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L-10-044

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License No. NPF-3
Response to Specific Issues on Tube Integrity Analyses Performed for the Large Break Loss-of-Coolant Accident at the Davis-Besse Nuclear Power Station, Unit No. 1

On June 25, 2009, the U.S. Nuclear Regulatory Commission (NRC) staff held a public meeting with representatives of the Pressurized-Water Reactor Owners Group (PWROG) at NRC Headquarters. The purpose of the meeting was to discuss issues related to PWROG Topical Report BAW-2374, Revision 2, "Risk-Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot-Leg Large-Bore Piping."

The NRC stated in letter dated July 6, 2009, (Accession No. ML091671135) that the generic PWROG Program is the proper place to address the issues related to Topical Report BAW-2374. By letter dated July 31, 2009 (Accession No. ML092120411), the NRC staff requested that the FirstEnergy Nuclear Operating Company (FENOC) provide a written response to a list of actions that resulted from the June 25, 2009 meeting. FENOC addressed the issues in the August 31, 2009, letter (Accession No. ML092450685) for the Davis-Besse Nuclear Power Station (DBNPS).

By letter dated November 6, 2009 (Accession No. ML 093010483), the NRC staff requested information on specific issues on the tube integrity analyses performed for the large break loss-of-coolant accident for DBNPS. On December 30, 2009, FENOC provided a written response to the DBNPS Resident Inspectors. The attachment to this letter provides the FENOC responses that were provided to the DBNPS Resident Inspectors. If the NRC staff requires further clarification, FENOC would welcome the opportunity for an onsite review of this documentation.

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There are no regulatory commitments contained in this letter. If there are any questions or if an onsite visit to review the documentation is requested, then please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

Sincerely,

Barry Allen

Barry Allen

Attachment:
Response to Specific Issues on Tube Integrity Analyses

cc: NRC Region III Administrator
NRC Resident Inspector
NRR Project Manager
Utility Radiological Safety Board

By letter dated November 6, 2009 (ADAMS Accession No. ML 093010483), the Nuclear Regulatory Commission (NRC) requested information on specific issues on the tube integrity analyses performed for the large break loss-of-coolant accident for Davis-Besse Nuclear Power Station (DBNPS). The NRC staff request is provided below in bold type followed by the FirstEnergy Nuclear Operating Company (FENOC) response for DBNPS.

1. Please discuss how the following issues were addressed in the condition monitoring and operational assessment of tube integrity following a large break loss-of-coolant accident (LBLOCA).

(a) Any flaw with a circumferential component including wear, intergranular attack, and circumferential cracking (including those outboard of the re-rolls) could potentially open up and possibly sever under the axial loads associated with a hot-leg LBLOCA. Please discuss how Davis-Besse Nuclear Power Station, Unit No. 1 addressed the amount of leakage for each of the flaw types with a circumferential component. Please include in the response the structural limit for each of these flaw types for an LBLOCA.

Response

All tubes at risk of exhibiting flaws with a circumferential component of degradation were inspected during the 2008 refueling outage using eddy current techniques capable of detecting such degradation in order to evaluate tube integrity under the high axial tensile loads developed under main steam line break (MSLB) and LBLOCA conditions.

Condition monitoring leakage integrity at LBLOCA yield point loading is demonstrated at 95/50 for freespan volumetric intergranular attack (IGA) and wear. Axial pull test results show that wear scar maximum depths must be in excess of 92 percent through-wall (TW) for pop-through to develop at yield point tube loads. The largest observed wear depth is 28 percent TW.

Leakage integrity of circumferential degradation near tube ends is the remaining issue. Two forms of degradation are of interest, circumferential primary water stress corrosion cracking (PWSCC) and volumetric IGA. Almost all of these indications were out board of re-rolls and thus outside of the normal pressure boundary.

In all cases leak paths are limited by radial gaps between rolled sections of tubing and the tubesheet. The driving force is a differential pressure of 50 pounds per square inch (psi). Leakage is modeled as laminar flow of sub-cooled liquid and is inversely proportional to the rolled length and directly proportional to the radial gap raised to the third power. For a rolled length of 1.0 inches and a radial gap of 1 mil (0.001 inches), the LBLOCA leak rate is 0.105 gallons per minute (gpm) per leaking indication for the tube-to-shell temperature difference and fluid temperature of interest.

The tube-to-tubesheet radial gap is a strong function of the tube's radial location on the tubesheet matrix. Tubesheet hole dilation is greatest at the periphery of the bundle. Gap values were developed by adjusting small break loss-of-coolant accident (SBLOCA) finite element calculations for the increased tube-to-shell temperature difference for the postulated LBLOCA event. The radial position must exceed 55 inches before a positive tube to tubesheet radial gap develops at LBLOCA loads. The previously reported LBLOCA leakage is 0.16 gpm for steam generator (SG) A and 0.0 gpm for SG B. This is a 50/50 estimate as required by the Davis Besse technical specification in force during the 2008 refueling outage. It is based on leakage occurring when the maximum depth exceeds 60 percent TW and the radial location is greater than 55 inches. When 95/50 nondestructive examination (NDE) sizing errors are included leakage develops at LBLOCA conditions when the radial position of tube end flaws exceeds 55 inches. The conservatively determined 95/50 LBLOCA leak rates are 0.40 gpm for SG A and 0.0 gpm for SG B.

The 50/50 structural limit for a 100 percent TW circumferential flaw at yield point loading is a total circumferential extent of 0.55 inches. When expressed in terms of percent degraded area (PDA) the 50/50 structural limit is a PDA value of 24. The corresponding 95/50 limits including material property and burst equation uncertainty are 0.40 inches for a 100 percent TW flaw or a PDA value of 13. Freespan volumetric IGA and wear are the only axial strength issues. Using a length uncertainty with a standard deviation of 0.1 inches and a maximum depth uncertainty with a standard deviation of 18.2 percent TW all freespan volumetric degradation at 2008 refueling outage meets LBLOCA structural integrity at 95/50. Wear scars could reach 100 percent TW and still meet LBLOCA structural integrity at 95/50.

(b) During an LBLOCA, axially oriented flaws that are through-wall and other repair hardware (e.g., plugs, sleeves) could potentially leak. Please discuss how these sources of leakage were accounted for in the analyses.

<u>Response</u>

The 95/50 LBLOCA leakage for all flaws with axial dimensions is 0.0 gpm. Axial loads do not increase opening areas and the driving pressure is only 50 psi for LBLOCA conditions. In addition axial loads produce compressive hoop strains lessening the chance of leakage for any possible axial openings, e.g., re-rolls due to tube end axial cracks. Repair products such as plugs, re-rolls of axial tube end flaws, and sleeves are evaluated at high axial loads and high MSLB pressures and lead to minimal leakage under these extreme conditions. The very low pressure of an LBLOCA event coupled with a tightening of leakage paths leads to a LBLOCA leakage contribution of 0.0 gpm.

2. In addition, please provide the following information:

(a) A high-level summary of your analysis methodology.

Response

Probabilistic functions were applied to develop structural limits for axial loads and pop-through events. Yield loads are necessarily correlated with tube flow strengths. This correlation was established along with the uncertainty in the correlation and included in the probabilistic functions used for axial strength and pop-through calculations. The degradation of interest for axial loading is volumetric IGA, circumferential PWSCC near tube ends and wear scars at tube support plate intersections. The dimensions of interest are maximum flaw depth and circumferential extent. The test data for axial loading of circumferential electrical discharge machining (EDM) slots is applicable to these forms of degradation.

(b) A summary of the technical basis for your analysis methodology.

Response

The hot-leg LBLOCA event leads to a maximum tube to shell temperature difference of 370 degrees Fahrenheit. The thermally imposed axial displacement of the full tube length is about 1.24 inches leading to an imposed total axial tensile strain of 0.00184. The 0.2 percent offset yield unflawed tube strength bounds the maximum possible axial load experienced by any tube since it is developed at a plastic strain of 0.002. The total axial strain at the 0.2 percent offset yield strength is about 0.0036. For displacement controlled loading the appropriate safety factor is applied to the strain or displacement level. The applied safety factor from use of the 0.2 percent offset yield load is well above the required value of 1.2.

Test data is available for the axial strength of OTSG tubing containing circumferential EDM slots of various circumferential lengths and depths. Test data is also available for pop-through of circumferential degradation under axial loads. Axial strength and pop-through equations will be published in the upcoming revision of the Electric Power Research Institute (EPRI) Flaw Handbook. This information was used in the evaluation of tube integrity under axial loads at the Davis-Besse 2008 refueling outage.

(c) The basis for any assumptions (e.g., assumptions on sizing uncertainty).

Response

The 50/50 structural limit for a 100 percent TW circumferential flaw at yield point loading is a total circumferential extent of 0.55 inches. When expressed in terms of percent degraded area (PDA) the 50/50 structural limit is a PDA value of 24. The corresponding 95/50 limits including material property and burst equation uncertainty are 0.40 inches for a 100 percent TW flaw or a PDA value of 13. Freespan volumetric IGA and wear are the only axial strength issues. Using a length uncertainty with a standard deviation of 0.1 inches and a maximum depth uncertainty with a standard deviation of 18.2 percent TW all freespan volumetric degradation at 2008 refueling outage meets LBLOCA structural integrity at 95/50. Wear scars could reach 100 percent TW and still meet LBLOCA structural integrity at 95/50. NDE sizing uncertainties are based upon test information for volumetric IGA.

(d) The primary-to-secondary leakage rate assumed in your accident analysis for a loss-of-coolant accident.

Response

The primary-to-secondary leakage rate assumed in the evaluation of the hot leg loss-of-coolant accident is one gpm.

(e) A copy of any references cited in the above information (if these references are not already publicly available).

Response

The following reference was cited and is available through the Electric Power Research Institute.

Steam Generator Management Program: Steam Generator Degradation Specific Management Flaw Handbook, Revision 1, EPRI Product ID # 1019037, December 22, 2009.