December 5, 2016 Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Response to Request for Additional Information Letter No. 8 for the review of Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 0.(CAC NO. RQ6002) dated October 7, 2016 (NRC Project No. 0769).

REFERENCES:

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” Revision 0, TR-0815-16497, dated October 29, 2015 (ML 15306A126).

2. NuScale Topical Report, “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” Revision 0, TR-0815-16497, dated October 29, 2015 (ML 15306A126).

3. Letter from U.S. Nuclear Regulatory Commission to NuScale Power, LLC, "Request for Additional Information Letter No. 8 for the Review of Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 0.(CAC NO. RQ6002) dated October 7, 2016 (NRC Project No. 0769, ML16281A103).

In a letter dated October 29, 2015 (Reference 1), NuScale Power, LLC (NuScale) submitted the topical report entitled “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” Revision 0 (Reference 2). In a letter dated October 6, 2016 (Reference 3), the NRC Staff submitted Requests for Additional Information (RAI) regarding the subject topical report.

The purpose of this letter is to provide NuScale’s response to the NRC RAIs. Enclosure 1 is the NuScale Response to Request for Additional Information Letter No. 8 for the review of Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments. Please feel free to contact Steven Unikewicz at 240-833-3015 or at sunikewicz@nuscalepower.com if you have any questions.

Sincerely,

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Enclosure 1: Response to NRC Letter "Request for Additional Information Letter No. 8 for the review of Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 0

NRC RAI Number: 8 NRC RAI Date: October 29, 2016

NRC Review of: Safety Classification of Passive Nuclear Power Plant Electrical Systems, TR-0815-16497, Revision 0.

The electrical power system presented in the Licensing Topical Report (LTR) depicts a design with no Class 1E power sources as the proposed reactor design does not require any safety-related electrical loads to support the safety analyses. However, 10 CFR 50.34(f)(2)(xx) calls for vital-bus-powered post- accident monitoring instrumentation with backup power from emergency power supplies. In order for the staff to be able to conclude that an electrical design such as the one presented in the TR provides equivalent protection to that prescribed in the regulation, the staff must be able to conclude that the proposed design is of similar (high) reliability. To that end, the staff requires the following additional information:

NRC RAI Question Number: 08.03.02-01

NRC RAI Question:

Table 3-2 of the TR states that Valve Regulated Lead Acid (VRLA) batteries will be used for the direct current (DC) power system. Based on various industry publications, including Institute of Electrical and Electronics Engineers (IEEE) Std. 1187, "Recommended Practice for Installation Design and Installation of Valve-Regulated Lead- Acid (VRLA) Batteries for Stationary Applications," the life of a VRLA battery can be seriously and suddenly reduced due to factors such as: 1) prolonged high ambient temperatures, 2) magnitude and frequency of discharge cycles, and 3) overcharging.

Please describe how these factors will be addressed in the design and operation of a passive reactor nuclear power plant that relies on VLRA battery systems to ensure high reliability DC power system.

NuScale RAI Question Response:

NuScale agrees that the life of a VRLA battery can be seriously and suddenly reduced due to prolonged high ambient temperatures. These effects are mitigated through the implementation of IEEE Std. 1187 and IEEE Std. 1188, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Valve-Regulated Lead- Acid (VRLA) Batteries for Stationary Applications" as noted in Table 3-2. Additionally, IEEE Std. 1187 refers to IEEE Std. 1491, "IEEE Guide for Selection and Use of Battery Monitoring Equipment in Stationary Applications," and IEEE Std. 1635, "IEEE/ASHRAE Guide for the Ventilation and Thermal Management of Batteries for Stationary Applications."

The use of IEEE Stds. 1187 and 1188 as supplemented by IEEE Stds. 1491 and 1635 provide reasonable assurance that the VLRA batteries will function as intended following exposure to prolonged periods of high ambient temperature. Further, the heating, ventilation, and air conditioning systems serving the battery and associated charger rooms are provided back-up

power from the backup power supply system to avoid prolonged periods of high ambient temperature.

NuScale agrees that the life of a VRLA battery can be seriously and suddenly reduced due to the magnitude and frequency of discharge cycles. Magnitude and frequency of discharge cycles are design considerations addressed in IEEE Std. 1187 and IEEE Std. 1188, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Valve-Regulated Lead- Acid (VRLA) Batteries for Stationary Applications" as noted in Table 3-2. Additionally, IEEE Std. 1187 refers to IEEE Std. 1491, "IEEE Guide for Selection and Use of Battery Monitoring Equipment in Stationary Applications." IEEE Std. 1491 provides monitoring criteria that may be used to detect and monitor a battery for degradation.

The use of IEEE Stds. 1187 and 1188 as supplemented by IEEE Std. 1491 provide reasonable assurance that the VLRA batteries are designed, constructed, and monitored considering the potential for magnitude and frequency of discharge cycles to degrade battery performance.

NuScale agrees that the life of a VRLA battery can be seriously and suddenly reduced due to overcharging. These effects are mitigated through the implementation of IEEE Std. 1187. IEEE Std. 1187 refers to IEEE Std. 1491, "IEEE Guide for Selection and Use of Battery Monitoring Equipment in Stationary Applications."

The use of IEEE Std. 1187 as supplemented by IEEE Std. 1491 provides reasonable assurance that the VLRA batteries will not be overcharged and that instances of potential overcharging will be detected prior to degrading a battery to a point where it is not able to perform its intended function.

<u>Impact of NRC RAI Question Response on Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems":</u>

This RAI Response does not require Licensing Document revisions.

<u>Attachments:</u>

None

NRC RAI Question Number: 08.03.02-02

NRC RAI Question:

Table 3-2 of the TR provides a comparison of the "Class 1E DC Electrical system" to the "Non Safety-Related DC Electrical System(s) Relied upon to Power Type B and Type C Accident Monitoring Instrumentation." Under the provision "Quality Assurance" in the Table 3-2, it stated that a Graded QA Program will be applied to the DC Electrical System, which will meet or exceed the augmented QA provisions specified in RG 1.155, Appendix A, "Quality Assurance Guidance for Non-Safety Systems and Equipment". RG 1.155, Appendix A provides QA guidance for meeting the requirements of 10 CFR 50.63 and not already explicitly covered by existing QA requirements in 10 CFR Part 50 in Appendix B or R.

Please describe the proposed quality assurance program in sufficient detail that will allow the staff to verify it meets or exceeds the provisions of RG 1.155.

NuScale RAI Question Response:

A COL applicant that references Topical Report 0815-16497 will be required to incorporate the guidance contained in RG 1.155 Appendix A, "Quality Assurance Guidance for Non-Safety Systems and Equipment as part of their Quality Assurance Program." It is not the intention of this LTR to provide an example quality assurance program as that is COL applicant specific. Verification of sufficient detail is considered a potential NRC COL review topic.

Impact of NRC RAI Question Response on Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems":

This RAI Response does not require Licensing Document revisions.

Attachments:

None

NRC RAI Question Number: 08.03.02-03

NRC RAI Question:

Table 3-2 of the TR, under the provision "Batteries," states that the VRLA batteries have augmented design, QA, and qualification provisions.

Please describe the methods and processes that will be used by a passive reactor nuclear power plant to verify that VRLA batteries will perform their intended function(s) during normal operation, operational occurrences and postulated design basis events.

NuScale RAI Question Response:

The VRLA batteries used in a passive reactor nuclear power plant design are not credited for use in mitigating the consequences of postulated design basis events.

To provide reasonable assurance that VRLA batteries will perform their intended function(s) when called upon, an applicant utilizing this TR shall implement a testing and monitoring program as described in IEEE Std. 1188, "Recommended Practice for Maintenance, Testing, and Replacement of Valve-Regulated Lead- Acid (VRLA) Batteries for Stationary Applications" and in IEEE Std. 1491, "IEEE Guide for Selection and Use of Battery Monitoring Equipment in Stationary Applications." These Standards provide for a wide variety of operating parameters to be monitored on a continuous basis including cell specific parameters.

Additionally, Table 3-2 of the TR notes that applicants are required to environmentally qualify their VRLA batteries in accordance with IEEE Std. 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and seismically qualify their batteries in accordance with IEEE Std. 344, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." Such qualification provides further assurance that the batteries will perform their intended functions.

NRC RAI Question (Continued):

Please also provide the industry standards or applicable references that will be used for verification purposes.

NuScale Response (Continued):

The industry standards that will be used for verification purposes include:

1. IEEE Std. 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" as endorsed by RG 1.89
2. IEEE Std. 344, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations" as endorsed by RG 1.100
3. IEEE Std. 1188, "Recommended Practice for Maintenance, Testing, and Replacement of Valve-Regulated Lead- Acid (VRLA) Batteries for Stationary Applications"
4. IEEE Std. 1491, "IEEE Guide for Selection and Use of Battery Monitoring Equipment in Stationary Applications"

Impact of NRC RAI Question Response on Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems":

This RAI Response does not require Licensing Document revisions.

Attachments:

None

NRC RAI Question Number: 08.03.02-04

NRC RAI Question:

The TR describes the presented dc power system as "highly reliable" and substantially equal in reliability to that of an analogous Class 1E dc power system. These statements have not been described adequately in the TR. In order for the staff to be able to fully evaluate the design and ultimately conclude on its acceptability as a highly reliable power system, the staff requests that NuScale provide a description of the methodology that will be used to compare the highly reliable DC system to be described in its design certification application to a Class 1E dc power system to show that the highly reliable DC system is substantially equal in reliability to a typical Class 1E dc power system.

NuScale RAI Question Response:

The LTR seeks NRC approval of the conditions of applicability, and the methodology and bases used in their development. The LTR further seeks NRC approval of the acceptability of a set of augmented design, qualification, and QA provisions to be applied by the conditions of applicability. The augmented provisions are intended to ensure suitable reliability for a direct current (DC) power system performing the nonsafety-related functions described in the LTR, analogous to a traditional licensee's application of the augmented provisions for a 1E power system, which has been judged acceptable without a quantitative reliability acceptance criterion.

The LTR terms the subject DC power system(s) as the "highly reliable DC electrical system(s)." The LTR further states that a comparison of specified augmented design, qualification, and QA provisions to a typical Class 1E DC electrical system "supports a determination that the augmented provisions result in an electrical system reliability substantially similar to that of a Class 1E DC power system." In using these descriptive phrases, NuScale intended to reflect, qualitatively, the attributes of a DC power system meeting the specified augmented design, qualification, and QA provisions. However, these descriptive phrases were not intended to define additional conditions for use of the LTR, distinct from the specified augmented provisions that a user of the report must implement.

However, while NuScale does not intend that a user of the LTR must, as a condition of its use, explicitly and quantitatively demonstrate "substantially similar" reliability to a typical Class 1E DC power system, NuScale intends to make available such a demonstration as one method of determining the system performs at a suitable reliability to perform the important functions addressed by the LTR. The NuScale example calculation shows that the highly reliable DC electrical system has a reliability that is approximately a factor of 5 better than that of a class 1E power system. In comparing reliability to a typical design, a user of the report should consider the specific nonsafety-related functions performed by their DC system, and the safety characteristics and risk profile of the overall plant design.

The method a Design Certification Applicant may use to compare the reliability of the highly reliable DC system to that of a typical Class 1E DC power system is as follows:

- First, define the required mission(s) the power system is required to support.

- Second, define the design and system boundaries that are needed to accomplish the required mission.

- Third, establish a measure of comparable reliability (i.e., how reliable should the system be) by reviewing a “typical” design. This typical system will be a Class 1E DC power system that supports a similar mission for a licensed facility. NuScale will build a reliability model for the typical design and assess that reliability.

- Fourth, build a reliability model for its highly reliable DC system design and assess the reliability in fulfilling the required mission.

- Finally, compare the reliability of the highly DC system to that of the typical Class 1E design. A determination that the reliability of the highly reliable DC system has reliability equal to or greater than the typical design is sufficient[1] to conclude that the reliability is acceptable for the required missions.

Impact of NRC RAI Question Response on Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems”:

This RAI Response does not require Licensing Document revisions.

Attachments:

None

[1] Reliability less than, but similar to, that of the typical system is not expected, but would require further evaluation to determine if it is adequate to support the required missions.

NRC RAI Question Number: 08.03.02-05

NRC RAI Question:

The regulation set forth in 10 CFR 50.55a(h)(3) requires that design certification applications under part 52 meet the requirements of IEEE Std. 603-1991, “Criteria for Safety Systems for Nuclear Power Generating Stations.” IEEE Std. 603-1991 provides a definition of "safety system" and states that the electrical portion of the safety systems, that perform safety functions, is classified as Class 1E. Included in the definition of safety system is a system that is relied upon to remain functional during and following a design basis event to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition.

Condition of Applicability Item I.1.b, contained in Table 3-1 of the TR, states that sufficient reactor coolant inventory and negative reactivity are assured during and following a design basis event to achieve and maintain safe shutdown. Additionally, the TR provides a clarifying example assessment to illustrate how the Conditions of Applicability would be demonstrated. This example assessment did not include a quantitative safety analysis to demonstrate the ability to insert sufficient negative reactivity during and following a design basis event to achieve and maintain safe shutdown.

SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” clarifies the conditions that constitute a safe shutdown as reactor sub-criticality, decay heat removal, and radioactive material containment. Additionally, SECY 94-084 states that an appropriate safety analysis can be used to demonstrate passive system capabilities to bring the plant to a safe stable condition and to maintain this condition. NRC staff is seeking to clarify whether Condition of Applicability Item is consistent with the description of safe shutdown provided in SECY-94-084. Additionally, NRC staff is seeking to clarify the requirements for demonstrating how Condition of Applicability Item I.1.b is satisfied. NRC staff requests the following additional information:

1. Specify the criteria that constitute a safe-shutdown as applied to Condition of Applicability Item I.1.b

NuScale RAI Question Response:

The criteria that constitute a safe shutdown are sub-criticality and decay heat removal in order to maintain fuel clad integrity (radioactive material containment).

These criteria are based on guidance for attaining safe shutdown in current generation reactors and for certified advanced reactors. NRC regulations that address safe shutdown do not include criteria for a safe shutdown condition or for the reliability of systems necessary to attain safe shutdown. What constitutes safe shutdown is addressed in SECY-94-084 for advanced reactors and in guidance such as RG 1.139 and BTP 5-4 for current generation reactors.

For current generation reactors that address RG 1.139 and BTP 5-4, safety analyses of design basis events are not typically relied on to demonstrate design capability to attain safe shutdown conditions. Rather, safety analyses of DBEs (as presented in Chapter 15 of a facility’s final safety analysis report) is focused on the short term reactor response to ensure that fuel integrity

is maintained for anticipated operational occurrences (AOO) and a coolable core geometry is maintained for accidents. The safety analyses thereby evaluate the capability of the reactivity control systems to perform their protection function, rather than their shutdown function.

The safety issue that underpins these NRC guidance documents (SECY-94-084, RG 1.139, and BTP 5-4) and their specification of a safe shutdown condition and the systems' capability to attain safe shutdown is relevant to GDC 34, in that systems or equipment failures resulting in insufficient heat removal capability can lead to core damage. Per RG 1.139, a risk evaluation of the heat removal capability of a typical pressurized-water reactor (PWR) and boiling water reactor (BWR) plant following a plant trip showed that

> *...systems or equipment failures that led to the inability to remove decay heat resulted in a higher probability of a core melt than that predicted for a large LOCA for both PWRs and BWRs. Consequently, a significant safety benefit will be gained by upgrading those systems and equipment needed to maintain the RCS at the hot-standby condition for extended periods or those needed to cool and depressurize the RCS so that the RHR system can be operated.*

To address the safety issue of system limitations or equipment failures resulting in insufficient heat removal capability for advanced designs, NRC staff proposed in SECY-94-084 that passive system capabilities can be demonstrated by:

1. *A safety analysis to demonstrate that the passive systems can bring the plant to a safe stable condition and maintain this condition, that no transients will result in the SAFDLs and pressure boundary design limit being violated, and that no high-energy piping failure being initiated from this condition will result in violation of 10 CFR 50.46 criteria.*
2. *A probabilistic reliability analysis, including events initiated from the safe shutdown conditions, to ensure conformance with the safety goal guidelines. The PRA would also determine the R/A missions of risk significant systems and components as a part of the effort for regulatory treatment of non-safety systems.*

Conservative assumptions are applied to Chapter 15 safety analysis of DBEs appropriate for the intended purpose of ensuring appropriate margins to protect fuel integrity or core coolability. Although these safety analyses can be used to demonstrate adequate shutdown capability per SECY-94-084, application of the same conservative assumptions may lead to excessive margin with respect to shutdown capability. Shutdown with additional margin due to conservative safety analysis assumptions may not be appropriate, considering a specific design's heat removal and shutdown capability and reliability.

NRC RAI Question (Continued):

2. Describe how a future passive plant applicant will demonstrate that electrical power is not necessary to achieve and maintain a safe shutdown for a minimum of 72 hours.

NuScale Response (Continued):

Electrical power is not necessary to achieve and maintain a safe shutdown condition for a minimum of 72 hours if the design includes safety-related capability to maintain a safe shutdown condition that does not depend on electrical power.

To demonstrate that capability, an applicant will evaluate the reactivity control systems to ensure sufficient shutdown function capability and evaluate the decay heat removal system to ensure sufficient heat removal capability. To ensure that safe shutdown capability is sufficient to address the safety issue of heat removal reliability, a probabilistic risk assessment is used to ensure that the reliability of systems used to achieve and maintain safe shutdown supports conformance to the commission's safety goal guidelines.

The response to RAIs 08.03.02-05, Question 1 and 2, describes an approach to meet the Conditions of Applicability. The design capability along with the approach to meet Conditions of Applicability is design specific and should be evaluated as part of an applicant's design certification or combined license application, rather than evaluating it within the scope of Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," which is intended to be design independent. Requiring a specific approach to meet Conditions of Applicability may be suitable for some designs but overly prescriptive for other designs.

Impact of NRC RAI Question Response on Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems":

This RAI Response does not require Licensing Document revisions.

Attachments:

None

NRC RAI Question Number: 08.03-02-06

NRC RAI Question:

GDC 15 requires the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Condition No. I.1 of the Conditions of Applicability, contained in Table 3-1 of TR-0815- 16497, states that for a design basis event, electrical power is not necessary to maintain the reactor coolant pressure boundary (RCPB) integrity for a minimum of 72 hours. Additionally, TR-0815-16497 provides a clarifying example assessment to illustrate how the Conditions of Applicability would be demonstrated. This example assessment includes a safety analysis showing an example passive plant response to an anticipated operational occurrence. The safety analysis shows that the example passive plant response to the anticipated operational occurrence includes establishing a direct coolant flow path between the reactor core and the containment, thereby removing a fission product barrier. This caused NRC staff to question if the items under Conditions of Applicability I.1 are sufficient to demonstrate RCPB integrity. Additionally, RIS 2005-29, discusses the design criteria for event non-escalation. NRC staff is questioning why the removal of a fission product barrier is not considered an event escalation.

NRC staff requests the following information:

1. Specify the criteria that constitute RCPB integrity as applied to Condition No. I.1 of the Conditions of Applicability.

NuScale RAI Question Response:

RCPB integrity refers to the structural integrity of RCPB components designed to retain pressure and contain reactor coolant. A loss of RCPB integrity or loss of structural integrity involves a mechanical failure in an RCPB component, for example a pipe. For an AOO, the RCPB integrity acceptance criterion is that pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values. For a postulated accident, the criteria for RCPB integrity is that pressure in the RCS is maintained below acceptable design limits, considering potential brittle as well as ductile failures.

Opening of a valve(s) that allows reactor coolant to pass into or out of the RCPB does not involve a mechanical failure in an RCPB component and does not constitute a loss of RCPB integrity. An interpretation that RCPB integrity is lost when opening a valve to allow fluid to pass through the RCPB is problematic in the following respects:

- It would preclude advanced designs that offer improvements in safety by relying on valves to depressurize the RCS for safe shutdown. As described in RG 1.139 and BTP 5-4, depressurization is one of the processes that support safe shutdown. The safety benefit of RCS depressurization through valves include: providing highly reliable means for depressurization; reducing the driving force for coolant out of the RCS; and reducing the driving force for fission products out of containment in the event of a loss in clad

integrity. Further, the ability to depressurize and provide long term heat removal using valves is consistent with the NRC's position to minimize the potential for an intersystem Loss of Coolant Accident (LOCA) outside of containment in advanced or evolutionary light-water reactors in SECY-90-016 and SECY-93-087 and their associated staff requirements memoranda. Lastly, the capability of advanced designs, such as NuScale, to safely depressurize the RCS reduces the importance of RCPB integrity to safety.

- It is not consistent with the licensing basis for PWRs and BWRs; it would imply that these designs do not comply with GDC 15. GDC 15 is relevant to Standard Review Plan (SRP) Section 15.0 "as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs." Interpreting that opening of a valve to allow fluid to pass into or out of the RCPB constitutes a breach of the RCPB would imply that licensed facilities do not meet GDC 15 in the following instances.

    - PWRs and BWRs rely on safety relief valves to prevent exceeding RCPB design limits for select AOOs.
    - PWRs and BWRs evaluate inadvertent opening of a pressure relief valve as an AOO in accordance with SRP 15.0 and SRP 15.6.1.
    - BWRs open valves (safety relief valve, reactor core isolation cooling) to route reactor coolant out of the RCPB to containment for the purpose of heat removal and RCS depressurization prior to transitioning to heat removal using the residual heat removal (RHR) system at low pressures.
    - The RHR systems for PWRs and BWRs are not part of the RCPB. Valves are opened upon RHR system actuation to cycle reactor coolant into and out of the RCPB and through the RHR system for the purpose of heat removal.

- It is not consistent with Appendix A to 10 CFR 50 which address maintaining structural integrity of RCPB components rather than preventing the opening of valves to allow fluid to pass into or out of the RCPB. Under Appendix A, GDC 14, "Protection by Multiple Fission Product Barriers," addresses RCPB integrity: "The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." Further, GDCs 15, 17, 28, 30, 31, 32, 33 and 34 include provisions to design, operate, and maintain the RCPB in order to prevent loss of structural integrity.

The opening of RCPB valves is addressed by 10 CFR 50.34(f)(1)(iv). The safety concern addressed by 10 CFR 50.34(f)(1)(iv), however, is the adverse impact on core damage frequency (CDF) due to frequent valve actuation, rather than a loss of RCPB integrity. 10 CFR 50.34(f)(1)(iv) was added after the TMI-2 accident when it was recognized that a loss of coolant from a stuck open PORV and other small-break LOCA contributors was more likely to lead to core damage that a large pipe break. The rule requires evaluation of the potential benefit from automatic PORV isolation for current generation PWR's in order to reduce CDF by reducing the demand on ECCS system. For advanced designs with a low CDF, a reduction in CDF by limiting deliberate or inadvertent RCPB valve actuation to reduce the CDF may not be warranted.

NRC RAI Question (Continued):

2. Explain why the removal of a fission product barrier during an anticipated operational occurrence is not considered an event escalation.

NuScale RAI Question Response (Continued):

Opening a valve to depressurize the RCS and establish long term cooling is not considered a removal of a fission product barrier, and thus not an event escalation, because the functions of the RCS barrier are not lost. The RCS barrier continues to provide a confined volume for reactor coolant which allows a flow path for cooling the core and thus, confining fission products to the fuel. The basis for this response is as follows.

As part of the analysis acceptance criteria for AOOs (p15.0-5, SRP 15.0),

> *The reviewer applies a third criterion, based on the American Nuclear Safety (ANS) standards to ensure that there is no possibility of initiating a postulated accident with the frequency of occurrence of an AOO.*

This review is performed under Acceptance Criterion 2.A.iii, based on the ANS standards referenced in SRP 15.0, which states:

> *An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.*

Based on SRP 15.0, the intent of the non-escalation criterion is to ensure that the consequences associated with accidents do not occur at the frequency of an AOO. Such a condition would lead to an unacceptable risk to the public, due to frequent events with more significant consequences. The two parts of the non-escalation criterion, preventing accidents generated by an AOO and protecting the functions of barriers, are intended to prevent such an increase in risk. Thus, events that do not result in unacceptable consequences or significantly increase the risk for radiological release do not challenge the intent of the non-escalation criterion.

With respect to whether opening of a valve to depressurize the RCS involves a “consequential loss of function of the RCS barrier,” it is helpful to review the regulatory history of this acceptance criterion. Acceptance Criterion 2.A.iii and the definition of event categories were first introduced in Rev. 0 of the SRP and were derived from the PWR and BWR ANS standards for nuclear safety. The PWR standard referred to in Rev. 0 of the SRP, ANSI N18.2, was reviewed to clarify what was meant with this acceptance criterion.

The SRP cites this acceptance criterion for ANSI N18.2 Condition II (“Incidents of Moderate Frequency”) and Condition III (“Infrequent Incidents”) events. ANSI N18.2 presents the following events as examples of Condition II and III events:

- Condition II example: “depressurization by spurious operation of an active element, for example, relief valve, pressurizer spray valve.”

- Condition III example: "loss of reactor coolant, such as from a small ruptured pipe or from a crack in a large pipe, which would prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only" (i.e., small-break loss of coolant accident (LOCA).

These ANSI N18.2 examples are also included in examples of AOOs in SRP 15.0. The ANSI N18.2 design requirement that is the basis for Acceptance Criterion 2.A.iii (3) in SRP 15.0 is:

> *A Condition III incident shall not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or reactor containment barriers.*

Thus, Condition II and III events do not in themselves involve a consequential (significant)[2] loss of function of the RCS barrier. Of particular interest, the two examples given above that result in continuously blowing down reactor coolant from the RCS, either through a valve (Condition II) or through a small-break LOCA (Condition III), do not result in a consequential loss of function of the RCS barrier.

A consequential loss of function of the RCS barrier is associated with only Condition IV ("Limiting Fault" or "Postulated Accident") events. Based on the Condition IV example provided in ANSI N18.2 and repeated in SRP 15.0, a loss of function of the RCS barrier is associated with a major pipe rupture. This interpretation is consistent with Appendix A to 10 CFR Part 50 Section II, "Protection by Multiple Fission Product Barriers" which includes general design criteria for the fission product barriers. GDCs 14 and 15 state the design criteria for the RCPB and the RCS, which are intended to have "an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture."

That is, the function of the RCS, as implemented by these GDCs, is not to form a leak-tight radionuclide barrier to the environment; in contrast the function of the containment as stated in GDC 16 is to "…establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment …."

Rather, the RCPB functions, which are equivalent to that of the RCS barrier, are stated in SRP 5.2.3 under the technical rationale for GDC 14:

> *The RCPB provides a fission product barrier, a confined volume for the inventory of reactor coolant, and flow paths to facilitate core cooling.*

A loss of these functions is also described in SRP Section 5.2.3 as a "gross failure of the RCPB resulting in substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, or interference with core cooling."

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[2] The word "consequential" can mean resultant or significant. "Significant" makes more sense in the context of Acceptance Criterion 2.A.iii. Under "Barrier Integrity Criteria" for the RCPB, ANSI N18.2 (3rd criterion, p8, Reference 9) states that the RCPB "shall withstand Conditions I, II, III and IV, including thermal transients associated with the operation of the emergency core cooling system, without significant consequential rupture (that is, if consequential rupture occurs, it shall not appreciably worsen the safety consequences)." The terms "result in a consequential loss of function" (Acceptance Criterion 2.A.iii) and "significant consequential rupture" (ANSI N18.2) are equivalent and are interpreted to mean "result in a significant loss of function due to RCPB rupture." In comparison, the design criterion for the "Containment Barrier" is: "The design pressure, temperature, and leakage rate of the reactor containment shall not be exceeded as a result of Conditions I, II, III, or IV."

Thus, gross failure is a necessary condition for such a substantial loss of function. Further, gross failure resulting in a substantial reduction of any one of the three functions of the RCPB constitutes a substantial or consequential loss in function of the RCS barrier. This is because the function of fission product confinement is integrated with the functions of inventory control and heat removal; i.e., the function of fission product confinement is maintained if the functions of inventory control and heat removal are maintained. “Fission product barrier” does not mean that leakage from fuel defects or activation products in RCS coolant must be confined in the RCS after all design basis events. It refers to maintaining integrity of the cladding. Absent the potential for fuel cladding failure, there are no significant radiological consequences associated with the event, and therefore no “consequential loss of function” of the RCS barrier.

Thus, opening a valve to depressurize the RCS and establish long term cooling does not result in a consequential loss of function of the RCS barrier, i.e. a substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, or interference with core cooling. Accordingly, opening such valve during an anticipated operational occurrence is not considered an event escalation.

Impact of NRC RAI Question Response on Topical Report 0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems”:

This RAI Response does not require Licensing Document revisions.

Attachments:

None