POLICY ISSUE
(Notation Vote)

October 14, 2021 | SECY-21-0088

FOR: The Commissioners

FROM: Daniel H. Dorman
Executive Director for Operations

SUBJECT: DENIAL OF PETITION FOR RULEMAKING ON ALTERNATIVE METHOD FOR CALCULATING EMBRITTLEMENT FOR STEEL REACTOR VESSELS (PRM-50-120; NRC-2019-0180)

PURPOSE:

The purpose of this paper is to request Commission approval to deny a petition for rulemaking (PRM) associated with an alternative method for calculating embrittlement for steel reactor vessels (PRM-50-120) and to publish a notice of the denial in the *Federal Register*. This paper does not address any new commitments or resource implications.

SUMMARY:

The U.S. Nuclear Regulatory Commission (NRC) received a petition requesting changes to a requirement for calculating the embrittlement for advanced reactor designs and to add an alternative formula. The staff has evaluated the petition and requests the Commission's approval to deny the PRM; the petitioner did not demonstrate the immediacy of any safety issues in the concerns raised in the petition and did not demonstrate any new information that would warrant revision of the NRC's regulations. In addition, the NRC recently completed its evaluation of the potential use of American Society for Testing and Materials Standard E900-15 in another NRC process and determined that the current practice continues to be adequate and appropriate for current and future reactor designs.

CONTACT: James O'Driscoll, NMSS/REFS
301-415-1325

SECY NOTE
This SECY Paper will be released to the public 5 working days after the dispatch of the letter to the petitioner.

BACKGROUND:

Thomas Bergman, on behalf of NuScale Power, LLC (NuScale), filed a petition with the NRC on August 19, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19254B848), requesting 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."

The NRC assigned docket number PRM-50-120 to this petition and published a notice of docketing in the *Federal Register* (84 FR 63819; November 19, 2019). The NRC received 6 public comment submissions from members of the public, interest groups, and an industry representative consisting of 38 comments. Five of the six comment submissions were opposed to the petition.

DISCUSSION:

Technical Background

Ductility is the ability of a material to undergo significant deformation (i.e., stretching or bending) before rupturing or breaking. A brittle material will rupture or break with little to no deformation. The temperature at which a material transitions from a brittle to a ductile response is called the nil ductility temperature (NDT), which is often managed through an index temperature such as the reference temperature $(RT)_{NDT}$. As a material cools toward and below the NDT, the material has a much greater tendency to fracture when under mechanical stress instead of bending or deforming. Regarding reactor pressure vessels (RPVs), RT_{NDT} is important because it is critical to assessing the rate at which a reactor may heat up or cool down for a given pressure.

Materials that respond in a ductile manner can still rupture or break, but the amount of energy required to cause such a failure is significantly larger than that required for a material responding in a brittle manner. Additionally, the amount of energy required to cause a ductile failure increases with rising temperature, up to a point. Eventually, the energy required reaches a plateau known as the Upper Shelf Energy (USE), which is a measure of a material's toughness past the brittle-to-ductile transition. Defining the USE is important for ensuring that components can survive large but temporary shocks such as seismic events without failing.

Both the RT_{NDT} and USE change as a result of accumulated exposure (i.e., neutron fluence). As neutron fluence increases, the RT_{NDT} increases while the USE decreases, which means that the temperature range over which an RPV responds in a brittle manner increases, and the amount of energy required to cause a failure when the RPV is responding in a ductile manner decreases. However, the USE does not directly affect the rate at which the RPV may heat up or cool down.

For the purposes of managing the changing RT_{NDT} of an RPV throughout an operating reactor's life, the RT_{NDT} of an unirradiated RPV is referred to as the $RT_{NDT(U)}$, and the net shift in the RT_{NDT} as a result of neutron fluence is called the ΔRT_{NDT}. An adjusted reference temperature (ART) is then calculated for a given point throughout a reactor's life as the sum of $RT_{NDT(U)}$, ΔRT_{NDT}, and an uncertainty margin component.

Different regression-derived formulas for predicting ΔRT_{NDT}, known as Embrittlement Trend Curves (ETCs), have been derived over the years, and they are generally a function of fluence and a material's copper and nickel content. The NRC's guidance and regulations (Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," issued May 1988 (ADAMS Accession No. ML003740284), and Section 50.61 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Fracture toughness requirements for protection against pressurized thermal shock events," respectively) currently use an ETC that was developed in the 1980s and is based on approximately 177 data points. ASTM E900-15, as referenced by the petitioner, provides another ETC that uses more contemporary data and a more recently developed model.

Current Regulatory Framework

The regulation in 10 CFR 50.61 contains fracture toughness requirements for protection against pressurized thermal shock (PTS) events. A PTS event is an event or transient that may occur in a pressurized-water reactor causing a severe overcooling (i.e., a thermal shock) concurrent with or followed by a significant pressure spike in the RPV. The screening criteria presented in 10 CFR 50.61 must be met for a specific evaluation of the ART known as the RT_{PTS}, which is the RT_{NDT} forecast for the end-of-life fluence at the inner surface of the RPV. The procedures in 10 CFR 50.61 and 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events," prescribe the calculation of the RT_{PTS}.[1] Compliance with 10 CFR 50.61 or 10 CFR 50.61a ensures that the RPV will function appropriately should a PTS event occur.

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary for any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. Normal operational transients (e.g., heatup and cooldown) and anticipated operational occurrences must be controlled to protect the RPV from nonductile (i.e., brittle) fracture events. To realize this, the reactor must be operated within appropriate pressure and temperature limits, which are prescribed in Table 1 of Appendix G to 10 CFR Part 50. Appendix G also contains requirements for the minimum value of the USE for both unirradiated and irradiated RPV materials.

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 specifies the requirements for the reactor vessel material surveillance program. That program monitors changes in the fracture toughness properties of ferritic material in the reactor vessel from exposure to neutron irradiation and the thermal environment. The monitoring of vessel materials is required to supplement or adjust the ART and RT_{PTS} predictions calculated in accordance with 10 CFR 50.61, 10 CFR 50.61a, and RG 1.99, Revision 2. The surveillance program requires that specimen capsules be placed in the RPV at startup and periodically removed and tested.

1°F/1°F Adjustment Methodology

The "1°F/1°F adjustment methodology" is a methodology that compensates for an observed difference between predicted ΔRT_{NDT} and the available data at temperatures less than 550 degrees Fahrenheit (°F). Studies have shown that for temperatures near 550°F, a

---

1 Licensees may voluntarily apply the requirements of 10 CFR 50.61a in lieu of those in 10 CFR 50.61; however, to date, no licensees have chosen to do so.

decrease in irradiation temperature of 1°F will result in an increase of approximately 1°F in ΔRT_{NDT}. The 1°F/1°F adjustment methodology, therefore, applies an increase of 1°F in the predicted ΔRT_{NDT} for every 1°F of reactor operating temperature below 550°F. RG 1.99, Revision 2, only recommends temperature adjustment below 525°F. The NRC has previously accepted applications that apply the 1°F/1°F adjustment below this temperature.

The NRC regulations do not require use of the 1°F/1°F adjustment methodology. However, 10 CFR 50.61(c)(2) states "…licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature…." This is further elaborated upon in paragraph 1.3.2 of RG 1.99, Revision 2, which states the following with regard to the ΔRT_{NDT} evaluation procedures discussed in 10 CFR 50.61:

> The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement. The correction factor used should be justified by reference to actual data.

Effectively, the NRC guidance and regulations require the consideration of plant-specific information that could affect the level of embrittlement, such as reactor vessel operating temperature, and that a correction factor to account for such effects should be justified by reference to actual data. The 1°F/1°F adjustment methodology is one means by which a licensee can account for plant-specific operating temperature effects, but alternative methods may be proposed. The NRC first described the 1°F/1°F adjustment methodology in a 1998 presentation (ADAMS Accession No. ML110070570), described further below.

<u>Summary of the Petition Evaluation</u>

The petitioner asserts that the regulations in 10 CFR 50.61 contain "an unnecessarily burdensome requirement" for calculating the mean value of the transition temperature shift (i.e., ΔRT_{NDT}). The petitioner requests 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 3 in 10 CFR 50.61(c)(1)(iv). In addition, the petitioner requests that the formula for calculating the mean value of the transition temperature shift in ASTM E900-15 be added for use as an alternative to Equation 2 in RG 1.99.

The staff identified five main issues raised in the petition, as summarized below:

(1) The methodology for calculating ΔRT_{NDT} in 10 CFR 50.61 and RG 1.99 is overly conservative and is based on outdated information.

(2) The 1°F/1°F adjustment methodology requires excessive compensation for irradiation temperatures lower than 525°F and has significant drawbacks.

(3) The staff required NuScale to comply with 10 CFR 50.61 and RG 1.99 and to use the 1°F/1°F adjustment methodology.

(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.

(5) The current methodology for determining embrittlement in 10 CFR 50.61, with 1°F/1°F adjustment, is unnecessarily burdensome, in that it would do the following:

- Result in unnecessarily restrictive heatup and cooldown rates during startups and shutdowns.

- Cause surveillance capsules to be withdrawn and tested prematurely.

Staff Technical Evaluation of Issues Raised in the Petition

Issue 1: The methodology for calculating ΔRT_{NDT} in 10 CFR 50.61 and RG 1.99 is overly conservative and is based on outdated information.

The petitioner states that additional information pertaining to neutron embrittlement has been collected since the NRC developed 10 CFR 50.61 and RG 1.99, Revision 2. The petitioner further states that, consequently, the methodology presented is unnecessarily burdensome and excessively conservative.

The staff evaluated this issue based on the information provided in the petition as well as other pertinent sources of information, including Technical Letter Report (TLR)-RES/DE/CIB-2019-2, “Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99,” issued July 2019 (ADAMS Accession No. ML19203A089); relevant sections of the NuScale design certification application (DCA) (ADAMS Accession No. ML20224A493) and Chapter 5 of the staff’s final safety evaluation of the NuScale DCA (ADAMS Accession No. ML20205L412). The staff previously evaluated this issue for a 40-year projected fluence state as a part of the NuScale DCA review, as described in Chapter 5 of the staff’s final safety evaluation (ADAMS Accession No. ML20205L412). The staff found that while a significantly larger body of data for neutron embrittlement is now available, the 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 for the 40-year case in the NuScale DCA. The petition presents no additional information or data to demonstrate that the current regulation is overly conservative. The staff’s evaluation of the need for rulemaking to enable generic use of the petitioner’s proposed alternative methodology for modeling the effects of neutron embrittlement is discussed in Issue 4.

Therefore, the staff determines that this issue does not warrant rulemaking because the petitioner does not provide any new information that would warrant the expenditure of limited NRC resources on rulemaking.

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

The petitioner asserts that the 1°F/1°F adjustment methodology over-adjusts embrittlement predictions for their specified operating conditions by requiring too large a penalty factor for the relatively low operating temperature of the NuScale reactor design.

The staff evaluated this assertion based on the information provided in the petition as well as other pertinent sources of information, including the staff’s presentation, “Generic Letter 92-01 and RPV Integrity Assessment,” given on February 12, 1998 (ADAMS Accession No. ML110070570), describing the 1°F/1°F adjustment methodology; and Chapter 5 of the staff’s final safety evaluation of the NuScale DCA review.

The petitioner states that the steel that would be used in manufacturing a NuScale RPV possesses superior embrittlement performance characteristics. Because the petitioner did not provide additional detail regarding the embrittlement performance characteristics, the staff assumed the petitioner's statement to mean the NuScale RPV would have low copper content. The staff determined that the steels NuScale proposed to use in its DCA, or those proposed in other light-water designs known to the NRC, are represented in the operating fleet so changes to the regulations are not needed. Neither the NuScale DCA nor the petition presents any pertinent new information on embrittlement performance characteristics of the proposed materials. Therefore, the staff determines that the NuScale design presents no unusual characteristics justifying a unique temperature-embrittlement relationship. In addition, the staff finds that 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), at 40-years of operation. The staff 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 staff determines that this issue does not warrant rulemaking because the petitioner does not provide any new information beyond that approved in the staff's final safety evaluation for the NuScale DCA. Therefore, the staff does not find that this concern has sufficient merit with regard to plant operation and, consequently, does not find the issue raised in the petition to be of high priority or warrant the use of limited NRC resources.

Issue 3: The staff required NuScale to comply with 10 CFR 50.61 and RG 1.99 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 1°F/1°F adjustment methodology deman[d]ed by the staff, requires an excessive compensation for irradiation temperature less than 525°F." Therefore, the petitioner asserts that the NRC required NuScale to comply with 10 CFR 50.61 and RG 1.99, Revision 2, as well as use the 1°F/1°F methodology.

The staff evaluated this assertion based on the information provided in the petition and the pertinent sections of the NuScale DCA and the staff's final safety evaluation of the NuScale DCA. NuScale originally proposed using several alternate ETCs in its DCA but declined to provide the necessary additional information clarifying the applicability and appropriateness of use of these ETCs for the NuScale design.

NuScale declined to support its initial proposal to use alternate methods for calculating RT_{NDT} in its DCA. NuScale could have requested an exemption from the requirements of 10 CFR 50.61 but did not choose to do so. Additionally, RGs do not constitute requirements. Use of the 1°F/1°F adjustment methodology is not required; rather, it is a methodology that the staff has previously accepted for specific applications. Consequently, NuScale could have proposed an alternate adjustment methodology for the temperature correlation.

The staff determines that this issue does not warrant rulemaking because the petitioner does not provide any new information beyond that described in the NuScale DCA.

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.

The petitioner asserts that ASTM E900-15 has many superior characteristics relative to the methodology in 10 CFR 50.61 and RG 1.99, Revision 2. The petitioner further asserts that ASTM E900-15 would provide wide-ranging benefits for "future advanced reactor designs."

The staff initiated a process to evaluate alternative ETCs to that of RG 1.99, Revision 2, after (TLR)-RES/DE/CIB-2019-2, "Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99," issued July 2019 (ADAMS Accession No. ML19203A089), was presented to the Advisory Committee on Reactor Safeguards (ACRS) on November 6, 2019 (ADAMS Accession No. ML20009C415). The ACRS responded with a letter to the staff on November 27, 2019 (ADAMS Accession No. ML19331A231). Preliminary results of the staff's evaluation were presented at a public meeting on May 19, 2020 (ADAMS Accession No. ML20139A030). During the Materials Information Exchange public meeting on July 14, 2020, the NRC gave a status update on the effort (ADAMS Accession No. ML20192A002) in which it indicated that it had decided not to pursue an alternative to RG 1.99, Revision 2.

At the July 14, 2020, public meeting, 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 decision-making 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 industry 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. The NRC also noted that an applicant could potentially use this documentation as the basis for an alternative method to predict embrittlement but use of the alternative method would require additional justification or modifications in order to use the methodology at the lower operating temperature.

On October 26, 2020, the NRC staff issued the first technical letter report TLR-RES/DE/CIB-2020-09, "RG 1.99 Revision 2 Update FAVOR Scoping Study" (ADAMS Accession No. ML20300A551). 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, are relatively low. Given the low risk, the NRC staff determined that there would be very little benefit to updating RG 1.99. The staff 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" (ADAMS Accession No. ML20345A003). 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. The report also describes the other elements of a potential alternative to the RG 1.99, Revision 2, guidance, including a method to adjust ETC predictions using surveillance data, margins, default input values, and limitations. In TLR-RES/DE/CIB-2020-11, the staff concluded that ASTM E900-15 does indeed exhibit many notable characteristics, including a broader range of applicability, as well as a superior method for estimating RT_{NDT} at a variety of temperatures. However, the staff determined that ASTM E900-15 cannot directly substitute for the methodologies described in 10 CFR 50.61 and RG 1.99, Revision 2, as proposed by the petitioner 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 staff that would operate with such a low temperature, and other advanced reactor designs that the NRC staff is currently aware of would operate at substantively 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 staff finds that the petitioner's claim that ASTM E900-15 would provide wide-ranging benefits for future advanced reactor designs is not supported.

The staff determines that this issue does not warrant rulemaking because the staff has evaluated the acceptability of using ASTM E900-15 for calculating RPV embrittlement trends and has determined that the current practice continues to be adequate and appropriate for current and future reactor designs.

Issue 5: The current methodology for determining embrittlement in 10 CFR 50.61, with 1°F/1°F adjustment, is unnecessarily burdensome, in that it would do the following:

- *Result in unnecessarily restrictive heatup and cooldown rates during startups and shutdowns.*
- *Cause surveillance capsules to be withdrawn and tested prematurely.*

The petitioner asserts that the use of 10 CFR 50.61 and RG 1.99, Revision 2, with the 1°F/1°F adjustment methodology is burdensome and would result in several adverse outcomes.

The staff evaluated this assertion based on the information provided in the petition and the pertinent sections of the NuScale DCA and the staff's final safety evaluation of the NuScale DCA. As noted in the evaluation of Issue 2, the staff determines that the specifics of the NuScale design submittal do not support the assertion that unnecessarily burdensome estimates of embrittlement would occur. The staff did not find evidence that the heatup and cooldown rates pertinent to NuScale as proposed in its DCA, or those proposed in other light-water designs known to the NRC, are unduly burdensome. Furthermore, the petitioner does not provide information to substantiate this assertion.

As discussed in the staff's evaluation of Issue 2, the estimation of embrittlement effects at intermediate fluences will vary when using RG 1.99, Revision 2, and 10 CFR 50.61, versus the ASTM E900-15 methodology. Consequently, the first surveillance capsule may be removed at a different time when estimating RT_{NDT} using the approach in 10 CFR 50.61 than the time when using the ASTM E900-15 methodology. The staff believes that it does not represent unnecessary burden to remove and test the first surveillance capsule because the cost is associated with the removal activity, rather than the timing of the capsule removal. The quality and utility of the information obtained from the surveillance capsule could be impacted when using the methodology in RG 1.99, Revision 2, but it is unlikely to be of significant consequence because information from low fluence capsules is generally more indicative of material variability and measurement uncertainty than embrittlement effects. More pertinently, ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," provides a recommended withdrawal schedule, and applicants may propose an alternate withdrawal schedule for surveillance capsules. Therefore, the staff does not agree that the adverse effects as described by the petitioner are significant.

The staff determines that this issue does not warrant rulemaking because the petitioner does not establish the merits of its assertions about the unnecessary burden imposed by using the methodology in RG 1.99, Revision 2, for determining the heatup and cooldown rates during startups and shutdowns. Likewise, the petitioner does not demonstrate the merits of the concern related to the withdrawal schedules for surveillance capsules.

Public Comments on the Petition

The notice of receipt of the PRM invited interested persons to submit comments. The comment period closed on December 19, 2019. The NRC received 6 comment submissions consisting of 38 comments. Of the 6 comment submissions, five opposed the petition and one supported the petition. Enclosure 2 contains details of the comments and the corresponding NRC evaluation of each comment.

RECOMMENDATION:

The staff recommends that the Commission deny PRM-50-120 because the petitioner did not demonstrate the merits of issues raised in the petition, and the staff recently completed its evaluation of the potential use of ASTM E900-15 in another NRC process and determined that the current practice continues to be adequate and appropriate for current and future reactor designs. In addition, the staff does not believe that the issues raised in the petition are of high priority or warrant the use of limited NRC resources. Therefore, the staff recommends no amendments to the NRC's regulations at this time. The *Federal Register* notice (Enclosure 1) provides a detailed reply to the petitioner's requests and responds to public comments on PRM-50-120.

The staff requests the Commission's approval to publish the *Federal Register* notice denying PRM-50-120. The enclosed letter for signature by the Secretary of the Commission (Enclosure 3) informs the petitioner of the Commission's decision to deny the petition. The staff will inform the appropriate congressional committees of the Commission's decision.

RESOURCES:

This paper does not address any new commitments or resource implications.

COORDINATION:

The Office of the General Counsel reviewed this package and has no legal objection to the denial of the petition.

Signed by Dorman, Dan
on 10/14/21

Daniel H. Dorman
Executive Director
for Operations

Enclosures:
1. *Federal Register* notice
2. NRC Response to Public Comments on PRM-50-120
3. Letter to the Petitioner

SUBJECT: DENIAL OF PETITION FOR RULEMAKING ON ALTERNATIVE METHOD FOR CALCULATING EMBRITTLEMENT FOR STEEL REACTOR VESSELS (PRM-50-120; NRC-2019-0180) DATED: October 14, 2021

ADAMS Accession Numbers:
ML20304A000 (Pkg.)
ML20304A001 (SECY Paper)
ML20304A002 (FRN)
ML20304A003 (Summary of Public Comments and Staff Evaluation)
ML20304A004 (Letter to Petitioner) *** Via email** **SECY-012**

OFFICE	NMSS/REFS/RRPB*	QTE*	NMSS/REFS/RRPB*	NMSS/REFS/RRPB*	NMSS/REFS/RASB*
NAME	JO'Driscoll	JDougherty	GLappert	IBerrios	CBladey
DATE	11/4/2020	11/4/2020	11/9/2020	11/16/2020	11/16/2020
OFFICE	NMSS/REFS*	NRR/DNRL*	NRR/DSS*	OGC*	NRR*
NAME	JTappert	ABradford (BCaldwell for)	JDonoghue	BHarris	AVeil (AKock for)
DATE	1/22/2021	1/22/2021	12/14/2020	7/21/2021	9/23/2021
OFFICE	OEDO				
NAME	DDorman				
DATE	10/14/2021				

OFFICIAL RECORD COPY