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NRC Meeting:
SMR-160 Reactor Pressure Vessel Embrittlement

February 22, 2023

Meeting Agenda

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- Introductions
- Purpose & Outcome
- Overview SMR-160 Reactor Pressure Vessel
- Overview of Regulations
- Discussion of Radiation Embrittlement Methodologies
- Questions provided to NRC
- Open Forum

Introductions

- NRC staff

- Holtec staff

Purpose & Outcome

PURPOSE: To give a high-level overview of the SMR-160 RPV design and to discuss various embrittlement trend curves and their potential application to SMR-160.

OUTCOME: To obtain feedback from the NRC staff on the reasonableness of applying different embrittlement trend curves in the generation of SMR-160 P-T curves.

Overview of SMR-160 Reactor Pressure Vessel

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- Cold leg/irradiation temperature [[ ]]
- Maximum estimated neutron fluence [[

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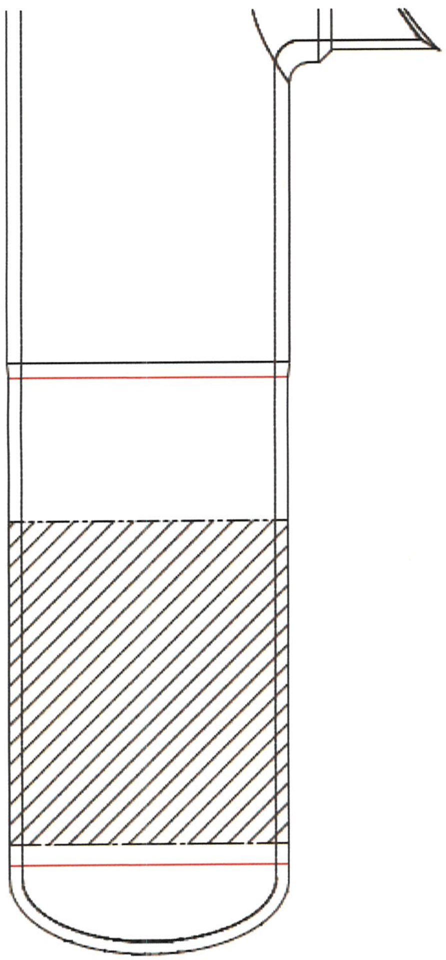
  - Fluence calculations based on RG 1.190 framework
  - Flux at RPV surface [[ ]]
  - Fluence for 40 years (38 EFPY) is [[ ]]

BTP 5-3 Fracture Toughness Requirements

- BTP 5-3 discusses the locations of concern for operating limitations for RPV fracture toughness:
  - ✓ Beltline materials are defined to be those materials directly surrounding the effective height of the active core and adjacent materials estimated to receive a neutron fluence of 1×10^{17} n/cm^2 or higher
  - ✓ RPV regions of higher stress than the beltline
  - ✓ RPV nozzles

RPV Beltline Materials and Fluence



- Previous maximum estimated fluence sent to NRC based on older RPV geometry
  - ✔ Increased RPV ID and downcomer water thickness
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General Design Criteria

- GDC 31, *Fracture Prevention of Reactor Coolant Pressure Boundary*, states:
  - The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

10 CFR 50 Appendix G

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Operating condition	Vessel pressure[1]	Requirements for pressure-temperature limits	Minimum temperature requirements
1. Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤20%	ASME Appendix G Limits	(2)
1.b Fuel in the vessel	>20%	ASME Appendix G Limits	(2) +90 °F(6)
1.c No fuel in the vessel (Preservice Hydrotest Only)	ALL	(Not Applicable)	(3) +60 °F
2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences:			
2.a Core not critical	≤20%	ASME Appendix G Limits	(2)
2.b Core not critical	>20%	ASME Appendix G Limits	(2) + 120 °F(6)
2.c Core critical	≤20%	ASME Appendix G Limits + 40 °F.	Larger of [(4)] or [(2) + 40 °F.]
2.d Core critical	>20%	ASME Appendix G Limits + 40 °F.	Larger of [(4)] or [(2)+160 °F]
2.e Core critical for BWR (5)	≤20%	ASME Appendix G Limits + 40 °F.	(2)+60 °F

[1] Percent of the preservice system hydrostatic test pressure.
[2] The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.
[3] The highest reference temperature of the vessel.
[4] The minimum permissible temperature for the inservice system hydrostatic pressure test.
[5] For boiling water reactors (BWR) with water level within the normal range for power operation.
[6] Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

- ASME BPV Code Section XI Appendix G provides a methodology for determining P-T limits based on the reference nil ductility temperature (RT_{NDT}) of the material. This methodology instructs users to consider irradiation effects, does not prescribe a specific method to follow.

RG 1.99 Radiation Embrittlement of Reactor Vessel Materials

- RG 1.99 Rev. 2 provides an NRC-approved methodology for determining the adjustment to reference transition temperature due to neutron radiation embrittlement effects (ΔRT_{NDT}). The calculated adjusted reference temperature (ART) is used in combination with the ASME Appendix G methodology to determine P-T limits.

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P-T Curve based on RG 1.99

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TLR-RES/DE/CIB-2019-2

- July 2019 NRC assessment of RG 1.99 Rev 2 (TLR-RES/DE/CIB-2019-2)
  - Fluence factor becomes increasingly inaccurate at fluences greater than 3 to 6 x 10^{19} n/cm^2 and invalid as fluence exceeds 10^{20} n/cm^2
  - "It is likely that use of the degree-per-degree adjustment becomes more inaccurate as one moves away from 288°C (550°F) and/or is too simple to fully account for a spectrum of Cu and temperature variances."
- "The NRC staff found that the ASTM E900-15 ETC provided the most accurate characterization of [the database reviewed by TR-RES/DE/CIB-2019-2]" – October 18, 2021 meeting (ML21270A002)

ASTM E900-21

- E900 provides a method for predicting values of reference transition temperature shift (TTS) for irradiated pressure vessel materials
- Calibrated to:
  - Irradiation temperature within the range from 491°F to 572°F
  - Fluence within the range from 1×10^{17} n/cm^2 to 2×10^{21} n/cm^2 (E > 1 MeV)
  - RPV material chemical content:
    - Cu up to 0.4 wt %
    - Ni up to 1.7 wt %
    - P up to 0.03 wt %
    - Mn between 0.55 and 2.0 wt %
- Only power reactor (PWR and BWR) surveillance data were used in the formulation of E900

Calculation of ART for E900

$$TTS = TTS_1 + TTS_2 \quad (1)$$

where:

$$TTS_1 = A \cdot \frac{5}{9} \cdot 1.8943 \times 10^{-12} \cdot \Phi^{0.5695} \left(\frac{1.8 \cdot T + 32}{550} \right)^{-5.47} \left(0.09 + \frac{P}{0.012} \right)^{0.216} \left(1.66 + \frac{Ni^{8.54}}{0.63} \right)^{0.39} \left(\frac{Mn}{1.36} \right)^{0.3} \quad (2)$$

$$A = \begin{pmatrix} 1.011 \text{ for forgings} \\ 1.080 \text{ for plates and SRM plates} \\ 0.919 \text{ for welds} \end{pmatrix} \quad (3)$$

$$TTS_2 = \frac{5}{9} \cdot \max[\min(Cu, 0.28) - 0.053, 0] \cdot M \quad (4)$$

$$M = B \cdot \max\{\min[113.87 (\ln(\Phi) - \ln(4.5 \times 10^{20})), 612.6], 0\} \cdot \left(\frac{1.8 \cdot T + 32}{550} \right)^{-5.45} \left(0.1 + \frac{P}{0.012} \right)^{-0.098} \left(0.168 + \frac{Ni^{0.58}}{0.63} \right)^{0.73} \quad (5)$$

$$B = \begin{pmatrix} 0.738 \text{ for forgings} \\ 0.819 \text{ for plates and SRM plates} \\ 0.968 \text{ for welds} \end{pmatrix} \quad (6)$$

$$SD = C \cdot TTS^D \quad (10)$$

$$C = \begin{pmatrix} 6.972 \text{ for forgings} \\ 6.593 \text{ for plates and SRM plates} \\ 7.681 \text{ for welds} \end{pmatrix} \quad (11)$$

$$D = \begin{pmatrix} 0.199 \text{ for forgings} \\ 0.163 \text{ for plates and SRM plates} \\ 0.181 \text{ for welds} \end{pmatrix} \quad (12)$$

$$ART = Initial\ RT_{NDT} + TTS + 2SD$$

Temperature Factor

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Flux Effects on Embrittlement

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Example Temperature Factor

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Example Temperature Factor

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Debarberis Temperature Factor

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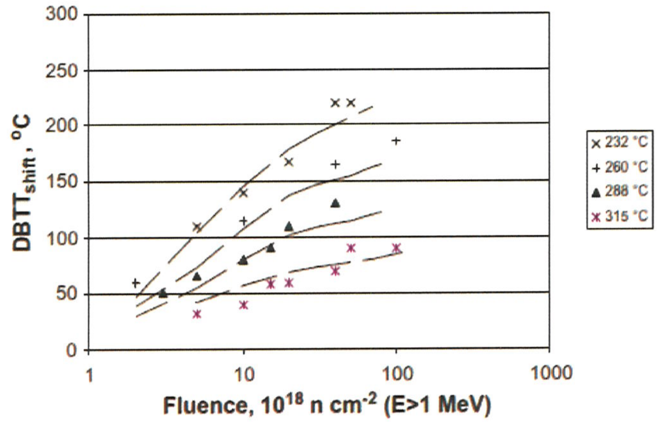

Fig. 1. Fitting of available AMES Report No. 6 data.

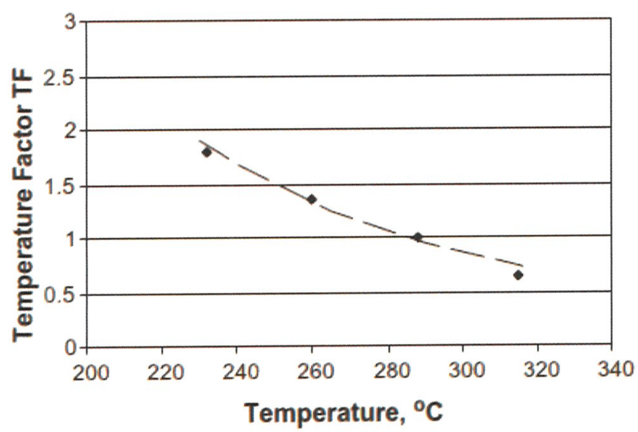

Fig. 2. Temperature dependence of the temperature factor.

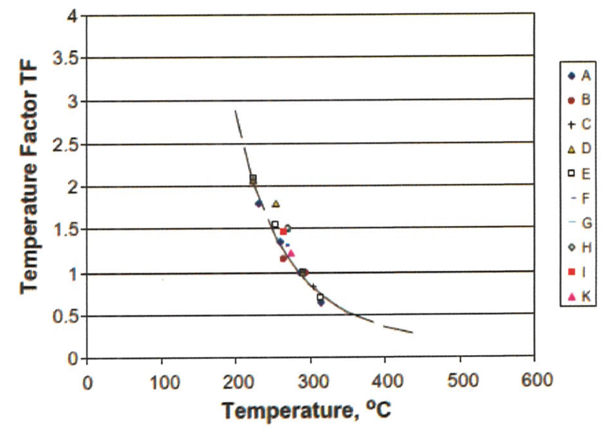

Fig. 3. Temperature factor versus temperature for data in the range 232–315 °C.

Application of TF

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SMR Actions Going Forward

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SECY-22-0019

Impacts on New and Advanced Reactors

In addition to the current operating fleet, the NRC staff investigated the impacts of the ETC in RG 1.99 on new light-water reactors. It is expected that all new known light-water-reactor designs will use modern RPV material chemistries having a low copper content. The staff has demonstrated in TLR-RES/DE/CIB-2019-2 that significant mispredictions of embrittlement may occur for these low-copper-content RPV material chemistries. While it is possible that the ETC in RG 1.99 produces mispredictions of embrittlement for low copper materials, the actual embrittlement of new reactor materials is likely to be low and will not result in a safety issue due to selection of low-copper materials. In addition, some small modular light-water reactor designs operate at temperatures outside of the temperature range for the data used in the development of the current ETC in RG 1.99, thus increasing the uncertainty in the embrittlement prediction. Since the current database used in the development of the ETC in RG 1.99 has no surveillance data for these non-light-water reactor designs, this ETC, and any other that is based on the current surveillance data from light-water reactors, is not applicable to those designs. However, a conservative correction factor may be used to compensate for the misprediction in embrittlement due to temperature, but like chemistry, the expected embrittlement of these new materials is also expected to be low and not a safety issue.

Questions

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Questions

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Open Forum

References

1. Debarberis et al., Effect of irradiation temperature in PWR RPV materials and its inclusion in semi-mechanistic model, Scripta Materialia, Volume 53, Issue 6, 2005, Pages 769-773.
2. Petrequin, A review of formulas for predicting irradiation embrittlement of reactor vessel materials, AMES Rep. No. 6, EUR 16455, Luxembourg: EC, 1996.