April 6, 2015

MEMORANDUM TO: John W. Lubinski Division of Engineering
Office of Nuclear Reactor Regulation

FROM: Robert O. Hardies, Sr. Level Advisor ***/RA by E-mail//***
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF FEBRUARY 19, 2015, PUBLIC MEETING TO DISCUSS REACTOR PRESSURE VESSEL ISSUES

On February 19, 2015, a Category 2 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of industry to allow an exchange of information about reactor pressure vessel integrity issues. A portion of the meeting related to the staff's recent evaluation of potential non-conservatisms in NRC Branch Technical Position (BTP) 5-3, "Fracture Toughness Requirements." Other topics addressed included reactor vessel surveillance programs, Title 10, *Code of Federal Regulations*, Part 50, (10 CFR 50) Appendices G and H evaluations, status of the NRC Regulatory Guide for Alternate Pressurized Thermal Shock (PTS) Rule Implementation, a discussion of ASME Codes requirements for pressure testing while the reactor vessel is critical, and a status of NRC work activities on the Reactor Embrittlement Archive Project (REAP) Database.

The NRC staff made a two-part presentation on the recent NRC activities performed regarding the potential non-conservatism of BTP 5-3. The first part of the presentation provided a review of definitions and estimates of unirradiated reference temperature (RT_{NDT}) and unirradiated upper shelf energy (USE) addressed in BTP 5-3, summarized the background of the recent questions concerning the potential non-conservatism of BTP 5-3, and summarized the objectives of the staff's analysis of BTP 5-3. The second part of the staff's presentation provided an assessment of the impact of the potential non-conservatisms in BTP 5-3 on U.S. plants with regard to PTS and pressure temperature (P-T) limit evaluations based on recent docketed information. The staff concluded that one PWR PTS evaluation is potentially affected, some PWRs and BWRs P-T limits are potentially affected, and that there are no immediate safety concerns. The staff provided a tentative schedule for completion of a report that documents the staff's findings.

The industry followed with three presentations on their recent activities performed regarding the potential non-conservatism of BTP 5-3. The first presentation provided background on the industry focus groups that have been formed to address the BTP 5-3 issues, the membership of the focus groups, and the focus groups activities that are currently underway. The second presentation provided a summary of the Materials Reliability Project (MRP)/Boiling Water Reactor Vessel and Internals Project (BWRVIP) focus group activities and results to-date. The objectives of the focus group activities include conducting a survey regarding use of BTP 5-3 in the PWR and BWR fleets, evaluating the BTP 5-3 procedures which had previously been identified as potentially non-conservative, determining if application of BTP 5-3 for defining reactor vessel P-T limits provides adequate margins against failure through the 60-year license period (EOLE), and, if needed, recommending alternative procedures to ensure that adequate margins against failure are maintained through EOLE. The focus group has completed a draft report that is under review with a target completion in June 2015. The focus group is considering similar work for a procedure developed by GE-Hitachi Nuclear Energy that is similar to BTP 5-3. The third presentation provided a summary of the Pressurized Water Reactor Owners Group (PWROG) activities regarding their Material Orientation Toughness Assessment (MOTA) for the purpose of mitigating BTP 5-3 uncertainties. The objective of the MOTA is to explore existing deterministic margin that may be potentially available in ASME Code Section XI, Appendix G and other NRC-approved sources to address potential non-conservatisms in BTP 5-3. The results of the industry's investigation to-date demonstrate that current methods for developing P-T limits are acceptable in light of the identified BTP 5-3 estimation uncertainties. The PWROG intends to document the MOTA in a final report later this year.

The NRC staff provided two presentations on the status of their 10 CFR 50 Appendix G research activities to-date. The first presentation provided a summary of overall activities to-date, and summarized the new version of the FAVOR computer code to address software errors previously identified by the industry. A release of FAVOR v15.1 that remedies the software errors is anticipated in Spring 2015. The second presentation summarized the staff's efforts regarding an advanced residual stress model under investigation for implementation into FAVOR. Further efforts are underway to explore other fracture mechanics models and to assess the ability to adopt an appropriate model into the FAVOR code at a later date.

The PWROG discussed an assessment of the margins associated with American Society of Mechanical Engineers (ASME) Section XI Appendix G pressure-temperature (P-T) limits for pressurized water reactor (PWR) nozzles. The assessment is intended to generically address 10 CFR 50 Appendix G requirements recently clarified by the NRC in Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components." This RIS clarifies that 10 CFR 50 Appendix G requires that P-T limits sufficiently address all ferritic materials of the reactor vessel, including the impact of structural discontinuities such as nozzles. The purpose of the industry's activity is to develop a basis for generically addressing nozzles supporting P-T limit submittals and to justify the use of the reactor vessel shell region with the highest embrittlement as the limiting region to be used for P-T limits.

The NRC staff presented their plans to begin rulemaking for revising 10 CFR 50 Appendix H based on recent commission approval to do so. The planned revisions will include changes to permit the use of updated and modern revisions of ASTM standards and, in response to industry requests, to allow an increase in the required time to submit surveillance program test results. The longer reporting time is intended to mitigate scheduler complications that modern integrated surveillance programs have experienced related to the logistics of withdrawal, shipping, testing and reporting results of testing of surveillance capsules.

The NRC staff presented a status and the latest tentative schedule for issuing a draft Regulatory Guide (RG) describing guidance for implementation of the Alternate Pressurized Thermal Shock Rule, 10 CFR 50.61a. The RG was published in the Federal Register for a 60-day public comment period on March 13, 2015.
The NRC staff presented an ASME Code item related to pressure testing with the reactor vessel core critical. Although NRC regulations in 10 CFR 50 specifically prevent pressure

testing of the pressure vessel while the core is critical, ASME Section XI repair/replacement activities allow pressure testing of other non-RPV Class 1 components while the reactor vessel core is critical. The staff is considering additional actions to clarify the NRC position on this item.

The NRC staff discussed the Reactor Embrittlement Archive Project. This project catalogues historical records of surveillance data and research irradiation information.

Action items captured during the meeting were as follows:

1. NRC and industry will explore opportunities to share longitudinal and transverse Charpy data sources to facilitate data verification.
2. The NRC will evaluate the need to request and review the EPRI sponsored evaluation of BTP 5-3 conservatism.
3. The NRC will evaluate the need to request and review the PWROG report on MOTA.
4. The NRC and industry will evaluate further communication opportunities at ASME Code meetings.

A list of attendees is enclosed. The slide presentations presented by the NRC staff and the industry representatives can be found in the Agencywide Documents Access and Management System (ADAMS) at Accession Number ML15061A072.

Enclosure:
List of Attendees

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Action items captured during the meeting were as follows:

5. NRC and industry will explore opportunities to share longitudinal and transverse Charpy data sources to facilitate data verification.
6. The NRC will evaluate the need to request and review the EPRI sponsored evaluation of BTP 5-3 conservatism.
7. The NRC will evaluate the need to request and review the PWROG report on MOTA.
8. The NRC and industry will evaluate further communication opportunities at ASME Code meetings.

A list of attendees is enclosed. The slide presentations presented by the NRC staff and the industry representatives can be found in the Agencywide Documents Access and Management System (ADAMS) at Accession Number ML15061A072.

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List of Attendees

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ADAMS Accession Nos.:

Package: ML15061A072

Meeting Summary: ML15096A128 * By E-mail

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DATE	04/06/2015

OFFICIAL RECORD COPY

List of Attendees

Public Meeting with U.S. Nuclear Regulatory Commission (NRC) Staff to Discuss Reactor Pressure Vessel Issues

February 19, 2015

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List of Attendees (concluded)

Public Meeting with U.S. Nuclear Regulatory Commission (NRC) Staff to Discuss Reactor Pressure Vessel Issues

February 19, 2015

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