NRC Response to Public Comments for PRM-50-120 [Docket ID NRC-2019-0180]

This document presents the U.S. Nuclear Regulatory Commission's (NRC's) responses to comments received on Petition for Rulemaking (PRM)-50-120. The NRC published the notice of docketing and request for comment in the *Federal Register* (84 FR 63819) on November 19, 2019.

Comments on this PRM are available electronically in the NRC's electronic Reading Room at https://www.nrc.gov/reading-rm/adams.html. From this page, the public can gain entry into the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. In addition, the public may view and download these comments electronically through the Federal e-Rulemaking Portal https://www.regulations.gov, Docket ID NRC-2019-0180.

The NRC received comments from the following individuals or groups:

Comment Submission Number	ADAMS Accession Number	Commenter	Affiliation	For or Opposed to PRM
1	ML19357A110	Anonymous	Individual	Opposed
2	ML19364A066	J. Brian Hall	ASTM E10.02 subcommittee chair	For
3	ML19364A067	Debra Higbee-Sudyka	Individual	Opposed
4	ML19364A063	Marianne Birkby	Radiation Free Lakeland	Opposed
5	ML19364A064	Sarah Fields	Uranium Watch	Opposed
6	ML19364A065	Anonymous	Individual	Opposed

In sum, the NRC received 6 comment submissions containing approximately 38 comments. The NRC binned comments according to the following 11 topics:

(1) Inadequate justification for the petition.

(2) NuScale reactor has not yet been built; there is a lack of data to determine the effect of the change on the NuScale design.

(3) Thinner reactor vessel shells pose a higher risk of adverse neutron embrittlement.

(4) The petition lacked a discussion of advanced materials or manufacturing technologies.

(5) Lack of basis in the petition for use of American Society for Testing and Materials (ASTM) E900-15.

(6) ASTM E900-15 would provide better prediction of embrittlement for NuScale than Title 10 of the *Code of Federal Regulations* (10 CFR) 50.61 or Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," issued May 1988.

(7) Proposed change should not be limited to new plants.

(8) Neutron embrittlement effects and resulting accident consequences are not adequately understood.

(9) There is an inappropriate application of mathematical models to predict the embrittlement of metal that includes the use of an arithmetic mean.

(10) Comments that were outside the scope of the PRM 50-120.

At the end of each comment, the NRC refers to the specific public comment letter containing that comment in the form [XX-YY], where XX represents the Submission ID in Table 1 of this document and YY represents individual, sequential comments as noted in the margin of the annotated copy of the public comments (see ADAMS Accession No. ML21074A309).

Comment Bin 1—Inadequate justification for the petition

Comment Bin 1 Summary: Several commenters expressed that the petitioner did not adequately explain or justify why the NRC should change the regulation for a variety of reasons.

One commenter asserted that the petitioner's desire for regulatory change constitutes evidence that their design is not safe. The commenter requested that the NRC increase the stringency of the NRC's safety standards based on the availability of lower risk alternative sources of power such as solar and wind energy.

Another commenter expressed opposition to the petition based on the lack of evidence, as presented by the NRC, of the consequences of granting the petitioner's request. The same commenter also opposed the petition because it was submitted by an industry entity instead of "the public." The same commenter expressed opposition to the petition on the grounds that the petitioner did not provide enough information on how the restrictions in plant operation, caused by the existing regulations, would impact public health and safety. This same commenter expressed opposition to the petition because the NRC should use the value that is the most safe and conservative, not the most convenient.

Two commenters expressed opposition to the petition on the grounds that there is no scientific basis for amending the rules.
(Comment 1-1, 3-3, 5-1, 5-2, 5-3, 5-10, 5-13, 5-17, and 5-18)

NRC Response to Comment Bin 1: The NRC agrees in part and disagrees in part with these comments.

The NRC agrees that the evaluation of the potential for and consequences of reactor embrittlement should use values appropriate to the safety significance and the uncertainty in the methodology and should not be selected for convenience. The NRC disagrees, however, that the values should be selected in order to produce the most conservative results.

The NRC disagrees that the petition constitutes evidence that a design is unsafe. The comments related to alternative energy sources and the NRC review of the NuScale Power, LLC (NuScale) design certification application are outside the scope of this petition
The NRC disagrees that certain entities should be excluded from submitting petitions for rulemaking. All members of the public including regulated entities are free to submit petitions

for rulemaking to request changes consistent with the NRC regulations. The NRC disagrees that the NRC is responsible for presenting evidence to support the change requested in the petition. The NRC's role is to evaluate the requested petition on its merits and based on the evidence presented by the petitioner and provided in timely comments.

The NRC agrees that the petitioner did not provide sufficient information for the NRC to support its request for rulemaking.

The NRC considered the petition and the public comments as described in the *Federal Register* notice denying the petition and in this document. For the reasons stated in the denial notice, the NRC is denying the petition.

Comment Bin 2—NuScale reactor has not yet been built; there is a lack of data to determine the effects of neutron embrittlement on the NuScale design

Comment Bin 2 Summary: One commenter stated that the estimation of embrittlement was inadequate because no data has been developed from a full-scale working NuScale reactor. Another commenter challenged the use of the ASTM E900-15 because the data may not include small modular reactors (SMRs) or other advanced reactor designs; the designer and potential applicants for the NuScale reactor do not have any appreciable experience in the construction, operation, and decommissioning of nuclear reactors; and the lack of analysis on potential synergistic effects from the co-located reactors. A third commenter asserted that the standard might not be appropriately conservative for NuScale and that the NuScale vessel might be especially susceptible to embrittlement due to the thinner walls.
(Comment 3-1, 5-6, 5-7, 5-11, 6-3, and 6-8)

NRC Response to Comment Bin 2:
The NRC disagrees in part and agrees in part with these comments. A petitioner or an applicant is not required to construct a demonstration or prototype plant in order to support a full-scale licensing process or to justify the petition for rulemaking. The NRC staff, therefore, disagrees with this comment. However, the NRC agrees that the petitioner did not adequately address the application of ASTM E900-15 in several aspects, as discussed further with regard to Comment Bin 7.

The NRC disagrees with the comment challenging the experience of a prospective applicant. The petition for rulemaking must be evaluated on its own merits. The qualifications and experience of a prospective applicant for a license will be evaluated in the licensing process, which is another regulatory process.

The NRC disagrees with the comment that factors such as the effect of multiple reactors located at the same site apply to this petition. The NRC evaluates the safety considerations of multiple reactors located at the same site in the licensing process, not in rulemaking. Because the number of reactors located at a particular site is a matter reviewed under another process, the NRC believes that the PRM should be evaluated on its own merits.

The NRC also disagrees that the NuScale design is substantively different from other operating reactor designs with respect to neutron embrittlement estimates. The NuScale design is a pressurized light-water reactor whose general geometry and nuclear physics are consistent with the operating fleet. Specifically, the NuScale reactor geometry, neutron spectrum, vessel material, coolant properties, neutron attenuation through the thickness of the vessel, and aging response are sufficiently similar to the operating fleet. The NRC notes that NuScale reactors operate at a temperature somewhat lower than that of most other power reactors. The required vessel surveillance program (described in Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Domestic Licensing of Production and Utilization Facilities," is designed to address plant-specific conditions that may affect the accuracy of estimation. The NRC has previous experience with temperature effects on neutron embrittlement.

Comment Bin 3—Thinner reactor vessel shells pose a higher risk of adverse neutron embrittlement

Comment Bin 3 Summary: Two commenters asserted that neutron embrittlement would impact thinner reactor vessels more adversely. Several commenters quoted from "Assessment of Materials Issues for Light-Water Small Modular Reactor," issued in 2013, on behalf of the U.S. Department of Energy (DOE) by Pacific Northwest National Laboratory. Specifically, one commenter noted that the NRC must evaluate this report to determine if thinner walled vessels would be subject to sudden through-wall cracking and vessel failure.
Two commenters noted that this report indicated that embrittlement would penetrate deeper into the thickness of the vessel due to the lesser thickness of the vessel compared to conventional light-water reactors and thus would pose an unanalyzed and unacceptable risk of reactor pressure vessel failure.

Two commenters asserted that the NuScale reactor pressure vessels are apparently more at risk than "regular" nuclear reactors to sudden catastrophic vessel failure.
(Comment 4-1, 5-12, 5-14, 6-1, 6-4, and 6-6)

NRC Response to Comment Bin 3: The NRC disagrees with these comments. The NRC has reviewed the DOE report referenced by the comment. Thus far, the NRC has not found that thinner walled reactor designs are substantively different from the designs of operating reactors. In the context of estimating neutron embrittlement, the NRC has not determined that additional study or modified regulatory treatment is necessary. The NRC evaluated this DOE report in the context of this petition and found it not to be generally applicable to the actual parameters of the NuScale design. Specifically, thinner vessels reduce the likelihood of "catastrophic rupture" due to lower thermal gradient induced stresses during transients. The NuScale design does not have a uniquely high end-of-life neutron fluence, and neutron attenuation through the thickness of a NuScale vessel is well understood and managed through the existing formula in RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, (as reconfirmed by the recent assessment of that RG). The NRC has routinely evaluated differences in vessel thickness due to both variations in ring thickness and the distinctions between pressurized- and boiling-water designs and was well prepared to evaluate the NuScale vessel.

Comment Bin 4—Lack of discussion of advanced materials or manufacturing technologies

Comment Bin 4 Summary: One commenter noted that the petitioner provided no information or discussion on "advanced manufacturing technologies" cited in the petition, despite the petition implying that these technologies were pertinent to the NuScale design. This commenter further suggested that "there is no data" concerning modern reactor pressure vessel (RPV) fabrication techniques. Another commenter cited a DOE technical report "United States Nuclear Manufacturing Infrastructure Assessment," issued in 2018, where it noted "SMR [small modular reactor] and advanced reactor designs require or may benefit from use of materials that do not

presently have sufficient irradiation test data," and "No ASME code cases have been accepted for [additive manufacturing] processes or parts."
(Comment 5-4, 6-5, and 6-9)

NRC Response to Comment Bin 4: The NRC agrees with the comment that the petitioner provided no basis for citing advanced manufacturing technologies. The NRC also agrees with the comment that some future small modular reactor designs may benefit from the use of advanced materials for which irradiation test data may be limited. The NRC's review of the NuScale design certification application, including issues related to the RPV material and manufacturing process, however, is outside the scope of this petition. The NRC notes that the NuScale RPV material and manufacturing processes are expected to be consistent with reactor fabrication techniques used worldwide. Nothing in the NuScale design certification application implies a change in practice, and all information provided to the NRC indicates that NuScale RPVs will be manufactured consistent with current worldwide practice.
The NRC also notes that it is following development of advanced manufacturing technologies[1] in order to remain positioned to evaluate and maintain safety if and when such techniques are applied to regulated components.

Comment Bin 5—Lack of basis in the petition for the use of American Society for Testing and Materials E900-15

Comment Bin 5 Summary: Two commenters noted that there is a lack of basis in the petition connecting the NuScale design with the "larger database" supporting the use of ASTM E900-15, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials."

One commenter asserted that because the NuScale design has not been built, it would be inappropriate to apply ASTM E900-15 to predict the performance of the design.
Another commenter asserted that much more damage would occur than predicted by ASTM E900-15 because the report is based upon the assumption of 31 years of operating experience for pressurized water reactors, and that NuScale proposes to operate its reactor for 60 years.

This same commenter also asserted that ASTM E900-15 is inappropriate for use because it does not account for differences in fluence rate and neutron energy spectra in power reactors and test reactors.
(Comment 5-5, 6-2, and 6-6)

NRC Response to Comment Bin 5: The NRC disagrees in part and agrees in part with these comments. "Adjunct for E900-15 Technical Basis for the Equation Used to Predict Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," published September 18, 2015, and available from ASTM, documents the development of ASTM E900-15. The NRC has reviewed this adjunct basis document in detail (TLR-RES/DE/CIB-2020-11, "Basis for a Potential Alternative to Revision 2 of Regulatory Guide 1.99," January 19, 2021 (ADAMS Accession No. ML20345A003)), and found ASTM E900-15 to contain an embrittlement trend curve of high quality, with a wide range of applicability based on data supporting a variety of chemistries, including the proposed chemistry of a NuScale vessel. Given the high degree of

[1] NRC Transmittal of Action Plan for Advanced Manufacturing Technologies, Revision 1 (ADAMS Accession No. ML19333B980)

similarity of the NuScale design to the operating fleet, the NRC does not agree with the comments that ASTM E900-15 has no applicability to the NuScale design.
The NRC, however, agrees that the petition does not adequately address the application of ASTM E900-15 in several other aspects that are discussed in the NRC response to Comment Bin 6.

Comment Bin 6—ASTM E900-15 would provide better prediction of embrittlement for NuScale than 10 CFR 50.61 or Regulatory Guide 1.99, Revision 2

Comment Bin 6 Summary: One commenter asserted that the embrittlement trend curve in ASTM E900-15 was developed from an extensive database of Charpy transition temperature shift measurements gathered from operating boiling water reactor and pressurized water reactor surveillance capsules, was developed via the ASTM consensus process, and represents the best understanding of embrittlement available at this time.

The commenter further asserted that ASTM E900-15 appears to include the NuScale operating conditions and would provide better predictions than the current NRC method found in 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," and RG 1.99, Revision 2.

The same commenter also noted that the uncertainty term in 10 CFR 50.61 and RG 1.99, Revision 2, is not applicable to the ASTM E900-15 embrittlement trend curve.
(Comment 2-1, 2-2, and 2-4)

NRC Response to Comment Bin 6: The NRC agrees in part and disagrees in part, with these comments.

The NRC has conducted a thorough analysis of ASTM E900-15 against available power reactor data (ADAMS Accession No. ML20345A003) and concurs that ASTM E900-15 represents an embrittlement trend curve of high quality, with a wide range of applicability, incorporating the broadest and largest power reactor dataset of any currently published trend curve. For a typical light-water reactor design, ASTM E900-15 is expected to provide the best estimation of embrittlement. Although several more narrowly applicable trend curves exist with better properties within their narrow range of application (e.g., only in Japan), these narrowly applicable trend curves do not perform as well as those from ASTM E900-15 over the broader range of properties found in the worldwide fleet or the U.S. domestic fleet.

The NRC evaluated this as part of the review of the NuScale design certification. The NRC notes that, due to the particulars of the NuScale design certification, the NRC found that the method in 10 CFR 50.61 and RG 1.99, as implemented with a temperature adjustment term, in the design certification would provide adequate estimates.

The NRC agrees that the petitioner proposed to insert references to ASTM E900-15 into 10 CFR 50.61 and RG 1.99 without thoroughly matching the rest of the methodological approach to the proposed ASTM E900-15 insertion. In particular, the ASTM E900-15 uncertainty term was calculated using a larger and more substantive body of data and consequently represents a more modern estimate of "scatter" than the current margin term used in the NRC methods. The NRC notes that the "scatter" term in ASTM E900-15 represents the scatter in data from the worldwide fleet and consequently may require additional consideration for domestic use by more careful calculation.

The NRC does not agree with the comment's assertion that, for the particular case of the NuScale design, which operates at an unusually low temperature, the ASTM E900-15 margin term would be sufficient as-is.

Comment Bin 7—Proposed change should not be limited to new plants

Comment Bin 7 Summary: One commenter noted that the petition should not have limited the proposed application of ASTM E900-15 to just new plants but should have proposed that the NRC change its regulations to allow its use for the operating fleet as well.
(Comment 2-3)

NRC Response to Comment Bin 7: The NRC agrees with this comment. Based on the NRC's evaluation of ASTM E900-15, the NRC agrees that a rulemaking that would limit ASTM E900-15 to new plants would be unnecessarily restrictive. For the reasons explained in denial notice, the NRC is, however, denying the petition.

Comment Bin 8—Neutron embrittlement effects and resulting accident consequences are not adequately understood

Comment Bin 8 Summary: One commenter noted that embrittlement of metals occurs due to effects other than neutron irradiation. The commenter noted that the NRC should consider how neutron irradiation affects the microstructure of exposed materials.
(Comment 5-15 and 5-16)

NRC Response to Comment Bin 8: The NRC agrees in part with the comments.
First, the NRC agrees that a variety of factors cause the embrittlement of RPVs over time. The NRC also agrees with the comment that the NRC must evaluate how exposure to neutrons affects materials. However, the NRC disagrees that the current requirements, guidance, and practice do not account for neutron irradiation and additional factors that can lead to reactor pressure vessel embrittlement. The NRC's current practice also accounts for current fleet operating experience developed from information gained from the current operating reactor recovered surveillance capsules. Data from these capsules continues to validate the NRC's current understanding of neutron embrittlement of reactor materials and supports the continued use of predictive models.

Comment Bin 9—There is an inappropriate application of mathematical models, including use of arithmetic mean

Comment Bin 9 Summary: Two commenters noted that the industry's use of mathematical models, including the arithmetic mean, is inappropriately applied to predict the embrittlement of metals.
(Comment 5-8, 5-9, and 6-7)

NRC Response to Comment Bin 9: The NRC disagrees with these comments. As previously stated, the NRC's current practice is to continually incorporate fleet operating experience from recovered capsules. Data obtained from these capsules have validated that the NRC's current understanding of neutron embrittlement of reactor materials is sufficient to support the predictive mathematical models currently used in industry guidance and practice.

Comment Bin 10—Comments that were outside the scope of the PRM 50-120

Comment Bin 10 Summary: One commenter raised concerns about the time provided for public comments. That same commenter asserted that the NuScale reactor is likely to behave similarly to the Lucens reactor in Switzerland.

Another comment expressed general opposition to nuclear power, stating that nuclear power should have been retired immediately following any of the nuclear accidents such as Fukushima. The commenter further stated that nuclear waste is highly toxic, and nuclear energy is too expensive compared to other forms of power.
(Comments 3-2, 6-10, and 6-11)

NRC Response to Comment Bin 10: These comments are outside the scope of PRM-50-120. The NRC did not receive any request to extend the public comment period from a person who was not able to provide a comment. Perceived similarities between the Lucens reactor and NuScale are outside the scope of PRM 50-120, which seeks to make a generic as opposed to design-specific change to the NRC requirements. General opposition to nuclear power is likewise outside the scope of this PRM but for information about post-Fukushima safety enhancements in the United States, see https://www.nrc.gov/reactors/operating/ops-experience/post-fukushima-safety-enhancements.html.