[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

[Docket No. PRM-50-120; NRC-2019-0180]

Alternative Method for Calculating Embrittlement for Steel Reactor Vessels

AGENCY: Nuclear Regulatory Commission.

ACTION: Petition for rulemaking; denial.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is denying a petition for rulemaking, dated August 19, 2019, submitted by Thomas A. Bergman on behalf of NuScale Power, LLC. The petition was docketed by the NRC on September 11, 2019, and was assigned Docket No. PRM-50-120. The petitioner requested that the NRC revise its regulations to add an alternative formula for calculating the mean value of the transition temperature shift described in American Society for Testing and Materials Standard E900-15 to the NRC's regulations and guidance documents. The NRC is denying the petition because the petitioner did not demonstrate the immediacy of any safety issues in the concerns raised in the petition and did not provide any new information that would warrant revision of the NRC's regulations.

DATES: The docket for the petition for rulemaking PRM-50-120 is closed on July 23, 2025.

ADDRESSES: Please refer to Docket ID NRC-2019-0180 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal Rulemaking Website:** Go to https://www.regulations.gov and search for Docket ID NRC-2019-0180. Address questions about NRC dockets to Helen Chang; telephone: 301-415-3228; email: Helen.Chang@nrc.gov. For technical questions, contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at https://www.nrc.gov/reading-rm/adams.html. To begin the search, select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, at 301-415-4737, or by email to pdr.resource@nrc.gov. For the convenience of the reader, instructions about obtaining materials referenced in this document are provided in the "Availability of Documents" section.

- **NRC's PDR:** The PDR, where you may examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time, Monday through Friday, except Federal holidays.

FOR FURTHER INFORMATION CONTACT: Aaron Kwok, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-1371, email: Aaron.Kwok@nrc.gov; or Dan Widrevitz, Office of Nuclear Reactor Regulation, telephone: 301-415-2620, email: Dan.Widrevitz@nrc.gov. Both are staff of the U.S. Nuclear Regulatory Commission, Washington DC 20555-0001.

SUPPLEMENTARY INFORMATION:

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I. The Petition

Section 2.802 of title 10 of the *Code of Federal Regulations* (10 CFR), "Petition for rulemaking—requirements for filing," provides an opportunity for any interested person to petition the Commission to issue, amend, or rescind any regulation. On August 19, 2019, the NRC received a petition for rulemaking (PRM) from Thomas A. Bergman on behalf of NuScale Power, LLC (NuScale). The petitioner requested that the NRC revise its regulations to add an alternative formula for calculating the mean value of the transition temperature shift described in American Society for Testing and Materials (ASTM) E900-15, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials."

On November 19, 2019 (84 FR 63819), the NRC published a notice of docketing and request for comment for PRM-50-120. The petitioner requested that the NRC amend its regulations in § 50.61(c)(1)(iv), with the first paragraph to read as follows: "ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to irradiation, and must be calculated using Equation 3. As an alternative, ΔRT_{NDT} may be determined in accordance with ASTM E900-15 instead of Equation 3, and Tables 1 and 2 of this section." Further, the petitioner requested that the formula for calculating the mean value of the transition temperature shift described in ASTM E900-15 be added for use as an alternative to Equation 2 in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The petitioner requested that the following text be added to paragraph 2 in Section 1.3 of RG 1.99, to read as follows: "For new plants electing to use ASTM E900-15 as allowed by Regulatory Position 3 for determining ΔRT_{NDT}, the correction factor is not required, provided that the irradiation temperature is within the ASTM E900-15 applicability range."

The NRC identified the following five main issues raised in the petition:

Issue 1: The methodology for calculating the mean value of the transition temperature shift (ΔRT_{NDT}) in § 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," and RG 1.99 is overly conservative and is based on outdated information.

Issue 2: The 1°F/1°F adjustment methodology requires excessive compensation for irradiation temperatures less than 525°F and has significant drawbacks.

Issue 3: The staff required NuScale to comply with § 50.61 and RG 1.99 and use the 1°F/1°F adjustment methodology.

Issue 4: ASTM E900-15 more accurately models the effects of irradiation temperature and does not suffer the drawbacks of the 1°F/1°F adjustment methodology.

Issue 5: The current methodology for determining embrittlement in § 50.61, with 1°F/1°F adjustment, is unnecessarily burdensome for reactors like NuScale, in that it would: a) result in unnecessarily restrictive heat-up and cool-down rates during startups and shutdowns, and b) cause surveillance capsules to be withdrawn and tested prematurely.

II. Public Comments on the Petition

The notice of docketing for PRM-50-120 requested interested persons to submit comments. The comment period closed on December 19, 2019. The NRC received 6 comment submissions consisting of 38 comments. The comments were received from private citizens, individuals affiliated with advocacy groups, and an individual affiliated with an industry group. The comments received on PRM-50-120 and the NRC's responses to them are available in ADAMS under Accession No. ML20304A003.

III. Reasons for Denial

The NRC is denying the petition because the petitioner did not demonstrate the immediacy of any safety issues in the concerns raised in the petition and did not provide any new information that would warrant revision of the NRC's regulations.

The NRC concludes that Issue 1 does not warrant rulemaking because the petitioner did not provide any new information that would warrant the expenditure of limited NRC resources for rulemaking. Specifically, the NRC found that while a significantly larger body of data for neutron embrittlement is now available, the core assertion that RG 1.99, Revision 2, with the use of the 1°F/1°F adjustment methodology, provides an overly conservative prediction is not correct in cases the NRC has evaluated such as the NuScale design certification application (DCA). The petition presents no additional information or data to demonstrate that the current regulation is overly conservative.

The NRC concludes that Issue 2 does not warrant rulemaking because the petitioner did not provide any new information beyond what is approved in the NRC's final safety evaluation for the NuScale DCA. The steels proposed to be used in the NuScale DCA, as well as those proposed in other light-water designs known to the NRC, are represented in the operating fleet. The petition did not present any pertinent new information regarding embrittlement performance characteristics of these materials. The NRC determined that the NuScale design presented no unusual characteristics justifying a unique temperature-embrittlement relationship for that design. In addition, RG 1.99, Revision 2, does not prescribe a temperature adjustment; rather, it states that any correction factor for operating conditions below 525°F should be “justified by reference to actual data.”

Embrittlement was previously evaluated by the staff for the specific case of a NuScale design, whose operating conditions include a relatively low operating temperature (the embrittlement impacts of which the 1°F/1°F adjustment compensates), for 40 years of operation. The NRC verified, during its review of the NuScale DCA, that a combination of the methodology in 10 CFR 50.61 and RG 1.99, Revision 2, together with the 1°F/1°F adjustment provides an appropriate estimate of RT_{NDT} based on a comparison to the publicly available information. While the NRC found that the ASTM E900-15 methodology may support improved accuracy at intermediate fluences, these were not proposed in the NuScale DCA, nor in the petition, and are bounded by the information presented in the NuScale DCA.

The NRC concludes that Issue 3 does not warrant rulemaking because the staff did not require NuScale to comply with § 50.61 and RG 1.99, Revision 2, and use the 1°F/1°F adjustment methodology. In Section IV of the petition the petitioner states, “The NuScale application of RG 1.99, Rev 2 ETC, plus the 1ºF/1ºF adjustment methodology deman[d]ed by the staff, requires an excessive compensation for irradiation temperature less than 525ºF.” In its design certification application, NuScale proposed but declined to support its initial proposal to use alternate methods for calculating RT_{NDT}. NuScale did not provide any new information beyond what is described in the NuScale DCA in the petition. Furthermore, the use of 1°F/1°F adjustment methodology is not required; rather, it is a methodology that the NRC has previously accepted for specific applications. Consequently, NuScale could have proposed an alternate adjustment methodology for the temperature correlation.

The NRC concludes that Issue 4 does not warrant rulemaking because ASTM E900-15 cannot be directly substituted for the methodologies described in § 50.61

and RG 1.99, Revision 2, as proposed by the petitioner. This is because the ASTM E900-15 embrittlement trend curve (ETC) is an embrittlement correlation; however, it lacks other pertinent features of RG 1.99, Revision 2, such as a methodology for utilizing plant-specific surveillance data to check prediction results. In addition, the paucity of data at NuScale's planned operating temperature within the dataset used to generate ASTM E900-15 would require further considerations prior to use. Furthermore, although NuScale asserts in its petition that ASTM E900-15 could also be used by advanced reactors and other small modular reactors, the ASTM E900-15 ETC is based mainly on data from light-water reactors, and its applicability is limited to the temperature range of the data used to develop the embrittlement trend curve. NuScale is the only light-water reactor design that has ever been reviewed by the NRC that would operate with such a low operating temperature, and the other advanced reactor designs the NRC is aware of would operate at substantially higher temperatures than are addressed by the current data, and therefore the NRC finds that ASTM E900-15 would not be useable for such high temperature reactors without additional adjustments. Therefore, the NRC finds that the petitioner's claim that ASTM E900-15 would provide wide-ranging benefits for future advanced reactor designs is not supported.

Additionally, the NRC determined that this issue does not warrant rulemaking because the NRC has evaluated the acceptability of using ASTM E900-15 for calculating reactor pressure vessel (RPV) embrittlement trends. The NRC provided details of this effort at a May 19, 2020, public meeting to discuss RG 1.99, Revision 2, and appendix H to 10 CFR part 50. During the Materials Information Exchange public meeting on July 14, 2020, the NRC gave a status update indicating that it had decided not to pursue an alternative to RG 1.99, Revision 2, at this time. As part of the status update, the NRC noted that it planned to document the results of its evaluation effort in two technical letter reports, and that it also would complete a holistic evaluation of RPV integrity, considering both the RG evaluation and RPV surveillance programs, using the principles of risk-informed decisionmaking from RG 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” The NRC also stated it would continue to engage with stakeholders on this topic. The NRC indicated that the documentation generated under this effort could be used by future licensees or applicants seeking an alternative to RG 1.99, Revision 2, based on the ASTM E900-15 ETC.

On October 26, 2020, the NRC issued the first technical letter report TLR RES/DE/CIB 2020 09, “RG 1.99 Revision 2 Update FAVOR Scoping Study.” In this report, the staff estimated the probability of potential reactor vessel cracking under a variety of plant operating transients relative to the degree of embrittlement underprediction (i.e., how much may risk increase if embrittlement was underpredicted). Estimates of embrittlement under RG 1.99, Revision 2 and the ASTM E900-15 were then generated for operating plant materials. This allowed for a comparison of the “risk” of using the older RG 1.99, Revision 2, correlation versus the ASTM E900-15 correlation. The technical letter report concluded that the risk associated with not updating the ETC of RG 1.99, Revision 2, is relatively low. Given the low risk, the NRC determined that there would be little benefit to updating RG 1.99, Revision 2. The NRC evaluated this conclusion based on the information included in the petition as well as the preliminary findings of the evaluation process described above.

On January 19, 2021, the NRC staff issued the second technical letter report, TLR-RES/DE/CIB-2020-11, “Basis for a Potential Alternative to Revision 2 of Regulatory Guide 1.99.” The report concluded that ASTM E900-15 is the best available alternative ETC to the RG 1.99, Revision 2 ETC, providing more accurate predictions when evaluated against the existing surveillance data. However, ASTM E900-15 cannot directly substitute for the methodologies described in 10 CFR 50.61 and RG 1.99, Revision 2, because the ASTM E900-15 ETC is an embrittlement correlation and lacks other pertinent features such as a methodology for using plant specific surveillance data to check prediction results. More specifically, the scarcity of data at NuScale's operating temperature within the BASELINE dataset used to generate ASTM E900-15 would require further considerations for use. NuScale is the only light-water reactor design reviewed by the NRC that would operate with such a low temperature, and other advanced reactor designs that the NRC is currently aware of would operate at substantially higher temperatures than are addressed by the current data and therefore the NRC finds that ASTM E900-15 would not be useable for such high temperature reactors without additional work. Therefore, the NRC finds that the petitioner's claim that ASTM E900-15 would provide wide-ranging benefits for future advanced reactor designs is not supported.

The NRC concludes that Issue 5 does not warrant rulemaking because the petition did not establish the merits of its assertions regarding unnecessary burden being imposed by the use of the RG 1.99, Revision 2, methodology for determining the heat-up and cool-down rates during startups and shutdowns. Consistent with the discussion concerning Issue 1, the NRC staff reviewed a forecasting of embrittlement for the NuScale DCA and found the application of the current approach to be acceptable and appropriate. With regards to the impact on heat-up/cool-down curves, the staff did not have a basis to conclude that these curves would have affected actual plant operation in a manner causing significant unnecessary burden. Likewise, the petitioner did not demonstrate the merits of the concern related to the withdrawal schedules for surveillance capsules. The specific timing of removal does not alter the associated burden of a removal and is not subject to specific regulatory requirements.

IV.Availability of Documents

The documents identified in the following table are available to interested persons through one or more of the following methods, as indicated.

DOCUMENT	ADAMS ACCESSION NO. / *FEDERAL REGISTER* CITATION / WEBSITE
NuScale, LLC Petition for Rulemaking to Revise 10 CFR Part 50 – Alternative Method for Calculating Embrittlement for Steel Reactor Vessels, August 19, 2019	ML19254B848
Alternative Method for Calculating Embrittlement for Steel Reactor Vessels, November 19, 2019	84 FR 63819
NRC Response to Public Comments for PRM-50-120, October 14, 2021	ML20304A003
Regulatory Guide 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” January 2018	ML17317A256
Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” May 1988	ML003740284
American Society for Testing and Materials, “Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials,” ASTM E 900-15e2, West Conshohocken, PA; ASTM International, 2015	https://doi.org/10.1520/E0900-15E02 https://www.astm.org/Standards/E900.htm
RG 1.99, Revision 2, and Reactor Vessel Surveillance Public Meeting, May 19, 2020	ML20168A008 (Package)
NuScale Standard Plant Design Certification Application, Chapter 5, “Reactor Coolant System and Connecting Systems,” July 2020	ML20224A493
American Society for Testing and Materials, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” ASTM E185-82e2, E 706 (IF). West Conshohocken, PA; ASTM International, 0 (July 1, 1982)	https://compass.astm.org/EDIT/html_historical.cgi?E185+02
PHASE 6 - NuScale DC Final Safety Evaluation Report (Complete with Appendices)	ML20023A318 (Package)
RG 1.99 Revision Evaluation Effort for Industry / U.S. Nuclear Regulatory Commission Materials Programs Technical Information Exchange Public Meeting, July 14, 2020	ML20192A002

RG 1.99 Revision 2 Update FAVOR Scoping Study, October 26, 2020	ML20300A551
TLR-RES/DE/CIB-2020-11, “Basis for a Potential Alternative to Revision 2 of Regulatory Guide 1.99,” January 19, 2021	ML20345A003

V. Conclusion

For the reasons cited in this document, the NRC is denying PRM-50-120. The NRC completed an evaluation of the petition and determined that the issues in the petition did not demonstrate the immediacy of any safety issues and did not provide any new information that would warrant revision of the NRC’s regulations. The NRC concludes that the arguments presented in the petition do not support the requested revisions to its regulations. Finally, the NRC reaffirms that its existing regulations continue to provide reasonable assurance of adequate protection of public health and safety.

Dated: July 21, 2025.

For the Nuclear Regulatory Commission.

/RA/

Carrie Safford,
Secretary of the Commission.