NuScale US460 Plant
Standard Design Approval Application

Exemptions

PART 7

Revision 2

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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 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi), which require high point vents for the reactor coolant system (RCS), reactor pressure vessel (RPV) head, and other systems required to maintain adequate core cooling. The underlying purpose of the requirements is to prevent the accumulation of noncondensible gases that may inhibit core cooling during natural circulation. The NuScale US460 standard design ensures core cooling without relying on high point vents in the RCS, RPV, and other systems required to maintain adequate core cooling. Therefore, the design meets the underlying purpose of the rules.

1.1.2 Regulatory Requirements

10 CFR 52.137(a) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *(4) …Analysis and evaluation of [emergency core cooling system] 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), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v)…*

10 CFR 50.46a states, in part:

> *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…*

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

> *(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 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) in their entirety.

1.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi).

1.2 Justification for Exemption

The underlying purpose of 10 CFR 50.46a, requiring high point vents for the RCS, RPV, and for other systems required to maintain adequate core cooling is to preclude an accumulation of noncondensible gases that may inhibit core cooling during natural circulation. 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 [emergency core cooling system] 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, NUREG-0737 Item II.B.1 states 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 Power Module 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 design, natural circulation is not inhibited by the accumulation of noncondensible gases and core cooling is not dependent on pump operation. Other systems required to maintain adequate core cooling do not require high point venting during an accident. Therefore, the underlying purpose of the requirements is met without the high point vents as specified in the rules.

1.2.1 Technical Basis

The design includes an RCS that is integral to the RPV; the core, steam generator, and pressurizer are contained in the RPV. The high point of the RCS 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 by use of the RPV high point degasification line.

As described in FSAR Section 5.4.4, the emergency core cooling system (ECCS) includes two reactor vent valves located on the top of the RPV that discharge to the containment upon ECCS actuation, thereby venting any noncondensible gases accumulated in the pressurizer space. The RCS does not include separate post-accident high point vent capability. As described in FSAR Sections 6.2 and 15.0, accumulated noncondensible gases vented to the containment vessel during ECCS operation do not challenge adequate core cooling.

During decay heat removal system (DHRS) cooling events, accumulation of noncondensable gases in the pressurizer does 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, and does not impede liquid circulation in the RPV. Noncondensible gas accumulation within the secondary system is calculated and considered in the DHRS performance analysis, summarized in FSAR Section 5.4.3, and determined not to challenge DHRS operation.

There are no other systems necessary to maintain adequate core cooling that require high point venting.

Therefore, the underlying purpose of 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) are met without the high point vents required by the rules.

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 does not impact the consequences of any design-basis event and does not create new accident precursors. The design does not rely on post-accident high point venting of the RCS, RPV, or other systems to accomplish safety functions. 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 that are necessary to maintain the secure status of the plant. This exemption has no impact on plant security or safeguards procedures. Therefore, this 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 an 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 from 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) for the US460 standard design approval.

2. 10 CFR 50.34(f)(2)(xvii) Combustible Gas Monitoring

2.1 Introduction and Request

2.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from 10 CFR 50.34(f)(2)(xvii)(C), which requires the capability for monitoring combustible gases during an accident. The underlying purpose of the rule is to support accident management and emergency planning for a significant beyond design-basis accident (BDBA) where hydrogen combustion could challenge containment integrity. The US460 standard design precludes combustion in containment during a significant BDBA by passively controlling the oxygen concentration to maintain an inert atmosphere. The capability to monitor hydrogen and oxygen concentrations is unnecessary to support mitigative actions or emergency planning. Moreover, the likelihood of a core damage event, where significant hydrogen could be generated, is very low. Therefore, the design meets the underlying purpose of the rules.

2.1.2 Regulatory Requirements

10 CFR 52.137(a) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *(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)(xvii) states:

> *Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)*

2.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50.34(f)(2)(xvii)(C).

2.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with 10 CFR 50.34(f)(2)(xvii)(C); post-accident hydrogen monitoring to satisfy the rule is not included in the design.

2.2 Justification for Exemption

10 CFR 50.34(f)(2)(xvii)(C) requires containment hydrogen monitoring capability. It is a Three Mile Island requirement that predates and has the same underlying purpose as 10 CFR 50.44(c)(4). The underlying purpose of 10 CFR 50.44, overall, is to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by the production and accumulation of combustible gases within containment following a BDBA. The rule's statements of consideration (at 68 FR 54130) explain that it addresses the risk from combustible gas generation during a BDBA:

> *Based upon the results of significant research into design-basis and beyond design-basis accidents, the NRC has determined that a design-basis combustible gas release is not risk-significant and certain beyond design-basis combustible gas releases are risk-significant. Therefore, the NRC is removing the requirements for combustible gas control systems that mitigate consequences of non-risk-significant design-basis accidents which are also not effective in reducing the risk from combustible gas releases in beyond-design-basis accidents.*

As discussed in FSAR Section 6.2.5, the NuScale Power Module (NPM) maintains an inert atmosphere in the containment during and following a BDBA. The design precludes the loss of containment structural integrity, safe shutdown functions, or accident mitigation features by hydrogen combustion.

10 CFR 50.44(c)(4), specifically, addresses the capability for containment hydrogen and oxygen monitoring for "water-cooled reactor designs with characteristics (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to" pre-existing light water reactors. Subparagraph (ii) requires hydrogen monitoring for all such containments. Subparagraph (i) requires oxygen monitoring for inert containments. Because the NPM is maintained inert, both provisions apply to the design.

As discussed in the rule's statements of consideration (68 FR 54136), the underlying purpose of combustible gas monitoring is to assess core damage and allow verification that combustible gas control systems perform their beyond design-basis functions, to support severe accident management and emergency planning:

> *Hydrogen monitors are required to assess the degree of core damage during beyond design-basis accidents. Hydrogen monitors are also used in conjunction with oxygen monitors to guide licensees in implementation of severe accident management strategies. Also, the NRC has decided to codify the existing regulatory practice of monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. If an inerted containment became de-inerted during a beyond design-basis accident, other severe accident management strategies, such as purging and venting, would need to be considered. Monitoring of both hydrogen and oxygen is necessary to implement these strategies.*

The statements of consideration (at 68 FR 54131) further link the purpose of monitoring to the potential for failure of combustible gas control measures:

> *Because hydrogen monitors are not needed to initiate or activate any mitigative features during these accidents, they are not risk-significant for reducing the combustible gas threat as long as the hydrogen igniters are operable. If the igniters are not operating (such as during station blackout) hydrogen monitoring does not reduce risk since the containment cannot be purged or vented without electrical power. Nevertheless, the amended rule requires licensees to retain hydrogen monitors (and oxygen monitors in Mark I and Mark II BWRs) for their containments because they are useful in implementing emergency planning and severe accident management mitigative actions for beyond design basis accidents.*

Thus, the statements of consideration explain (68 FR 54126):

> *If an inerted containment was to become de-inerted during a significant beyond design-basis accident, then other severe accident management strategies, such as purging and venting, would need to be considered.…*
>
> *The hydrogen monitors are required to assess the degree of core damage during a beyond design-basis accident and confirm that random or deliberate ignition has taken place…. If an explosive mixture that could threaten containment integrity exists, then other severe accident management strategies, such as purging and/or venting, would need to be considered.*

As discussed in FSAR Section 6.2.5, the NPM relies on a safety-related passive autocatalytic recombiner to maintain the containment inert. In the NuScale design oxygen is the limiting reactant for the PAR function. The NPM is not susceptible to de-inerting. The design utilizes radiation monitors under the bioshield and core exit thermocouples to assess core damage. As such, combustible gas monitoring is not necessary for the NPM to guide implementation of the emergency plan and severe accident management mitigative actions.

Therefore, the design meets the underlying purpose of 10 CFR 50.34(f)(2)(xvii)(C).

2.2.1 Technical Basis

The underlying purpose of the combustible gas monitoring requirements is to enable the assessment of core damage and verification that combustible gas control systems perform their beyond design-basis functions, to support severe accident management and emergency planning.

As discussed in FSAR Section 6.2.5, the NPM relies on a passive autocalyctic recombiner (PAR) to maintain the containment inert through the continuous recombination of oxygen and hydrogen. The PAR is designed to maintain the containment inert following both design basis and beyond design basis events, but design basis events are limiting. In the NuScale design oxygen is the limiting reactant for the PAR function. Unlike hydrogen igniters and similar mitigation features, the PAR does not rely on electric power or moving parts to function. The PAR is a safety-related passive device that self-actuates to recombine oxygen and hydrogen present in the surrounding environment. The PAR is designed to function in environments for which it is intended.

The NPM is not susceptible to de-inerting. The only sources of oxygen are from the initial quantities in the reactor coolant system controlled by the primary chemistry control program and through radiolytic decomposition of water. Inerting is accomplished solely by the PAR recombining oxygen; no inert gas is added to the containment during operations or post-accident. The PAR has adequate capacity to maintain the containment oxygen concentration below four percent by volume.

The design does not rely on hydrogen monitoring to assess core damage. As described in FSAR Section 7.1, the radiation monitors under the bioshield and the core exit thermocouples provide the ability to detect and assess core damage.

In summary, the design relies on passive limiting of oxygen concentration to preclude combustible gas mixtures from forming in the containment environment. The PAR is highly reliable and the NPM is not susceptible to de-inerting. Accordingly, combustible gas monitors are not needed to support other severe accident management strategies, such as purging or venting, in the event of containment de-inerting. Hydrogen monitoring is not needed to assess the degree of core damage in a BDBA or verify that combustible gas control features are functioning. Therefore, the design meets the underlying purpose of the combustible gas monitoring rules.

2.2.2 Risk Considerations

During the Advisory Committee on Reactor Safeguards (ACRS) review of the design certification application for the predecessor, NuScale US600 design-which does not include the capability to preclude hydrogen combustion-the ACRS observed ("NuScale Combustible Gas Monitoring," April 28, 2020):

> *Continuous monitoring of combustible gases would allow operators to minimize the chance of a detonation that could challenge containment integrity. This core damage event is of very low probability because it requires failure of normal heat removal, failure of the passive decay heat removal system, and failure of the emergency core cooling system valves that provide another passive means to remove decay heat.*

The ACRS observed the "the risk tradeoff between unisolating the NuScale containment to enable long-term hydrogen and oxygen monitoring" because such monitoring would entail "circulating large portions of containment volume through" nonsafety-related piping. The ACRS observed that "alternatives that may not require such monitoring" should be considered and concluded:

> *The need for post-accident monitoring might be greatly reduced and an exemption might be possible based on the low risk (probability and consequence) of this type of scenario.*

The US460 standard design further reduces the likelihood of hydrogen generation and combustion during severe accidents. The design includes a PAR to ensure an inert containment atmosphere. Therefore, the need for monitoring is further decreased: the "risk tradeoff" of unisolating the NPM to monitor combustible gases more strongly favors an exemption from combustible gas monitoring requirements.

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)). 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. This exemption concerns only the capability to monitor combustible gases during a BDBA; the design precludes combustion. 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 is not necessary to achieve the underlying purpose of the rule. The design precludes combustion that could challenge containment structural integrity, safe shutdown functions, or accident mitigation features. Combustible gas monitoring is not necessary to support severe accident management and emergency planning.

Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that there is present a material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption. The design has a very low likelihood of core damage that would lead to significant amounts of combustible gases within containment, and the design passively controls oxygen levels to preclude combustion. Combustible gas monitoring would require unisolating the containment during the response to an accident, where containment isolation is essential to both severe accident prevention and mitigation. Therefore, the difference in "risk tradeoff" is a material circumstance not considered when the regulation was adopted; the exemption avoids unnecessary containment unisolation, which is in the public interest.

2.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption from 10 CFR 50.34(f)(2)(xvii)(C) for the US460 standard design approval.

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 underlying purpose of 10 CFR 50.62 is to reduce the risk associated with ATWS events. The US460 standard plant design reduces the risk of an ATWS event via redundancy, diversity, and independence within the module protection system (MPS). The MPS design reduces the probability of a failure to scram. When combined with the 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 basis. Therefore, the underlying purpose of the rule is met without the diverse turbine trip capabilities specified in 10 CFR 50.62(c)(1).

The US460 standard plant design does not include an auxiliary or emergency feedwater system. Therefore, the portion of 10 CFR 50.62(c)(1) requiring diverse and automatic auxiliary feedwater system (AFWS) initiation is not applicable to the design.

3.1.2 Regulatory Requirements

10 CFR 52.137(a)(15) requires a standard design approval application final safety analysis report to include, in part:

> *Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram (ATWS) 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 from the reactor trip system 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 US460 standard plant 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 US460 standard design does not conform with 10 CFR 50.62(c)(1). The design does not have equipment diverse from the MPS to initiate a turbine trip. The portion of 10 CFR 50.62(c)(1) requiring diverse AFWS initiation is not applicable because the design does not include an AFWS.

3.2 Justification for Exemption

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 (SECY-83-293 and NUREG-1780). The value-impact calculations presented in SECY-83-293 were derived from the fundamental constraints of then-existing plant designs that limited the options for design enhancements under consideration when the rule was promulgated. Digital instrumentation and control designs as employed in the US460 standard plant design are not considered. As discussed below, the NuScale design process integrated risk reduction for ATWS events during initial design activities, unconstrained by an existing reactor trip system design. To meet the underlying purpose of the rule, the MPS is designed to limit the risk from common-cause failures leading to a failure to scram.

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. 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. For the US460 standard plant design, the protection system is the MPS. Diversity within the MPS reduces 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." As described in FSAR Section 19.1.9, the ATWS contribution to single module core damage frequency is less than the target of 1.0E-5 per reactor year. The NuScale Power Module (NPM) response to an ATWS event does not rely on diverse turbine trip functionality to reduce ATWS risk. 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 US460 standard plant design, which does not include an AFWS.

<u>MPS Diversity</u>

The design achieves acceptable ATWS risk using a robust reactor trip system that has internal diversity. As discussed in FSAR Section 7.1 and Section 7.2, the MPS utilizes the highly integrated protection system (HIPS) platform. The HIPS platform encompasses the principles of independence, redundancy, predictability and repeatability, and diversity and defense-in-depth. These key design concepts of the HIPS platform contribute to simplicity in both the functionality of the MPS and in its implementation.

Internal diversity within the MPS achieves a similar outcome as a diverse scram system, which is a requirement for other reactor designs (10 CFR 50.62(c)(2) and SECY-90-016). The diversity within MPS provides a simpler solution than a diverse scram system by achieving sufficient ATWS risk reduction without the addition of a separate scram system.

ATWS Response

As described in FSAR Section 15.8, the 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 plant response to postulated ATWS events, the event sequences are modeled in the plant PRA to include ATWS as discussed in FSAR Section 19.2.2. The plant response to an ATWS event, considering features such as passive cooling and a low power density core, protects against fuel damage, thereby limiting the risk of an ATWS. Without diverse turbine trip and without a diverse scram system, the ATWS contribution to single module core damage frequency is less than the target of 1.0E-5 per reactor year of SECY-83-293 as demonstrated in FSAR Section 19.1.9.

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 design incorporates diversity within the MPS, reducing the risk from common-cause failures leading to a failure to scram. The 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)(ii)) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The design does not rely on diverse turbine trip functionality to reduce the risks associated with ATWS. The 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 the design 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 design features prescribed by 10 CFR 50.62(c) 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. The prescribed design features were delineated for large pressurized water reactors based on the risk reduction they 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.

The plant designs that were considered during the 10 CFR 50.62 rule making and their responses to an ATWS event differ from the US460 standard plant design. The design includes enhanced safety features that reduce the risk from ATWS events and also maintains a simpler instrumentation and control configuration than the separate turbine trip 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 US460 standard design approval from the portion of 10 CFR 50.62(c)(1) requiring diverse turbine trip equipment. The portion of 10 CFR 50.62(c)(1) requiring diverse AFWS initiation is not applicable to the 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 US460 standard 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 US460 standard design uses passive safety systems and features to accomplish safety-related functions without reliance on electrical power. Therefore, NuScale meets the underlying purpose of the rule.

NuScale further requests exemptions from GDC 18 and from the portions of GDCs 34, 35, 38, 41, and 44 addressing electric power as conforming changes. These requirements are intended to ensure sufficient electric power is available to accomplish the safety functions of the respective systems. Because the design does not rely on electric power to perform safety functions, these requirements are unnecessary to apply to the design.

4.1.2 Regulatory Requirements

10 CFR 52.137(a)(3) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *(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;*

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, GDCs 34, 35, 38, 41, and 44 require that the residual heat removal system, emergency core cooling system, containment heat removal system, containment atmosphere cleanup systems, and cooling water system, respectively, each 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, assuming a single failure.

4.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests exemptions from GDCs 17 and 18 in their entirety.

NuScale requests exemptions from the provisions of GDCs 34, 35, 38, 41, and 44 addressing capabilities for various systems with respect to electric power. For each of these GDCs, the exemption is 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)." GDC 33, which includes the same provision, is the subject of a separate exemption request.

4.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with GDCs 17 and 18; GDCs 17 and 18 are not applicable design criteria.

The design conforms to design-specific principal design criteria (PDCs) instead of GDCs 34, 35, 38, 41, and 44. The plant design bases include the PDCs as set forth in FSAR Section 3.1. In each of these PDCs, 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)…" is eliminated from the GDC.

4.2 Justification for Exemption

The underlying purpose of GDC 17 is to provide onsite and offsite electric power systems to assure sufficient power to accomplish safety functions. 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.*

The US460 systems and features credited for safe shutdown, core cooling, containment isolation and integrity, and reactor coolant pressure boundary (RCPB) integrity do not rely on electric power to perform their safety-related functions. The underlying purpose of GDC 17 is thus achieved because the specified functions are performed without onsite and offsite power.

The underlying purpose of GDC 18 is to provide capability for periodic inspection and testing of the power systems that are subject to GDC 17, to ensure power system capability to perform its safety functions. Because electric power does not perform a safety-related function, GDC 18 is unnecessary to apply to the design.

GDCs 34, 35, 38, 41, and 44 specify reliability-related criteria for various systems that perform plant safety functions, including the ability to function with a loss of either onsite or offsite electric power. The safety-related functions addressed by these GDCs are accomplished by passive systems or inherent design characteristics. Therefore, the electric power provisions of these GDCs are unnecessary to apply to the design.

4.2.1 Technical Basis: GDC 17

General

In operating light-water reactor plant designs, safety-related systems require electric power to function. Because of the importance of electric power in supporting multiple safety-related functions, GDC 17 requires redundant means of supplying electric power to perform those plant safety-related functions (an onsite and offsite electric power system), and prescribes reliability criteria for each.

As described in FSAR Chapter 8, the US460 standard design does not rely on electric 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 (DBE). If electric power becomes unavailable, the safety-related systems actuate and their continued operation relies on fundamental physical and thermodynamic principles that do not require electric power (e.g., gravity; natural circulation; convective, radiative, and conductive heat transfer; condensation; and evaporation). Therefore, electric power is not required to actuate or operate systems or components that perform safety-related functions.

Onsite power systems provide power to the plant loads during all modes of plant operation. The onsite power systems include independent alternating current (AC) power systems and direct current (DC) power systems. The plant safety-related functions are achieved and maintained without reliance on electrical power; therefore, neither the AC power systems nor the DC power systems are safety-related (Class 1E). Additionally, the on-site power systems do not perform risk-significant functions. 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 offsite power system includes one or more connections to a transmission grid, micro-grid, or dedicated service load. The design does not depend on offsite electric power, including that from the transmission grid, for safe operation. The availability of electric power from an offsite power source does not impact the ability to achieve and maintain safety-related functions. A loss-of-voltage condition, degraded-voltage condition, or other electrical transients on the nonsafety-related AC power systems does not have an adverse effect on the ability to achieve and maintain safe-shutdown conditions.

Qualified isolation devices, described in FSAR 7.1.2.2, provide electrical isolation between electric power systems and safety-related equipment.

FSAR Chapter 15 demonstrates that, with electric power unavailable (lost at event initiation), (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. Because the design does not require electric power to meet the acceptance criteria of GDC 17, the design satisfies the purpose of the rule.

Onsite DC Power Availability

As described in FSAR Section 8.3.2, the augmented DC power system (EDAS) comprises two DC subsystems that provide a continuous, failure-tolerant source of DC power to assigned plant loads during normal plant operation and for a specified minimum duty cycle following a loss of AC power. The module-specific subsystem (EDAS-MS) includes the function to preclude unnecessary ECCS valve actuation for a minimum of 24 hours following a postulated loss of AC power, unless a valid ECCS actuation signal is received. The ECCS actuates upon a loss of EDAS power.

FSAR Chapter 15 analyzes DBEs both with and without power available from the AC or DC power supplies, including EDAS. Safety analysis demonstrates that electric power is not relied upon to remain functional during a DBE to perform safety-related functions, and the US460 standard design has appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including effects of anticipated operational occurrences. The safety analysis demonstrates that plant safety functions are fulfilled with EDAS unavailable (lost at event initiation) and with EDAS available for the event duration.

Safety analyses with EDAS unavailable satisfies the requirements of GDCs 34, 35, 38, 41, and 44 if those GDCs were applied to the design (i.e., not subject to an exemption). For example, GDC 34 would otherwise require for the DHRS “suitable redundancy in components and features…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.” Because the design does not rely on either onsite or offsite power, this provision could be simplified to require that the DHRS performs its safety function with electric power unavailable. GDCs 34, 35, 38, 41, and 44 require power unavailability to be assessed, and they require a single failure of each safety system (not of the power system) to be assumed. Loss of EDAS at event initiation satisfies this intent.

If one of those safety systems could not function without electric power, then onsite power would be safety-related, and GDC 17 would in turn require that the onsite power system tolerate a single failure. In some cases, nonconsequential single failures of safety-related SSC are postulated at limiting times during an event progression. Because the US460 standard design is shown to perform all safety functions with electric power unavailable, the single failure criterion for safety-related systems is not applicable to EDAS.

Separately from the single failure criterion, nonsafety-related systems are generally assumed unavailable in the safety analysis, in order to show that the nonsafety-related system is not “relied upon” in a DBE to assure one of the three safety functions specified by the safety-related definition (10 CFR 50.2). The Chapter 15 safety analyses are consistent with this practice in considering EDAS unavailable at event initiation.

Further, large light-water reactors also generally consider the potential for a loss of offsite power (LOOP) coincident with turbine trip. This consideration is not explicitly required by the GDCs, but is reflected in GDC 17's requirement "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." For such designs, a LOOP can be a consequence of a turbine trip (for example, SECY-01-0133, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)," Attachment 1, discusses causes of a LOOP following a LOCA). Therefore, LOOP upon turbine trip may be considered part of the DBE progression as a consequential failure. For the US460 standard design, there is no analogous causal relationship between DBEs and a loss of EDAS (FSAR 15.0.0); failures during event progression are unnecessary to assume with respect to mechanistic event progressions.

In sum, the assumed unavailability of EDAS at event initiation satisfies the requirements of GDCs 17, 34, 35, 38, 41, and 44 (if those GDCs were applied to the design) and the safety-related definition, and consequential EDAS failures following event initiation are not applicable to the design. Therefore, because safety-related functions are performed and acceptance criteria are satisfied with electric power unavailable during DBEs, application of GDC 17 and the electric power provisions of GDC 34, 35, 38, 41, and 44 is not necessary to achieve the underlying purpose of those rules.

4.2.2 Technical Basis: GDC 18

The requirements of GDC 18 further the electric power reliability purpose of GDC 17 by requiring certain design provisions for inspection and testing of the electric power systems. In describing the technical rationale for compliance with GDC 18, NUREG-0800 discusses the scope of GDC 18 in terms of Class 1E systems. For example, Standard Review Plan Section 8.3.1, Revision 4, states "the AC power system should provide the capability to perform integral testing of Class 1E systems on a periodic basis."

As discussed above and in FSAR Chapter 8, the US460 standard design AC and DC power systems are nonsafety-related and non-Class 1E. The electric power systems are not relied on to perform safety functions or meet the acceptance criteria of GDC 17. Therefore, conformance with the inspection and testing provisions of GDC 18 is unnecessary to verify electric power system capabilities.

As described in FSAR Section 8.3.2, the EDAS is designed to permit appropriate periodic inspection and testing to assess the operability and functionality of the system and the condition of its components.

4.2.3 Technical Basis: GDCs 34, 35, 38, 41, and 44

GDCs 34, 35, 38, 41, and 44 specify reliability-related criteria for various systems that perform safety-related functions, including the ability to function with a loss of either onsite or offsite electric power. The NuScale Design Specific Review Standard 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 DBEs for 72 hours, without operator action, following a loss of both offsite and onsite ac power sources."

The safety-related functions addressed by these GDCs are accomplished by passive systems or inherent design characteristics. As discussed in the FSAR within the applicable system descriptions, the safety-related functions addressed by these GDCs (except GDC 33, which is subject to a separate exemption request) are provided in the design and are performed without electric power available. The design has the capability to automatically establish and maintain safe-shutdown conditions after DBEs for 72 hours, without operator action, following a loss of both offsite and onsite AC power sources. Therefore, the electric power provisions of these GDCs do not apply.

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 these regulations. Therefore, the exemption is authorized by law.

The requested exemption will not present an undue risk to the public health and safety (10CFR50.12(a)(1)). This exemption does not impact the consequences of any DBE or create new accident precursors. The design does not rely on electric power 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 (10CFR50.12(a)(1)). This 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 does not impact the security power system. 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 regulations in the particular circumstances not necessary to achieve the underlying purpose of the rules. The underlying purpose of GDC 17 (together with GDC 18 and the power provisions of GDC 34, 35, 38, 41, and 44) is to ensure safety functions that rely on electric power are reliable. The design does not rely on electric power to accomplish safety-related functions or meet the GDC 17 acceptance criteria, and therefore the underlying purpose of the GDCs is met without the applying these requirements to the design.

4.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant exemptions for the US460 standard design from GDCs 17 and 18 and from 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). The US460 standard design does not require makeup to protect against small breaks in the RCPB. The NuScale Power Module (NPM) design preserves reactor coolant inventory by isolating containment at specified safety setpoints. The design, without relying on makeup, ensures that fuel integrity is not challenged by a small break in the RCPB. Therefore, the design meets the underlying purpose of the rule.

5.1.2 Regulatory Requirements

10 CFR 52.137(a)(3) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

(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;

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 GDC 33 in its entirety.

5.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with GDC 33; GDC 33 is not an applicable design criterion.

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 makeup 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."

During a small break in the RCPB, the NPM retains sufficient reactor coolant system (RCS) inventory such that, with actuation of the emergency core cooling system (ECCS) and isolation of the chemical volume and control system from the RCS, adequate core cooling is maintained. Without relying on makeup, a small RCPB break does not exceed specified acceptable fuel design limits (SAFDLs). Therefore, the design meets the underlying purpose of GDC 33.

5.2.1 Technical Basis

The design includes a chemical and volume control system that maintains RCS inventory during normal operation (FSAR Section 9.3.4).

During off-normal transients, chemical and volume control system makeup is not relied on to protect against exceeding minimum critical heat flux ratio. Instead, reactor coolant inventory is preserved within the NPM by isolating connected systems at safety setpoints. For small RCPB breaks that lead to ECCS actuation, the ECCS protects SAFDLs by maintaining core coolant inventory and core coolability. For small RCPB breaks that do not actuate the ECCS, the decay heat removal system cools the core to meet SAFDLs.

Thus, the NPM relies on the retention of sufficient RCS coolant and the operation of safety systems as an alternative means of maintaining reactor coolant inventory and coolability during a small break of the RCPB. The design meets the purpose of GDC 33 without relying on the 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 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 US460 standard design incorporates design provisions to retain adequate reactor coolant inventory such 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 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 that are 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 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 US460 standard design incorporates design provisions to retain adequate reactor coolant inventory such 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 design meets 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 GDC 33 for the US460 standard design.

6. 10 CFR 50.60 Acceptance Criteria for Fracture Prevention Measures

6.1 Introduction and Request

6.1.1 Summary

NuScale Power LLC (NuScale), requests an exemption from 10 CFR 50.60, which requires that light water reactors meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary (RCPB) set forth in 10 CFR 50, Appendices G and H. The underlying purpose of 10 CFR 50.60 is to reduce the risk associated with the effects of neutron and thermal embrittlement of the RCPB. Appendices G and H include a methodology for calculating the nil-ductility reference temperature (RT_{NDT}). The NuScale Power Module (NPM) design reduces the susceptibility to the effects of neutron and thermal embrittlement by using austenitic stainless steel rather than ferritic materials in the lower reactor pressure vessel (RPV), which includes the beltline region. The austenitic stainless steel used in the lower RPV is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. NuScale evaluated data on irradiated austenitic stainless steel to conclude that the effects of neutron and thermal embrittlement are minor even when these materials are exposed to a fluence level exceeding the design life peak fluence of the NPM lower RPV. Therefore, the underlying purpose of 10 CFR 50.60 is met without applying to the lower RPV the fracture toughness and material surveillance program requirements for ferritic materials of 10 CFR 50, Appendices G and H.

6.1.2 Regulatory Requirements

10 CFR 52.137(a)(14) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 10 CFR 50.60 and 50.61.*

10 CFR 50.60 requires all light water nuclear power reactors to:

> *…meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.*

10 CFR 50, Appendix G, states in part:

> *This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and*

system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The ASME Code forms the basis for the requirements of this appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components."

…The requirements of this appendix apply to the following materials:

I.A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G-2110 of Appendix G of Section XI of the latest edition and addenda of the ASME Code incorporated by reference into § 50.55a(b)(2).

I.B. Welds and weld heat-affected zones in the materials specified in paragraph I.A of this appendix.

10 CFR 50, Appendix G, defines ferritic material as:

Ferritic material means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.

10 CFR 50, Appendix G, defines RT_{NDT} as:

RT_{NDT} means the reference temperature of the material, for all conditions. (i) For the pre-service or unirradiated condition, RT_{NDT} is evaluated according to the procedures in the ASME Code, Paragraph NB-2331. (ii) For the reactor vessel beltline materials, RT_{NDT} must account for the effects of neutron radiation.

10 CFR 50, Appendix G, defines beltline or beltline region as:

Beltline or Beltline region of reactor vessel means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation *damage to be considered in the selection of the most limiting material with regard to radiation damage.*

10 CFR 50, Appendix H, states in part:

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Section IV of Appendix G to Part 50.

10 CFR 50.60, Appendix H, also states:

III.A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 *(E > 1 MeV).*

6.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50.60 as applied to the lower RPV.

6.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with 10 CFR 50.60. The NPM lower RPV does not conform with the fracture toughness and material surveillance program requirements for the RCPB set forth in 10 CFR 50, Appendices G and H. The lower RPV RT_{NDT} cannot be calculated for the austenitic stainless steel lower RPV and because the requirements and methodology for a material surveillance program for an austenitic stainless steel RPV do not exist; 10 CFR 50.61, which uses the RT_{NDT}, is addressed by a separate exemption. Ferritic materials in the RCPB comply with 10 CFR 50.60.

6.2 Justification for Exemption

The underlying purpose of 10 CFR 50.60 is to reduce the risk associated with the effects of neutron and thermal embrittlement of the RCPB. Appendices G and H include a methodology for calculating the nil ductility reference temperature (RT_{NDT}), which allows calculation of fracture toughness requirements for protection against pressurized thermal shock in 10 CFR 50.61.

The requirements of 10 CFR 50, Appendices G and H reduce the risk of abnormal leakage, of rapidly propagating failure, and of gross rupture of the RCPB (pursuant to 10 CFR 50 Appendix A General Design Criteria (GDCs) 14 and 31); and ensure that the reactor coolant system and associated auxiliary, control, and protection systems contain sufficient margin that design conditions of the RCPB are not exceeded during operation (pursuant to GDC 15). The reactor vessel surveillance program (RVSP) requirements of 10 CFR 50, Appendix H further address GDC 32 by specifying an RSVP that is appropriate for ferritic materials in the RPV.

To meet the underlying purpose of 10 CFR 50.60, the NPM RCPB is designed with consideration for the effects of neutron and thermal embrittlement. The lower RPV, including the beltline region, is made of austenitic stainless steel, which has superior ductility and is less susceptible to neutron and thermal embrittlement effects than ferritic materials typically used in light water RPVs. NuScale evaluated data on irradiated austenitic stainless steel to conclude that the effects of neutron and thermal embrittlement are minor even when these materials are exposed to a fluence level exceeding the design life peak fluence of the NPM lower RPV. The decreased susceptibility of austenitic stainless steel to the effects of neutron and thermal embrittlement compared to ferritic materials increases the integrity and safety of the RCPB. Therefore, the underlying purpose of 10 CFR 50.60 is met without applying to the lower RPV the fracture toughness and material surveillance program requirements for ferritic materials of 10 CFR 50, Appendices G and H.

6.2.1 Technical Basis

10 CFR 50.60 requires light water reactors to meet fracture toughness and material surveillance program requirements in 10 CFR 50, Appendices G and H. Appendices G and H specifically address fracture toughness and material surveillance of ferritic materials, due to their susceptibility to the effects of neutron and thermal embrittlement. Appendix G requires that ferritic beltline materials be tested in accordance with Appendix H and be subject to requirements supplementing the ASME Code pertaining to Charpy upper-shelf energy and pressure-temperature limits. The beltline portion of the RPV is the limiting part of the RPV when evaluating fracture toughness and material embrittlement because it is exposed to the highest fluence. Under Appendix H, ferritic materials in the RPV beltline are subject to an RVSP if design life peak fluence exceeds 10^{17} n/cm^2 (E > 1 MeV).

10 CFR 50, Appendices G and H use RT_{NDT} evaluations based on ASME BPVC Section III, Paragraph NB-2331, along with data based only on ferritic materials. ASME BPVC Section III, Paragraph NB-2331, follows ASME BPVC Section III, Paragraph NB-2311. There are no impact test requirements for austenitic stainless steels in ASME BPVC Section III, Paragraph NB-2311, because these materials do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials used for ASME BPVC Section III Class 1 pressure-retaining components. The NRC endorsed the ASME BPVC 2017 version in 10 CFR 50.55a.

To satisfy with the intent of 10 CFR 50.60, the US460 standard design RPV is subject to the following manufacturing requirements for austenitic stainless steel used in the lower RPV.

- The lower RPV only uses forgings to avoid vertical pressure-retaining welds.
- SA-965 Grade FXM-19 is in the solution annealed condition prior to circumferential welding.
- The maximum carbon content for the forgings and weld filler metal is limited to 0.04 percent to minimize sensitization concerns during welding.
- Consistent with ASME BPVC Section III, Paragraph NB-4622, there is no post-weld heat treatment because SA-965 Grade FXM-19 is a P8 material.

In addition, NuScale evaluated total elongation data post-irradiation and concluded that the effects of neutron embrittlement on solution-annealed FXM-19 is minor for a fluence level that is 28 percent higher than the design life peak fluence of the NPM lower RPV. NuScale also evaluated mill-annealed FXM-19, which remains highly ductile post-irradiation when subjected to a fluence level that is 94 percent higher than the design life peak fluence for the NPM lower RPV. Finally, NuScale evaluated fracture toughness test results of mill-annealed FXM-19 post-irradiation and concluded that there is a minor reduction in average plane-strain fracture toughness values at a fluence level that is over 200 percent of the design life peak fluence for the NPM lower RPV.

NuScale also evaluated FXM-19 and associated weld materials (SFA 5.4 E209 or E240 and SFA 5.9 ER209 or ER240) against Type 3XX austenitic stainless steels. Two publications proposed maximum threshold fluence levels for austenitic stainless steel base metal and weld metal, both of which are greater than the design life peak fluence for the NPM lower RPV. Therefore, since the design life peak fluence for the NPM lower RPV is well below the recommended threshold fluences, the embrittlement effects are minimal for FXM-19 and the associated weld materials.

The data and evaluations supporting this technical basis are in NuScale technical report TR-130721, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel."

The available data for austenitic stainless steel supports the conclusion that austenitic stainless steel is highly ductile and less susceptible to the effects of neutron and thermal embrittlement than ferritic materials. Because the NPM lower RPV uses materials that increase the integrity and safety of the RCPB, the NPM lower RPV design satisfies the intent of 10 CFR 50.60 without applying 10 CFR 50, Appendices G and H.

6.3 Regulatory Basis

6.3.1 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 NPM uses a material for the lower RPV that is less susceptible to the effects of neutron and thermal embrittlement than the ferritic materials used in the operating light water reactor fleet. The RPV design has a lower risk of abnormal leakage, rapidly propagating failure, gross rupture, and fracture of the RCPB, which is a boundary that protects the public from radiation release. Therefore, exemption from the provisions of 10 CFR 50.60, including those in 10 CFR 50, Appendices G and H, regarding fracture toughness calculations and a material surveillance program for ferritic materials in the RPV 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 that are necessary to maintain the secure status of the plant. This exemption has no impact on plant security or safeguards procedures. Therefore, this 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 NPM design provides adequate RCPB fracture toughness, and the effects of neutron and thermal embrittlement are not concerning for austenitic stainless steel in the lower RPV. Application of 10 CFR 50, Appendices G and H to the lower RPV is therefore not required to meet the underlying purpose of the 10 CFR 50.60.

Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that other material circumstances are present that were not considered when the regulation was adopted. 10 CFR 50, Appendices G and H, applicable via 10 CFR 50.60, establish requirements for ferritic materials in the RCPB because the population of nuclear power plant RPVs were composed of these materials at the time the regulations were promulgated. Thus the regulations did not address an RPV made of a non-ferritic material. The ferritic RPVs considered during the 10 CFR 50.60 rulemaking and their responses to the effects of neutron and thermal embrittlement differ significantly from the NPM design. The NPM design includes a more robust material that sufficiently reduces the risk of abnormal leakage, rapidly propagating failure, gross rupture, fracture of the RPV, or breach of the RCPB. Therefore, it is in the public interest to grant an exemption from the fracture toughness and material surveillance program requirements in 10 CFR 50.60.

6.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the US460 standard design approval from 10 CFR 50.60, regarding fracture toughness calculations and a material surveillance program for ferritic materials in the RPV, as applied to the NPM lower RPV.

7. 10 CFR 50, Appendix 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 the capability for testing to verify containment leakage integrity, in order to ensure that containment leakage does not exceed allowable limits. The NuScale Power Module (NPM) relies on the containment vessel (CNV) design, factory inspection and testing, the capability for inservice inspection and examination, and the capability for leak testing other than ILRT that together provide adequate means to verify containment leakage integrity. Therefore, the design meets the underlying purpose of the rule.

7.1.2 Regulatory Requirements

10 CFR 52.137(a)(3) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *(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;*

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 Part 50, Appendix J, "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.

7.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from GDC 52 in its entirety.

10 CFR Part 50 Appendix J (hereafter "Appendix J") requires the performance of integrated leak rate ("Type A" testing). Appendix J is not applicable to the US460 standard design approval and is therefore not within scope of this exemption. Appendix J is applicable to operating and combined licenses and will be the responsibility of a license applicant to request an exemption.

7.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with GDC 52; GDC 52 is not an applicable design criterion.

7.2 Justification for Exemption

The underlying purpose of GDC 52 is to provide for the ability to conduct ILRT, which is one aspect of ensuring containment leakage integrity is maintained during its service life. Appendix J identifies containment leakage rate inspection and testing requirements for licensees, including preoperational and periodic ILRT (Type A tests) and 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 its requirements 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 CNV design allows for testing and inspection, other than ILRT, to meet the underlying purpose of the rule to assure CNV leakage integrity. The CNV design and provisions for testing and inspection provide adequate means to verify leakage integrity, and thus the design meets the underlying purpose of GDC 52.

7.2.1 Technical Basis

NuScale technical report TR-123952, "NuScale Containment Leakage Integrity Assurance," describes the design features and programmatic elements that ensure leakage integrity for the CNV. As described therein, those design features and programmatic elements ensure that containment leakage does not exceed allowable values without conducting ILRT. Therefore, the design meets the underlying purpose of GDC 52.

Replacing ILRT with other means of verifying CNV leakage integrity benefits public health and safety by maintaining occupational radiation doses as low as reasonably achievable (ALARA). As described in TR-123952, the NPM design presents unique challenges to performing ILRT at containment design pressure. Accessibility constraints and the 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 exposure is unnecessary because Type B and C testing can be used to quantify containment leakage for the NPM design.

As described in TR-123952, ILRT requirements do not reflect the unique challenges of the NPM design. The impacts of temperature and pressure fluctuations on Type A testing and associated acceptance criteria for the NPM 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 design compared to the LLWR designs contemplated by the regulation. Because the NPM relies on other means of ensuring leakage integrity, designing for the capability to perform ILRT presents an undue hardship.

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 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 CNV design and provisions for inspection and testing provide adequate means to ensure that no unknown leakage pathways exist. Type B and C tests quantify CNV leakage to ensure it is within the allowable leakage rate. 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 that are 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 is not necessary to achieve the underlying purpose of the rule. The CNV design and provisions for inspection and testing provide adequate means to verify leakage integrity. Therefore, the design meets the underlying purpose of GDC 52.

Special circumstances are present (10 CFR 50.12(a)(2)(iii)) in that compliance would result in undue hardship. The prescriptive Appendix J, Type A testing requirements and acceptance criteria are impractical for the design. Application of Type A testing requirements to the CNV would likely yield inaccurate leakage results because of the limited effectiveness of Type A acceptance criteria when applied to the design. Therefore, designing the CNV to permit ILRT pursuant to GDC 52 is an undue hardship.

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. The CNV design and provisions for inspection and testing provide adequate means to verify leakage integrity. These methods maintain occupational radiation doses ALARA by avoiding unnecessary tests, which benefits 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 Type A testing reflect containments different from the NPM design. The CNV design (e.g., an ASME Class 1 pressure vessel with all surface areas and welds accessible for inspection) represents material circumstances not considered when the regulation was adopted.

7.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption from GDC 52 for the US460 standard 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. The underlying purpose of GDC 40 is to provide the capability for testing to verify the operability and performance of the containment heat removal system. For the US460 standard design, inspections of the passive components comprising the containment heat removal function are adequate to verify operability and performance. Therefore, the design meets the underlying purpose of the rule.

8.1.2 Regulatory Requirements

10 CFR 52.137(a)(3) requires a standard design approval application final safety analysis (FSAR) report to include, in part:

(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;

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 in its entirety.

8.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with GDC 40; GDC 40 is not an applicable design criterion.

8.2 Justification for Exemption

The underlying purpose of GDC 39, which covers design for inspection, and GDC 40, which addresses design for testing, is to ensure appropriate means for verifying the operability and performance of the containment heat removal system. GDCs 39 and 40 address typical containment heat removal systems, such as a containment spray system, where a combination of inspection and testing is needed to ensure integrity, operability, and performance of the components and system. The NuScale Power Module (NPM) does not have a containment heat removal system; the components that serve the containment heat removal function do not require periodic pressure or functional testing to their ensure operability and performance. Therefore, the design meets the underlying purpose of GDC 40 without designing for periodic pressure and functional testing of the containment heat removal function.

8.2.1 Technical Basis

In the US460 standard design, containment heat removal is an inherent characteristic ensured by the materials and physical configuration of the NuScale Power Module (NPM) being partially immersed in the reactor pool, which functions as the ultimate heat sink. This configuration directly removes heat from containment without a containment heat removal system. Containment heat removal is performed without reliance on electrical power, valve actuation, cooling water flow, or other active system or component operations. Further design details of are described in FSAR Section 6.2.2.

Periodic pressure and functional testing, as specified by GDC 40, is not necessary because performance of the containment heat removal function is assured through other means. Periodic inspection of the containment heat removal surfaces, as addressed by GDC 39, assesses surface fouling or degradation that could potentially impede heat transfer from the containment vessel (CNV). Such inspections are sufficient to ensure the operability and performance of the containment heat removal function. Inspections and conformance with GDC 39 are discussed in FSAR Section 6.2.2.

Structural and leakage integrity of the CNV is addressed by GDC 50, 51, and 53. GDC 52 is subject to a separate exemption request (Part 7, Section 7). Testing and inspection for CNV integrity with respect to GDC 50, 51, and 53 is addressed in FSAR Section 6.2.1. 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 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 a design basis event, and does not create new accident precursors. The containment heat removal function does not have active components that require periodic pressure or functional testing to ensure operability and performance. 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. This exemption has no impact on plant security or safeguards procedures. Therefore, this 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 underlying purpose of GDC 40 is to provide the capability for testing to verify the operability and performance of the containment heat removal system. Containment heat removal is an inherent characteristic of the design, performed without reliance on electrical power, valve actuation, cooling water flow, or other active system or component operations. Operability and performance of the passive containment heat removal function is ensured by periodic inspections of the CNV. Therefore the design meets 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 GDC 40 for the US460 standard design.

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 (GDCs) 55, 56, and 57, as applied to several containment penetrations in the US460 standard design. GDCs 55, 56, and 57 specify containment isolation provisions for piping system lines penetrating primary containment, and generally require one isolation barrier inside containment and one outside containment. The underlying purpose of GDCs 55, 56, and 57 is to ensure reliable containment isolation capability. With respect to GDC 55 and 56, the 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 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 52.137(a)(3) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

(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;

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, in part:

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:*

(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.

10 CFR 50, Appendix A, GDC 56 states, in part:

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:

(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 GDC 55 for the lines with penetrations CNV6, CNV7, CNV13, and CNV14.

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.

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50, Appendix A, GDC 57 for the lines with penetrations CNV 1, CNV 2, CNV3, CNV4, CNV22, and CNV23.

9.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with GDCs 55, 56, and 57 for the specified lines that penetrate containment. The GDC 55 and GDC 56 lines subject to this exemption have two CIVs outside containment rather than locating one of the CIVs inside containment. The GDC 57 lines subject to this exemption use a closed system outside containment in lieu of a CIV.

9.2 Justification for Exemption

The underlying purpose of GDCs 55, 56, and 57 is to ensure reliable containment isolation capability to support the function of containment as an "essentially leak-tight barrier against the uncontrolled release of radioactivity." 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. GDCs 55, 56, and 57 prescribe specific containment isolation provisions for lines penetrating containment that are part of the reactor coolant pressure boundary (RCPB), connected directly to the containment atmosphere, or closed inside containment, respectively. To achieve that purpose, GDCs 55, 56, and 57 generally require redundant isolation barriers (a CIV or a closed system), physically separated by the primary containment boundary.

For the lines subject to GDCs 55 or 56 within scope of this exemption, the NuScale Power Module (NPM) includes redundant CIVs outside containment, with appropriate design provisions to ensure reliable isolation. For the lines subject to GDC 57 within scope of this exemption, the NPM relies on two closed systems as isolation barriers, which ensure reliable containment isolation without an isolation valve. Therefore, the containment isolation barriers satisfy the underlying purpose of GDCs 55, 56, and 57.

9.2.1 Technical Basis

GDC 55 and GDC 56 Lines

Lines penetrating containment that are part of the RCPB or connected directly to the containment atmosphere include two primary system containment isolation valves (PSCIVs) in series outside containment. As discussed in FSAR Section 6.2.4, each set of two PSCIVs shares a single valve body welded to a containment isolation test fixture valve, which is welded to a nozzle safe-end on the outside of the containment vessel (CNV). This design precludes CIV bypass due to a pipe break outside containment. 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 control rod drive system supply (CNV12) and return (CNV5) lines penetrate primary reactor containment and are neither part of the RCPB nor connected directly to containment atmosphere. However, the lines inside containment are not credited as isolation barriers and are conservatively treated as subject to GDC 56.

FSAR Section 6.2.4 further describes the design and quality provisions applied to the penetrations subject to GDCs 55 or 56. Although the lines subject to this exemption request are not part of an engineered safety feature system or required for safe shutdown, the isolation provisions otherwise meet the intent of the alternative allowable isolation provisions defined by NuScale Design-Specific Review Standard (DSRS) Section 6.2.4, acceptance criterion 4. Therefore, the isolation provisions meet the purpose of GDCs 55 and 56.

GDC 57 Lines

The lines with penetrations 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 (which isolates the main steam and feedwater lines), the lines comprise 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 DHRS outside, the steam generator system (SGS), and connecting piping.

Inside containment the SGS and connected piping comprise a closed system consistent with NuScale DSRS 6.2.4, acceptance criterion 15. As discussed in FSAR Section 5.4.1, a relief valve in the feedwater header for each steam generator protects against thermally induced overpressure. The relief valves discharge from the SGS to the inside of containment, are qualified for that purpose, and are designed to re-seat following the overpressure transient.

Outside containment the DHRS and connected piping comprise a closed system outside containment that functions as the second containment barrier. Consistent with the intent of GDC 57, the closed system serves as a redundant barrier to the closed system inside 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 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 Standard Review Plan Section 3.6.2. No single failure causes 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 GDCs or by regulatory guidance. Although not directly applicable to a GDC 57 line, the DHRS outside containment otherwise meets the criteria for a closed system outside containment as described in NuScale DSRS 6.2.4, acceptance criterion 5. Further, NuScale DSRS 6.2.4 acceptance criterion 1 recognizes the use of closed systems both inside and outside containment as acceptable alternate containment isolation barriers for instrument lines, and acceptance criterion 6 accepts other types of sealed-closed barriers in place of isolation valves. Therefore, the isolation provisions meet the underlying purpose of GDC 57.

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, this 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 NPM includes alternative isolation provisions that ensure reliable containment isolation capability, consistent with the purpose of GDCs 55, 56, and 57. 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 that is 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 regulations in the particular circumstances is not necessary to achieve the underlying purpose of the rules. The NPM includes isolation provisions for each line penetrating containment that are redundant and designed to ensure reliability. Therefore, the design meets the underlying purpose of the rules.

Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that, with respect to GDC 57, there are other material circumstances not considered when the rule was adopted. GDC 57 does 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 with 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 exemptions for the US460 standard design from GDCs 55, 56, and 57 as applied to the containment penetrations specified herein.

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 certain portions of 10 CFR Part 50, Appendix K ("Appendix K") regarding features that are required of the emergency core cooling system (ECCS) evaluation model (EM). The underlying purpose of Appendix K is to ensure that the ECCS EM conservatively calculates the consequences of postulated loss-of-coolant accidents (LOCAs). Certain phenomena addressed by Appendix K provisions are not encountered in design-basis LOCAs for the NuScale Power Module (NPM) and are not relevant to the ECCS EM. By precluding those phenomena during a LOCA, the ECCS EM remains a conservative method of calculating LOCA consequences without the features subject to this exemption. Therefore, the ECCS EM meets the underlying purpose of the rule.

10.1.2 Regulatory Requirements

10 CFR 52.137(a)(4) requires a standard design approval application final safety analysis report (FSAR) 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)(ii) states:

> *Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of 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.

10.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the requirements of paragraphs I.A.4, I.A.5, I.B, I.C.1.b, and I.C.5a of Appendix K.

10.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design ECCS EM does not conform with the requirements of 10 CFR 50 Appendix K identified in Section 10.1.3. Those features are excluded from the ECCS EM.

10.2 Justification for Exemption

The underlying purpose of Appendix K is to ensure that the ECCS EM conservatively calculates the consequences of postulated LOCAs from a spectrum of pipe break sizes and locations (design-basis LOCAs). The Appendix K requirements are based on the event progression and phenomena encountered during a LOCA for a traditional light water reactor design. NuScale's ECCS EM (referred to as the LOCA EM) only includes the phenomena and processes encountered in the NPM during design-basis accident conditions. Appendix K elements that are precluded by the design of the NPM are not modeled. By precluding those phenomena during a LOCA, the LOCA EM remains a conservative method of calculating LOCA consequences without the features subject to this exemption. The LOCA EM also uses an updated decay heat model versus that required by Appendix K. Therefore, the LOCA EM meets the underlying purpose of the rule.

10.2.1 Technical Basis

NuScale Topical Report TR-0516-49422 describes the LOCA EM for the analysis of design-basis LOCAs in the NPM. The topical report includes a description and sample calculations of LOCA 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 EM conservatively calculates the consequences of design-basis LOCAs and the underlying purpose of 10 CFR 50 Appendix K is met.

Exemption from the specific Appendix K required features subject to this exemption request is 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. Instead, the LOCA EM uses an implementation of Draft ANS-5.1/N18.5, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors," dated 1973 with 20 percent uncertainty added. 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 below the level where cladding oxidation occurs on the timescale of a LOCA for the NPM. Maintaining core coverage and avoiding critical heat flux (CHF) during design-basis LOCAs precludes the occurrence of significant transient cladding oxidation. Therefore, the LOCA EM excludes the required features of paragraph I.A.5.

Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters (I.B): Appendix K specifies requirements for predicting cladding swelling and rupture. Calculated cladding temperatures for design-basis LOCAs in the NPM are below the threshold for cladding swelling and rupture. Peak cladding temperature does not increase during a LOCA in the NPM. Therefore, the LOCA EM excludes the required features for predicting cladding swelling and rupture.

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 LOCA EM uses the single-phase critical flow model instead of the Moody model. The single-phase critical flow model is conservative and an acceptable alternative to the I.C.1.b 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 design-basis LOCAs, so heat transfer beyond CHF is not encountered. Therefore, the LOCA EM excludes the required features for post-CHF regimes.

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, this 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. Some features required by Appendix K are not relevant to the NPM LOCA EM. Other features of the LOCA EM are conservative as compared to Appendix K requirements. The LOCA EM conservatively models the processes and phenomena experienced by the NPM during a design-basis LOCA. 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 that are necessary to maintain the secure status of the plant. This exemption has no impact on plant security or safeguards procedures. Therefore, this 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 underlying purpose of Appendix K is to ensure that the ECCS EM conservatively calculates the consequences of postulated LOCAs. The Appendix K required features subject to this exemption are unnecessary because the design precludes the underlying phenomena or because the LOCA EM uses model features that are acceptable alternatives to those prescribed. Therefore, the LOCA EM meets the underlying purpose of Appendix K.

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 during design-basis LOCAs. 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 LOCA compared to the light water reactors underlying Appendix K required features.

10.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption from the identified provisions of Appendix K for the US460 standard design approval.

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) applicable to pressurizer level indicators. 10 CFR 50.34(f)(2)(xx) specifies power requirements for pressurizer relief valves, block valves, and level indicators. The underlying purpose of the rule is to enable natural circulation core cooling in a loss of offsite power condition. The US460 standard design does not rely on pressurizer level indication to achieve and maintain natural circulation in a loss of electric power condition, and therefore meets the underlying purpose of the rule.

The design does not include pressurizer relief valves or pressurizer block valves. Therefore, the portions of 10 CFR 50.34(f)(2)(xx) applicable to such valves are not technically relevant.

11.1.2 Regulatory Requirements

10 CFR 52.137(a)(8) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *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)(2) states, in part:

> *(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 [pressurized water reactors] 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.

The portions of 10 CFR 50.34(f)(2)(xx) applicable to pressurizer relief valves and pressurizer block valves are not technically relevant to the design, and therefore not within the scope of this exemption request.

11.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the design does not conform with the provisions of 10 CFR 50.34(f)(2)(xx) related to pressurizer level indicators. The design does not include pressurizer relief valves or pressurizer block valves.

11.2 Justification for Exemption

The underlying purpose of 10 CFR 50.34(f)(2)(xx) is to ensure design capabilities for enabling and maintaining natural circulation core cooling in a loss of offsite power condition. Per 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:

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.

The NuScale Power Module (NPM) is designed to automatically achieve and maintain natural circulation core cooling upon a loss of electric power. The pressurizer level instrumentation is not necessary for this function, thus the design achieves 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 decay heat removal system (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 electric power to actuate or operate. Because the NPM is designed to achieve and maintain natural circulation core cooling in the event of loss of electrical power, the underlying purpose of the rule is met.

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 does 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 US460 standard design does not rely on pressurizer level indication to achieve or maintain natural circulation cooling upon the loss of electric 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)). This 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 is not necessary to achieve the underlying purpose of the rule. The underlying purpose of 10 CFR 50.34(f)(2)(xx), to ensure the capability to achieve and maintain natural circulation core cooling upon loss of offsite power, is accomplished by passive design features that do not require electric power to operate or actuate.

11.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the US460 standard design approval from the portions of 10 CFR 50.34(f)(2)(xx) applicable to pressurizer level indicators. The portions of 10 CFR 50.34(f)(2)(xx) applicable to pressurizer relief valves and pressurizer block valves are not technically relevant to the design.

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 providing power supplies for pressurizer heaters and associated motive and control interfaces to establish and maintain natural circulation in hot standby conditions. The underlying purpose of the rule is to enable natural circulation core cooling in a loss of offsite power condition. The NuScale US460 standard design does not rely on pressurizer heaters to achieve and maintain natural circulation in a loss of electric power condition, and therefore meets the underlying purpose of the rule.

12.1.2 Regulatory Requirements

10 CFR 52.137(a)(8) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *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)(2) states, in part:

> *(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 [pressurized water reactors] only) (II.E.3.1)*

12.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50.34(f)(2)(xiii) in its entirety.

12.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the design does not conform with 10 CFR 50.34(f)(2)(xiii).

12.2 Justification for Exemption

The underlying purpose of 10 CFR 50.34(f)(2)(xiii) is to ensure design capabilities for enabling and maintaining natural circulation core cooling in a loss of offsite power condition. Per 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:

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.

The NuScale Power Module (NPM) is designed to maintain natural circulation core cooling upon a loss of electric power. The pressurizer heater is not necessary for this function, thus the design achieves 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 decay heat removal system (DHRS) removes post-reactor trip residual and core decay heat from the core during operating conditions and transitions the NPM to safe shutdown conditions without reliance on the pressurizer heaters. The DHRS design, as discussed in FSAR Section 5.4.3, is a passive system and does not require electric power to actuate or operate. Because the NPM is designed to achieve and maintain natural circulation core cooling in the event of loss of electric power, the underlying purpose of the rule is met.

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 design does not rely on pressurizer heaters to achieve or maintain natural circulation cooling upon the loss of electric 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)). This 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 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 underlying purpose of 10 CFR 50.34(f)(2)(xiii), to ensure the capability to achieve and maintain natural circulation core cooling upon loss of offsite power, is accomplished by passive design features that do not require electric power to operate or actuate.

12.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption from 10 CFR 50.34(f)(2)(xiii) for the US460 standard design approval.

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 other plant parameters may not generate a containment isolation signal during an accident sequence. The NuScale Power Plant design meets the purpose of the rule by ensuring isolation of CES upon any event involving radiological consequences inside of the containment vessel (CNV). Because alternate means to prevent radiological release from the CES to the environs are provided, 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 52.137(a)(8) requires a standard design approval application final safety analysis report to include, in part:

> *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)(2) states, in part;

> *(xiv) Provide containment isolation systems that…(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 CES of the NuScale Power Plant US460 standard design will not conform with 10 CFR 50.34(f)(2)(xiv)(E). Automatic isolation of CES is initiated by parameters other than high radiation.

13.2 Justification for Exemption

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 during events where other parameters may not generate a containment isolation signal. The discussions related to 10 CFR 50.34(f)(2)(xiv)(E) within NUREG-0578, NUREG-0660, and NUREG-0737 address isolation of containment purge and vent systems, which provide a path to the environs. NRC identified a need to initiate isolation of these systems in core damage scenarios with radiological release into the containment, but that may not initiate timely containment isolation--specifically, a sequence where containment pressure does not reach the isolation setpoint for a small loss of coolant flowrate into containment.

The design satisfies the intent of 10 CFR 50.34(f)(2)(xiv)(E) by automatically initiating a containment isolation signal from several monitored parameters, including pressurizer level. Any event that could lead to core damage will initiate a containment isolation signal prior to core damage. Therefore, CES isolation upon a high radiation signal is not required in core damage scenarios to prevent a direct release path to the environs.

13.2.1 Technical Basis

The design differs from the large light water reactor designs considered in development of the TMI action plan because core damage cannot occur without generating a containment isolation signal due to pressurizer level. In the design, the pressurizer is integral to the upper region of the reactor pressure vessel and is located well above the level of the reactor core. Any decrease in reactor vessel inventory to the level of the reactor core would necessitate emptying of the pressurizer.

FSAR Chapter 15 demonstrates that all design basis events meet their radiological release acceptance criteria without isolating CES on a high radiation signal. None of these events result in degraded or damaged core conditions.

FSAR Section 19.2 demonstrates the design cannot suffer core damage without the preceding generation of a containment isolation signal due to pressurizer level. FSAR Table 19.2-2 summarizes the status of mitigating systems for each of the core damage simulations. FSAR Tables 19.2-4 through 19.2-10 show all severe accident events result in a containment isolation 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.

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 isolating CES on a high radiation signal.

13.3 Regulatory Basis

13.3.1 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 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 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)). 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 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 the design includes alternate means to isolate CES prior to any event leading to core damage or degradation.

13.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the US460 standard design from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES.

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 10 CFR 50.46 concerning zircaloy or ZIRLO as acceptable fuel rod cladding materials. The NuScale Power Module (NPM) fuel design uses Framatome's M5® zirconium alloy for the fuel rod cladding material. The underlying purpose of 10 CFR 50.46 is to ensure that a light-water reactor's emergency core cooling system (ECCS) performs adequately in the event of a loss-of-coolant accident (LOCA). 10 CFR 50.46 implies that only zircaloy or ZIRLO are to be used as the fuel rod cladding material. M5® is an alternative cladding material and, when evaluated pursuant to 10 CFR 50.46, achieves the underlying purpose of the rule.

14.1.2 Regulatory Requirements

10 CFR 52.137(a)(4) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *Analysis and evaluation of ECCS cooling performance … following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of 10 CFR 50.46…*

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 [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 requests an exemption from 10 CFR 50.46 as it concerns the use of zircaloy or ZIRLO as fuel rod cladding materials. 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 (SDA application Part 7, Section 10).

14.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale US460 standard design does not conform with the fuel rod cladding materials specified by 10 CFR 50.46. The design uses M5® fuel rod cladding material.

14.2 Justification for Exemption

The underlying purpose of 10 CFR 50.46 is to ensure that nuclear power facilities have an ECCS with adequately demonstrated cooling performance by meeting defined acceptance criteria with an acceptable evaluation model. Evaluation of Advanced Cladding and Structural Material (M5®) in PWR Reactor Fuel (Reference 14.5-1) demonstrates that the 10 CFR 50.46(b) ECCS acceptance criteria are acceptable for reactors using M5® cladding material. Reference 14.5-2 and Reference 14.5-3 demonstrate the applicability of Reference 14.5-1 to the NPM 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. SDA application Part 7, Section 10 requests an exemption from 10 CFR 50 Appendix K Paragraph I.A.5 because it is not applicable to NuScale's ECCS evaluation model.

14.2.1 Technical Basis

Reference 14.5-1, Section 4.2, demonstrates that the effectiveness of the ECCS is not 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. FSAR Section 4.2 evaluates the use of M5® cladding in the NPM fuel system. Therefore, 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.

Reference 14.5-1 demonstrates that operation with M5® fuel rod cladding does not increase the probability of occurrence or the consequences of an accident, and 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)]. 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 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 underlying purpose of 10 CFR 50.46 is to ensure that nuclear power facilities have an ECCS with adequately demonstrated cooling performance. 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 US460 standard design approval to allow the use of M5® fuel rod cladding material.

14.5 References

14.5-1 AREVA Inc., "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," BAW-10227P-A, Revision 1, June 2003.

14.5-2 NuScale Power, LLC, "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-P-A, Revision 1.

14.5-3 NuScale Power, LLC, "Framatome Fuel and Structural Response Methodologies Applicability to NuScale," TR-108553-P-A, Revision 0.

15. 10 CFR 50.61 Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

15.1 Introduction and Request

15.1.1 Summary

NuScale Power LLC, (NuScale) requests an exemption from 10 CFR 50.61, which provides fracture toughness requirements to protect against pressurized thermal shock (PTS) events. The underlying purpose of 10 CFR 50.61 is to prevent conditions that could result in reactor pressure vessel (RPV) failure in the event of a PTS event in a pressurized water reactor (PWR), wherein severe overcooling in conjunction with significant reactor vessel pressurization occurs that could, in the presence of an initiating flaw, cause brittle fracture of the RPV. The NuScale Power Module (NPM) lower RPV materials are made of austenitic stainless steel and do not contain copper; the chemistry factor tables in 10 CFR 50.61 that are used in calculating the PTS screening criterion (RT_{PTS}) do not apply to the lower RPV materials. Therefore, the screening methodology in 10 CFR 50.61 cannot be used for austenitic stainless steel. As an alternative, NuScale evaluated data on irradiated austenitic stainless steel to conclude that the effects of neutron and thermal embrittlement are minor even when these materials are exposed to a fluence level exceeding the design life peak fluence of the NPM lower RPV. The reduced susceptibility of the lower RPV materials to the effects of neutron and thermal embrittlement increases the integrity and safety of the RCPB. Because the lower RPV, which contains the beltline, cannot be screened for PTS events, the upper RPV made of ferritic steel is evaluated. Together these considerations satisfy the underlying purpose of 10 CFR 50.61 without using the rule's methodology that was developed for ferritic RPV materials.

15.1.2 Regulatory Requirements

10 CFR 52.137(a)(14) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in 10 CFR 50.60 and 50.61.*

10 CFR 50.61(b)(1) states in part:

> *…For pressurized water nuclear power reactors for which a construction permit is issued under this part after February 3, 2010 and whose reactor vessel is designed and fabricated to an ASME Code after the 1998 Edition, or for which a combined license is issued under Part 52, the projected values [of RT_{PTS} or RT_{MAX-X}] must be in accordance with this section. When determining compliance with this section, the assessment of RT_{PTS} must use the calculation procedures described in paragraph (c)(1) and perform the*

evaluations described in paragraphs (c)(2) and (c)(3) of this section. The assessment must specify the bases for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation for each beltline material. This assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility.

10 CFR 50.61(b)(2) states:

The pressurized thermal shock (PTS) screening criterion is 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be evaluated according to the procedures of paragraph (c) of this section, for each weld and plate, or forging, in the reactor vessel beltline. RT_{PTS} must be determined for each vessel beltline material using the EOL fluence for that material.

15.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50.61 in its entirety.

15.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the US460 standard design does not conform with 10 CFR 50.61. The RT_{PTS} is not calculated for, and the PTS screening criterion is not applied to, the NPM lower RPV materials. The methodology of 10 CFR 50.61 is intended for ferritic materials, which are not present in the RPV beltline region.

15.2 Justification for Exemption

The underlying purpose of 10 CFR 50.61 is to prevent conditions that could result in RPV failure in the event of a PTS event in a PWR, wherein severe overcooling in conjunction with significant reactor vessel pressurization occurs that could, in the presence of an initiating flaw, cause brittle fracture of the RPV. Evaluating materials in the RPV for a PTS event reduces the risk of abnormal leakage, of rapidly propagating failure, and of gross rupture of the reactor coolant pressure boundary (RCPB) (pursuant to General Design Criteria 14 and 31); and ensures that the reactor coolant system and associated auxiliary, control, and protection systems contain sufficient margin to assure that design conditions of the RCPB are not exceeded during operation (pursuant to General Design Criterion 15).

To meet the underlying purpose of the rule, the NPM lower RPV is made of austenitic stainless steel; the 10 CFR 50.61 PTS screening methodology is not applicable to the chemical composition of non-ferritic materials. NuScale evaluated data on irradiated austenitic stainless steel to conclude that the effects of neutron and thermal embrittlement are minor when these materials are exposed to a fluence level exceeding the design life peak fluence of the NPM design. The austenitic stainless steel used in the lower RPV is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. Upper RPV materials, which are outside the beltline, would not require PTS screening because the design life peak fluence is less than the threshold of 10^{17} n/cm^2 (E > 1 MeV). Therefore, the NPM design satisfies the underlying purpose of 10 CFR 50.61 by protecting against PTS events using lower RPV materials with superior ductility and reduced risk of the effects of neutron and thermal embrittlement.

15.2.1 Technical Basis

10 CFR 50.61 requires PTS screening for the RPV beltline region of PWRs. A PTS event is an event or transient in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV. The beltline portion of the RPV is the limiting part when evaluating PTS events; RT_{PTS} is to be calculated for each beltline material considering EOL Fluence (a defined term).

The RT_{PTS} calculation uses the acceptance criteria in 10 CFR 50.61(b)(2). The RT_{PTS} value calculations use chemistry factors in Table 1 and Table 2 of 10 CFR 50.61 that are based on copper and nickel content to identify the susceptible material. The RT_{PTS} evaluations are based on American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III, Paragraph NB-2331, along with data based only on the chemical composition of ferritic materials. ASME BPVC Section III, Paragraph NB-2331, follows ASME BPVC Section III, Paragraph NB-2311, which specifically excludes austenitic stainless steel from impact test requirements. There are no impact test requirements for austenitic stainless steels in ASME BPVC Section III, Paragraph NB-2311, because these materials do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials used for ASME BPVC Section III Class 1 pressure-retaining components. The NRC endorsed the ASME BPVC 2017 version in 10 CFR 50.55a.

The PTS screening methodology in 10 CFR 50.61 uses a calculated RT_{PTS}. The calculated RT_{PTS} is the nil-ductility reference temperature evaluated for end of design life peak fluence for each RPV beltline material using Equation 4 in 10 CFR 50.61. Equation 4 of 10 CFR 50.61 uses $RT_{NDT(U)}$, which is the nil-ductility reference temperature in an unirradiated condition, and then derates that reference temperature through a series of fracture toughness equations that are based on the chemistry factors and irradiation levels applicable to the beltline material. A portion of the fracture toughness of the material is established through impact testing according to ASME BPVC Section III, Paragraph NB-2311. Because ASME BPVC Section III, Paragraph NB-2311 does not require impact testing for austenitic stainless steel, the methodology in 10 CFR 50.61 cannot be used for austenitic stainless steel.

Therefore, the 10 CFR 50.61 PTS screening methodology is not applicable to, and cannot be used for, the austenitic stainless steel in the NPM lower RPV. Instead, the lower RPV, which includes the beltline, is protected against PTS events by using materials less susceptible to the effects of neutron embrittlement as discussed below. Although outside the beltline, NuScale considered the ferritic upper RPV under the 10 CFR 50.61 methodology. However, the 57 effective full-power year (EFPY) peak fluence for the top of the lower flange surface of the lower RPV is less than the 10 CFR 50, Appendix H surveillance criterion of 10^{17} n/cm^2 (E > 1 MeV). Therefore, the upper RPV is not subject to embrittlement concerns.

Because the PTS screening methodology does not apply to the NPM lower RPV, NuScale evaluated total elongation data post-irradiation to conclude that the effects of neutron embrittlement of solution-annealed FXM-19 is minor for a fluence level that is 28 percent higher than the design life peak fluence of the NPM design. NuScale also evaluated mill-annealed FXM-19, which remained highly ductile post-irradiation when subjected to a fluence level that is 94 percent higher than the design life peak fluence for the NPM design. Finally, NuScale evaluated fracture toughness test results of mill-annealed FXM-19 post-irradiation to conclude that there is minor reduction in average plane-strain fracture toughness values at a fluence level that is over 200 percent higher than the design life peak fluence for the NPM design.

NuScale also evaluated FXM-19 and associated weld materials (SFA 5.4 E209 or E240, and SFA 5.9 ER209 or ER240) against Type 3XX austenitic stainless steels. Two publications proposed maximum threshold fluence levels for austenitic stainless steel base metal and weld metal, both of which are greater than the design life peak fluence for the NPM design. Therefore, since the design life peak fluence for the NPM design is well below the threshold fluences, the embrittlement effects are minimal for FXM-19 and the associated weld materials.

The data and evaluations supporting this technical basis are in NuScale technical report TR-130721, “Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel.”

The available data for austenitic stainless steel supports the conclusion that austenitic stainless steel is highly ductile and less susceptible to the effects of neutron and thermal embrittlement than ferritic materials. Together with consideration for PTS events on the upper RPV, the NPM design satisfies the underlying purpose of 10CFR 50.61 without applying a PTS screening criterion to the beltline materials in the lower RPV.

15.3 Regulatory Basis

15.3.1 Criteria of 10 CFR 50.61, 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, this 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 NPM design uses a material for the lower RPV that is less susceptible to the effects of neutron and thermal embrittlement than the ferritic materials used in the operating light water reactor fleet. The RPV design thus has a lower risk of abnormal leakage, rapidly propagating failure, gross rupture, and fracture of the RCPB, which is a boundary that protects the public from radiation release. Therefore, exemption from beltline PTS screening pursuant to 10 CFR 50.61 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)). This 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. This exemption has no impact on plant security or safeguards procedures. Therefore, this 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 US460 standard design provides adequate protection to PTS events because the effects of neutron and thermal embrittlement are not a concern for austenitic stainless steel in the lower RPV. The provisions of 10 CFR 50.61 requiring PTS screening of the lower RPV materials are therefore not necessary 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 that were not considered when the regulation was adopted. 10 CFR 50.61 establishes requirements for ferritic materials in the RPV because the population of nuclear power plant RPVs were composed of these materials at the time the regulation was promulgated. Thus the regulation does not address an RPV made of a different material. The ferritic RPVs considered during the 10 CFR 50.61 rulemaking and their responses to the effects of neutron and thermal embrittlement differ significantly from the NPM design. The NPM design includes a more robust material that sufficiently reduces the risk of abnormal leakage, rapidly propagating failure, gross rupture, fracture, or breach of the RPV. Therefore, it is in the public interest to grant an exemption from the PTS screening requirements in 10 CFR 50.61.

15.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the US460 standard design approval from 10 CFR 50.61 requiring PTS screening for RPV materials.

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), which requires certain capabilities 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 US460 standard 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. Use of these alternative means benefits public health and safety by reducing operator dose, preventing the spread of contamination, and reducing the potential for radioactive leaks and spills.

16.1.2 Regulatory Requirements

10 CFR 52.137(a)(8) requires a standard design application FSAR to include, in part:

> *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;

> *. . .*
>
> *(viii)Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term[11] radioactive materials 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 10 CFR 50.34(f)(2)(viii) in its entirety.

16.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the design does not conform with 10 CFR 50.34(f)(2)(viii).

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 aid 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, 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 (ALWR) Designs." In the SRM for SECY-93-087, 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, plant owners' groups submitted topical reports for NRC review to eliminate PASS requirements. As stated by the Combustion Engineering (CE) Owners Group in CE NPSD-1157, Revision 1:

> *[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.*

NRC 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, 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, in the US460 design the information that could be obtained from post-accident sampling is not necessary for accident management and emergency response, because the design allows for sufficient information collection 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 US460 design, this capability is provided by radiation monitors under the bioshield and by core exit thermocouples. The 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.

In major accident scenarios, including core damage events, the NuScale Power Module (NPM) is designed to preserve primary coolant inventory and contain the potential post-accident source term by isolating containment. The process of taking a sample from the primary coolant or containment would require unisolating containment and extracting potentially radioactive post-accident material to the outside of containment. In lieu of such a process, the design relies upon other means to indicate the presence of core damage, namely radiation monitors under the bioshield and core exit thermocouples. This design philosophy results in a lower potential for facility contamination and personnel radiation exposure. The specific sampling capabilities required by 10 CFR 50.34(f)(2)(viii) are addressed below.

<u>Primary Coolant Dissolved Gases (Including Hydrogen):</u>

The NPM is insusceptible to an accumulation of noncondensable gases interfering with post-accident natural circulation (Part 7, Section 1, and FSAR Section 5.4.4). Therefore, grab sampling of reactor coolant for dissolved gas analysis is unnecessary to ensure post-accident natural circulation capability.

<u>Containment Hydrogen and Oxygen:</u>

The NPM has features that support containment hydrogen and oxygen monitoring using the process sampling system (PSS) during normal operations (FSAR Section 9.3.2). The design precludes a combustible atmosphere following a beyond design basis event by using a passive autocatalytic recombiner to limit oxygen concentration. Therefore, sampling of containment hydrogen and oxygen is unnecessary to ensure containment integrity. FSAR Section 6.2.5 contains additional information on combustible gas control capability.

Primary Coolant Chlorides:

The purpose of sampling the reactor coolant for chlorides is to ensure that chloride-induced stress corrosion cracking of stainless steel components will not occur post-accident in the long term. As opposed to typical light water reactors, the NPM design does not employ automatic safety injection or other coolant makeup, and does not utilize large quantities of chlorinated cable insulation inside containment. Therefore, the amount of reactor coolant chlorides during ECCS recirculation remains unchanged and post-accident reactor coolant sampling for chlorides is unnecessary.

Primary Coolant Boron Concentration:

The purpose of sampling the reactor coolant for boron is to ensure that there is adequate shutdown margin to maintain safe shutdown during long term emergency cooling. The capability to ascertain the RCS boron concentration is important where makeup water, other than the original reactor coolant inventory, is used to refill the reactor vessel or to flood the containment during an accident. Because the NPM design does not employ automatic safety injection or other coolant makeup, the total boron concentration in the primary coolant does not decrease. FSAR Section 15.0.5 addresses long-term boron concentration and reactor shutdown capability. Therefore, post-accident boron sampling is not necessary.

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 is not breached during an accident, or to assess the degree of core damage if cladding is breached. The capability to measure reactor coolant radionuclides also supports the Emergency Action Level (EAL) classification in the Site Emergency Plan. The design utilizes radiation monitors under the bioshield and core exit thermocouples to assess core damage. Therefore, post-accident radionuclide content sampling is not necessary.

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 design-basis event, and does not create new accident precursors. The information that would available by post-accident sampling is provided by other means or not necessary for the design; the exemption avoids unisolating containment unnecessarily. 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 is not necessary to achieve the underlying purpose of the rule. 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 and asses system conditions following an accident. The 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. Other post-accident sampling capabilities are not necessary tor the design. Therefore, compliance with 10 CFR 50.34(f)(2)(viii) is not necessary to achieve the underlying purpose of the rule.

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. Analysis of post-accident samples for other variables that indicate presence of core damage is unnecessary for the design. By maintaining containment isolation as the preferred accident response, the spread of potentially highly radioactive material to systems outside of the NPM is prevented, avoiding unnecessary operator dose, preventing the spread of contamination to systems outside of the NPM, and reducing the potential for leaks and spills that could result in additional dose to the public.

16.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption from 10 CFR 50.34(f)(2)(viii) for the US460 standard design approval.

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 the portion of General Design Criterion (GDC) 19 requiring the capability to achieve cold shutdown from equipment outside the control room. The underlying intent of the remote shutdown requirements of GDC 19 is to ensure means for operators to place and maintain the reactor in a safe condition in the event of a control room evacuation. The US460 standard design does not require the capability to establish cold shutdown from outside the main control room (MCR) to ensure a long-term safe, stable condition following MCR evacuation. The design conforms to a principal design criterion ensuring the capability for safe shutdown in the event of MCR evacuation, thereby meeting the underlying purpose of the rule.

17.1.2 Regulatory Requirements

10 CFR 52.137(a)(3) requires a standard design approval application final safety analysis report (FSAR) to include, in part:

> *(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;*

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 19 states, in part:

> *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.

17.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the portion of GDC 19 that requires equipment outside the control room providing "a potential capability for subsequent 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 design conforms to Principal Design Criterion (PDC) 19, as set forth in FSAR Section 3.1. PDC 19 requires the capability for safe shutdown from equipment outside the control room, in lieu of GDC 19's requirements for "design capability for prompt hot shutdown" and "potential capability for subsequent cold shutdown." PDC 19 also clarifies the requirements for control room radiation protection consistent with GDC 19.

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 plant safely under normal conditions and to maintain it in a safe condition under accident conditions, while providing adequate radiation protection for operators. Additionally, GDC 19 requires capabilities for remote shutdown: "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 requirements is to provide means for maintaining the reactor in a safe condition in the event of a control room evacuation. As originally proposed, GDC 19 (proposed as GDC 11), would have required the capability "to shut the reactor down and maintain it in a <u>safe condition</u> 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 proposed requirement could be interpreted to require a second control room (SECY-R 143, January 28, 1971). In response, NRC clarified final GDC 19 to separate 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 basic intent of maintaining a "safe condition." NRC guidance is consistent with the interpretation that the "cold shutdown" language was not directed at a particular reactor coolant system temperature, but rather a safe, stable shutdown condition in the event of long-term control room unavailability. 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 <u>safe shutdown</u> of the plant."

Therefore, 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 shutdown condition. Thus, the ability to maintain the reactor in a long-term safe shutdown condition in the event of control room evacuation satisfies the underlying purpose of GDC 19.

17.2.1 Technical Basis

The design addresses GDC 19's intent with respect to control room evacuation in two ways. First, the MCR is designed with the ability to place and maintain the reactors in safe shutdown in the event of an MCR evacuation event. As described in FSAR Sections 7.1.1 and 7.2.2, prior to evacuating the MCR, operators trip the reactors, initiate decay heat removal, and initiate containment isolation. These actions result in passive cooling that achieves and maintains safe shutdown of the reactors. Second, operators can also achieve safe shutdown of the reactors from outside the MCR in the instrumentation and controls (I&C) equipment rooms within the Reactor Building.

Following shutdown and initiation of passive cooling from either the MCR or the I&C equipment rooms, the design does not rely on operator action, instrumentation, or controls outside of the MCR to maintain safe shutdown. The design includes alternate operator work stations in various locations that allow operators to monitor the modules in a safe shutdown condition.

PDC 19's requirement for 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 Section 7.0:

> *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 includes the footnote:

> *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)).*

Therefore, by requiring the "equipment at appropriate locations outside the control room … with a design capability for safe shutdown of the reactor," PDC 19 satisfies the underlying purpose of GDC 19's remote shutdown provisions. Additional changes to PDC 19 are incorporated to improve clarity of the design criterion.

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 design-basis events, and does not create new accident precursors. The design includes the ability to achieve and maintain long-term a safe shutdown condition from outside the MCR. Therefore, this 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. This exemption has no impact on plant security or safeguards procedures. Therefore, this 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 underlying purpose of the remote shutdown provisions of GDC 19 is to ensure 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. Therefore, conformance with PDC 19 achieves the underlying purpose of the rule.

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 would require NuScale to incorporate additional features, increasing complexity of the design. 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 condition for the design, there is no 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 US460 standard design approval from the portion of GDC 19 requiring the capability to achieve cold shutdown from equipment outside the control room.

18. 10 CFR 50.46 and GDC 35, LOCA Break Spectrum

18.1 Introduction and Request

18.1.1 Summary

NuScale Power, LLC (NuScale) requests an exemption from the requirement of 10 CFR 50.46(a)(1)(i) that "the most severe postulated loss-of-coolant accidents are calculated." This rule requires loss-of-coolant accidents (LOCAs) of different sizes, locations, and other properties be postulated to ensure the "most severe" LOCA is evaluated in the "acceptable evaluation model," where LOCAs are breaks in pipes in the reactor coolant pressure boundary (RCPB). Using the acceptable evaluation model, an applicant must demonstrate specified emergency core cooling system (ECCS) performance criteria are met for the spectrum of postulated LOCAs.

The design evaluates a variety of postulated LOCAs using an ECCS evaluation model (ECCS EM) developed in conformance with 10 CFR 50 Appendix K, with exemptions. Those postulated LOCAs include the largest reactor coolant pressure boundary (RCPB) pipe inside containment, for which adequate ECCS performance is demonstrated. However, several potential LOCA locations are excluded from the LOCA break spectrum: (1) the connection between each of the ECCS valves and the reactor pressure vessel (RPV), and (2) components and component connections between the containment vessel (CNV) and the second containment isolation valve (CIV) in each of the chemical and volume control system (CVCS) lines outside the CNV. Although there is no pipe in these locations, the excluded break locations include piping system components and their connections that precedent indicates are to be considered as potential LOCAs within the scope of 10 CFR 50.46.

This exemption implements a design-specific, risk-informed approach in excluding those break locations, similar to the NRC's incomplete effort to risk-inform 10 CFR 50.46. Each of the excluded break locations are specified to rigorous design, quality, and inspection standards to minimize their likelihood of failure. The effect of this exemption is to exclude these break locations from the scope of design-basis events (DBEs). The excluded break locations are instead addressed as beyond-design-basis events (BDBEs) in the US460 standard design licensing basis. Breaks postulated within the excluded break locations are analyzed to demonstrate adequate core cooling conservatively but more realistically than DBE LOCA analyses. The BDBE breaks are shown to not challenge containment design limits. Additionally, nonsafety-related means are available to mitigate the breaks. The foregoing is sufficient to assure very low risk and maintain defense-in-depth, but as additional assurance the BDBE break locations are also shown to yield offsite doses less than the limits prescribed by 10 CFR 52.137(a)(2)(iv).

The performance and evaluation requirements of 10 CFR 50.46 implement the general ECCS requirements of General Design Criterion (GDC) 35. Although the US460 standard design is subject to a design-specific principal design criterion (PDC) 35 in lieu of GDC 35, this exemption request addresses departure from GDC 35 in the selection of the LOCA break spectrum.

Because postulating the excluded break locations as DBE-LOCAs pursuant to 10 CFR 50.46 and GDC 35 is not necessary to reasonably assure adequate protection or to meet the underlying intent of the rules, NuScale requests an exemption from those rules as applied to the excluded break locations.

18.1.2 Regulatory Requirements

10 CFR 52.137(a)(4) requires a standard design approval application FSAR to include, in part:

An analysis and evaluation of the design and performance of SSC 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 SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of 10 CFR 50.46 and 50.46a.

10 CFR 50.46(a)(1)(i) requires, 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. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.

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

Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.

10 CFR 50.46(c)(1) provides:

[LOCAs] are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

10 CFR 50.2 provides:

Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the reactor coolant system, or

(2) Connected to the reactor coolant system, up to and including any and all of the following:

(i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,

(ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,

(iii) The reactor coolant system safety and relief valves.

10 CFR 52.137(a)(3) requires a standard design approval application FSAR to include, in part:

(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;

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 35 provides:

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.

18.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from 10 CFR 50.46(a)(1)(i) and GDC 35 to exclude from the LOCA break spectrum certain postulated locations described herein.

The NuScale US460 standard design implements a design-specific PDC 35 via exemption request in Part 7 Section 4 to address the electrical power provisions of GDC 35. This exemption request justifies an additional departure from GDC 35 in the implementation of PDC 35 with respect to the LOCA break spectrum.

18.1.4 Effect on Regulatory Compliance

As a result of this exemption, the US460 standard design excludes certain RCPB locations from the scope of LOCAs subject to GDC 35 and 10 CFR 50.46 requirements (the LOCA break spectrum). The excluded break locations are:

- Each connection between each ECCS main valve (i.e., the reactor vent valves (RVVs) and reactor recirculation valves (RRVs)) and the RPV, and
- Each connection and component body in each CVCS line between and including the connection to the CNV nozzle and the second primary system containment isolation valves (PSCIV).

Rather than DBEs events analyzed with the Appendix K-based ECCS EM to meet the performance criteria of 10 CFR 50.46(b), certain breaks in the excluded locations are postulated as BDBEs in the licensing basis. Event-specific acceptance criteria, derived from existing regulatory criteria for DBEs, are applied to the BDBE breaks in order to demonstrate reasonable assurance of adequate protection.

18.2 Justification for Exemption

This exemption addresses excluded break locations of two types. First are the connections of the ECCS valves to the RPV, which are flanged connections inside the CNV. Although there is no pipe in between the RPV and the ECCS valve bodies, the flanged connections are treated, according to precedent, as piping components within the definition of LOCA for 10 CFR 50.46. If an ECCS valve is postulated to break free of the RPV, the RCS blowdown rate through the break could somewhat exceed flow through an inadvertent actuation of the same valve (inadvertent ECCS valve opening is analyzed as an anticipated operational occurrence in the licensing basis). However, that blowdown would be retained within the CNV and recirculated through the reactor core upon successful actuation of the ECCS. Thus, these excluded breaks are different only in potential fluid flow rate from LOCAs within the design basis, not in plant response.

The second type of excluded breaks are the group of CVCS components and their connections outside the CNV. Here also there is no pipe, but excluded locations would otherwise be treated by precedent as piping components within the scope of LOCAs for 10 CFR 50.46. The connections in these excluded break locations are not larger than the same piping systems inside containment that are analyzed as LOCAs. However, the break flow would occur outside containment and potentially challenge the ordinary LOCA response by losing inventory that cannot be recirculated by ECCS.

18.2.1 Design Basis

The NuScale Power Module (NPM) includes a steel CNV that is partially submerged in the ultimate heat sink pool. The RPV is located within the CNV, and includes integral ECCS valves. The ECCS includes four main valves; two RVVs located on the upper RPV, and two RRVs located on the lower RPV. Each ECCS main valve is a solenoid pilot-operated relief valve that is hydraulically closed, spring-assist to open, normally closed, and fails open. Upon ECCS actuation, the RVVs open, allowing steam to vent out of the upper RPV. The steam condenses on the inner surface of the partially submerged CNV, flows to the lower CNV, and eventually flows back into the RPV through the RRVs. This cycle allows continual, passive recirculation that does not require electric power nor operator action.

The CNV includes several containment penetrations for process fluids, electrical power supply, access penetrations, etc. Process fluid penetrations that extend beyond the containment boundary include CIVs. The CIVs include passive stored energy that allows containment isolation in the event of an actuation signal or loss of power. There are four process fluid penetrations that extend beyond the containment boundary and are connected to the reactor coolant pressure boundary (RCPB), each of which are associated with CVCS. The CVCS penetrations are located on the CNV upper head. Part 7 Section 9 includes an exemption request for GDC 55 for these four lines. As discussed in the exemption request and FSAR Section 6.2.4, lines penetrating containment that are part of the RCPB include two PSCIVs in series outside containment. The PSCIVs on the CVCS lines satisfy the intent of GDC 55 for isolation of lines penetrating the containment that are part of the RCPB. Each set of two PSCIVs share a single valve body that is welded to a containment isolation test fixture (CITF), which is welded to a nozzle safe-end on the outside of the CNV. This approach removes the need to locate a hydraulically-operated valve inside of containment (including the post-accident atmosphere in the CNV). Thus, the primary break areas of concern are the weld between the CNV nozzle and safe-end, the weld between the safe-end and the CITF, and the weld between the CITF and PSCIV body. Notwithstanding, the technical basis and break exclusion discussed below are applicable to components and component bodies in the area of concern (e.g., the CITF body). The CVCS penetrations are located on the CNV upper head.

18.2.2 Regulatory History and Risk-Informed Approach Overview

GDC 35 requires that an ECCS be provided in light-water reactors to mitigate "any loss of reactor coolant." Thus, ECCS is provided as a layer of defense-in-depth for losses of reactor coolant. 10 CFR 50.46 was enacted after and in specific implementation of GDC 35; it imposes strict, conservative analysis methods and acceptance criteria for the demonstration of ECCS performance over a range of postulated LOCAs.

ECCS is intended to prevent LOCAs from progressing to core melt accidents. As described in the proposed rule "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" (70 FR 67,598), the ECCS requirements are part of a deterministic regulatory approach that "assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs) for which specified acceptance criteria must be satisfied. Each

DBE encompasses a spectrum of similar but less severe accidents. The deterministic approach then requires that the licensed facility include safety systems capable of preventing and/or mitigating the consequences of those DBEs to protect public health and safety." While these regulations are deterministic, they contain "implied elements of probability (qualitative risk considerations), from the selection of accidents to be analyzed to the system level requirements for emergency core cooling (e.g., safety train redundancy and protection against single failure)."

Early power reactors were required to have long-term core cooling to help mitigate LOCAs, but it was assumed that a large LOCA would cause fuel melting. Rather than preventing core damage, the third layer of defense-in-depth—containment—was the primary means for addressing design-basis LOCAs. By maintaining an integral containment at a design-basis leak rate and reducing fission product inventory available to leak, reactor licensees could demonstrate acceptable doses at the site boundary if the assumed core melt were to occur. "The earliest commercial reactor containments were designed to confine the fluid release from a double-ended guillotine break (DEGB) of the largest pipe in the reactor coolant system (RCS)" (70 FR 67,599).

As reactor sizes grew, concern arose that core melting during a LOCA could result in containment basemat melt-through, causing a failure of the containment and loss of the primary means for mitigating large LOCAs. This concern resulted in the GDC 35 requirement for a reliable, high-capacity ECCS to prevent design-basis LOCAs from progressing to core melt accidents. Here again, a DEGB of the largest RCS pipe was deterministically assumed so as to encompass a spectrum of less severe accidents. Implied probabilistic judgements excluded even more severe accidents (e.g., catastrophic RPV failures) from the ECCS and containment design bases.

After the GDCs were finalized, the Commission promulgated 10 CFR 50.46 and Appendix K to specify, respectively, detailed acceptance criteria for ECCS performance and required and acceptable features for ECCS EMs. Consistent with the deterministic regulatory approach, the "the causes, probability and [radiological] consequences of a LOCA" were not considered in the rulemaking, "which started with the assumption that the highly unlikely LOCA had occurred, for whatever reason" (CLI-73-39, 6 AEC 1085 at 1087). The ECCS acceptance criteria and EM requirements contain considerable margin reflecting the uncertainty in LOCA phenomena and ECCS performance at the time of the rulemaking.

In one effort to address EM conservatism, a 1998 revision to 10 CFR 50.46 enables licensees to implement a best-estimate ECCS EM in lieu of an Appendix K-based model. However, that rulemaking did not address other conservatisms in the ECCS design basis, such as break selection and the single failure and electric power assumptions.

In one of the risk-informed rulemaking efforts following the Commission's "Policy Statement on the Use of Probabilistic Risk Assessment" (60 FR 42,622), the NRC undertook an effort to more holistically risk-inform GDC 35, 10 CFR 50.46, and Appendix K. Following several feasibility studies and research reports, the Commission ultimately provided clear and succinct direction on the scope and approach of a risk-informed alternative to the ECCS rules in Staff Requirements

Memorandum (SRM) SECY-04-0037. The following general principles of that rulemaking directive inform this design-specific exemption request:

- Develop an appropriate break size alternative to the DEGB based on initiating event frequencies from the expert elicitation process supported by historical data, fracture mechanics analysis, and other relevant information. Assure that the selection of the maximum break size is risk-informed and conforms to Regulatory Guide 1.174 safety principles.
- Retain the capability to mitigate LOCAs larger than the new maximum DBE LOCA. Severe accident mitigation strategies ensured by a performance-based requirement are one acceptable approach to mitigating BDBE LOCAs.
- Ensure containment integrity to mitigate potential consequences from a BDBE LOCA.
- Required analyses should comport to LOCA classification. “For example, design-basis LOCA analysis should continue to meet the requirements of 10 CFR 50.46... while the appropriate mitigation capabilities for beyond design-basis LOCAs need not meet the single failure criterion nor would the models used to demonstrate mitigation capabilities need to be 50.46 evaluation models.”

With respect to an event frequency consistent with risk-informed principles, the Commission provided explicit guidance: “For example, a frequency of 1 occurrence in 100,000 reactor years is an appropriate mean value for the LOCA frequency guideline for selecting the maximum design-basis LOCA since it is complemented by the requirement that appropriate mitigation capabilities … must be retained for the beyond design-basis LOCA category.”

This exemption request differs from the rulemaking initiative in that instead of using a risk-informed approach to classify different break sizes within the same pipe, the licensing basis classifies certain break locations as BDBE breaks. However, the exemption and resulting licensing basis are otherwise consistent with the risk-informed rulemaking approach:

- Break locations are excluded with consideration of initiating event frequency. Stringent design, quality, and inspection provisions ensure integrity of the excluded break locations and support a qualitative determination that a frequency of LOCA in the excluded locations is below 1 occurrence in 100,000 reactor years. Quantitative frequency estimates are also considered for the CVCS breaks.
- Mitigation capability is retained for the excluded break locations. The ECCS continues to provide reliable, passive mitigation capability for BDBE LOCAs. Other design capabilities are also available as defense-in-depth.
- Containment integrity is ensured.
- Analyses consistent with BDBE classification are performed to demonstrate adequate mitigation capabilities. Those analyses allow alternatives to aspects of the EM required by 10 CFR 50.46 and do not require the single failure criterion for ECCS.

The following sections detail the exemption technical basis.

18.2.3 Integrity of Excluded Break Locations

18.2.3.1 ECCS Valve Connections

The ECCS is described in FSAR Section 6.3, including the ECCS main valves. The RVVs and RRVs include a bolted connection to the RPV by mating bolted penetrations that are integral to the vessel shell, and the connections do not include a physical piping length. The valves include an inlet venturi, limiting blowdown flow during postulated inadvertent actuation where there is high pressure differential between the RPV and the CNV.

Final Safety Analysis Report Section 3.6.2 discusses the connection of the ECCS main valves to the RPV including augmented and inservice inspection requirements for break exclusion. Final Safety Analysis Report Section 3.13 discusses the design, materials, and inspection requirements for threaded fasteners. Final Safety Analysis Report Section 5.2 discusses the reactor coolant pressure boundary. The vessel flanges, threaded inserts, and threaded fasteners are within the jurisdictional boundary of the RPV. They are constructed in accordance with ASME BPVC, Section III, Subsection NB and are classified with the RPV as a Class 1 vessel. Prior to operation, VT-1 examinations are performed on the threaded inserts and seal welds, penetrant testing is performed to verify the final surface condition of the seal welds, and flatness requirements are applied to the flange faces.

The selection, design, fabrication, installation, and inspection of the RRV and RVV threaded fasteners meet the requirements of 10 CFR 50.55a, including application of ASME BPVC Class 1 criteria (Section III, Subsection NB). As discussed in FSAR Section 3.13, the threaded fasteners are nickel-based Alloy 718, which is resistant to corrosion. The threaded fasteners meet the relevant requirements of ASME BPVC Section III for Class 1 components. The Class 1 threaded fasteners are subject to various controls during fabrication, including heat treatment controls, preservice inspections performed in accordance with ASME BPVC, Section XI, Subsection IWB-2200, and additional augmented surface and ultrasonic examinations beyond the ASME code requirements. Fracture toughness requirements in accordance with ASME BPVC, Section III, Subsection NB-2300 are met, as well as the additional requirements set forth in 10 CFR 50, Appendix G. Final Safety Analysis Report Section 3.6.2 describes that ECCS main valve threaded fasteners are subject to a fatigue evaluation utilizing ASME BPVC, Section III, Appendix I requirements, including application of a fatigue strength reduction factor for high strength bolting.

Degradation mechanisms (e.g., corrosion) associated with industry piping failures are shown to be either not applicable or not credible to the bolted-flange connections, as described in FSAR Section 3.6.2. Notwithstanding, inservice inspection requirements provide assurance of early detection and an extremely low likelihood of gross rupture. As discussed in FSAR Section 3.6.2 and Section 3.13, RRV and RVV threaded fasteners are inspected per ASME BPVC, Section III, NB-2581 and NB-2584, ASME BPVC, Section XI, Table IWB-2500-1, and augmented requirements beyond that of the ASME BPVC. Exceptions in ASME code requirements that allow only a sample of bolting to be inspected are not followed, and all flange bolts are inspected during each inspection interval. The threaded inserts and seal welds are subject to augmented examination requirements and stricter ISI that exceed the requirements imposed by 10 CFR 50.55a and the ASME Code inservice inspection requirements. Additionally, leakage detection is provided by the NPM. As discussed in FSAR Section 5.2.5, the NPM supports low leakage rate detection capability through both CNV pressure and the containment evacuation system. The evacuated containment design allows for CNV pressure monitoring to detect and quantify leakage that is directly correlated to CNV pressure. Leak detection capability provides additional assurance that gross rupture is extremely unlikely.

18.2.3.2 CVCS Lines between Nozzles and PSCIVs

Part 7 Section 9 includes an exemption request for GDC 55. As discussed in the exemption and FSAR Section 6.2.4, the lines penetrating containment that are part of the RCPB include two PSCIVs in series outside containment. The PSCIVs on the CVCS lines satisfy the intent of GDC 55 for isolation of lines penetrating the containment that are part of the RCPB. Each set of two PSCIVs share a single valve body that is welded to a CITF, which is welded to a nozzle safe-end on the outside of the CNV. This approach minimizes piping between valves, between the vessel and the valve, and minimizes RCPB welds outside of containment. This approach also removes a hydraulically-operated valve from being located inside of containment and not exposing the valve to the post-accident atmosphere in the CNV. A hypothetical break in the limited area between the CNV and the associated PSCIVs for CVCS lines presents a potential non-isolable break.

Final Safety Analysis Report Section 6.2.4 describes the CVCS PSCIVs as Quality Group A components with design, fabrication, construction, testing, and inspection in accordance with ASME BPVC, Section III, Subsection NB, and Seismic Category I criteria. Materials, including weld materials, conform to fabrication, construction, installation, and testing requirements of Subsection NB (e.g., welding procedure qualification in accordance with Subarticle NB-4300). Final Safety Analysis Report Section 3.6.2 describes that conservative stress and fatigue limits are met for Class 1 piping in the containment penetration area (per relevant provisions of Branch Technical Position 3-4), the length of piping is minimized, and there is a minimum number of circumferential and no longitudinal welds, among other requirements. Thus, in accordance with the criteria summarized above, the lines described above are designed, manufactured, and constructed to preclude breaks.

Final Safety Analysis Report Section 5.2.4 describes preservice and inservice inspection between the CNV and the CVCS PSCIVs is performed in accordance with ASME BPVC Section XI, pursuant to 10 CFR 50.55a. Additional augmented inspection and examination controls are applied to the welds, piping, and valve body (e.g., a volumetric examination is performed on the CITF-to-PSCIV weld that is in addition to the required surface examination during each inspection interval consistent with Branch Technical Position 3-4). In addition to the design integrity of piping between the CNV and CVCS PSCIVs, the inspection and examination requirements summarized above provide assurance that potential defects (e.g., through-wall leaks) are identified prior to gross pipe rupture. As discussed in Final

Safety Analysis Report Section 5.5, the NPM supports leakage detection, and technical specifications provide operational controls for pressure boundary leakage and unidentified leakage. Other potential leak identification capability (e.g., under-the-bioshield temperature indication) provides early identification of potential leakage outside the CNV, precluding a break between the CNV and PSCIVs for CVCS.

18.2.4 Consequences if Excluded Breaks Occurred

18.2.4.1 ECCS Valve Connections

Though the RRVs and RVVs are designed to preclude breaks, a break could allow bypass of the valve venturi, allowing a greater flow rate than what is analyzed in DBE LOCA evaluations. In the hypothetical scenario where such a break occurs, the ECCS still passively operates and allows continued core cooling throughout the transient. However, using the prescriptive methods used to satisfy 10 CFR 50.46 (e.g., reactor pool level temperatures above the technical specification limit assuming single failure of ECCS valves, etc.), prescriptive limits of minimum critical heat flux ratio (MCHFR) may be temporarily violated. The possible MCHFR limit violation occurs on the order of one second, with peak clad temperature (PCT) occurring shortly thereafter. However, PCT does not approach the 10 CFR 50.46 limit, collapsed liquid level is maintained above the top of active fuel, and containment integrity is not challenged.

18.2.4.2 CVCS Lines between Nozzles and PSCIVs

Though the area between the CNV and PSCIVs are designed to preclude breaks for CVCS penetrations, a break would be a non-isolable loss of primary coolant outside of the NPM. In the postulated scenario, ECCS actuation occurs due to low or low-low RPV riser water level, depressurizing the RCS to terminate break flow and thereby passively mitigating the transient. However, using the prescriptive methods used to satisfy 10 CFR 50.46, the collapsed liquid level above the top of active fuel may not satisfy limits implicit to the LOCA EM as referenced in FSAR Section 15.0. For liquid-space breaks, injection line breaks are more limiting than discharge line breaks. For vapor-space breaks, high point vent line breaks are more limiting than pressurizer spray breaks. Therefore, only injection line breaks and high point vent breaks are analyzed.

Without crediting active event mitigation (e.g., makeup):

1) For a spectrum of injection line breaks, PCT occurs at event initiation.

2) For a spectrum of high point vent line breaks, PCT occurs at event initiation.

With credit for nonsafety-related event mitigation, the collapsed liquid level remains above the top of active fuel and PCT does not increase from the initial steady-state value.

Using the modified approach as described in FSAR Section 15.6.5.6, analyses demonstrate that for both the ECCS valve breaks and CVCS line break, core cooling is maintained, containment integrity is not challenged, and offsite radiological consequences (i.e., dose at the exclusion area boundary and the low population zone) are below applicable limits.

18.2.5 Additional Risk Considerations

18.2.5.1 ECCS Valve Connections

Final Safety Analysis Report Section 19.1.4 identifies the capability to maintain core cooling using the passive ECCS, regardless of size or location. A hypothetical break at an ECCS valve does not result in inventory loss outside of the CNV and therefore probabilistic risk assessment (PRA) insights conclude that core damage does not occur with successful ECCS operation.

18.2.5.2 CVCS Line Between Nozzle and PSCIVs

Using weld failure rates from the probabilistic fracture mechanics (PFM) analysis (per NUREG/CR-2189, Volume 5), probabilistic risk assessment (PRA) sensitivities show the failure likelihood of welds in the non-isolable region is extremely low. The analysis does not account for design details aimed at precluding degradation (e.g., chemistry control). The associated frequency for a non-isolable break on each CVCS line due to weld failure is orders of magnitude less than the frequency of isolable breaks and orders of magnitude less than the 1E-5 frequency recommended in SRM-SECY-04-0037.

Because of the extremely low failure rate, potential non-isolable line breaks do not impact the risk insights from the PRA.

Disregarding the extremely low failure rate, the PRA discussed in FSAR Chapter 19 considers sequences where containment isolation fails for an injection line break outside containment, which results in a similar plant response to a break in the non-isolable region. As discussed in FSAR Section 19.1, core damage is avoided if all ECCS valves and a single train of the decay heat removal system functions. Core damage is also avoided if a single RVV, a single RRV, and the containment flooding and drain system (CFDS) function. These capabilities demonstrate defense-in-depth mitigation for a potential non-isolable CVCS line break.

18.2.6 BDBE LOCA Analyses

In general, BDBEs are considered in the PRA to address regulatory requirements and policy, such as determining risk insights and implementation of reactor safety goals.

To address the excluded break locations, the licensing basis postulates BDBE LOCAs within each of the two types of break locations to demonstrate adequate mitigation capabilities: a hypothetical failure at the connection of one RVV or one RRV to the RPV, or a hypothetical failure in a connection between components in the CVCS lines outside the CNV. Consistent with the direction of SRM-SECY-04-0037, analyses consistent with the BDBE classification are performed to demonstrate event mitigation capabilities. Those analyses apply event-specific acceptance criteria and allow certain, prescribed deviations from the 10 CFR 50.46-compliant ECCS EM used for DBE LOCAs.

In addition to following the risk-informed ECCS rulemaking approach, the BDBE LOCA licensing basis is analogous to the treatment of other beyond-design-basis “special events” that are specifically addressed by regulation. For example, an anticipated transient without scram (ATWS) is a special event addressed by 10 CFR 50.62 and included in FSAR Chapter 15 by Standard Review Plan (SRP) section 15.8. For certain designs, an applicant has the option of demonstrating “acceptable consequences” for ATWS events to address the ATWS rule. NRC's guidance provides acceptable consequences for ATWS as maintaining coolable core geometry, maintaining RCPB integrity, and maintaining containment integrity.

To demonstrate ECCS mitigation capability is retained for the BDBE LOCA locations, adequate core cooling is shown by either:

1) MCHFR is maintained above 1.15 and the liquid level remains above the top of active fuel

   or

2) Peak cladding temperature remains below the 10 CFR 50.46 limit

To demonstrate containment integrity, the containment response (i.e., peak containment pressure and peak containment temperature) is shown to remain below containment design pressure and temperature limits.

Additionally, the BDBE LOCA licensing basis employs a third acceptance criteria not specified in SRM-SECY-04-0037: acceptable offsite doses. Quantitative dose evaluations are not normally included in the licensing basis for BDBEs; rather, potential for core damage and large release is addressed in the PRA. However, in the case of the excluded breaks, the underlying purpose of 10 CFR 50.46 suggests that quantitative dose evaluations against the dose limits normally applied to design-basis accidents are appropriate. That is, the ECCS rules are intended to prevent a LOCA from exceeding the offsite dose limits, such as might occur if core melt damaged containment. Verifying acceptable offsite doses for BDBE breaks that are less likely than DBE LOCAs, but based on mechanistic event response rather than an assumed core melt, is consistent with that underlying purpose.

18.3 Regulatory Basis

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 DBE and will not create new accident precursors. As a result of this exemption, certain postulated breaks are excluded from DBEs and addressed as BDBEs instead. Mitigation capabilities, analysis methods, and acceptance criteria applied to these very low probability breaks ensure minimal risk from the BDBE breaks. 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.

18.3.1 Special Circumstances Regarding Both Types of Excluded Break Locations

1. *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 rules.

The design-basis LOCA spectrum required by GDC 35 and 10 CFR 50.46 is a deterministic requirement informed by implicit probabilistic considerations of designs in use and development at the time the rules were adopted. The DEGB assumption is intended to conservatively bound a spectrum of less severe, more likely LOCAs. Even more severe LOCAs were judged to be too unlikely to constitute design-basis requirements for the ECCS, although ECCS would nevertheless provide some protection for them. The analysis methods, acceptance criteria, and reliability requirements applied to DBE LOCAs by GDC 35 and 10 CFR 50.46 provide a high level of confidence that ECCS will perform its safety function for DBE LOCAs.

Treatment of the excluded breaks as DBEs, and thus application of certain GDC 35 and 10 CFR 50.46 requirements to those breaks, is not necessary to achieve the underlying purpose of those rules. Those rules are part of the deterministic regulatory framework that uses deterministic criteria to ensure adequate protection of public health and safety. As stated in the Commission's policy statement on the "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (60 FR 42,622), "the deterministic approach contains implied elements of probability (qualitative risk considerations), from the selection of accidents to be analyzed as design-basis accidents (e.g., reactor vessel rupture is considered too improbable to be included) to the requirements for emergency core cooling (e.g., safety train redundancy and protection against single failure)."

This exemption applies qualitative and quantitative probabilistic considerations to conclude certain break locations are too improbable to be included as DBEs. Analyses are performed to demonstrate that ECCS mitigation capability is maintained for these BDBE LOCAs, containment integrity is maintained, and offsite doses would not exceed accident dose limits. Nonsafety-related features provide additional defense-in-depth mitigation capabilities. This limited, risk-informed exception to the design-basis LOCA spectrum otherwise required by GDC 35 and 10 CFR 50.46 yields a level of protection for LOCAs equivalent to or greater than that contemplated when the rules were promulgated. Therefore, the underlying purpose of the rules is achieved.

2. *Special circumstances are present (10 CFR 50.12(a)(2)(vi))* in that there is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption.

GDC 35 and 10 CFR 50.46, including the design basis LOCA break spectrum applicable to the rules, are deterministic rules containing implied elements of probability. The rules were informed by understandings of designs (e.g., RCS designs and resulting qualitative break frequencies) and their risk profiles (e.g., potential for core melt, containment failure, and large radiation releases) of plants under consideration at the time the rules were promulgated.

The attributes of the design, and the resulting risk from potential breaks in the RCPB, present a material circumstance not considered when the rules were adopted. Specifically, the design and prescribed programmatic requirements reduce the likelihood of the excluded breaks, and provides appropriate capabilities to mitigate them, to render the risk from such a break insignificant.

In consideration of this material circumstance, it is in the public interest to grant the exemption. In the policy statement on the “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities,” the Commission concluded that safety decision-making is enhanced by the use of PRA insights (risk information) in regulatory decisions. Risk-informed application of these regulations helps to ensure appropriate protection against the BDBE LOCAs without diverting resources and attention from more safety-significant aspects of the design.

18.3.2 Special Circumstances Applicable to ECCS Valve Break Locations

1. 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.

The underlying purpose of the strict ECCS performance and evaluation requirements imposed by GDC 35 and 10 CFR 50.46 relates to a concern--in designs then under consideration--that core melt could lead to containment failure and unacceptable doses to the public. An ECCS valve break in the design is not relevant to ECCS capability because the transient is a short-term challenege (on the order of seconds) to fuel criteria due to rapid blowdown. In other words, improving ECCS capability would not affect the outcome of the event. In this respect it is similar to reactivity excursion events like steam line rupture, rather than the LOCAs that are postulated to dictate ECCS design and reliability. Thus, classification as a BDBE meets the underlying intent of 10 CFR 50.46 by reasonably assuring core cooling and containment integrity for such a break in consideration of its very low likelihood of occurrence.

2. Special circumstances are present (10 CFR 50.12(a)(2)(vi)) in that there is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption.

The NuScale design is unique, and such a design was not considered upon adoption of GDC 35 nor 10 CFR 50.46. The reactor coolant loop is integrated in the RPV and the ECCS main valves are attached directly to the RPV. Therefore, the ECCS design allows core cooling capability that is included directly with the NPM, requires no piping penetrations through buildings or vessels, and does not include physical piping length. The integral ECCS design (coupled with other passive design features) leads to a passively safe design that exceeds Commission safety goals by orders of magnitude. A postulated break conservatively analyzed may exceed design-basis fuel acceptance criteria, but the safety advantages of the integral ECCS design result in net benefit to the public health and safety. Unlike certain postulated LOCA breaks in traditional large light water reactors (e.g., breaks associated with forced recirculation piping), an ECCS main valve connection break does not reduce the functionality of ECCS. Rather, it results in a short-lived potential violation of MCHFR limits when applying a prescriptive design-basis LOCA EM. The hypothetical break allows for continual passive recirculation of coolant through the reactor coolant system.

18.3.3 Special Circumstances Applicable to CVCS Break Locations

1. Special circumstances are present (10 CFR 50.12(a)(2)(iii)) in that compliance would result in undue hardship.

Treating a break in CVCS outside containment as a DBE LOCA would impose substantial cost. Prevention or mitigation of the event would require safety-related capabilities meeting all of the design basis reliability requirements (e.g., single failure) and demonstrated using the overly conservative ECCS EM. For example, one approach to addressing the event as a DBE would require at least two safety-related, active isolation valves inside the CNV for each CVCS line. By comparison, the associated frequency for a non-isolable break on each CVCS line due to weld failure is orders of magnitude below 1E-5 failures per year. While the avoided cost resulting from this exemption is not quantified nor directly compared to the avoided risk if the exemption were not granted, it is clear at initiating event frequencies in this range, a meaningful change to the design to address these breaks as DBEs is undue. Treatment of the excluded CVCS breaks as BDBEs using the approach specified in the licensing basis achieves a similar degree of protection of public health and safety without undue hardship.

18.4 Conclusion

For both the ECCS main valve connections and the non-isolable CVCS locations, the design and construction (e.g., material selection, application of ASME BPVC, etc.), operational requirements (e.g., inservice inspection), and leakage detection capabilities provide assurance that the probability of gross rupture is extremely low. Leakage detection capability provides additional assurance that potential failure mechanisms are detected prior to onset of a significant failure. Moreover, risk considerations specific to the design and event-specific analyses and acceptance criteria ensure these potential break locations do not pose undue risk to the public health and safety. For these reasons, NuScale requests an exemption from GDC 35 and 10 CFR 50.46 to treat the excluded break locations as BDBEs for the US460 standard design.