From:	Carolyn Lauron
To:	Justin Hawkins
Cc:	Greg Cranston; Andrew Brenner; Jordan Glisan; Michael Dudek
Subject:	NRC Staff Response to Questions regarding RG 1.99 Revision 2, Radiation Embrittlement of Reactor Vessel Materials
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Hi Justin –

Below is the NRC staff response to the questions regarding Regulatory Guide 1.99, Revision 2, on Radiation Embrittlement of Reactor Vessel Materials.
If you have questions or need more information, please let us know.

Thanks,
Carolyn Lauron
US NRC

The SMR-160 team had some questions related to Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, that came up during development of our Pressure-Temperature (P-T) curves:[1]

1. **<u>Context for the Question below:</u>** RG 1.99, Section 1.3 Limitations – Limitation number 2 states, "*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*."

   SMR-160 irradiation temperature of the reactor vessel beltline will be approximately 465°F, which is outside the limitation of the RG 1.99 procedure. SMR-160 plans to address this by following approved methodology found in a staff presentation (ML110070570), as referenced in NRC eRAI No.: 9118 and discussed in NuScale's response to the stated eRAI (ML18096B882).[2], [3] **This consists of adding a degree-for-degree to the results from RG 1.99, Revision 2, starting from 525°F, or approximately 60°F (=525-465).**

   SMR-160 notes this approach was used for the NuScale design which assumes an irradiation temperature of 497°F.

   <u>Question 1:</u> Is this approach still acceptable for the SMR-160 design given the assumed irradiation temperature of 465°F?

NRC Staff Response to Question 1:

As noted in the question the NuScale operating temperature is 497°F, while the proposed SMR-160 temperature is 465°F. The temperature effect on embrittlement is known to increase below the traditional operating window of the current nuclear fleet. The proposed operating temperature for the SMR-160 is substantially lower than where the NRC understands the degree-per-degree approximation to be appropriate. Consequently, the basis for acceptance of the NuScale operating temperature would likely not be applicable due to the significantly lower proposed SRM-160 temperature. The NRC staff would require substantial data-based support to extend the NuScale approach to the SMR-160 as proposed here.

2. **Context for the Question below:** RG 1.99, Section 1.3, Limitation number 3 states, "*Application of these procedures to fluence levels or to copper or nickel content beyond the ranges given in Figure 1 and Tables 1 and 2 or to materials having chemical compositions beyond the range found in the data bases used for this guide should be justified by submittal of data*."

   SMR-160 preliminary evaluations provide a bounding fluence at 80 years of approximately **1.6 E20 n/cm^2**, which is beyond the range given in Figure 1 of RG 1.99 (limit is 1 E20 n/cm^2).

   Question 2: The curve in RG 1.99 - Figure 1 appears to be a fit to data to correlate fluence factor to fluence. Is the data used to create Figure 1 available, and if data exists at a fluence beyond 1E20 n/cm^2, can that be used to inform an acceptable fluence factor for SMR-160?

NRC Staff Response to Question 2:

Figure 1 is based on curve fitting. The information cited in response to Question 3 is pertinent to this question. The data from which RG 1.99, Revision 2, curves were drawn does not appear directly applicable to the SMR-160 based on the information provided in Question 1. In addition, RG 1.99, Revision 2, is known to have reduced accuracy at high fluences, with the effect increasingly pronounced at high fluence. The information linked in response to Question 3 provides further elaboration.

The NRC has issued a SECY related to this high-fluence issue.[4]

   Question 3: The NRC noted in a 2014 review of RG 1.99 (ML13346A003) that, while Revision 2 is acceptable for continued use, a detailed evaluation of embrittlement prediction methodologies, data, and understandings to assess their impact on RG 1.99 was intended for publication in approximately two years' time (~2016).[5] Was this evaluation published and, if not, is the data that would inform the evaluation publicly available?

NRC Staff Response to Question 3:

The staff issued its report in 2019.[6] The data used to perform the analyses presented in the report is available from the American Society of Testing and Materials (ASTM) in the ADJE090015-EA package.[7] The ASTM Subcommittee E10.02 on Behavior and Use of Nuclear Structural Materials maintains a database of relevant data that has been updated since this package was released.

The NRC also conducted several pertinent public meetings concerning RG 1.99, Revision 2, and related topics on May 19, 2020, and October 18, 2021. Slides and background materials are referenced in the public meeting summaries available in ADAMS.[8], [9]

The May 19, 2020, slides contain draft thoughts concerning the potential use of an alternate trend curve, ASTM E900-15, for example. The NRC has not revised RG 1.99, Revision 2 to date and the SECY referenced above provides relevant information concerning the NRC's current activities on this topic.

References:

1. US NRC, RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003740284) https://www.nrc.gov/docs/ML0037/ML003740284.pdf
2. US NRC, "Generic Letter 92-01 And PV Integrity Assessment, Status, Schedule and Issues - Presentation Given at NRC/Industry Workshop Held February 12, 1998," dated February 12, 1998. (ML110070570) https://www.nrc.gov/docs/ML1100/ML110070570.pdf
3. NuScale Power, LLC, "NuScale Power, LLC Response to NRC Request for Additional Information No. 234 (eRAI No. 9118) on the NuScale Design Certification Application. (Non-Proprietary)," April 6, 2018. (ML18096B882) https://www.nrc.gov/docs/ML1809/ML18096B882.pdf
4. US NRC, SECY-22-0019, "Rulemaking Plan for Revision of Embrittlement and Surveillance Requirements for High-Fluence Plants in Long-Term Operation," dated March 8, 2022. (ML21314A215) https://www.nrc.gov/docs/ML2131/ML21314A215.pdf
5. US NRC, "2013-12-11- enclosure - RG1 99 Staff Review," dated January 9, 2014. (ML13346A003) https://www.nrc.gov/docs/ML1334/ML13346A003.pdf
6. US NRC, "Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99 - Technical Letter Report," dated July 31, 2019. (ML19203A089) https://www.nrc.gov/docs/ML1920/ML19203A089.pdf
7. ASTM Standard ADJE090015-EA, "Adjunct for E900-15 Technical Basis for the Equation Used to Predict Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials."
8. US NRC, "May 19, 2020, Summary of the Category 2 Public Meeting on Regulatory Guide 1.99 Revision 2 and Reactor Vessel Material Surveillance Program," dated June 15, 2020. (ML20168A009) https://www.nrc.gov/docs/ML2016/ML20168A009.pdf
9. US NRC, "10/18/2021-Public Meeting Summary RE: Reactor Pressure Vessel Embrittlement Monitoring and Prediction in Long-Term Operation," dated November 10, 2021. (ML21309A304) https://www.nrc.gov/docs/ML2130/ML21309A034.pdf