Davis-BesseNPEm Resource

From: CuadradoDeJesus, Samuel
Sent: Friday, July 29, 2011 9:56 AM
To: Doutt, Clifford
Subject: RE: Teleconference call request

Don't worry about this e-mail. It wasn't for you but for Cliff Custer

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From: CuadradoDeJesus, Samuel
Sent: Friday, July 29, 2011 9:40 AM
To: Doutt, Clifford; dorts@firstenergycorp.com
Subject: Teleconference call request

Cliff:

Let me know at what time we can have a teleconference next Tuesday on the following topics.

1) **Follow-up Clarification (see attachment) Related to the DB Response to RAIs 3.1.2.2-1 and RAI 3.1.2.2-2**

2) **DB Response to RAI 3.3.2.3.14-1**

Question RAI 3.3.2.3.14-1

In LRA Table 3.3.2-14, the applicant identified loss of material and cracking as aging effects for steel bolting exposed to an external environment of raw water.
As identified in EPRI **NP-5769** and **NUREG-1833,** loss of pre-load for bolting can occur in any environment.
In LRA Table 3.3.2-14, the applicant did not identify loss of pre-load for steel bolting exposed to an external environment of raw water.
Justify why loss of pre-load is not identified as an aging effect for steel bolting in an environment of raw water.

RESPONSE RAI 3.3.2.3.14-1

Loss of pre-load is not identified as an aging effect requiring management for the submerged steel bolting in the Fire Protection System that is exposed to a raw water environment, as described below.
The aging management review for the Fire Protection System was conducted with the guidance provided in EPRI Technical Report 1010639 (the "Mechanical Tools"). In accordance with the Mechanical Tools, loss of pre-load is an applicable aging effect as a result of thermal effects, gasket creep, embedment (including cyclic load embedment), and/or self-loosening.

Loss of pre-load can be promoted by thermal effects (high temperature) through a process called stress relaxation. However, stress relaxation is only a concern at extremely high temperatures (above 700°F for low-alloy steels), although there may be bolting of some grades that could be susceptible to stress relaxation at temperatures slightly lower than 700oF. The submerged bolting in the Fire Protection System is

associated with the diesel fire pump column that is submerged in raw water supplied by Lake Erie. The normal temperature of this water is no greater than 85 ^{0}F, which is well below the temperature at which stress relaxation occurs. Therefore, for the submerged steel bolting in the Fire Protection System, loss of pre-load due to thermal effects is not an aging effect requiring management.

Loss of pre-load may occur as a result of gasket creep. However, gasket creep has a very small effect on pre-load (2 - 5%) and occurs, and will be evident, very soon after initial loading (10 - 20 minutes). Therefore, for the submerged steel bolting in the Fire Protection System, loss of pre-load due to gasket creep is not an aging effect requiring management.

Loss of pre-load may occur after initial loading as surfaces (e.g., threads in the bolts and joint members), which are initially in contact only on high spots, settle in together, a process called embedment. However, the effect of embedment is considered to be small and to have minimal effect on the integrity of the bolted connection. Additionally, bolted connections subjected to large cyclic loads will embed and relax more than those under static loads. However, the diesel fire pump is normally in a standby mode, and, being located in a relatively stagnant atmospheric pool, is not subject to large thermal, vibrational, or pressure-induced cyclic loading. Therefore, for the submerged steel bolting in the Fire Protection System, loss of pre-load due to embedment is not an aging effect requiring management.

Loss of pre-load due to self-loosening may occur as a result of vibration, flexing of the joint, cyclic shear loads, thermal cycles and other factors. In addition to the discussions of these factors above, self-loosening is precluded by good bolting practices and, if it occurs, is usually detected and corrected early in the service life of the component, as during maintenance activities. Therefore, for the submerged steel bolting in the Fire Protection System, loss of pre-load due to self-loosening is not an aging effect requiring management.

3) DB Response to RAI B.2.34-01

Question RAI B.2.34-1

Background:
The preventive actions program element of Generic Aging Lessons Learned (GALL), Rev. 2, aging management program (AMP) XI.M3, “Reactor Head Closure Stud Bolting,” references the guidance outlined in Regulatory Guide (RG) 1.65, Materials and Inspections for Reactor Vessel Closure Studs,” and NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants.” AMP XI.M3 states that one of the preventive measures that can reduce the potential for stress-corrosion cracking includes using bolting material for closure studs that has an actual measured yield strength less than 150 ksi. During its audit, the U.S. Nuclear Regulatory Commission (NRC or the staff) noted that the FirstEnergy Nuclear Operating Company's (FENOC or the applicant) program basis document for its Reactor Head Closure Studs Program states that the reactor head closure studs and nuts are manufactured from SA-540, Grade 23 material.

Issue:
License renewal application (LRA) Section B.2.34 and the applicant's program

basis document do not include the preventive action of using stud materials with an actual measured yield strength level less than 150 ksi. The staff needs to confirm the actual measured yield strength of the applicant's reactor head closure stud material to determine whether the applicant's program is adequate to manage stress-corrosion cracking.

Request:
The staff requests the following information:
1) Clarify whether the actual measured yield strength of the reactor head closure stud material is less than 150 ksi. If the reactor head closure stud material has a measured yield strength level greater than or equal to 150 ksi, justify the adequacy of the AMP to manage stress-corrosion cracking in the high-strength material.

2)Clarify if preventive actions will be added to the Reactor Head Closure Studs Program that would preclude the future use of replacement closure stud bolting fabricated from material with actual measured yield strength greater than or equal to 150 ksi. If not, and in view of the greater susceptibility of the studs for stress-corrosion cracking, describe any preventative actions to avoid exposure of the studs to environments conducive to stress-corrosion cracking. Otherwise, justify why preventative measures to mitigate stress-corrosion cracking of high strength studs will not be required.

RESPONSE RAI B.2.34-1

1. As confirmed by the certificate of material test report (CMTR), the actual measured yield strength ranges from 151 to 159 ksi, and tensile strength ranges from 166 to 171 ksi for the Davis-Besse reactor head closure studs. The Davis-Besse stud material is SA-540 Grade B-23. As provided in Regulatory Guide 1.65, this material when tempered to a maximum tensile strength of 170 ksi, is relatively immune to stress corrosion cracking (SCC). In addition, the Reactor Head Closure Studs Program provides for examination of the reactor vessel stud assemblies in accordance with the examination and inspection requirements specified in the ASME B&PV Code, Section XI, Subsection IWB (1995 Edition through the 1996 Addenda) and approved ASME Code Cases. Specifically, each stud is volumetrically examined once per each 10-year Inservice Inspection Interval. No unacceptable indications were noted in these examinations.
Reactor Head Closure Studs Program preventative measures to mitigate SCC are listed as follows:

a. There are no metal platings applied to the closure studs, nuts, or washers.
b. A manganese-phosphate coating was applied to the studs, nuts and washers during fabrication to act as a rust inhibitor.
c. An enhancement to the program provides for selection of an alternate stable lubricant that is compatible with the fastener material and the environment. A specific precaution against the use of compounds containing sulfur (sulfide), including molybdenum disulfide (MoS_2), as a lubricant for the reactor head closure stud assemblies will be included in the program.

2. An enhancement will be added to the Reactor Head Closure Studs Program to preclude the future use of replacement closure stud bolting fabricated from material with actual measured yield strength greater than or equal to 150 ksi except for use of the existing spare reactor head closure stud bolting.

The exception to allow future use of the existing spare reactor head closure stud bolting (2 each) is justified based on Davis-Besse plant-specific operating experience of over 30 years that has not experienced SCC of the reactor head closure stud bolting. The existing spare bolting, if used as a future replacement, would experience less than 30 years of service to the end of the period of extended operation.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Thanks

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