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General Comment

Code Case (CC) N-809 should not be accepted. CC 809 provide fatigue crack growth (FCG) rate that was developed through test result. ASME CCN-809 used only mean fit of the test data and completely ignored data points above the mean value. The FCG rate should address the test population standard deviation (σ). Using the FCG rate from CC N-809 will underestimate the actual crack size.

see attached file

Attachments

3-31-21Public Comment on ASME Section XI Code Case N-809-Final

Comment:

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Safety Implications Regarding Code Case N-809

- Code Case N-809 represents the **FIRST TIME** where section XI is proposing an approach for evaluation of Fatigue Crack Growth Rate (FCGR) in PWR water environment.
- Employing an unconservative approach by underestimating FCGR by a factor of 2 (use of mean fit rather than 95% confidence upper bound fit) is inappropriate Shaving off-existing design margin goes against the basic principle of "defense in depth" and will likely lead to premature cracking and failure due to unanticipated mechanisms not accounted for in the original design. Operational experience failures (e.g., Davis Basse, Summer, SONGS, Palo Verdi, Quad Cities) is a testament.
- The only focus and rational approach for judging the acceptability of proposed code changes should be based on whether it meets sound engineering principles and accepted documented practice.
- Code Case N-809 is designated for Austenitic stainless steel components exposed to PWR water environments **safety related class 1 components**. These include class 1 RCS piping and its class 1 branch lines which are **most important to safety.**
- CC N-809 was approved by ASME in 2015. The code members who developed this code case used mean curve fit to FCG data instead of using 95% confidence upper bound curve fit. This leads to significant underestimation in FCG rate by factor of 2 as shown in the FCG rate figures of PVP 2015-45884 [ref.1] and PVP 2015-45967 [ref.2], which are in log-log scale. The figures illustrate so much of data that fall above the mean fit curve with significant under-estimation of FCG rate.
- The non-conservatism in N-809 is recognized in PVP 2019-93563 [ref.10] that notes FCG rate coefficient should be 2.78E-5 (at 95% confidence) and 1.72E-5 (at 50% confidence).
- Dealing with data that has significant scatter, one should not underestimate by using 50% confidence. 95 % confidence (2σ) is used by several ASME published papers on FCG. In manufacturing, 99.5% confidence (or 3σ limits) are used.

ASME Section III and Section XI are based on Deterministic approach. All known uncertainties should be accounted for.

- Interactions with code members from Japan [ref. 7,8,9], Korea [ref.5], French, Canada, & UK [ref.10,11] indicate that their Codes are similar to ASME Code, and in some portions are more stringent by applying 95% confidence (e.g. JSME [ref.6])
- Over the years, **some relaxations took place** and were implemented subsequent to design, such as higher damping values, from RG 1.61 values (1% and 2%, to CC N-411 variable damping to 4%, AP1000 DC approved 5% damping). Also, the construction gaps between pipe and supports were increased from 1/16" cold gap to 1/16" hot gap as approved in AP1000 DC, thus, introducing uncertainty regarding actual pipe response during operational events. During the FCG calculation, the analysis does not address piping geometry deviation of the 12.5 % manufacturer's tolerance on wall thickness (potential 12.5 % reduction in installed piping wall thickness is acceptable according to ASTM). Therefore, the load capacity is lower, thus, unconservative. Additionally, by using this non-conservative Code Case, **inspection intervals** for Class 1 safety related components may be extended.
- Section III original Design is for 40 years, however, during operation cracks were found even in the first inspection interval (10 years itself) , which indicates the presence of unknown uncertainties. Moving forward ~~for~~ with 60 years (License Renewal) and 80 years (Subsequent License Renewal) life extensions, **there are unknown or unquantifiabe uncertainties**, such as potential synergistic effects of PWSCC, FCG, IASCC & FCG that may aggravate the crack growth and potential for leakage in safety significant nuclear Class 1 RCS & RCS connected piping and components.
- Reduction in layers of conservatism (e.g., geometry, damping, factors of safety against failure) would lead to reduction in available margin, thus, the use of mean fit will lead to underestimation of the crack size.
- It is imperative for NRC as a regulator to place a condition on CC N-809 that is applicable to RCS system **most important to safety** that is currently under NRC Rulemaking review. How can NRC overlook significant under-estimation **(technical and safety perspective)** by as much as a factor of two in FCG?
- When deciding on standards, and Code Cases that eventually become part of standards, **the focus should be the most technically justifiable and defensible argument while taking into account the range of variability of testing data**.
- CC N-809 is currently included as endorsed for the **first time** in Proposed Draft RG 1.147 that is issued for public comments. The public comment period ends on April 5, 2021. This is the opportunity for the public to provide comments for NRC approval of CC N-809 with a condition to account for 95% confidence. Use the following conditions for constant C when $\Delta K \geq \Delta K_{th}$

1. For Type 304 and Type 316 stainless steels and associated weld metals use C= 8.86×10^{-7} instead of 4.43×10^{-7} in U.S. Customary Units and use C= 1.82×10^{-5} instead of 9.1E-6 in SI Units.
2. For Type 304L and Type 316L stainless steels and associated weld metals use C= 1.35×10^{-6} instead of 6.75E-7 in U.S. Customary Units and use C= 2.78×10^{-5} instead of 1.39×10^{-5} in SI Units.

- From **technical and safety perspective** and moving forward for safety related most important class 1 RCS system, and connected class 1 branches exposed to PWR water environment, a condition to use 95% confidence upper bound fit is **prudent** for CC N-809 in the current rulemaking. (This proposal is for going forward only as this is the first time this codes case is up for endorsement).

ENCLOSURE: OPERATIONAL EXPERIENCE - FAILURES

Not specifically related to CC N-809, the following few examples of failures from operating experience due to **unknown mechanisms** not considered in design that may interact with fatigue, signify the need to address **all known uncertainties**:

1) V C SUMMER: RCS Leak at Hot Leg Nozzle due to PWSCC.

2) PALO VERDE: After Steam generator replacements severe vibrations of Shut Down Cooling line connected to RCS was identified and required major modifications in the SDC system. The mechanism not anticipated was acoustic resonance of stagnant portion Shut Down Cooling Line off of RCS.

3) SONGS steam generator tube failures and RCS Leakage (that led to closure of the plant) due to a new mechanism (In-plane fluid elastic Instability caused by minor changes in gaps or clearances), an unknown mechanism not considered in the design.

4) Quad cities steam dryer failures due to FIV-acoustic resonances which were unknown and not considered in the initial design but manifested during EPU implementation.

5) DAVIS BESSE: RPV Vessel Head CRDM Penetration Boric Acid Leakage not considered in the design.

References:

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2) "Example Analysis for Environmental Fatigue Crack Growth in Austenitic Stainless Steel Piping Using Code Case N-809", ASME

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3) Development of Safety factors to be used for evaluation of Cracked Nuclear Components, B. Brickstad, and M. Bergman (see page 1) SAQ/FoU-Report 96/07, ISSN 1401-5331, SAQ KONTROLL AB, Stockholm, Sweden, 1996.
4) ASME Code Case N-809: Reference Fatigue Crack Growth Rate Curves for Austenitic Stainless Steels in Pressurized Water Reactor Environments Section XI, Division 1
5) PVP 2017-65563: Analysis of Environmental Fatigue Crack Growth Behavior of Type 347 Stainless Steels Under Simulated PWR Water Conditions; Seokmin Hong,, Ki-Deuk Min, Soon-Hyeok Jeon, Bong-Sang Lee, Korea Atomic Energy Research Institute Daejeon, Korea
6) The Japan Society of Mechanical Engineers, 2008, Codes for Nuclear Power Generation Facilities, “Rules on Fitness for Service for Nuclear Power Plant”, JSME S NA 1-2008..
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