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877-4SI-POWER

ASME Code, Section XI, Appendix G P-T Limits

By: *TIM GRIESBACH*
tgriesbach@structint.com

CHRIS LOHSE
clohse@structint.com

ASME Section XI, Appendix G defines the ASME Code criteria and methodology to establish heatup and cooldown pressure-temperature (P-T) limits and pressure test conditions. Recently there have been questions from the NRC about the adequacy of the Appendix G method for protection of brittle fracture for all ferritic components in the reactor coolant pressure boundary. This may require some plants to identify and characterize the vessel material properties (e.g., RT_{NDT}) that are outside the traditional vessel beltline region.

The draft NRC Regulatory Issue Summary 2014-XX (ADAMS Search No. ML13301A188) identifies the issue as such:

- In determining P-T limits, reactor vessel materials with the highest reference temperature may not always be limiting because the consideration of stress levels from structural discontinuities (such as nozzles) may be more limiting. Licensees should ensure that P-T limits (including NRC-approved PTLRs) sufficiently address all ferritic materials of pressure-retaining components of the RCS. Specifically, all ferritic components within the RCPB must be considered for any materials that are predicted to experience end-of-license neutron exposure greater than 1x1017 n/cm2 (E > 1 MeV).

- When developing new P-T limit curves, or when evaluating the applicability of the existing curves and pressure test temperatures for extended plant operation, it is likely that these concerns will need to be addressed in more detail. Specifically, for those areas not currently included in evaluations for P-T limits for the reactor pressure vessel, additional considerations are needed from both a materials perspective and stress concentrations due to nozzle corners or discontinuities.

Structural Integrity has an approved PTLR (licensing approved generic method) for use by BWRs and we have developed many plant's P-T limits for heatup/cooldown and leak tests. We've also developed PWR operating P-T limit curves and performed third party reviews for utilities preparing licensing submittals. We are available to assist utilities with Appendix G issues.

NRC RAIS FOR WESTINGHOUSE AND CE PLANTS ON RVI AMPS/INSPECTION PLANS

By: CHRIS LOHSE
clohse@structint.com

The industry developed MRP-227 to address aging of reactor vessel internals (RVI) for PWR plants. MRP-227 was then submitted for NRC review on January 12, 2009. The NRC had a number of comments that were addressed through Request for Additional Information (RAI) responses between the industry and the NRC.

The NRC issued an Safety Evaluation (SE) for MRP-227 in June of 2011 and the final approved version of MRP-227 (MRP-227A) was issued in December of 2011.

The SE of MRP-227 did not give a blanket approval of MRP-227A for use of the document. In the SE, the NRC added a number of Applicant/Licensee Action Items (ALAI) that each utility must address for their plant. These ALAIs are normally addressed during development of the RVI AMP. For Westinghouse and CE plants, there were a number of questions on whether the assumptions made in MRP-227 were applicable for all plants. Meetings were held between the industry and the NRC and a definitive conclusion could not be made that the MRP-227 assumptions were applicable and bounding to all CE and Westinghouse plants. Therefore, the NRC is issuing two plant-specific RAIs to all Westinghouse and CE plants that have submitted RVI AMPs/ inspection plans for review.

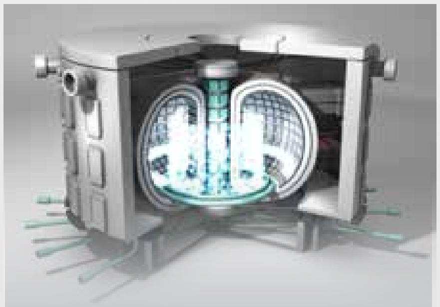
These RAIs are:

1. Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)
2. Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

Westinghouse and EPRI worked together with the NRC to develop a guidance document for addressing these RAIs. It was issued October 24, 2013 as Letter MRP 2013-025. These issues are currently being addressed by other utilities following the guidance document (where applicable) and as outlined below:

1. For RAI#1
   a. Develop a list of all stainless steel components in the RVIs. This can be performed by looking at the license renewal application where all stainless steel components are located.
   b. Review all the drawings and purchase specifications for these components to determine material type, grade, and fabrication requirements.
   c. Based on the list, determine if the material had potential to be cold worked. If cold working is potentially found, then those materials could screen in (based on operating stress) and potentially require inspections based on the guidance in MRP-191 and MRP-227.

2. For RAI#2
   a. For each complete core loading/core design, show that :
      i. Active fuel to fuel alignment plate distance > 12.4 inches.
      ii. Average core power density < 110 Watts/cm^3
   b. For addressing core loading/design, show that:
      i. Heat generation figure of merit, $F \leq 68$ Watts/cm^3



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The guidance provided in MRP 2013-025 describes what must be done to address these RAIs but it does not adequately define how the analyses are to be performed, particularly RAI#2 that depends on plant-specific core loading/core design patterns. Structural Integrity has recently assisted plants in addressing these issues. Our experience in these areas is extensive, especially with the recent creation of the nuclear fuels group at ANATECH. In addition, we can assist in support for both the technical and licensing aspects of preparing and submitting the RAI responses.

ESTIMATION OF INITIAL RTNDT USING BTP 5-3

By: TIM GRIESBACH
tgriesbach@structint.com

Reactor pressure vessel materials for nuclear power plants are qualified per Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Code) and Appendix G of Title 10 (Energy) Code of Federal Regulation Part 50 (10 CFR 50). Prior to the Summer 1972 Addenda of the ASME Code, the materials were qualified based on Charpy V-notch tests from specimens oriented in the longitudinal direction (parallel to the principal working direction). The Summer 1972 Addenda to the Code modified the requirements for qualifying a material to include Charpy V-notch tests from specimens oriented in the transverse direction (normal to the principal working direction) and the reference nil-ductility transition temperature (RT_{NDT}). Thus, nuclear power plants whose reactor vessel order dates fell prior to July 1, 1974 have materials that were not ordered to the new specifications.

According to Appendix G of 10 CFR 50, material requirements that had been changed were also applied to older plants. Therefore, it became necessary to develop correlations between longitudinally-oriented Charpy V-notch specimen data and data from the newly required tests in order to estimate these data for the older plants. The Materials Engineering Branch of the Division of Engineering for the U.S. Nuclear Regulatory Commission prepared Branch Technical Position 5-3 which provided criteria and guidelines to evaluate the older plant test data and index temperatures with respect to the new requirements. These criteria are contained in Paragraphs B1.1(1) - (4) and B1.2 of MTEB 5-2. As part of each plant's licensing basis, all reactor vessel beltline materials initial RT_{NDT} values were determined using one of the estimation approaches, or an alternative endorsed by NRC. Recently, it has come to the attention of the NRC staff that several of the Branch Technical Position methods for plates and forgings may give unconservative results for initial RT_{NDT} values. EPRI recently sent a survey to utilities and the NRC is looking more closely at the potential impact on plant P-T limit curves and pressurized thermal shock (RT_{PTS}) evaluations. Understanding the basis for the licensing values for initial RT_{NDT} will help to resolve questions that may arise about use of the estimation methods in the Branch Technical Position.

Structural Integrity has experience in this area and we are prepared to assist utilities in resolving issues or concerns related to reactor vessel material properties.

Structural Integrity Associates, Inc.®

11515 Vanstory Drive Suite 125
Huntersville, NC 28078

APPLICANT/LICENSEE ACTION ITEM NO. 7 (A/LAI NO.7) FOR PWR VESSEL INTERNALS

Under MRP-227, PWR plants with CASS materials are required to develop plant-specific analyses to demonstrate that the CASS components (e.g., C-E lower support columns, and Westinghouse lower support column bodies) will maintain their functionality during the period of extended operation. Because of the lack of guidance in this area, utilities have relied on industry programs to provide some direction. In order to obtain clarification on what is required to resolve this issue, a series of meetings and conference calls have taken place recently with the NRC staff, the most recent meeting was held on July 15, 2014 at NRC headquarters. Following this meeting between the NRC staff; the BWRVIP/MRP CASS Working Group, and after subsequent discussions such as the PWROG Reactor Internals focus group on August 4th, A/LAI No.7 will be rewritten by the NRC staff to remove some or most of the burden associated with functionality analysis of Westinghouse and C-E lower support columns fabricated from CASS. The intent is to limit any functionality analysis to situations where both thermal embrittlement and neutron irradiation embrittlement are at issue, while eliminating functionality analysis for something much simpler when only neutron irradiation embrittlement dominates. The latter was shown by the industry to be the case for the top few inches of the lower support columns, and the NRC staff seemed to be amenable to a much simpler demonstration – referred to as a safety evaluation – rather than a full-fledged functionality analysis in such a case.

The industry cannot be completely sure how the rewrite of A/LAI No. 7 will come out, but the expectation is that industry submittals for both Westinghouse and C-E vessel internals will be based on something along the following lines. First, the top few inches of the lower support columns will be an Expansion Category item, linked to a Primary Category item that is more readily inspected, but which can be shown to “lead” the top few inches of the lower support columns with respect to IASCC both in terms of neutron irradiation embrittlement exposure and in stress level, while retaining adequate similarity with respect to delta ferrite content. The most likely Primary Category item for Westinghouse internals is the lowest girth weld in the core barrel, while the most likely option for C-E plants is to identify another component as the Expansion Category link.

The steps to be followed in the safety assessment include:

1. a review of the delta ferrite content for the lower support columns showing that average delta ferrite content is in the range to be expected in the lowest core barrel girth weld;
2. a review of the fabrication records for the lower support columns showing that adequate workmanship acceptance criteria were used in the pre-service examinations; and
3. some form of discussion of the absence of credible flaw growth during service with an emphasis on the relative preponderance of compressive stresses. While some of the details may still to be further refined, this path forward appears to be acceptable to the NRC staff.