UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

November 27, 2019

Ms. Margaret M. Doane
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99

Dear Ms. Doane:

During the 668th meeting of the Advisory Committee on Reactor Safeguards, November 6-8, 2019, we completed our review of the staff's technical letter report, TLR-RES/DE/CIB-2019-2, "Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide (RG) 1.99." Our Metallurgy & Reactor Fuels Subcommittee reviewed this technical letter report on August 22, 2019. During these meetings, we had the benefit of discussions with the staff and the Electric Power Research Institute (EPRI). We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATION

1. The embrittlement trend correlation (ETC) in RG 1.99, Revision 2 (the RG) has a number of deficiencies, the most significant of which is increasing error beyond a fluence of 6×10^{19} n/cm^2 (E >1 MeV).

2. The American Society for Testing and Materials (ASTM) Subcommittee E10.02, Behavior and Use of Nuclear Structural Material, has performed an extensive review of several ETCs. It concluded that the correlation in ASTM E900-15, that is based on a much more extensive database, overcomes the deficiencies in the RG and provides the best fit at higher fluences.

3. A staff working group has been established and has identified a path forward for addressing this issue.

4. A staff oversight group has also been established to guide the implementation of a revision to the RG to correct its deficiencies. This group should consider each plant's situation to eliminate unnecessary burden on plants for which reactor pressure vessel (RPV) limits are not challenged.

BACKGROUND

Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, describes methods that may be used to predict the effects of radiation embrittlement of RPVs.

Specifically, neutron irradiation of the RPV steel results in material property changes making the steel more brittle and potentially susceptible to rapid failure under high-stress conditions. This effect increases with neutron fluence. The embrittlement of RPV steels can pose a safety challenge that impacts operational pressure-temperature limits. An embrittlement trend correlation is used to calculate the shift in reference nil-ductility temperature (ΔRT_{NDT}) as a function of fast neutron fluence.

The most recent revision of this guide (Revision 2) was published in 1988. At that time, the number of data points available for development of the correlation was 177. It was expected at the time of publication that the regulatory guide would be updated and refined as more material data became available. The current data base now contains approximately 1900 data points.

The staff recently evaluated predictions using RG 1.99, Revision 2 for higher fluences that will be experienced during subsequent license renewal (SLR) periods. Results demonstrate that the correlation in this RG introduces significant errors that are non-conservative at higher fluence. The adoption of new guidance regarding prediction of the effects of embrittlement may have significant impact on all operating Pressurized Water Reactors (PWRs) that previously used RG 1.99, Revision 2 to develop Pressure-Temperature (P-T) curves, Low Temperature Overpressure Protection setpoints, and pressurized thermal shock (PTS) limits.

In response to the increasing number of plants that have applied for SLR, the industry has embarked on an extensive program of data gathering at high fluence. It is expected that actual plant data will be available for verification of embrittlement trends well before the existing PWRs will require it for extended operation.

DISCUSSION

Regulatory Guide 1.99, Revision 2 ETC Deficiencies

The technical letter report identifies several deficiencies with the current Revision 2. These are:

- Non-conservativism at high fluence for base metals
- Inaccuracies for reactor vessel materials with low copper content
- Underestimated standard deviation relative to the current database
- Conservative bias at low-to-mid fluences
- Lack of temperature adjustment

These deficiencies raise potential safety margin issues because resulting estimates of reference nil-ductility transition temperature shift (ΔRT_{NDT}) may be non-conservative for plants with vessels exposed to higher levels of neutron fluence anticipated during subsequent license renewal. Figure 1 contains plots of the residual of ΔRT_{NDT} for both welds and base metal. Residuals are computed as the difference between the RG 1.99, Revision 2 predicted value and the measured value from the current large embrittlement data base, incorporating both US and international data. A negative residual value indicates non-conservatism. The technical letter report analysis suggests that between fluences of 3×10^{19} n/cm^2 and 6×10^{19} n/cm^2 (E > 1 MeV), the mean residual becomes increasingly non-conservative.

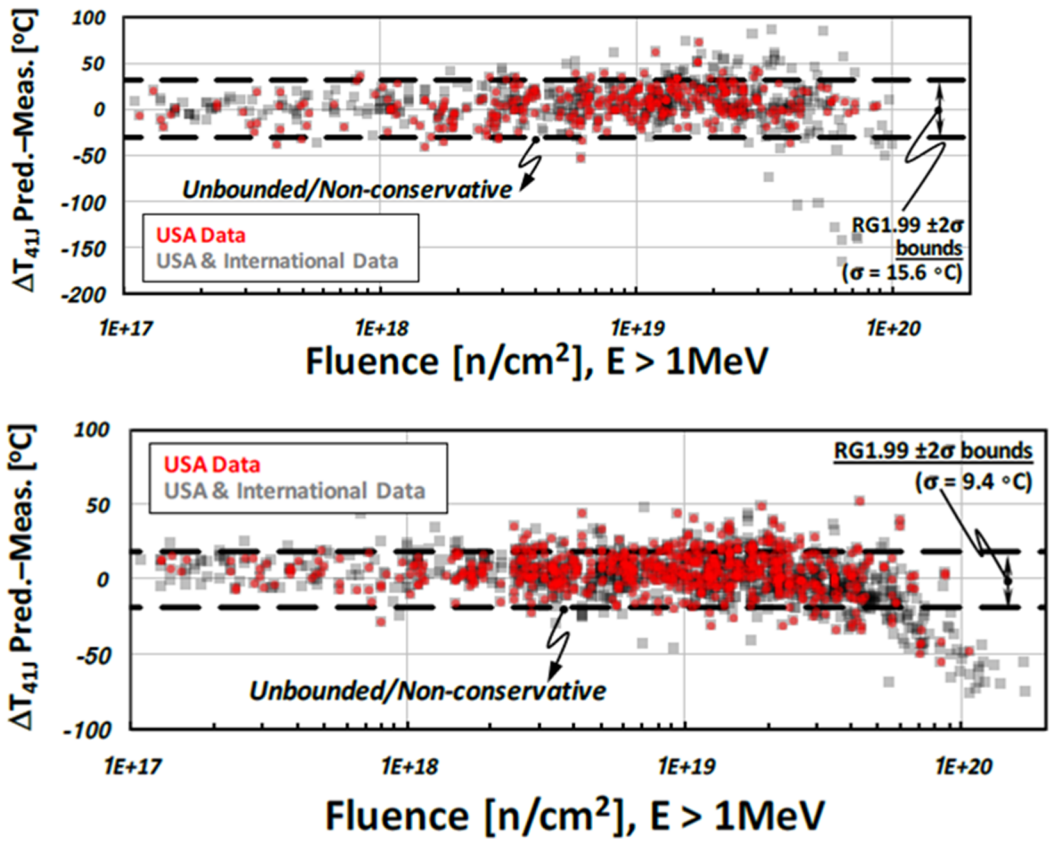

Figure 1. Estimates of the residual (RG 1.99, Revision 2 predicted minus measured values) of ΔRT_{NDT} using the current US and international database for (a) welds and (b) base metal.

Standard deviation values from the RG are used as a defined margin in RPV embrittlement calculations (horizontal dash lines in Figure 1). The current large embrittlement database has a larger standard deviation (greater scatter) than the original database used to develop the ETC in the RG. This represents another potential non-conservativism.

Evaluation of Modern Embrittlement Trend Correlations

The ASTM Subcommittee E10.02, Behavior and Use of Nuclear Structural Materials, has performed an extensive review and comparison of several ETCs, including that in the RG, and the correlation used for the alternate PTS rule (10 CFR 50.61a). The ASTM E900-15,"Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," correlation was found to represent the best fit to the current expanded data base and is more predictive of the behavior at the higher fluences expected during subsequent license renewal. The ASTM E900-15 ETC also has a built-in temperature adjustment term, addressing another of the deficiencies identified in the RG ETC.

Impact on Operating Plants

Electric Power Research Institute (EPRI) estimated the effects of adopting the ASTM E900-15 correlation for RPVs with high or low copper content. In the case of boiling water reactors (BWRs), where the fluence to the reactor vessel is an order of magnitude lower than PWRs, use of the ASTM E900-15 correlation will not be necessary. For PWRs, the effect can

be significant in some cases. Additionally, the limiting RPV component (weld or plate) may change. As projected fluences increase from 10^{19} to 10^{20} n/cm^2 (E > 1 MeV), EPRI results indicate increases in ΔRT_{NDT} of 25° to 75°F above that predicted using the RG (irrespective of the copper content). The NRC staff and EPRI estimate that none of the existing PWR fleet will be affected until circa 2025.

Working and Oversight Groups

The NRC staff has established a working group to complete the following tasks:

- Recommend an alternative ETC.
- Determine limitations of ETC implementation.
- Determine how to apply individual plant surveillance data.
- Determine margins on ETC.
- Determine default values for inputs that are not available.
- Write a draft revised RG for internal review.

The staff working group has adopted the correlation in ASTM E900-15. Completion of the tasks identified by the working group will provide a sound basis for revision of the RG. The working group also presented a schedule for completion of the identified tasks and development of a draft revised guide.

The oversight group is charged with overall supervision of the effort and to provide guidance on implementation. Given that each plant has a different operating temperature, material chemistry, and projected end-of-life fluence, the oversight group should develop an implementation path forward considering each plant's situation. This would eliminate unnecessary burden on plants for which RPV limits are not challenged.

CONCLUSIONS

The ETC in RG 1.99, Revision 2 has a number of deficiencies, the most significant of which is increasing error at high fluence. The correlation in ASTM E900-15, that is based on a much more extensive database, overcomes the deficiencies in this RG and provides the best fit at higher fluences.

A staff working group has been established and has identified a path forward for addressing this issue. A staff oversight group has also been established to guide the implementation of a revised RG.

We look forward to reviewing the updated RG and implementation plan.

Sincerely,

/RA/

Peter Riccardella
Chairman

REFERENCES

1. Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99 Technical Letter Report, TLR-RES/DE/CIB-2019-2, July 31, 2019 (ML19203A089)

2. Assessment of Predictions of RTNDT and Upper Shelf Energy made using Branch Technical Position 5-3, March 23, 2017 (ML16341B108)

3. Regulatory Guide 1.99 Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 31, 1988 (ML003740284)

4. SECY-09-0059, Final Rule to Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a) (RIN 3150-A101) April 9, 2009 (ML083470915)

5. NUREG-2163, Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule – Final Report, September 30, 2018 (ML18255A118)

6. BWRVIP-86, Revision 1-A: BWR Vessel and Internals Project, “Updated BWR Integrated Surveillance Program (ISP) Implementation Plan,” Final Report, May 2013 (ML13176A097)

7. ASTM E900-15, Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, February 1, 2015

8. ADJE090015-EA, Technical Basis for the Equation used to Predict Radiation Induced Transition Temperature Shift in Reactor Vessel Materials

9. 10 CFR Appendix G to Part 50 – Fracture Toughness Requirements

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