the operator can display information on the video display units in a manner which is comparable to the way the information is displayed in the control room. The operator normally selects an appropriate set of displays based on the particular operational goals being monitored by the operator at the time. The operability of the remote shutdown display functions provides sufficient information to monitor the passive safety system performance, and verify that the unit transitions to MODE 3, is passively cooled, and remains stable. Under normal operating conditions where the MCR is intact and fully functional, the RSS is secured and equipment therein is in a standby state.

17.3 Regulatory Basis

17.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR Part 50 regulations are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the proposed exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any design basis event, and does not create new accident precursors. Therefore, the exemption will not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). The exemption does not affect the design, function, or operation of structures or plant equipment that is necessary to maintain the secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures. Therefore, the requested exemption is consistent with the common defense and security.

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying intent of the remote shutdown portion of GDC 19 is to provide means for operators to place and maintain the reactor in a safe condition in the event of a control room evacuation. NRC guidance recognizes that for passive plant designs, "safe shutdown" is a long-term safe stable shutdown condition. Thus, as a result of this exemption, NuScale's design basis will satisfy the underlying intent of GDC 19.

Special circumstances are present (10 CFR 50.12(a)(2)(iv)) in that the exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption. Application of the remote cold shutdown provision of GDC 19, as originally prescribed for active plants, would require NuScale to incorporate additional features increasing complexity of the I&C and shutdown systems. The NRC's Policy Statement on the Regulation of Advanced Reactors recognizes simplified, passive safety features, including highly reliable and less complex shutdown and decay heat removal systems, as a benefit to the public health and safety. Because safe shutdown is a long term, safe, stable shutdown condition, there is no identified decrease in safety as a result of this exemption.

17.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the portion of GDC 19 pertaining to potential capability to achieve cold shutdown from equipment outside the control room. PDC 19 maintains the required control room and remote shutdown capabilities, but clarifies that safe shutdown is the necessary reactor condition to achieve and maintain from outside the control room.