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June 17, 2011
L-11-203

10 CFR 54

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 Number NPF-3
Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640), and License Renewal Application Amendment No. 9

By letter dated August 27, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102450565), FirstEnergy Nuclear Operating Company (FENOC) submitted an application pursuant to Title 10 of the *Code of Federal Regulations*, Part 54 for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). By letters dated April 5, 2011 (ADAMS Accession No. ML110820490) and May 2, 2011 (ADAMS Accession No. ML111170204), the Nuclear Regulatory Commission (NRC) requested additional information to complete its review of the License Renewal Application (LRA).

By letter dated May 5, 2011 (ADAMS Accession No. ML11131A073), FENOC responded to 19 of the 41 Batch 1 requests for additional information (RAIs) in NRC letter dated April 5, 2011. By letter dated May 24, 2011 (ADAMS Accession No. ML11151A090), FENOC responded to 21 of the 41 RAIs in NRC letter dated April 5, 2011. The remaining RAI (XI.S8-1) in NRC letter dated April 5, 2011 was discussed with Mr. Brian Harris, NRC Project Manager, and upon mutual agreement, the response to this request was deferred to be submitted with this letter, and is contained herein. The Attachment provides the FENOC reply to that RAI. The NRC request is shown in bold text followed by the FENOC response.

By letter dated May 5, 2011 (ADAMS Accession No. ML11131A073), FENOC responded to three of the RAIs (3.6-1, 3.6-2 and 3.6-3) in NRC letter dated May 2, 2011. By letter dated June 3, 2011 (L-11-166), FENOC responded to 49 of the 79 Batch 3 RAIs in NRC letter dated May 2, 2011. Twenty-six of the remaining 27 of 79 RAIs (3.3.2.3.14-3, 3.5.2.2.2-1, 4.1-2, 4.1-3, and 4.3-1 through 4.3-22 inclusive) were discussed with Mr. Samuel Cuadrado de Jesus, NRC Project Manager, and the responses to these

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requests were deferred to a mutually agreeable submittal date of June 17, 2011, and are contained herein. The Attachment provides the FENOC reply to those RAIs. The NRC request is shown in bold text followed by the FENOC response. The last RAI of the 79 Batch 3 RAIs in NRC letter dated May 2, 2011 was discussed with Mr. Samuel Cuadrado de Jesus, NRC Project Manager, on June 14, 2011 and it was mutually agreed to defer the response to that request to a later date.

This letter also clarifies and expands information provided in FENOC letter, dated May 24, 2011 (ADAMS Accession No. ML11151A090) as discussed in a June 7, 2011 telephone conference call between FENOC and the NRC.

Enclosure A provides Amendment No. 9 to the DBNPS LRA. Enclosure B provides a supporting document.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Clifford I. Custer, Fleet License Renewal Project Manager, at 724-682-7139.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 17, 2011.

Sincerely,

Barry S. Allen

Barry S. Allen

Attachment:
Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), License Renewal Application, Sections 3.3, 3.5, 4.1, 4.3 and B.2.7

Enclosure:
A. Amendment No. 9 to the DBNPS License Renewal Application
B. AREVA Report No. 51-9157140-001, "DB-1 Design CUFs and NUREG/CR-6260 Screening for License Renewal," dated 6/10/2011

cc: NRC DLR Project Manager
NRC Region III Administrator

cc: w/o Attachment or Enclosure
NRC DLR Director
NRR DORL Project Manager
NRC Resident Inspector
Utility Radiological Safety Board

Question RAI XI.S8-1

The GALL Report states that proper maintenance of protective coatings inside containment (defined as Service Level I in Nuclear Regulatory Commission Regulatory Guide [RG] 1.54, Revision 1) is essential to ensure operability of post-accident safety systems that rely on water recycled through the containment sump/drain system. Degradation of coatings can lead to clogging of strainers, which reduces flow through the sump/drain system.

The DBNPS LRA does not credit the protective coating monitoring and maintenance program for aging management. Although the licensee does not credit the program for aging management, there needs to be adequate assurance that there is proper management and maintenance of the protective coatings in containment, such that they will not degrade and become a debris source that may challenge the Emergency Core Cooling System and Containment Spray System performance.

The staff requests the following information:

1. Discuss why XI.S8, "Protective Coating Monitoring and Maintenance Program," is not credited for aging management.

2. Discuss in detail whether DBNPS has a coatings monitoring and maintenance program. Describe the program if one is used.

3. Describe how DBNPS will ensure that there will be proper maintenance of the protective coatings inside containment such that they will not become a debris source that could impact the operability of post-accident safety systems that rely on water recycled through the containment sump or drain system in the PEO.

If a program is used, describe the frequency and scope of the inspections, acceptance criteria, standards used, and the qualification of personnel who perform containment coatings inspections.

RESPONSE RAI XI.S8-1

1. GALL Program XI.S8, "Protective Coating Monitoring and Maintenance Program," was not credited for aging management because coatings were not credited for protecting structures, systems or components from aging effects.

2. FirstEnergy Nuclear Operating Company (FENOC) monitors and maintains coatings within the DBNPS containment vessel with an existing Nuclear Safety-Related Protective Coatings Program. FENOC also monitors and maintains the containment vessel coating inside containment within the Inservice Inspection (ISI) Program - IWE. The 1995 edition of American Society of Mechanical Engineers (ASME) Section XI, IWE-3510.2, states that the inspected area, when painted or coated, shall be examined for evidence of flaking, blistering, peeling, discoloration and other signs of distress, and areas that are suspect shall be accepted by engineering evaluation or corrected by repair or replacement.

3. FENOC will implement the Nuclear Safety-Related Protective Coatings Program as a license renewal plant-specific aging management program during the period of extended operation.

See Enclosure A to this letter for the changes to the LRA.

<u>Section 3.3.2</u>

Question RAI 3.3.2.3.14-3

SRP-LR Revision 2 Table 3.3-1, item 112, recommends that steel piping, piping components, and piping elements exposed to concrete do not need to be age managed, provided that the attributes of the concrete are consistent with ACI 318 or ACI 349 and that plant operating experience indicates no degradation of the concrete. LRA Table 3.3.2-14, item 54 (fire protection system), Table 3.3.2-26, item 56 (service water system), Table 3.3.2-31, item 48 (station plumbing, drains, and sumps system), and Table 3.5.2-12, item 7 (yard structures), state that steel components exposed to concrete do not need to be age managed. LRA Section B.2.39, "Structures Monitoring Program," includes several incidents of operating experience where water leakage through the concrete has occurred.

It is not clear to the staff whether concrete degradation has occurred in the vicinity of in-scope components described in the request such that the steel components would be exposed to water and thus be subject to corrosion.

The staff requests the following information:

1. **State whether concrete degradation has occurred such that water may have intruded into the concrete that surrounds the steel components in the fire protection system, service water system, station plumbing, drains, and sumps system, and yard structures. If water intrusion has occurred, state how the aging of the steel components will be managed.**

2. **State how the Structures Monitoring Program, or other plant-specific program, will address water intrusion into concrete to ensure that resulting aging of embedded steel components will be effectively managed during the period of extended operation.**

RESPONSE RAI 3.3.2.3.14-3

1. FENOC conducted a review of the Davis-Besse plant specific operating experience for License Renewal. Concrete degradation has not occurred such that water may have intruded into the concrete that surrounds the subject steel components in the fire protection system, service water system, station plumbing, drains, and sumps system and yard structures. Plant-specific operating experience has identified limited areas where water has leaked through concrete. Review of the plant-specific operating experience does not suggest that any of the identified leakage has had any effect on embedded piping, or on the embedded emergency diesel generator fuel oil tank hold down restraints.
2. The Structures Monitoring Program, with the enhancements described in the responses to RAI B.2.39-3 and RAI B.2.39-6, will effectively manage water intrusion into concrete. The RAI responses were provided in FENOC Letter, dated May 24, 2011 (ADAMS Accession No. ML11151A090). The response to RAI B.2.39-3 includes concrete core bore evaluation and the response to RAI B.2.39-6 includes enhancement of the acceptance criteria for visual inspection of concrete.

Section 3.5.2

Question RAI 3.5.2.2.2-1

SRP Table 3.5.1, Item 3.5.1-33 recommends further evaluation for any concrete elements that exceed the specified temperature limits of 150°F general and 200°F local.

LRA Section 3.5.2.2.2.3 notes that several localized areas in the upper regions of the containment internal structures have maximum temperatures exceeding 150°F.

The staff is unclear how concrete, having temperatures above the limits in the SRP Table 3.5.1, Item 3.5.1-33, will be managed during the period of extended operation. The staff requests the following information:

1. **Provide a listing of locations where concrete temperature exceeds SRP Table 3.5.1, Item 3.5.1-33 limits for general or local areas.**

2. **For each of these locations, provide the extent of the region of concrete impacted and the maximum temperature experienced by the concrete.**

3. **Provide a description of how these locations will be managed during the period of extended operation or an assessment of the impact of the elevated temperature on concrete to demonstrate that the concrete properties have not been adversely impacted.**

The staff needs the above information to confirm that the effects of aging such as noted above will be adequately managed so that the intended function of impacted structural members will be maintained consistent with the current licensing basis for the period of extended operation as required by 10 CFR 54.21(a)(3).

RESPONSE RAI 3.5.2.2.2-1

1. The Davis-Besse Technical Specifications require that containment average air temperature shall be less than or equal to 120 degrees Fahrenheit (°F). Therefore, there are no locations that have been identified in containment where general concrete temperatures exceed the SRP Table 3.5.1, Item 3.5.1-33 specified temperature limit of 150°F general.

   There is one location in containment which is considered to exceed the SRP Table 3.5.1, Item 3.5.1-33 specified temperature limit of 200°F local. That location is at the top of the primary shield wall that encircles the reactor vessel, just below the Permanent Canal Seal Plate that was installed during the Cycle 13 refueling outage. Based on calculated temperatures, a volume of concrete in the Primary Shield Wall was determined to exceed the SRP Table 3.5.1, Item 3.5.1-33 specified temperature limit of 200°F local.

2. The extent of the region of concrete impacted is based on calculated temperatures. The concrete that forms the upper four feet of the Primary Shield Wall is affected. The elevated temperature is localized in the upper corner of the Primary Shield Wall and will drop off rapidly since the air temperature above the Permanent Canal Seal Plate is at the containment general air temperature.

   A bounding calculation was performed using an assumption that the maximum local concrete temperature would be 207°F. The maximum local concrete temperature was later calculated as 205°F based on the use of conservative assumptions and a measured local "hot spot" concrete temperature of 155°F.

3. An assessment of the impact of the elevated temperature on the affected concrete was performed by FENOC. The concrete that is considered to exceed the SRP Table 3.5.1, Item 3.5.1-33 specified temperature limit of 200°F local, was assessed with a calculation that concluded that the concrete is fully capable of performing its functions with no detrimental effects. The calculation shows that the affected

localized area of concrete has low mechanical stresses. From a material property standpoint, the calculation shows that reduction in the compressive strength of the affected concrete at 207°F will be more than offset by the concrete strength gain due to concrete aging.

Section 4.1

Question RAI 4.1-2

LRA Section 4.3.2.2.4 discusses the fatigue TLAA for the reactor coolant pump (RCP) casings and states that they were analyzed for fatigue by the OEM to meet the requirements of the ASME Code Section III, 1968 Edition through Winter-1968 Addenda. LRA Table 3.1.1 item 3.1.1-55 states that these pump casings will be managed by the applicant's Inservice Inspection Program.

The applicant's licensing basis includes a flaw tolerance analysis for the RCP casings that was used to support ASME Code Case N-481's alternate augmented visual inspection bases for the RCP casings. The staff noted that this flaw tolerance analysis is documented in Structural Integrity Associates (SIA) Topical Report No. SIR-99-040, Revision 1, "ASME Code Case N-481 of Davis Besse Reactor Coolant Pumps." (ADAMS Accession No. ML011200090, dated April 23, 2001).

The staff noted that the evaluation in Report No. SIR-99-040 includes a cycle-dependent fatigue flaw growth analysis for the pump casings welds that is based on a 40-year design life; however, the applicant did not identify this analysis as a TLAA.

Justify why the fatigue flaw growth analysis for the RCP pump casing welds in SIA Topical Report No. SIR-99-040, Revision 1, does not need to be identified as a TLAA in accordance with 10 CFR 54.21(c)(1).

RESPONSE RAI 4.1-2

Code Case N-481 requires an evaluation to demonstrate the safety and serviceability of the pump casings. The evaluation for the Davis-Besse RCPs required by Code Case N-481 is Structural Integrity Associates (SIA) report SIR-99-040. This analysis assumed a quarter thickness flaw, with length six times its depth, and showed that the flaw will remain stable considering the stresses and material properties of the pump casing. Though not required by the Code Case, the analysis also showed that a small initial assumed flaw will not grow to quarter thickness during plant life.

There are two potential time-dependencies in this analysis.

1. The fracture toughness of the cast austenitic stainless steel is not time dependent as the analysis used a lower bound fracture toughness of 139 ksi√in that bounds the saturated fracture toughness of the Davis-Besse material.
2. Although the optional flaw growth analysis is based on the design transients, it is not based on the cycles expected in 40 years. The analysis examined the design cycles and decided there were 240 cycles that were significant to flaw growth in the RCPs. Then 2000 cycles were conservatively analyzed, and flaw growth remained well below the postulated flaw. As this optional analysis was analyzed well above the design basis number of transients, it is not based on the design life of the plant (neither 40 years nor 60 years) and therefore is not a TLAA.

Question RAI 4.1-3

The LRA Table 4.1-1 identifies "RCS Loop 1 Cold Leg drain line weld overlay repair," as a plant- specific TLAA with its disposition discussed in the LRA Section 4.7.5.1. The Section 4.7.5.1 states that, even though there is no time dependency in the weld overlay design that is a full structural overlay assuming the as-found flaw to be 100% through-wall 360-degree, fatigue analysis for the repaired configuration was performed by conservatively estimating cycles for 60 years; as such the analysis is based on a specific number of cycles and so it is a TLAA.

The staff could not identify any other instances of similarly repaired piping and nozzle locations being considered in the LRA as plant-specific TLAAs. From the LRA, it is not clear to the staff if this item in Table 4.1-1 is the only weld overlay repair where fatigue analysis was performed.

Clarify if and why the RCS loop 1 cold-leg drain line weld overlay repair is the only one to include the cycle-based or time dependent flaw growth assumptions. If there are other instances of repairs with similar analyses justify their exclusion from TLAA identification.

RESPONSE RAI 4.1-3

The weld overlay on the RCS Loop 1 cold leg drain was given special mention in the LRA because it was installed as a result of an unacceptable Inservice Inspection examination. As stated in LRA Section 4.7.5.1, there are no time dependencies related to the "flaw growth" in the weld overlay on the RCS Loop 1 cold leg drain. The weld repair used a bounding assumption of the flaw going through wall, 360 degrees, rather than perform a flaw growth analysis and credit any of the remaining pipe wall for structural integrity.

Other weld overlays have been preemptively installed to mitigate any potential primary water stress corrosion cracking (no known flaw) on various dissimilar metal welds located in the Davis-Besse reactor coolant pressure boundary. These weld overlays are "full structural weld overlays" in which the overlay itself provides adequate structural support for the pipe, assuming that there was 100% failure of the original. This bounding assumption eliminates the need for a flaw growth analysis and thus eliminates time dependency from the overlay design.

However, the weld overlay evaluations have a time dependency relative to the fatigue analysis of the new configuration. As with any modification, the fatigue analyses of record for the applicable piping system is revised to address the new configuration. For license renewal, review of the Davis-Besse fatigue analyses is addressed in LRA Section 4.3.2, "Class 1 Fatigue."

Section 4.3

Question RAI 4.3-1

LRA Section 4.3.2.2 states that:

> **Cumulative usage factors for the Class 1 components are calculated based on normal and upset design transient definitions contained in the component design specifications. The design transients used to generate cumulative usage factors for Class 1 components are discussed in Section 4.3.1 above. In accordance with Davis-Besse Technical Specification 5.5.5, the Allowable Operating Transient Cycles Program (Fatigue Monitoring Program) provides controls to track the updated safety analysis report (USAR) Section 5 cyclic and transient occurrences to ensure that components are maintained within the design limits.**

The staff noted that USAR Table 5.1-8 includes the classification for transients by the plant condition (e.g., normal, upset, emergency, faulted, or test). LRA Table 4.3-1, which is in LRA Section 4.3.1, includes additional transients that are not listed in USAR Table 5.1-8 and the transient classification is also not provided.

The aforementioned statement in LRA Section 4.3.2.2 implies that LRA Table 4.3-1 lists only normal and upset design transients. However, the staff noted that Transient #9, "Rapid Depressurization" in LRA Table 4.3-1 is classified as an "Emergency" transient in USAR Table 5.1-8 and it is not clear to the staff if LRA Table 4.3-1 includes all emergency transients that were used in the fatigue analyses.

The staff requests the following information:

1. **Clarify whether all fatigue significant transients, that have been included in the fatigue TLAAs, have been included in the LRA Table 4.3-1. Identify the plant condition (e.g., normal, upset, emergency, faulted, or test) for each transient listed in LRA Table 4.3-1.**

2. **Confirm whether the CUF analyses of record included emergency and test conditions in addition to the normal and upset condition. If necessary, clarify and revise the aforementioned statement in LRA Section 4.3.2.2.**

RESPONSE RAI 4.3-1

1. Table 4.3-1 of the DB-1 LRA includes all fatigue significant transients that are included in the fatigue TLAAs. Table 4.3-1 is consistent with the FENOC Allowable Operating Transient Cycle procedure, which is based on the Davis-Besse RCS Functional Specification. The RCS Functional Specification is the primary source of design transients for the Babcock and Wilcox (B&W)-supplied RCS components. Table 4.3-1 of the LRA has been previously amended to include the applicable ASME category for the event. See FENOC response to RAI B.2.16-1 (FENOC Letter L-11-166) for the amendment to LRA Table 4.3-1.

2. The design CUFs of record for the DB-1 Class 1 components are reported in AREVA document 51-9157140-000, which is included in the enclosure to this letter in response to RAI 4.3-12. From a review of the design report summaries for RCS components, the fatigue analyses include test transients, normal and upset transients identified in the amended LRA Table 4.3-1, and operational basis earthquakes (30 earthquakes—650 cycles total). The only CUF reported in 51-9157140-000 that included an emergency event was for the RV studs where the design CUF of 0.70 was conservatively increased by 0.026 to include 20 natural circulation cooldown events. The incremental fatigue due to the emergency event is not required by ASME IIII, NB-3224.5. The primary contribution to fatigue of DB-1 NSSS components is attributed to normal and upset service loadings; however, the statement in Section 4.3.2.2 requires a change to more accurately reflect the ASME III requirements.

   See Enclosure A to this letter for the change to the LRA.

Question RAI 4.3-2

LRA Section 4.3.1.2, "Projected Cycles," states that the analysis of the high-pressure injection (HPI) nozzles determined that the elbowlets in HPI nozzles 1-1 and 1-2 were limited to 13 cycles for Transients 9A and 9B, respectively. The applicant stated that the current cycles are at 9 and 8 for HPI nozzles 1-1 and 1-2, respectively.

In LRA Table 4.3-1, Transients 9A to 9D, labeled "Rapid RCS Depressurization" are listed in the USAR Table 5.1-8 as Transient #8. During its audit, the staff noted discrepancies in the cycle count for Transient #8 of USAR Table 5.1-8, as described in the applicant's existing Fatigue Monitoring Program (identified as "AOTC" by the applicant) logs. In the AOTC log, dated February 1990, it stated that a total of 11 cycles were recorded for this transient, out of the design limit of 13. Furthermore, an AOTC log, dated May 2003, stated that the recorded cycle count for this transient was 9.

In addition, the staff noted, during its audit, that the cycle count from the AOTC log dated February 1990 for this transient exceeded the applicant's 75% action limit, which is based on the design cycle limit of 13 cycles. It is not clear to the staff if the applicant's procedures required corrective actions and the associated results for any corrective actions that may have been taken.

The staff noted, during its audit, that the elbowlets in HPI nozzles 1-1 and 1-2 have a design CUF of 0.981 (with the limit of 13 cycles of Transients 9A and 9B). It is not clear to the staff, if there were other transients that are a significant contributor to fatigue and the number of analyzed cycles in the design CUF calculation for these components.

The staff requests the following information:

1. **Describe and justify the discrepancy between cycle counts for Transients 9A to 9D, which are listed in LRA Table 4.3-1, and the cycles counts in the AOTC logs dated February 1990 and March 2003.**

2. **Based on the AOTC log dated February 1990, clarify whether corrective actions were taken, based on the cycle count exceeding the applicant's 75% action limit. If corrective actions were taken, describe the actions taken and the associated results of these actions. If corrective actions were not taken, explain why no action was required.**

3. **Identify the design transients and associated cycle limits that were used in the fatigue analysis of the HPI nozzles and elbowlets.**

RESPONSE RAI 4.3-2

Please note that by letter dated June 3, 2011 (L-11-166), in the response to RAI B.2.16-1, FENOC amended LRA Table 4.3-1.

LRA Table 4.3-1 was revised to include transient numbers 1C, 8C, 9A, 9B and 25 (AOTC Program Transient 33). Previous listed transients 9A through 9D are renamed as the HPI System Pressure Isolation Integrity Tests, and are now grouped under transient number 22 A2 (HPI Nozzles 1-1, 1-2, 2-1 and 2-2). The Rapid RCS Depressurization (Upset) event is now monitored as transient 9A, and the Rapid RCS Depressurization, trip RCS Pumps (Emergency) event is now monitored as transient 9B. LRA Table 4.3-1 was further revised to provide clarification and align transient descriptions with the RCS Functional Specification and the AOTC Program.

For the below discussion, the new transient number will be shown in parentheses following the old number.

1. During the review of the AOTC program as part of the Cycle 13 refueling outage (ended March 27, 2004) restart effort, the AOTC Status Log was updated based upon review of the AOTC Event Log. The Status Log dated 5/22/2003 replaced the Status Log dated January 25, 2003. The updated Status Log included the latest Transient 9 (now Transient 22 A2) cycle counts and limits (13 cycle limit for Train 1 and 40 cycle limit for Train 2) based on that review. The cycles for the individual nozzles were separated commencing with this updated log. As a result of the review and update, the resulting event counts as of 5/22/2003 were as follows: 9 cycles for HPI Nozzle 1-1, 8 cycles for HPI Nozzle 1-2 , 20 cycles for HPI Nozzle 2-1 and 15 cycles for HPI Nozzle 2-2.

2. The February 19, 1990 Rev. 01, AOTC Status Log showed a total of 11 events for Transient 9 (now Transient 22 A2). A review of the AOTC Event Logs up to that date shows 11 cycles logged for nozzle 2-1, the normal Makeup flow path at that time. Additionally nozzle 2-2 shows 2 cycles logged. Nozzle 1-1 shows three cycles and nozzle 1-2 shows 2 cycles. At that time, the cycles for the different nozzles were not separated in the Status Log. The Train 2 nozzles (2-1 & 2-2) cycle counts were well below the 40 cycle limit. Additionally, the Train 1 nozzles (1-1 & 1-2) were also below the 13 cycle limit for that train. The design limit of 13 cycles listed in the log only applies to the Train 1 nozzles. Therefore, none of the transient counts exceeded the 75% action limit.

3. Transient 9 (now Transient 9A), "rapid RCS depressurization (upset)," in Table 4.3-1 of the LRA, is defined in the Davis-Besse RCS Functional Specification as an upset event that includes short term, rapid cooling of the RCS by the Steam Generators (SGs) to reduce the RCS pressure to a value less than the design pressure (1065 psia) of the SGs within 15 minutes. The initial conditions at the start of the transient are assumed to be hot standby with core decay heat removal by the SGs

dumping steam to the condenser. The turbine bypass control pressure is assumed to be 1050 psia. This gives an average RCS temperature of about 550°F.

The objective of the rapid depressurization is to isolate a steam generator tube leak. Transient 9 (now Transient 9A) results in the actuation of high pressure injection and is the only upset event in the RCS Functional Specification that results in HPI actuation. The design cycle limit for this transient is 40. The 40-cycle limit was reduced to 13 (for HPI lines 1-1 and 1-2 with elbolet as the limiting location) in 1983 by a Bechtel evaluation of the HPI lines in response to IEB 79-14. HPI lines 2-1 and 2-2 were qualified for 40 cycles.

The only other RCS Functional Specification normal condition that results in HPI actuation is Transient 22 (now Transient 22 A1), HPI System Test, and includes HPI flow through all 4 HPI nozzles for 10 seconds with RCS pressure of 2200 psig and RCS temperature of 550°F. This transient has 40 design cycles. The HPI pump shutoff head is approximately 1600 psig and therefore, the pumps are recirculated back to the Borated Water Storage Tank during the HPI System Test. Since no inventory is added to the Reactor Coolant System, Transient 22 (now Transient 22 A1) is not applicable to Davis-Besse, but is conservatively included in the fatigue evaluations of the HPI nozzles and HPI elbolet.

In 1987 Davis-Besse initiated an HPI system test entitled, "HPI System Pressure Isolation Integrity Test-Back-to-Back Check Valves." This test isolated makeup flow to one of the 4 HPI nozzles (i.e., the HPI nozzle used for reactor makeup) with RCS pressure at 2155 psig and RCS temperature of 532°F. Makeup flow was isolated for approximately 15 minutes and then resumed. The purpose of the test is to ensure that the HPI/MU check valves work properly and isolate the HPI/MU system from the RCS. This test did not fit the RCS Functional Specification definitions for Transient 9 (now Transient 9A) or Transient 22 (now Transient 22 A1) and was considered a new transient with the number of test cycles defined as 40. These new transients were included as transients 9A through 9D (now Transient 22 A2 for each of the HPI Nozzles) in the AOTC Program.

Question RAI 4.3-3

LRA Section 4.3.2.2.2.1 states that the applicant has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts. In addition, LRA Section 4.3.2.2.2.1 states that the reactor vessel internals are designed to meet the stress requirements of ASME Section III, they are not code components. Consequently, a fatigue analysis of the reactor vessel internals was not performed as part of the original design.

LRA Table 3.1.2-2, Row Nos. 42 and 110, for upper thermal shield bolts and flow distribution bolts, respectively, credit a TLAA to manage cumulative fatigue damage.

It is not clear to the staff what TLAA is being referenced by LRA Table 3.1.2-2 Row Nos. 42 and 110, when LRA Section 4.3.2.2.2.1 states that fatigue analyses were not performed for the reactor vessel internals.

Clarify the fatigue TLAA that is being credited to manage cumulative fatigue damage of the components identified by the AMR line items in LRA Table 3.1.2-2 Row Nos. 42 and 110.

RESPONSE RAI 4.3-3

As provided in LRA Section 4.3.2.2.2.1, Davis-Besse has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts. Therefore, a correction is required to row numbers 42 and 110 of LRA Table 3.1.2-2.

See Enclosure A to this letter for the changes to the LRA.

Question RAI 4.3-4

LRA Section 4.3.2.2.1 "Reactor Vessel" states that the design CUFs for the limiting reactor vessel assembly locations were calculated to be less than 1.0 based on the design transients. The applicant also dispositioned these fatigue TLAAs in accordance with 10 CFR 54.21(c)(1)(iii). The staff noted that the bottom head of the reactor vessel assembly is penetrated by the instrumentation nozzles which were analyzed for fatigue due to flow-induced vibrations and discussed in LRA Section 4.3.2.2.2.3. LRA Section 4.3.4.2 discusses the nickel-based incore instrument nozzle and addresses the effect of reactor coolant environment on component fatigue life.

During its audit, the staff noted that the applicant's basis documents, for metal fatigue TLAAs, lists CUF values for the instrument nozzle weld locations that vary from 0 to 0.323. LRA Section 4.3.2.2.2.3 states that the incore instrumentation nozzles were analyzed for fatigue due to flow-induced vibrations (FIV) with the resulting CUF of 0.59 for a 40-year life and was projected to have a CUF of 0.885 for a 60-year life. LRA Section 4.3.4.2 states that the maximum design CUF for nickel-based alloy incore instrument nozzle is 0.77.

The LRA does not indicate the locations that are considered to be limiting, the specific CUF values that are associated with these locations and the design transients used to determine the CUF values. In addition, it is not clear to the staff

whether the generic reference of "Instrument Nozzles" in the applicant's basis documents and the LRA refer to the same locations.

The staff requests the following information:

1. **Clarify the location(s) that are being referenced by the "Instrument Nozzle" CUFs in LRA Sections 4.3.2.2.1, 4.3.2.2.2.3, 4.3.4.2, and the applicant's basis documents for the metal fatigue TLAA.**
2. **Clarify which of these locations for the instrument nozzle of the reactor vessel assembly support the aforementioned statement in LRA Section 4.3.2.2.1 and is considered the limiting location. In addition, provide the corresponding limiting CUF values.**

RESPONSE RAI 4.3-4

1. The following instrument nozzle CUFs appear in the Davis-Besse LRA or in Davis-Besse basis documents. The basis documents are the reactor vessel stress report summary and the environmentally assisted fatigue analysis that was done to support License Renewal.

Document	CUF	Title	Comment
LRA Section 4.3.2.2.1	None	Reactor vessel	4.3.2.2.1 mentions the incore nozzles, but only says all vessel CUFs are less than 1.0. This is consistent with the other sections of the LRA and all bases documents.
LRA Section 4.3.2.2.2.3	0.59	FIV	Review of the source documents determined that this value was reported in error, see discussion following this table.
LRA Section 4.3.4.2	0.77, 0.206, 0.857	EAF	This section confirms that 0.77 is currently the highest CUF of record for the incore nozzles, and it pertains to the nozzle to vessel weld. This nozzle to vessel weld has been re-analyzed as part of the EAF evaluation, and the CUF was reduced from 0.77 to 0.206 by applying the alternating stresses from the original design calculation to the new in-air design curve for stainless steel in NUREG/CR-6909. The CUF of 0.206 was multiplied by an F_{en} of 4.16 and resulted in an environmentally adjusted CUF of 0.857. When the License Renewal Application is approved, this 0.857 will be the new CUF of record for the incore instrument nozzle.

Document	CUF	Title	Comment
Reactor vessel stress report summary	0.000-0.323	Nozzles	CUFs were calculated for two locations on two styles of nozzle bodies. Those CUFs were 0.0, 0.0, 0.269, and 0.323. These CUFs are the nozzle bodies, not the nozzle to vessel weld, and are never discussed in the LRA because they are not the limiting location.
Reactor vessel stress report summary	0.770	Nozzle to vessel weld	The highest CUF of record in the basis documents is 0.77 for the nozzle to vessel weld. This agrees with the CUF in LRA Section 4.3.4.2.
EAF analyses for License Renewal	0.857	Nozzle to vessel weld	Agrees with LRA Section 4.3.4.2.

Review of the source documents determined the CUF of 0.59 reported in LRA Section 4.3.2.2.2.3 was a typical CUF for B&W-designed plants. In addition, flow induced vibration of the incore instrument nozzles was evaluated using the endurance limit approach as discussed in LRA Section 4.3.2.2.2.2. Therefore, corrections to the LRA are required.

See Enclosure A to this letter for the changes to the LRA.

2. The CUFs for the incore instrument nozzles identified in the table above are less than the Code design limit of 1.0 and therefore, support the statement in LRA Section 4.3.2.2.1 that reactor vessel CUFs are less 1.0.

   Currently, the highest design CUF of record for the incore instrument nozzles is the 0.77 reported in LRA Section 4.3.4.2. This CUF is for the weld between the incore instrument nozzle and the reactor vessel lower head. When the LRA is approved, the highest CUF of record for the incore instrument nozzles will be the environmentally adjusted CUF of 0.857 reported in LRA Section 4.3.4.2.

Question RAI 4.3-5

LRA Section 4.3.2.2.2.2 discusses the fatigue of reactor vessel internals subject to the flow-induced vibrations. In addition, the fatigue TLAA discussion is based on the endurance limit approach, which establishes the allowable stress limit for infinite fatigue life, The staff noted that ASME Code Section III (Mandatory Appendix I) provides the design fatigue curves.

The applicant stated that the ASME Code fatigue curve was extended to 1E+12 cycles because the 60-year projection used in the vessel internals fatigue TLAA exceeds the Code design curves. The applicant stated that an extrapolation of the curve(s) was necessary to obtain the allowable stress limit. It is not clear to the staff which Appendix I design curve was used by the applicant and the method of extrapolation used to establish the endurance limit for the 40-year analysis and the 60-year projection.

The staff requests the following information:

1. **Clarify and justify the ASME Code Section III (Mandatory Appendix I) design curves used in the extrapolation described in LRA Section 4.3.2.2.2.2 for all the vessel internal materials subject to the flow-induced vibration.**

2. **Describe and justify the method of extrapolation for the design fatigue curves used in establishing the endurance limits. Provide the allowable stresses and the calculated peak stress intensities for fatigue of the components/locations discussed in the LRA Section 4.3.2.2.2.2.**

RESPONSE RAI 4.3-5

1. The curve that was extrapolated is the ASME design fatigue curve for austenitic steel, Sa, 104 psi versus 10n cycles. The specific curve extrapolated in the flow induced vibration (FIV) analysis was Figure I-9-2 of the 1971 edition of the ASME code.

2. The FIV analysis extended the ASME Code curve from 1E+6 cycles to 1E+12 cycles based on the curve fit for the data found in the ASME transactions. This fit resulted in a decrease of 4% per decade on the fatigue curve for austenitic stainless steel, and an endurance limit of 20,400 psi for 1E+12 cycles, or 40-years of operation. Although the endurance limit was 20,400 psi, the FIV analysis conservatively assumed only 18,000 psi for the endurance limit. The maximum calculated peak stress intensity provided in the FIV analysis is 8260 psi for the upper thermal shield support blocks, still well below the 18,000 psi endurance limit.

   For License Renewal, the ASME Code fatigue curve was extended from 1E+12 cycles (the upper bound on the number of cycles for a 40-year design life) to 1.5E+12 cycles (the upper bound on the number of cycles for a 60 year design life). The extrapolated fatigue curve at 1.5E+12 cycles is approximately 20,200 psi, still above the 18,000 psi that was used as the endurance limit in the FIV analysis.

Question RAI 4.3-6

LRA Section 4.3.2.2.4 states that the reactor coolant pumps (RCP) were analyzed for fatigue by the original equipment manufacturer. The applicant stated that the design CUF for the limiting coolant pump locations were calculated based on the design transients and are all less than 1.0. The LRA also states that the fatigue TLAA for the reactor coolant pumps will be managed for the period of extended operation by the Fatigue Monitoring Program, in accordance with 10 CFR 54.21(c)(1)(iii).

During its audit, the staff reviewed the applicant's basis documents for the metal fatigue TLAAs and noted that the cooling hole ligament location of the pump cover has a CUF value of 0.56. The staff also noted that the applicant's basis documents stated the CUF was calculated with an exception to the ASME Code rules. It is not clear to the staff what the exception was, and whether the exception affects the applicant's disposition for this TLAA. The staff noted that LRA Section 4.3.2.2.4 did not discuss the particular location.

Clarify the exception used for the fatigue analysis of cooling hole ligament of the RCP cover and justify that the exception does not affect the TLAA disposition of the reactor coolant pump casing fatigue evaluation.

RESPONSE RAI 4.3-6

The Davis-Besse Unit 1 reactor coolant pumps were designed and analyzed in accordance with the 1968 Edition of the ASME Code Section III with Addenda through the Winter 1968, but the original components provided by the manufacturer (Byron-Jackson) were not code stamped. The cumulative usage factor calculated by the pump vendor (Byron-Jackson) for the cooling hole ligament in the original design reports was less than 1.0.

During the course of investigation into thermal cracking of RC pump covers in the mid 1980s, reanalysis of the ligament region at the cooling holes revealed stresses that are higher than those calculated in the original design report. The manufacturer (B-J) evaluated the revised stresses using the models used in the original design report. The updated stresses could not be shown to be acceptable based on the original acceptance criteria in ASME Section III, 1968 Edition, and Nuclear Code Case 1441-1. The revised CUF for the cooling hole ligament exceeded 1.0.

To demonstrate that the fatigue life of the cover cooling hole ligament is acceptable for the current term of operation, B&W utilized the vendor stress analysis and developed alternate simplified elastic-plastic methodology for locations where the primary plus secondary stress exceeded 3Sm based on consideration of the 1968 Edition of ASME III, USAS B31.7, Nuclear Code Case 1441, and Nuclear Code Case 1441-1. Specifically, a stress and fatigue analysis of the pump cover cooling hole ligament was performed in accordance with the procedure described in Paragraph N-415 of the 1968 Edition of ASME III, except that the procedure was modified as follows.

The limit on the range of primary-plus-secondary stress intensity may be waived if:

1. There are not more than 1000 cycles of primary-plus –secondary stress intensity range greater than $3S_m$.

2. The value of S_a used for entering the design fatigue curve is increased by the factor K_e as defined below:

   K_e = 1.0 for $S_n < 2S_b$

   K_e = 1.0+ (1-n)/n(m-1) * ($S_n/2S_b$ -1) for $2S_b < S_n < 2mS_b$

   K_e = 1/n for $S_n > 2mS_b$

   S_n = calculated range of primary-plus-secondary stress intensity

   S_b = cyclic strain-hardened yield strength
   = 93.9 KSI at 550°F

   m = 1.7

   n = 0.5

3. The stresses produced by the equivalent linear portion of the radial thermal gradients are classified as secondary instead of peak. The equivalent linear portion of a radial gradient is defined a linear radial gradient that develops the same thermal moment as the actual radial gradient.

4. The rest of the fatigue evaluation stays the same as required in N-415 of Section III except that the procedure of N-417.5 (b) need not be used.

The above alternate procedure was applied to the heatup and cooldown transients (205 cycles) and to the combined hydrostatic tests and heatup transient (35 cycles). The cumulative usage for the remaining transients was calculated in accordance with ASME N-415. The total usage of 0.56 included 0.41 for heatup/cooldown, 0.09 for hydrotest/heatup, and approximately 0.06 for the remaining events. As shown in LRA Table 4.3-1, the number of design transients used in the alternate procedure remains valid for 60 years of operation and therefore, the CUF of 0.56 for the cooling hole ligament is applicable for 60 years of operation.

Question RAI 4.3-7

LRA Section 4.3.2.6.1 states that the steam generators were analyzed for fatigue by the original equipment manufacturer (OEM) and that the CUFs for limiting locations were calculated to be less than 1.0 based on the design transients.

LRA Section 4.3 states that the new design cycle limit for the remotely welded plugs was reduced to 33 cycles (Transient 32 in LRA Table 4.3-1). During its audit, the staff noted in the applicant's basis documents for the metal fatigue TLAA, that manually welded plugs may also be limited to 33 cycles although no specific analysis was performed at the time. The staff also noted that there were other once through steam generator (OTSG) tube plug types that did not need to be qualified to the OEM equipment specification requirements. Furthermore, the staff noted that by letter dated November 3, 2003, the applicant responded to the staff's request for additional information regarding the 2002 steam generator tube inspection (ADAMS Accession No. ML033100370) and stated that there are 36 construction-era welded plugs and two of them were repaired in 2003 with remote welded plugs.

It is not clear to the staff if other types of weld plugs, such as the 36 construction-era welded plugs and the two repaired welded plugs that were not discussed in the LRA Section 4.3.2.6.1, have applicable fatigue design analysis. It is also not clear to the staff whether these other types of plugs are bounded by the remotely welded plugs which have a limit of 33 cycles for Transient 32.

Clarify whether there are other types of plugs, other than remote welded plugs, for the steam generator. If so, clarify whether these other types of plugs have applicable fatigue design analysis and provide the applicable design transients and associated limits for these plugs.

RESPONSE RAI 4.3-7

As provided in LRA Section 4.3.2.2.6.1, the steam generator remote weld plugs have a limited design life of 33 heatup/cooldown cycles to maintain a fatigue usage of less than 1.0.

The once through steam generators tube repairs include explosive tube plugs, welded U-cup plugs, rolled tube plugs, sleeve plugs, mechanical plugs and welded tube plugs. Only the welded tube plugs, which includes construction era welded plugs and repaired welded plugs, whether welded remotely or manually, have fatigue analyses. The remote welded plugs with a design life of 33 cycles are the most limiting and therefore, bound the other welded tube plugs.

Question RAI 4.3-8

LRA Section 4.3.2.2.6.3 states that "The analysis of the auxiliary feedwater thermal sleeve stresses provided a basis for demonstrating that the auxiliary feedwater thermal sleeve is capable of withstanding 300 cycles of auxiliary feedwater injection transients." The applicant also stated that auxiliary feedwater (AFW) initiations (Transients 30A and 30B in LRA Table 4.3-1) are currently at 196.5 and 224.5 cycles, respectively. The staff noted that Transients 30A and 30B are projected to a maximum of 387 and 442 cycles, respectively, through the period of extended operation. These 60-year projections are less than the 875 design cycles for the riser flange attachment but exceed the 300 design cycles for the auxiliary feedwater thermal sleeve.

The staff noted that Transients 30A and 30B in LRA Table 4.3-1 are identified as "Auxiliary Feedwater Bolted Nozzle" (1-1 and 1-2). It is not clear to the staff whether these auxiliary feedwater injection transients refer to those transients identified in LRA Table 4.3-1.

During its audit, the staff noted that the applicant's basis documents for the metal fatigue TLAA indicated that the 3-inch auxiliary feedwater nozzles are limited to 1447 cycles of AFW initiation based on the CUF of 1.0 for the studs. It is not clear to the staff whether the design cycle limit of 1447 cycles for "AFW initiation" is tracked in the applicant's Fatigue Monitoring Program.

The staff requests the following information:

1. **Clarify how the "auxiliary feedwater injection transient" for the modified AFW thermal sleeve design is related to the "Auxiliary Feedwater Bolted Nozzle 1-1," Transient 30A in LRA Table 4.3-1, and "Auxiliary Feedwater Bolted Nozzle 1-2," Transient 30B in LRA Table 4.3-1.**

2. **Clarify the cycle limit of 1447 for the "AFW initiations" transient discussed in the basis document for the metal fatigue TLAA and whether this "AFW initiation" transient is monitored by the Fatigue Monitoring Program during the period of extended operation. If not, justify why the "AFW initiations" transient does not need to be monitored by the Fatigue Monitoring Program during the period of extended operation.**

RESPONSE RAI 4.3-8

The auxiliary feedwater injection transient used to evaluate the AFW nozzle thermal sleeves, AFW nozzle studs, and AFW nozzle flange (see Figure 1 of this response) assumes on/off control of the AFW system at a temperature of 40°F, flow duration per cycle of 425 seconds at 1000-1400 gallons per minute per steam generator, and 600 second interval from flow stop to flow start.

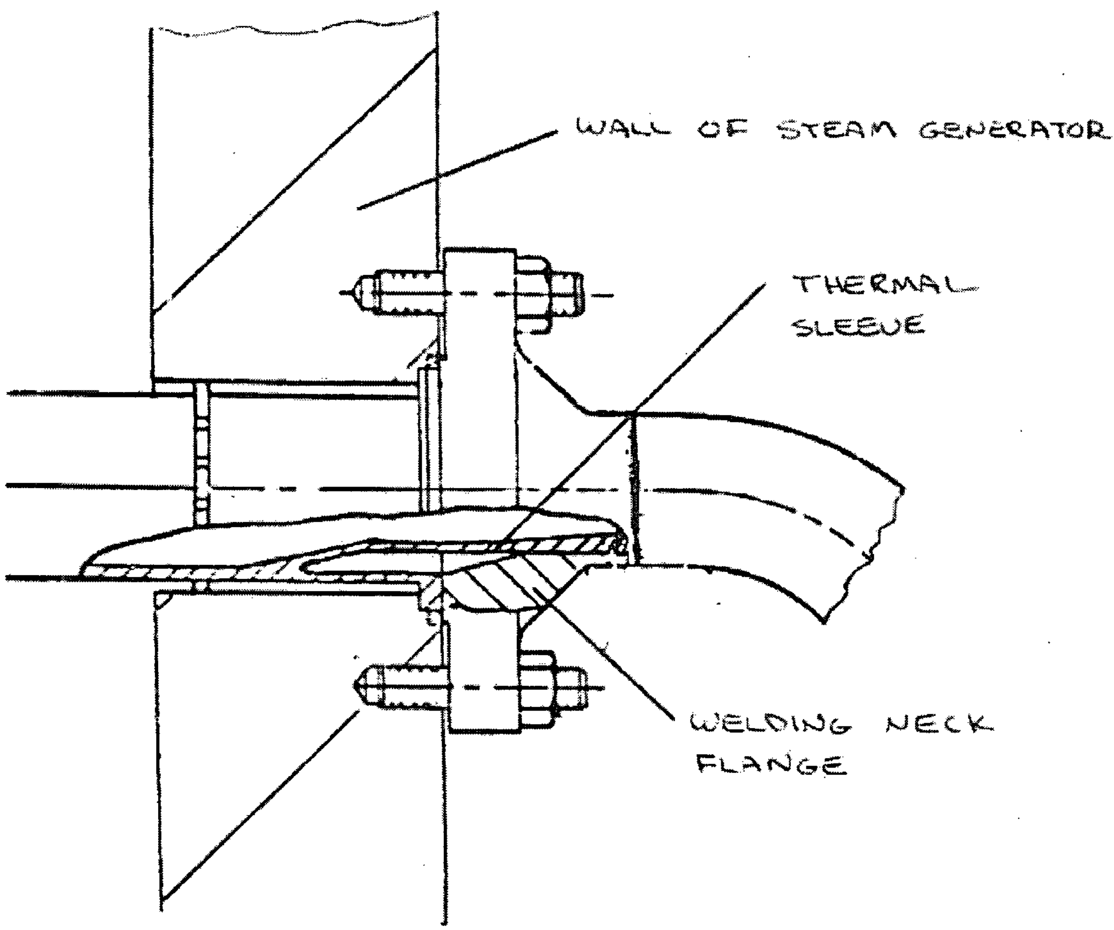


FIGURE 1 : AUXILIARY FEEDWATER NOZZLE GEOMETRY

The AFW nozzle thermal sleeves were initially qualified for 300 AFW cycles in July 1982 using conservative analytical techniques (hand calculations). The thermal sleeves were reanalyzed in December 1982 using numerical methods and re-qualified for 40,000 AFW cycles. Thermal loads were as described above from AFW actuations with an initial bulk fluid temperature of 570°F. The fluid temperature changes from 570°F to 40°F instantaneously upon AFW actuation and returns to 570°F after AFW is stopped. Mechanical loads assumed in the evaluation of the thermal sleeves included transverse drag and oscillating lift from steam flow in the SG annulus, and hydrostatic pressure. NSSS design transients were not included in the thermal sleeve evaluation since it is not a pressure retaining item.

The AFW nozzle stud fatigue analysis included the following bounding NSSS design transients: heatup/cooldown, boltup/unbolt and AFW initiation. Mechanical loads included piping moment loads on the flange. The cycles were reduced from 7,000 to 1,447 to obtain a CUF for the studs of less than 1.0.

The AFW nozzle flange fatigue analysis included the following bounding NSSS design transients; heatup/cooldown, boltup/unbolt and AFW initiation. Mechanical loads included piping moment loads on the flange. The cycles were reduced from 7,000 to 875 to obtain a CUF for the flange of 0.55.

Transients 30A and 30B in LRA Table 4.3-1, identified as "Auxiliary Feedwater Bolted Nozzle" (1-1 and 1-2), are applicable to the AFW nozzle flanges. The limiting component is the flange and therefore, the transient design cycle limit is set to 875. The studs have a higher CUF but are analyzed for 1,447 cycles.

LRA change is required to shown that the AFW nozzle thermal sleeve is qualified for 40,000 cycles and that the AFW nozzle flange is qualified to 875 cycles of heatup/cooldown, boltup/unbolt and AFW initiation.

See Enclosure A to this letter for the change to the LRA.

Question RAI 4.3-9

LRA Section 4.3.1.2 indicates that the number of cycles accrued as of February 2008 were compiled and linearly extrapolated to the 60 years of operation to determine whether the incurred cycles would remain below the number of design cycles.

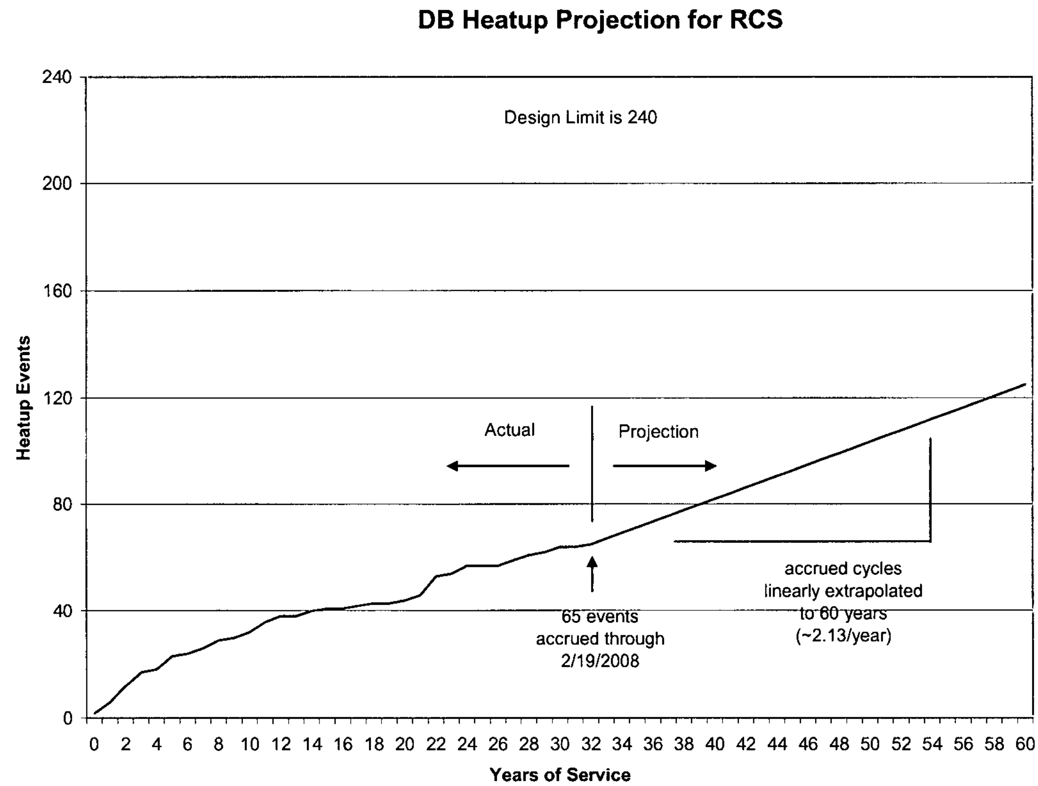
The applicant did not justify the use of a linear extrapolation to determine the number of cycles for 60 years and whether it is conservative, based on its plant-specific operating history.

Explain the methodology used for the linear extrapolation of design transients and justify that the use of a linear extrapolation to determine the number of cycles for 60 years is valid and conservative, based on the plant-specific operating history.

RESPONSE RAI 4.3-9

Davis-Besse plant-specific operating history is similar to the operating history of other commercial nuclear power plants of similar vintage. Early in plant life, transients were frequent. As issues were resolved, the transient frequency decreased. In addition, the Davis-Besse fuel cycle has been increased to 2 years in duration resulting in a further decrease in transient cycles. Therefore, linear extrapolation of cycles that have occurred over the entire operating history of the plant to project 60-year cycles is conservative.

For example, the RCS heatup transients have occurred at a decreased frequency over time as shown below.



From plant startup (August 12, 1977) to February 19, 2008 the plant accrued 65 plant heatups. Using a linear extrapolation of these cycles, 65 cycles divided by 30.5 years, results in a rate of approximately 2.13 cycles/year. Using the assumption of 2.13 heatups per year for the remaining 29.5 years of operation, from February 19, 2008 through the period of extended operation, results in 128 heatups for 60 years of operation as shown in LRA Table 4.3-1.

Question RAI 4.3-10

LRA Table 4.3-1 states that Transients #19, #20A, #20B, #200, #23A, #23B, #23C, and #23D are not fatigue significant events. LRA Table 4.3-1 also states that Transients #25A and #25B are not fatigue events. Therefore, the applicant concluded that the monitoring of these transients is not needed

The applicant did not provide a discussion to explain and justify why these transients are not fatigue significant events or fatigue events.

Justify why these transients are not considered fatigue significant events or fatigue events. In addition, justify why these transients do not need to be monitored by the Fatigue Monitoring Program during the period of extended operation.

RESPONSE RAI 4.3-10

Transient 19 is a feed and bleed operation wherein RCS boron concentration change is made by introducing borated or deborated water through the makeup system and with letdown to the letdown storage tank or the waste disposal system. The stress analysis of the makeup nozzle was reviewed and this transient has no fatigue contribution.

Transient 20 is included to account for unknown transients on the makeup and spray line piping and nozzles during normal operation. These transients have very little impact on fatigue due to the number of expected cycles compared to the large number of design cycles.

Transient 23 includes Steam Generator Filling, Draining, Flushing, and Cleaning. These transients are conducted at temperatures less than 225°F and are expected to have little or no contribution to fatigue of the steam generators.

Transient 25 is applicable to the pressurizer electrical heaters and has no contribution to fatigue of the pressurizer or the pressurizer heater elements.

Question RAI 4.3-11

LRA Table 4.3-1 indicates that Transient 22A "Test-High Pressure Injection System" corresponds to Transient 12 in USAR Table 5.1-8. The applicant indicated that Transient 3 "Power change 8-100%" and Transient 4 "Power change 100-8%" correspond to Transient #3 in USAR Table 5.1-8. The applicant stated that these transients are not monitored and provided technical justifications in LRA Table 4.3-1.

The staff noted that cycle counting of the applicant's design basis transients in USAR Table 5.1-8 is required by its Technical Specification (TS) 5.5.5, unless the USAR specifically explains why the design basis transient is not monitored. The staff noted that the Revision 26 of USAR Table 5.1-8 indicates that these transients are applicable to TS 5.5.5 and the USAR does not identify the transients listed above as not requiring cycle counting.

The staff requests the following information:

1. **Confirm that the "Test-High Pressure Injection System", "Power change 8-100%", and "Power change 100-8%" transients are the only transients, listed both in LRA Table 4.3-1 and USAR Table 5.1-8 that require counting per TS 5.5.5, but are not counted by the Fatigue Monitoring Program. If not, identify any additional transients that require counting per TS 5.5.5, but are not counted by the Fatigue Monitoring Program.**
2. **Clarify whether USAR Table 5.1-8 currently does not require the "Test-High Pressure Injection System", "Power change 8-100%", and "Power change 100-8%" transients from the cycle monitoring requirements of TS 5.5.5.**
3. **Explain and justify why the monitoring of transients can be omitted without justification in USAR Section 5.2, USAR Table 5.1-8 and the applicant's cycle counting procedure.**

RESPONSE RAI 4.3-11

By letter dated June 3, 2011 (L-11-166), FENOC responded to the above request for information. In response to RAI B.2.16-1, the technical justification for transients not monitored was provided along with any required changes to USAR Table 5.1-8 and LRA Table 4.3-1.

Question RAI 4.3-12

The LRA does not provide the CUF values for ASME Code Section III Class 1 components described in LRA Section 4.3.2. Without these values, the staff is not able to ascertain whether the CUF value for these locations exceeded the allowable limit or evaluate the applicant's dispositions of these TLAAs in accordance with 10 CFR 54.21(c).

Provide the design-basis 40-year CUF values for all components and/or critical locations that are applicable to the dispositions discussed in LRA Sections 4.3.2.

RESPONSE RAI 4.3-12

Design (40-year) CUFs for all DB-1 Class 1 components are provided in Tables 3-1 through 3-9 of AREVA document 51-9157140-001, "DB-1 Design CUFs and NUREG/CR-6260 Screening for License Renewal," dated 6/10/2011.

See Enclosure B to this letter for a copy of report 51-9157140-001.

Question RAI 4.3-13

LRA Section 4.3.2.3.3 states that the CUF analyses for Class 1 High Energy Line Break (HELB) locations TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). The applicant also stated that the effect of fatigue on the HELB location selection will be managed by the Fatigue Monitoring Program during the period of extended operation.

The staff noted that a CUF value less than 0.1 is one of the HELB location selection criteria discussed in the Standard Review Plan (NUREG-0800) Sections 3.6.1 and 3.6.2, including Branch Technical Position MEB 3-1. The staff also noted that a CUF value less than 1.0 is one of the cumulative fatigue damage design criteria in ASME Code Section III.

The staff noted that it may be possible that the design cycle limit applicable to HELB piping locations can be less than the "Design Cycles" identified in LRA Table 4.3-1. In addition, the "acceptance criteria" program element in the Fatigue Monitoring Program did not address how the acceptance criteria will be different for HELB and cumulative fatigue damage. The staff noted that the Fatigue Monitoring Program indicates that when the accumulated cycles approach the design cycles, corrective actions will be taken to ensure the analyzed number of cycles is not exceeded. However, the Fatigue Monitoring Program does not discuss the situation when the accumulated cycles approach the limit in the HELB analyses.

The staff requests the following information:

1. **Identify the ASME Code Class 1 piping locations discussed in USAR Section 3.6.2 that are within the scope of LRA Section 4.3.2.3.3. Provide the design-basis transients and associated cycle limits that are applicable to each HELB piping location that are within the scope of LRA Section 4.3.2.3.3.**

2. **Justify that the Fatigue Monitoring Program can adequately ensure the CUF for HELB locations remain below 0.1 by using systematic counting of plant transient cycles associated with HELB analysis. Provide any appropriate revisions to the program elements of the Fatigue Monitoring Program, as needed, to incorporate activities for ensuring that the CUF for HELB locations remain below 0.1.**

RESPONSE RAI 4.3-13

1. High energy line break (HELB) postulation based on fatigue usage is applicable to ASME Code Class 1 piping listed as follows:
    - Low Pressure Injection Lines
    - Core Flooding Lines
    - Letdown Line
    - Decay Heat Removal Lines

    The fatigue analyses for the Low Pressure Injection Lines, Core Flooding Lines, Letdown Line, and Decay Heat Removal Lines considered the design transients as shown in the table below. The 60-year projected cycles are from LRA Table 4.3-1 as revised by FENOC Letter L-11-166, dated June 3, 2011.

Program Transient#	***Transient***	***HELB Analyses Analyzed Cycles***	***LRA Table 4.3-1 60-year Projected Cycles***
1 A	RCS Heatup from 70F to 8% Full Power (Normal) [USAR Transient # 1A]	240	128
1 B	RCS Cooldown from 8% Full Power (Normal) [USAR Transient # 1B]	240	128
2 A	Power change 0 to 15% (Normal) [USAR Transient # 2]	1440	205
2 B	Power change 15 to 0% (Normal) [USAR Transient # 2]	1440	94
3	Power Loading 8% to 100% (Normal) [USAR Transient # 3]	48000	Transients are not monitored. Davis-Besse is not a load following plant and therefore; transients 3 and 4 could not credibly approach the number of analyzed cycles during the period of extended operation.
4	Power Unloading 100-8% (Normal) [USAR Transient # 4]	48000	
5	10% Step Load Increase (Normal) [USAR Transient # 5]	8000	67
6	10% Step Load Decrease (Normal) [USAR Transient # 6]	8000	140

<table>
<tr><th>Program Transient#</th><th>Transient</th><th>HELB Analyses Analyzed Cycles</th><th>LRA Table 4.3-1 60-year Projected Cycles</th></tr>
<tr><td>7 A</td><td>Step Load Reduction 100-8% from Turbine Trip (Upset)
[USAR Transient # 7A]</td><td rowspan="2">310</td><td>8</td></tr>
<tr><td>7 B</td><td>Step Load Reduction 100-8% from Electrical Load Rejection (Upset)
[USAR Transient # 7B]</td><td>4</td></tr>
<tr><td>8 A</td><td>Reactor Trip - Low RCS flow directly causes Rx trip (Upset)
[USAR Transient # 8A]</td><td rowspan="3">288</td><td>4</td></tr>
<tr><td>8 B</td><td>Reactor Trip - High RCS outlet temperature, high RCS pressure or overpower trip (assumes a turbine trip occurs without automatic control system action) (Upset)
[USAR Transient # 8B]</td><td>47</td></tr>
<tr><td>8 C</td><td>Reactor Trip - High RCS pressure resulting from loss of feedwater (Upset)
[USAR Transient # 8C]</td><td>26</td></tr>
<tr><td>9 A</td><td>Rapid RCS Depressurization (Upset)
[USAR Transient # 9A]</td><td>80</td><td>4</td></tr>
<tr><td>11</td><td>Rod withdrawal accident (Upset)
[USAR Transient # 11]</td><td>40</td><td>40</td></tr>
<tr><td>12 A</td><td>Hydrotest – RCS components except OTSG Secondary (includes 5 shop tests) (Test)
[USAR Transient # 12A]</td><td>20</td><td>9</td></tr>
<tr><td>14</td><td>Control Rod Drop (Upset)
[USAR Transient # 14]</td><td>40</td><td>18</td></tr>
<tr><td>15</td><td>Loss of Station Power (Upset)
[USAR Transient # 15]</td><td>40</td><td>6</td></tr>
</table>

2. As shown in the above table, the 60-year projected cycles are bounded by the analyzed cycles and therefore, the Class 1 HELB postulations remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

   FENOC has elected to disposition the Class 1 HELB postulations in accordance with 10 CFR 54.21(c)(1)(i) and therefore, will not credit the Fatigue Monitoring Program for managing the effects of fatigue on the high energy line break postulations.

   An LRA change is required to revise the disposition of the TLAA. In addition, a correction was required as to the applicable Section of the USAR and the guidance cited for determining the break locations.

   See Enclosure A to this letter for the changes to the Davis-Besse LRA.

Question RAI 4.3-14

In LRA Section 4.3.4, the applicant discussed the methodology to determine the locations that require environmentally assisted fatigue (EAF) analyses consistent with NUREG/CR-6260 "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The staff recognized that, in LRA Table 4.3-2, there are fifteen plant-specific locations listed, based on the six generic components identified in NUREG/CR-6260.

The GALL Report AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary" states that the impact of the reactor coolant environment on a sample of critical components should include the locations identified in NUREG/CR-6260 as a minimum, and that additional locations may be needed. It was not clear to the staff whether the applicant verified that the plant-specific locations listed in the LRA Table 4.3-2 were bounding for the generic NUREG/CR-6260 components. Furthermore, the staff noted that the applicant's plant-specific configuration may contain locations that should be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260.

The staff requests the following information:

1. **Confirm and justify that the plant-specific locations listed in LRA Table 4.3-2 are bounding for the generic NUREG/CR-6260 components.**

2. **Confirm and justify that the LRA Table 4.3-2 locations selected for environmentally assisted fatigue analyses consists of the most limiting locations for the plant (beyond the generic locations identified in the NUREG/CR-6260 guidance). If these locations are not bounding, clarify the locations that require an environmentally assisted fatigue analysis and the actions that will be taken for these additional locations.**

RESPONSE RAI 4.3-14

A response to the above request for information has been previously provided in FENOC Letter L-11-166. In response to RAI B.2.16-2, FENOC addressed application of environmentally assisted fatigue (EAF) for locations beyond the generic locations identified in the NUREG/CR-6260 guidance and provided the required changes to LRA Sections A.1.16 and B.2.16 and Table A-1.

In addition, in response to RAI 4.3-12 of this letter, a copy of AREVA document 51-9157140-001, "DB-1 Design CUFs and NUREG/CR-6260 Screening for License Renewal," dated 6/10/2011, is provided in Enclosure B to this letter. In this document, a list was compiled of all design CUFs multiplied by a maximum EAF correction factor (F_{en}) to obtain a list of bounding EAF CUF values.

Question RAI 4.3-15

LRA Section 4.3.1.2 states that "Transients 9C, 9D, and 32 are the only transients affecting Class 1 components where the 60-year projected cycles exceed the design cycles".

The applicant stated that HPI nozzles 2-1 and 2-2 are limited to 40 cycles for Transients 9C and 9D, respectively, and it will manage cumulative fatigue damage of these nozzles for the period of extended operation. However, it is not clear to the staff if there are other components that have Transient 9C or 9D in the design-basis fatigue calculation and whether these components will be affected if the 60-year projected cycles are exceeded.

Clarify whether there are other components that include Transients 9C or 9D in their design- basis fatigue calculation. If there are other components that use Transient 9C or 9D in their design-basis fatigue calculations, identify the number of design cycles in those fatigue calculations. Discuss and justify the fatigue TLAA disposition of these components.

RESPONSE RAI 4.3-15

Transients 9C (now Transient 22 A2, HPI Nozzle 2-1) and 9D (now Transient 22 A2, HPI Nozzle 2-2) are only applicable to HPI Nozzles 2-1 and 2-2, respectively.

Please note that by letter dated June 3, 2011 (L-11-166), in the response to RAI B.2.16-1, FENOC amended LRA Table 4.3-1 such that previous listed transients 9A through 9D are renamed as the HPI System Pressure Isolation Integrity Tests, and are now grouped under transient number 22 A2 (HPI Nozzles 1-1, 1-2, 2-1 and 2-2).

Question RAI 4.3-16

LRA Section 4.3.4.2 and LRA Table 4.3-2 states that the in-air design CUFs were adjusted by reducing conservatism in the original design calculations and/or by refining the material specific F_{en} factor. LRA Table 4.3-2 provided a summary of the adjusted CUFs and environmentally- adjusted U_{en} factors.

Specific to the reactor vessel inlet and outlet nozzles and the pressurizer surge nozzle safe-end, the applicant stated that incremental fatigue contributions were identified and reduced based on the 60-year projected cycles. Specific to the high pressure injection/makeup nozzle and stainless steel safe-end, the applicant stated that although conservatism in the design analysis was removed and it still maintained the full-set of 40-year NSSS design transients.

It is not clear to the staff which incremental contributions were reduced based on the 60-year projected cycles, which transients were used and the number of cycles that were used in the analysis. Furthermore, it is not clear to the staff which variables in the original design calculations were adjusted, what elements of conservatism were reduced and the basis for these adjustments and reductions.

The staff requests the following information:

1. **For each location in which the incremental fatigue contributions were reduced based on the 60-year projected cycles, provide the following:**

    a. **Identification of the transients used in the original design CUF calculation.**

    b. **The analyzed number of cycles used for the transients identified above in the CUF calculation.**

    c. **Clarification on how the incremental fatigue contribution was adjusted.**

2. **Clarify if there are other variables and elements of the original design calculations that were used to reduce the conservatism in the original CUFs of record. Describe and justify the reduction of conservatism for each variable and element in the original CUFs of record.**

RESPONSE RAI 4.3-16

1. From Table 4.3-2 of the LRA, the locations where the design CUFs were reduced are as follows:

    - RV Inlet Nozzle

    The overall maximum cumulative usage factor (CUF) for the RV inlet nozzle was reduced from 0.829 to 0.146 by utilizing the current design cycles for Transients 3 and 4 and 60-year projections for Transients 5 and 6. The design CUF for the RV inlet nozzle utilized all of the NSSS design cycles defined at the time of the analysis (i.e., 1990). The largest contribution to fatigue was found to be due to Transients 3 through 6. Specifically, the analysis used 48,000 cycles for Transients 3 and 4 (48,000 cycles is very conservative since the current design cycles for Transients 3 and 4 are 1,800), and 8,000 cycles for Transients 5 and 6. The CUF reduction was obtained by using 1,800 cycles for Transients 3 and 4, and projected cycles at 60 years for Transients 5 and 6 at 67 and 140, respectively. Therefore, the CUF reduction for the RV inlet nozzle was obtained by reducing the incremental fatigue contribution for Transients 3 through 6 only. The CUF contribution for the remaining NSSS design transients listed in LRA Table 4.3-1 is unchanged. Therefore, the design CUF for the RV inlet nozzle was reduced from 0.829 to 0.146 for the F_{en} evaluation reported in Table 4.3-2 of the DB-1 LRA.

- RV Outlet Nozzle

Similar to the RV inlet nozzle, the maximum CUF for the RV outlet nozzle was reduced from 0.768 to 0.335. The CUF reduction was obtained by using 1,800 cycles for Transients 3 and 4, and projected cycles at 60 years for Transients 5 and 6 at 67 and 140, respectively. Therefore, the CUF reduction for the RV outlet nozzle was obtained by reducing the incremental fatigue contribution for Transients 3 through 6 only. The CUF contribution for the remaining NSSS design transients listed in LRA Table 4.3-1 is unchanged. Therefore, the design CUF for the RV outlet nozzle was reduced from 0.768 to 0.335 for the F_{en} evaluation reported in Table 4.3-2 of the DB-1 LRA.

- Pressurizer Surge Nozzle Safe End

The design CUF for the pressurizer surge nozzle safe end consists of 0.108 from heatup and cooldown and 0.000 from all other NSSS design transients. Using the 60-year projection of 128 heatups and cooldowns, the design CUF for the pressurizer nozzle safe end was reduced from 0.108 to 0.0581 for the F_{en} evaluation reported in Table 4.3-2 of the DB-1 LRA.

- HPI/MU Nozzle

<u>Carbon Steel Nozzle</u>

The design CUF for the carbon steel HPI nozzle is 0.589. One of the major contributions to this CUF is from Transient 12 (Hydrotest) with 10 cycles and Transient 23 (Steam generator filing, draining, flushing and cleaning) with 540 cycles. The stress for Transient 23 was conservatively based on the same pressure as the hydrotest (i.e., 3125 psig*1.25) plus stresses from thermal moments and mechanical loads. In accordance with the RCS Functional Specification, the permitted pressure range for Transient 23 is 485 psig. Therefore, the stress due to pressure for Transient 23 was reduced by a factor of (500/3125) with stresses due to thermal moments and mechanical loads unchanged. The usage factor contributions from the other NSSS transients were not changed. The overall usage factor for the carbon steel nozzle was reduced from 0.589 to 0.348 for the F_{en} evaluation reported in Table 4.3-2 of the DB-1 LRA.

Stainless Steel Safe End

The design CUF for the HPI nozzle safe end is 0.660. The transient events considered include Transient 9 (Rapid Depressurization), Transient 22 (HPI Test), full range of earthquake, and all other conditions excluding Transients 9 and 22. Transient 22 as defined in the RCS Functional Specification cannot occur at Davis-Besse. The HPI pump shutoff head is approximately 1600 psig and therefore, the pumps are recirculated back to the Borated Water Storage Tank during the HPI System Test. Since no inventory is added to the Reactor Coolant System, fatigue usage due to Transient 22 was eliminated. The usage factor contributions from the other NSSS transients were not changed. The overall usage factor for the HPI stainless steel safe end was reduced from 0.664 to 0.550 for the F_{en} evaluation reported in Table 4.3-2 of the DB-1 LRA.

2. The RV inlet nozzle, RV outlet nozzle, pressurizer surge nozzle safe end, HPI/MU nozzle and HPI/MU nozzle safe end are the only locations where selected 60-year transient projections were used to reduce the CUFs.

Question RAI 4.3-17

LRA Section 4.3.4.2 states that the surge line piping and high pressure injection/makeup (HPI/MU) nozzle and safe end were evaluated using an integrated F_{en} approach consistent with EPRI Technical Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application," Revision 1, Section 4.2.

The staff noted that consistent with MRP-47, Section 4.2, the CUF and U_{en} are computed for each load pair and an effective F_{en} is calculated by dividing the U_{en} by the CUF. LRA Section 4.3.4 states that the maximum U_{en} is calculated with a global F_{en} and the adjusted CUF is then obtained by dividing the U_{en} by the global F_{en}.

The staff noted that EPRI Technical Report MRP-47 has not been reviewed and approved by the NRC. Furthermore, the applicant stated that in footnote 2 of LRA Table 4.3-2 the global F_{en} is calculated using the method from Section 4.2 of MRP-47. However, the term "global F_{en}" is not discussed in MRP-47. The staff further noted that the process of calculating global F_{en} is not discussed in the LRA.

Therefore, it is not clear to staff how the applicant determined the environmentally adjusted CUF for the surge line piping and HPI/MU nozzle and safe end.

The staff requests the following information:

1. **Justify that use of the integrated F_{en} approach in the EPRI MRP-47 is applicable and adequately conservative to calculate U_{en} for the period of extended operation.**

2. **Clarify the term "global F_{en}" and how it is calculated for each component. Provide its relationship with U_{en} calculation methodology discussed in MRP-47.**

RESPONSE RAI 4.3-17

Question 1

Surge Line Piping:

The surge line piping was evaluated using an integrated F_{en} approach consistent with MRP-47, Revision 1, Section 4.2, since there is no specific NRC guidance provided for application of F_{en} reported in NUREG/CR-5704 to an ASME fatigue evaluation. The environmental fatigue analysis of the pressurizer surge line involved computation of a separate F_{en} multiplier for each transient pair in the analysis. For the computation of F_{en}, a value for F_{en} was computed using a bounding strain rate of 0.0004%/S, oxygen content of <0.05 ppm, and appropriate values for the remaining variables depending upon operational conditions, such as temperature, associated with each transient type. The projected number of NSSS design cycles for 60 years reported in Table 4.3-1 of the LRA were used (except for best estimate 60-year project cycles of 114 used for HU/CDs events) and the pairings were performed in accordance with ASME Section III rules.

An F_{en} multiplier was applied to each pairing. This approach is conservative since the F_{en} factor used in the analysis considers the worst strain rate of 0.0004%/s and dissolved oxygen level of 0.05 ppm, which results in higher (conservative) F_{en} values. Thus, the use of this method is conservative and justified.

HPI Safe End

The Integrated Strain Rate approach from Section 4.2.2 of MRP-47, Revision 1, was used in the calculation since there is no specific NRC guidance provided for application of F_{en} reported in NUREG/CR-5704 to an ASME fatigue evaluation. It was assumed that the thermal strain is proportional to the service temperature difference (ΔT) since the major contribution of total strain comes from thermal strain for the HPI transients.

For this approach, F_{en} is computed in an integrated fashion at different temperatures, and an overall F_{en} is integrated over the entire temperature range considered. The integrated F_{en} was determined for each of the transient events that apply to the HPI nozzle safe end. That is Transient 9 rapid depressurization and the effect of HPI injection with temperature of the safe end changing from 650°F to 35°F. These integrated F_{en} were applied to the incremental CUF associated with each transient. The EAF CUF at this location was calculated to be 4.417, which is not acceptable. Therefore, FENOC provided commitment number 23 in Appendix A of the LRA to replace all four high pressure injection /makeup nozzle safe ends prior to the period of extended operation. (Note - See RAI 4.3-18 and its associated revision to license renewal future commitment No. 23.)

Question 2

Surge Line Piping

The "global F_{en}" is a term that is used to give an idea of the severity of the environmental effect at a specific location.

In this analysis, the in air CUF for 60 years is first calculated for each location along the surge line using 60 year projected number of cycles and using the same methodology used for the 40 year fatigue evaluation.

Also, the effect of the environment is taken into effect by performing fatigue analysis in which an F_{en} factor was determined for each transient pair, the U_{en} for each pair is determined by multiplying the in air usage for that transient pair by the F_{en} calculated for that pair. The U_{en} for each transient pair are added to come up with cumulative U_{en} for that specific location.

The "global F_{en}" is then calculated by dividing the U_{en} by the in air CUF. As it was mentioned above, the "global F_{en}" is an output that was calculated to give an idea of the severity of the environmental effect at a specific location. The "global F_{en}" is not used as an input in the analysis.

HPI Safe End

The "global F_{en}" is a term that is used to give an idea of the severity of the environmental effect at the HPI safe end. The global F_{en} is obtained by dividing the EAF CUF of 4.417 described above by the adjusted design CUF of 0.550 for a Global F_{en} of 8.03 versus the maximum F_{en} of 15.35 for stainless steel.

Question RAI 4.3-18

In LRA Appendix A, Table A-1, Commitment No. 23, the applicant committed to evaluate the environmental effects on the replacement high pressure injection (HPI) nozzle safe ends and associated welds in accordance with NUREG/CR-6260 and the guidance of EPRI Technical Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application," Revision 1, Section 4.2

EPRI Technical Report MRP-47 has not been reviewed and approved by the NRC. In addition, the applicant does not specify the specific portions of MRP-47 that will be used as part of this evaluation of environmental effects on the replacement HPI nozzle safe ends and associated welds. The staff noted that the applicant's Fatigue Monitoring Program with enhancements, in which the applicant stated is consistent with GALL AMP X.M1, addresses the effects of the reactor coolant environment on component fatigue life.

Justify that the use of EPRI Technical Report MRP-47 will properly evaluate the environmental effects on the replacement HPI nozzle safe ends and associated welds, in lieu of performing the evaluation and managing cumulative fatigue damage as part of the Fatigue Monitoring Program, which is consistent with the recommendations of the GALL AMP X.M1.

RESPONSE RAI 4.3-18

FENOC uses Section 4.2 of MRP-47, Revision 1, in the environmentally assisted fatigue calculations since there is no specific NRC guidance provided for application of F_{en} reported in NUREG/CR-5704 to an ASME fatigue evaluation.

As revised in FENOC Letter L-11-166, the Fatigue Monitoring Program (LRA Section B.2.16) prevents the fatigue TLAAs from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable.

LRA Appendix A, Table A-1, Commitment No. 23, previously revised in FENOC Letter L-11-107, is changed to credit the Fatigue Monitoring Program to evaluate the environmental effects and manage cumulative fatigue damage for the replacement high pressure injection (HPI) nozzle safe ends and associated welds.

See Enclosure A to this letter for the changes to the LRA.

Question RAI 4.3-19

LRA Section 4.3.4.2, specifically the discussion of the environmental fatigue usage evaluation for the stainless steel surge line piping, states that the 60-year transient projections were used for the evaluation with the exception of the 60-year projection of heatup/cooldowns (HU/CDs), where a best estimate number of 114 total cycles were used.

The staff noted that LRA Table 4.3-1 states that the 60-year projection cycles for HU and CDs are each 128 cycles, which is based on the linear extrapolation method described in the LRA Section 4.3.1.2.

In LRA Appendix A, Table A-1, Commitment No. 9, the applicant committed to monitor any transient where the 60-year projected cycles were used in an environmentally-assisted fatigue evaluation and establish an administrative limit that is equal to or less than the 60-year projected cycles. However, in this particular analysis for the stainless steel surge line piping, the staff noted that the analyzed number of cycle for HU/CDs is less than the 60-year projected cycle.

The staff requests the following information:

1. **Provide the basis of using 114 total HU/CDs in the environmental fatigue usage evaluation for the stainless steel surge line piping. Justify that the Fatigue Monitoring Program and Commitment No. 9 ensure that corrective actions are taken prior to the HU/CDs transients exceeding the analyzed number of cycles of 114 for each transient.**

2. **Clarify whether there are any additional locations in which the analyzed transient cycles are less than the 60-year projected cycles listed in LRA Table 4.3-1. If so, identify these locations and the associated analyzed cycles and the 60-year projected cycles for the applicable transients. In addition, justify that the Fatigue Monitoring Program ensures that corrective actions are taken prior to the applicable transients exceeding the analyzed number of cycles.**

RESPONSE RAI 4.3-19

1. Davis-Besse plant-specific operating history is similar to the operating history of other commercial nuclear power plants of similar vintage. Early in plant life, transients were frequent. As issues were resolved, the transient frequency decreased. In addition, the Davis-Besse fuel cycle has been increased to 2 years in duration resulting in a further decrease in transient cycles. Therefore, linear extrapolation of cycles that have occurred over the entire operating history of the plant to project 60-year cycles is conservative. Using this approach, the 60-year cycle projection of heatup transients is projected as follows. From plant startup (August 12, 1977) to February 19, 2008 the plant accrued 65 plant heatups. Using a

linear extrapolation of these cycles, 65 cycles divided by 30.5 years, results in a rate of approximately 2.13 cycles per year. Using the assumption of 2.13 heatups per year for the remaining 29.5 years of operation, from February 19, 2008 through the period of extended operation, results in 128 heatups for 60 years of operation as shown in LRA Table 4.3-1.

However, FENOC was unable to show the surge line piping environmentally assisted fatigue evaluation is acceptable for 60 years of operation using the above conservative approach, relative to projection of the heatup and cooldown cycles. Therefore, FENOC used best-estimate 60-year projected cycles for the heatup and cooldown cycles based on more recent operating experience versus the entire operation history of the plant. Using the recent operating experience (i.e. period from March 31, 2000 to February 19, 2008), the rate of occurrence of the subject transients are bounded by 1.5 cycles per year. This rate of occurrence resulted in best-estimate 60-year projected cycles of 114 for the heatup transient. The cooldown transient will occur an equal number of times and therefore, is set to 114 cycles.

Since projected cycles were used in the Davis-Besse environmentally assisted fatigue evaluations, these TLAAs were dispositioned using 10 CFR 54.21 (c)(1)(iii) where the effects of environmentally assisted fatigue will be managed by the Fatigue Monitoring Program (LRA Section B.2.16). As revised in FENOC Letter L-11-166, the Fatigue Monitoring Program prevents the fatigue TLAAs from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable.

2. The Davis-Besse environmentally assisted fatigue evaluations used 60-year projected cycles reported in Table 4.3-1 of the LRA for the NUREG/CR-6260 locations with the exception of the surge line piping evaluations that used best-estimate 60-year projected heatup and cooldown cycles.

Question RAI 4.3-20

LRA Section 4.3.2.2.6.4 states the CUF for the 3/8" tube stabilizers is calculated using both high cycle (flow-induced vibration) and low cycle (transients) fatigue. The applicant also stated that the cumulative usage factors are only 0.12 for the tube-to-stabilizer weld and 0.07 for the nail. In addition, the applicant stated that the flow induced vibration portion of these cumulative usage factors can be increased by 1.5 for 60 years and the cumulative usage factors will remain below 1.0.

The applicant stated that in accordance with 10 CFR 54.21(c)(1)(ii), the TLAA associated with the flow induced vibration of the steam generator tubes and tube stabilizers has been projected through the period of extended operation.

It is not clear to the staff whether the CUF values of 0.12 and 0.07 for the tube-to-stabilizer weld and the nail, respectively, include both high cycle and low cycle fatigue.

It is also not clear to the staff why only the flow induced vibration portion of these CUF values are increased by 1.5 to demonstrate that the TLAA is valid for the period of extended operation and how the low cycle (transient) portion of the CUF value is being dispositioned in accordance with 10 CFR 54.21(c).

The staff requests the following information:

1. **Clarify whether the CUFs of 0.12 and 0.07 are calculated considering both high cycle and low cycle fatigue.**

2. **Justify why the low cycle (transients) fatigue portion of the CUF values for the tube-to-stabilizer weld and nail do not need to be increased by 1.5 to determine if they will remain below 1.0. In addition, provide the disposition in accordance with 10 CFR 54.21(c)(1) for the low cycle (transient) portion of the fatigue TLAA for the tube-to-stabilizer weld and nail.**

RESPONSE RAI 4.3-20

1. FENOC confirms that the CUFs for the 3/8 inch tube stabilizers are calculated using both high cycle (flow inducted vibration) and low cycle (thermal transients) fatigue. The cumulative usage factors are 0.12 for the tube-to-stabilizer weld and 0.07 for the nail.

2. LRA Section 4.3.2.2.6.4 applies to the disposition of the high cycle fatigue TLAA for the steam generator tubes and stabilizers.

   The low cycle fatigue TLAA of the once through steam generator locations, this includes the stabilizers, is addressed in LRA Section 4.3.2.2.6.1. As provided in Section 4.3.2.2.6.1, the cumulative usage factors for the limiting primary and secondary side steam generators locations were calculated based on design transients, and are all less than 1.0. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Question RAI 4.3-21

LRA Section 4.3.4.2 states that an environmentally assisted fatigue correction factor, F_{en} was determined using material specific guidance contained in NUREG/CR-6583 "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels" and in NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials."

LRA Section 4.3.4.2 states that the following bounding F_{en} values calculated are: 1.74 for carbon steel, 2.45 for low-alloy steel and 4.16 for the nickel-based alloy incore instrument nozzles.

The staff noted that based on the guidance in NUREG/CR-6583 and NUREG/CR-6909, the F_{en} factor can vary based on sulfur content, temperature, dissolved oxygen, and strain rate. The staff noted that for nickel-based alloy components, per the guidance in NUREG/CR-6909, the F_{en} factor can be as high as 4.52. In addition for carbon and low-alloy steel components, per the guidance in NUREG/CR-6583, the F_{en} factor can vary significantly depending on the plant's history of dissolved oxygen content.

It is not clear to the staff, how the applicant determined the bounding F_{en} factors for the carbon and low-alloy steel and nickel-based alloy components that are described in LRA Section 4.3.4.2 and LRA Table 4.3-2.

The staff requests the following information:

1. **Clarify how the bounding F_{en} factors for the carbon and low-alloy steel and nickel-based alloy components were determined.**

2. **Justify any assumptions, on the parameters such as sulfur content, temperature, dissolved oxygen, and strain rate, which were used in determining the F_{en} factors for these components. As part of the justification, specifically for carbon and low-alloy steel, confirm that dissolved oxygen remained less than 0.05ppm since initial plant operation. If it has not, justify that the F_{en} factors are bounding.**

3. **Justify that the dissolved oxygen content will remain less than 0.05ppm during the period of extended operation, such that the F_{en} factors are bounding for the conditions at the plant.**

RESPONSE RAI 4.3-21

1. The lower bound F_{en} for carbon steel (1.74) and low-alloy steel (2.54) were calculated from NUREG/CR-6583, Equations 6.5a and 6.5b, respectively. For carbon and low-alloy steels in a PWR environment the dissolved oxygen level is less than 0.05 ppm at reactor coolant system (RCS) temperatures > 150 degrees

Celsius (°C) (302°F). Therefore, with RCS temperature > 150°C (302°F) the transformed dissolved oxygen is 0.0 and the bounding F_{en} for carbon steel and low-alloy steel are 1.74 and 2.54, respectively. At temperatures below 150°C (302°F) the dissolved oxygen may increase above 0.05 ppm but the transformed metal service temperature is 0.0 and bounding F_{en} for carbon steel and low alloy steel remain at 1.74 and 2.54, respectively.

The method used to calculate a nominal F_{en} for the incore instrument nozzles made from nickel based alloy is described in Appendix A, Page A.2 of NUREG/CR-6909. The transformed temperature, strain rate, and dissolved oxygen used in the evaluation are discussed below.

> The incore instrument nozzles are at the bottom of the reactor vessel. For DB-1 assume a temperature of the incore instrument nozzles of 582°F = 305.6°C. A temperature of 582°F corresponds to the average RCS temperature at 15% power (relative to current rated power of 2819 MWt) for steady state operations in the RCS Functional Specification. At 15% power the reactor inlet temperature (same as cold leg temperature) is ~577°F and the reactor outlet temperature is ~586°F for an average temperature of 582°F. As power increases the reactor inlet temperature decreases to 556.5°F at full power. Since Davis-Besse operates primarily at full load conditions the unit is operated for very short periods of time at 15% power with reactor inlet temperature as high as ~577°F. Therefore, assuming a constant reactor inlet temperature of 582°F for the EAF evaluation of the incore instrument nozzles is conservative for operation at rated power of 2819 MWt. Therefore, the transformed temperature is conservatively calculated as (307/325) = 0.945.
>
> The transformed strain rate is assumed to be the most limiting, and therefore the strain rate is calculated as ln(0.0004/5.0) = -9.43
>
> The transformed oxygen is 0.16 (PWR or HWC BWR water).

2. FENOC has confirmed that dissolved oxygen in the RCS at Davis-Besse has in general historically been less than 0.05 ppm with RCS temperatures > 150°C (302°F). The only exception is short periods of time during selected heatups where the pressurizer temperature was elevated to approximately 425°F with the RCS temperature at approximately 100°F. See number 3 below.

   Transformed strain rate was not relevant for carbon steel and low alloy steel since dissolved oxygen is less than 0.05 ppm at RCS temperatures > 150°C (302°F). The most limiting transformed strain rates were used for nickel based alloy. Transformed sulfur was not relevant for carbon steel and low alloy steel since dissolved oxygen is less than 0.05 ppm at RCS temperatures > 150°C (302°F) and transformed metal service temperature is 0.0 at RCS temperatures < 150°C.

3. The Davis-Besse PWR Water Chemistry Program invokes the EPRI water chemistry guidelines (TR-1014986 Revision 6, "Pressurized Water Reactor Primary Water Chemistry Guidelines"). These guidelines require action when dissolved oxygen exceeds 0.005 ppm at power operation (Modes 1 and 2). Historically the dissolved oxygen levels have exceeded the EPRI guidelines of less than 100 ppb during heatup of the RCS where the pressurizer temperature was elevated to approximately 425°F with the RCS temperature at approximately 100°F. FENOC determined that pressurizer dissolved oxygen requirements could not be met during heatup without changing the method for filling and venting the RCS and heating the Pressurizer. It was further determined that in order to meet dissolved oxygen requirements a method of adding hydrazine directly to the Pressurizer was needed. To ensure success in meeting pressurizer dissolved oxygen requirements for future plant heatups, an alternate method for confirming that pressurizer dissolved oxygen is within limits was developed. This method consists of 1) adding hydrazine to the pressurizer prior to heatup in excess of the amount necessary to consume the dissolved oxygen present, 2) heating the pressurizer to a temperature band of 235-245°F, and 3) holding temperature within that band for the time necessary to ensure dissolved oxygen is <100 ppb prior to continuing heatup beyond 250°F. This method was successfully employed during pressurizer heatup following the cycle 16 refueling outage. Use of this method along with the sampling frequency and dissolved oxygen limits specified in the Davis-Besse PWR Water Chemistry Program, provides reasonable assurance that reactor coolant dissolved oxygen levels will continue to be maintained below 50 ppb (.05 ppm) at temperatures above 250°F for the period of extended operation.

Question RAI 4.3-22

LRA Section A.2.3, Metal Fatigue, is divided into the following subsections:

- **Section A.2.3.1, Class 1 Code Fatigue Requirements**
- **Section A.2.3.2, Class 1 Fatigue Analyses**
- **Section A.2.3.3, Non-Class 1 Fatigue Analyses**
- **Section A.2.3.4, Generic Industry Issues on Fatigue**

10 CFR 54.21(d) requires that UFSAR supplement contain an appropriate summary description of all TLAA evaluations in the LRA.

The staff noted that LRA Section A.2.3.1 discusses the fatigue requirements for the reactor vessel and its components, Class 1 piping, and the once-through steam generator (OTSG) components. However, LRA Section A.2.3.2 does not include a summary description for all of the Class 1 components that received fatigue analysis in LRA Section 4.3.2 and its subsections. Specifically, the staff

noted LRA Section A.2.3 does not include a summary description subsection for the following Class 1 components:

- **Reactor vessel (RV) assembly shell components (LRA Section 4.3.2.2.1 has the corresponding analysis basis RV assembly components)**
- **Class 1 piping designed to ANSI B31.7 requirements (LRA Section 4.3.2.3.1 has the corresponding analysis basis)**
- **OTSG primary and secondary shell components (LRA Section 4.3.2.2.6.1 has the corresponding analysis basis)**

Justify why LRA Section A.2.3 does not include a summary description for the RV shell assembly and its components, the Class 1 piping designed to ANSI B31.7 requirements, and the OTSG primary and secondary shells and their components.

RESPONSE RAI 4.3-22

LRA Section A.2.3.2 is revised to include summary descriptions of the TLAA evaluations for the reactor vessel, the Class 1 piping, and the once through steam generators.

See Enclosure A to this letter for the change to the LRA.

Section B.2.7

Question RAI B.2.7-1, part 3

During a telephone conference call between FENOC and the NRC on June 7, 2011, the NRC requested a new license renewal commitment associated with part 3 of RAI B.2.7-1.

RESPONSE RAI B.2.7-1, part 3

By letter dated May 24, 2011 (L-11-153 – ADAMS Accession No. ML11151A090), FENOC responded to RAI B.2.7-1. No change to that response is necessary. LRA Table A-1 is revised to include a new license renewal future commitment as follows:

> The EDG Fuel Oil Storage Tanks (DB-T153-1 and DB-T153-2) and the in-scope fuel oil and Service Water buried piping will be cathodically protected in accordance with NACE SP0169-2007 or NACE RP0285-2002.

This new license renewal future commitment will be implemented prior to entering the period of extended operation.

See Enclosure A to this letter for the change to the LRA.

Question RAI B.2.7-1, part 10

During a telephone conference call between FENOC and the NRC on June 7, 2011, the NRC requested clarification of the term "reasonable assurance" in the original response to part 10 of RAI B.2.7-1.

RESPONSE RAI B.2.7-1, part 10

LRA Section A.1.7 states: Degradation or leakage found during inspections is entered into the Corrective Action Program to ensure evaluations are performed and appropriate corrective actions are taken. If adverse indications are detected, additional buried in-scope piping inspections will be performed in order to provide reasonable assurance of the integrity of buried piping.

The response to RAI B.2.7-1 stated: Degradation or leakage found during inspections is entered into the Corrective Action Program to ensure evaluations are performed and appropriate corrective actions are taken. If adverse indications are detected, additional buried in-scope piping inspections will be performed in order to provide reasonable assurance of the integrity of buried piping. The selection of components to be examined will be based on previous examination results, trending, risk ranking, and areas of cathodic protection failures or gaps, if applicable. Additional sampling continues until reasonable assurance of the integrity of buried piping is provided.

Further clarification of reasonable assurance is as follows: Evaluation within the Corrective Action Program determines the potential extent of the degradation observed. Expansion of sample size may be limited by the extent of piping or tanks subject to the observed degradation mechanism. When an adverse condition is detected that is not limited by the degradation mechanism, inspection sample sizes within the affected piping categories are doubled. If adverse indications are found in the expanded sample, the inspection sample size is again doubled. This doubling of the inspection sample size continues as necessary.

Enclosure A

Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS)

Letter L-11-203

<u>Amendment No. 9 to the</u>
<u>DBNPS License Renewal Application</u>

License Renewal Application
Sections Affected

Table 3.1.2-2	A.2.3.2.2
Table 3.5.1	A.2.3.2.7
Table 4.1-1	A.2.3.2.9
4.3.2.2.2.2	A.2.3.2.10
4.3.2.2.2.3	A.2.3.2.11
4.3.2.2.6.3	A.2.3.2.12
4.3.2.3.3	Table A-1
A.1.42	Table B-1
A.1.43	Table B-2
A.2.3.2	B.2.42

This Enclosure identifies the change to the License Renewal Application (LRA) by Affected LRA Section, LRA Page No., and Affected Paragraph and Sentence. The count for the affected paragraph, sentence, bullet, etc. starts at the beginning of the affected Section or at the top of the affected page, as appropriate. Below each section the reason for the change is identified, and the sentence affected is printed in *italics* with deleted text ~~*lined-out*~~ and added text <u>*underlined*</u>.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.1.2-2	**Page 3.1-72**	**Rows 42 and 110**

In response to RAI 4.3-3, rows 42 and 110 of LRA Table 3.1.2-2, "Aging Management Review Results – Reactor Vessel Internals," are revised as follows:

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals									
Row No.	**Component Type**	**Intended Function(s)**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
42	~~*CSA, Core Barrel; Bolt - Thermal Shield (UTS) and Lower Internals to Core Barrel*~~ *Not used.*	~~*Support*~~	~~*Stainless Steel*~~	~~*Borated Reactor Coolant (Internal)*~~	~~*Cracking fatigue*~~	~~*TLAA*~~	~~*IV.B4-37*~~	~~*3.1.1-05*~~	~~*A*~~
110	~~*CSA, Flow Distributor; Bolt – Shell Forging to Flow Distributor*~~ *Not used.*	~~*Support*~~	~~*Stainless Steel*~~	~~*Borated Reactor Coolant (Internal)*~~	~~*Cracking fatigue*~~	~~*TLAA*~~	~~*IV.B4-37*~~	~~*3.1.1-05*~~	~~*A*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.5.1	**Page 3.5-42**	**Rows 3.5.1-25 Discussion column**

In response to RAI XI.S8-1, the discussion column of row 3.5.1-25 of LRA Table 3.5.1, "Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801," is revised as follows:

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-25	All Groups except Group 6: steel components: all structural steel	Loss of material due to corrosion	Structures Monitoring Program. If protective coatings are relied upon to manage the effects of aging, the structures monitoring program is to include provisions to address protective coating monitoring and maintenance.	Yes, if not within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. Loss of material is managed by the Structures Monitoring Program for the affected steel structural components. Protective coatings are not relied upon to manage the effects of aging. *However, protective coatings inside the containment vessel are managed by the Nuclear Safety-Related Coatings Program.* ~~Davis-Besse has provided responses to the NRC regarding Generic Letter 2004-02. Containment~~

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801					
Item Number	**Component/Commodity**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
					~~coating condition assessment inspections are performed each refueling outage to identify and correct degraded coating materials under the current licensing basis. Containment coatings are subject to ongoing oversight that addresses their current status, which will continue to address their status over the period of extended operation.~~ Further evaluation is documented in Section 3.5.2.2.2.1.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 4.1-1	**Page 4.1-3**	**Metal Fatigue section, Class 1 Fatigue subsection, third and fourth rows**

In response to RAI 4.3-4, the third and fourth rows of the Class 1 Fatigue subsection of the Metal Fatigue section of Table 4.1-1 is revised to read:

Table 4.1-1 Time-Limited Aging Analyses

Results of TLAA Evaluation by Category	54.21(c)(1) Paragraph	LRA Section
Metal Fatigue		4.3
Class 1 Fatigue		4.3.2
Reactor Vessel internals <u>*and incore instrumentation nozzles*</u> *– flow induced vibration*	(i)	4.3.2.2.2.2
~~*Incore Instrumentation Nozzles and*~~ *Surveillance Capsule Holder Tubes – flow induced vibration*	(ii)	4.3.2.2.2.3

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 4.1-1	**Page 4.1-3**	**Metal Fatigue section, Class 1 Fatigue subsection, last row**

In response to RAI 4.3-13, the last row of the Class 1 Fatigue subsection of the Metal Fatigue section of Table 4.1-1 is revised to read:

Table 4.1-1 Time-Limited Aging Analyses

Results of TLAA Evaluation by Category	54.21(c)(1) Paragraph	LRA Section
Metal Fatigue		4.3
Class 1 Fatigue		4.3.2
High Energy Line Break Postulations	~~*(iii)*~~ <u>*(i)*</u>	4.3.2.3.3

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.3.2.2	**Page 4.3-7**	**Second paragraph, first sentence**

In response to RAI 4.3-1, the first sentence of the second paragraph of Section 4.3.2.2 is revised to read:

Cumulative usage factors for the Class 1 components are calculated based on the service and test loading ~~normal and upset design transient~~ definitions contained in the component design specifications.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.3.2.2.2.2	**Page 4.3-9**	**Section title**

In response to RAI 4.3-4, the section title of Section 4.3.2.2.2.2 is revised to read:

4.3.2.2.2.2 Reactor Vessel Internals and Incore Instrument Nozzles Flow Induced Vibration

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.3.2.2.2.2	**Page 4.3-9**	**Second paragraph, last sentence**

In response to RAI 4.3-4, the last sentence of the second paragraph of Section 4.3.2.2.2.2 is revised to read:

Therefore ~~the 18,000 psi endurance limit used for~~ the flow induced vibration ~~analyses~~ analysis of the reactor vessel internals and the incore instrument nozzles remains valid for the period of extended operation.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.3.2.2.2.2	**Page 4.3-9**	**Disposition**

In response to RAI 4.3-4, the Disposition subsection of Section 4.3.2.2.2.2 is revised to read:

Disposition: 10 CFR 54.21(c)(1)(i) ~~*The endurance limit for flow induced vibration of the reactor vessel internals remains valid to the end of the period of extended operation.*~~ *The flow induced vibration analysis of the reactor vessel internals and the incore instrument nozzles remain valid for the period of extended operation.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.3.2.2.2.3	**Page 4.3-10**	**Entire section**

In response to RAI 4.3-4, Section 4.3.2.2.2.3 is replaced in its entirety to read:

4.3.2.2.2.3 Surveillance Capsule Holder Tubes Flow Induced Vibration

The re-designed surveillance capsule holder tubes (re-designed holder tubes are installed at Davis-Besse) were analyzed for fatigue due to flow induced vibration. The resulting cumulative usage factor (CUF) is 0.00042. To project the flow induced vibration analysis from 40 years to 60 years of operation, 0.00042 was multiplied by 1.5 resulting in a CUF of 0.00063. The 60-year projected CUF is below the Code design limit of 1.0. Therefore, the surveillance capsule holder tubes flow induced vibration analysis has been satisfactorily projected to the end of the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(ii) The flow inducted vibration analysis for the surveillance capsule holder tubes has been projected to the end of the period of extended operation.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
4.3.2.2.6.3	**Page 4.3-13**	**Third paragraph**

In response to RAI 4.3-8, the third paragraph of Section 4.3.2.2.6.3 is revised to read:

The analysis of the auxiliary feedwater thermal sleeve stresses provided a basis for demonstrating that the auxiliary feedwater thermal sleeve is capable of withstanding ~~300~~ 40,000 cycles of auxiliary feedwater injection transients. This analysis was performed in accordance with the requirements of the ASME Code for Class I components. *The riser flange attachment (auxiliary feedwater nozzle flange) to the steam generator shell was also analyzed per ASME Code requirements, and was acceptable for a design life of 875 cycles (heatup/cooldown, boltup/unbolt and AFW initiation) ~~of auxiliary feedwater initiation. Auxiliary feedwater initiations,~~ Transients 30A and 30B in Table 4.3-1, are currently only at 196.5 and 224.5 cycles respectively.* Transients 30A and 30B are projected to a maximum of 387 and 442 cycles, respectively, through the period of extended operation. *These 60-year projections are ~~below~~ less than the 875 design cycles for the riser flange ~~attachment but exceed the 300 design cycles for the auxiliary feedwater thermal sleeve~~. However, ~~The~~ the number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached.* As such, the effects of aging due to fatigue are managed for the period of extended operation.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.3.2.3.3	**Page 4.3-17**	**Entire section**

In response to RAI 4.3-13, Section 4.3.2.3.3 is revised to read:

USAR Section ~~3.6.2.1~~ 3.6.2.2 indicates that the criteria given in ~~Standard Review Plan Sections 3.6.1 and 3.6.2, including Branch Technical Position MEB 3-1,~~ Regulatory Guide 1.46 was ~~were~~ used in determining the pipe break locations for pipe whip restraint design. This allows the elimination of potential break locations based on cumulative usage factors being less than 0.1, if other stress criteria are also met. *~~The cumulative usage factors calculated for Davis-Besse piping were based on the design transients that are counted by the Fatigue Monitoring Program. If any of the design cycles are approached, the Fatigue Monitoring Program will require action prior to the design cycles being reached. That action will include a review of the high energy line break location selections. As such, the effects of fatigue on the high energy line break location selection will be managed for the period of extended operation.~~ FENOC performed a comparison of the analyzed cycles that were used in the Class 1 HELB break location determinations to the 60-year projected cycles provided in LRA Table 4.3-1 and determined that the analyzed cycles bound the 60-year projected cycles. Therefore, the Class 1 HELB postulations remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).*

The identification of high energy line break locations for the hot and cold leg piping was replaced by leak-before-break criteria in 1990. See Section 4.7.1 below for a discussion of leak-before-break.

Disposition:	*10 CFR 54.21(c)(1)~~(iii)~~(i)*	*~~The effects of fatigue on the high energy line break location selection will be managed for the period of extended operation by the Fatigue Monitoring Program.~~ Class 1 HELB postulations remain valid for the period of extended operation.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.1.42 & A.1.43	**Page A-25**	**New section / Title Revision**
Appendix A Table of Contents	**Page A-4**	**A.1.42 & A.1.43**

In response to RAI XI.S8-1, a new LRA section is created to include a plant-specific aging management program. LRA Section A.1.42 is renamed from "References" to "Nuclear Safety-Related Coatings Program." The "References" section is renumbered as Section A.1.43, "References." Although not shown below, LRA Appendix A, "Updated Safety Analysis Report Supplement," "Table of Contents" on LRA Page A-4 is revised accordingly to include the renumbered sections.

New LRA Section A.1.42 reads:

A.1.42 Nuclear Safety-Related Coatings Program

The Nuclear Safety-Related Protective Coatings Program monitors the performance of Service Level 1 coatings inside containment through periodic coating examinations, condition assessments and remedial actions, including repair or testing. The Nuclear Safety-Related Protective Coatings Program defines roles, responsibilities, controls and deliverables for monitoring the condition of coatings in containment. This program also ensures that the Design Basis Accident (DBA) analysis limits with regard to debris loading from failed coatings will not be exceeded for the Emergency Core Cooling Systems (ECCS) suction strainers.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.3.2	**Page A-37**	**First paragraph**

In response to RAI 4.3-13, the first paragraph of Section A.2.3.2 is revised to read:

The Fatigue Monitoring Program monitors the number of plant transient cycles to ensure that action is taken before the number of design cycles is exceeded. As such, the effects of aging due to fatigue are managed for the period of extended operation for the Class 1 piping and components. ~~*The effects of fatigue on the high energy line break analyses are also managed by the Fatigue Monitoring Program.*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.3.2.2	**Page A-38**	**Entire section**

In response to RAI 4.3-4, Section A.2.3.2.2 is revised to read:

A.2.3.2.2 Reactor Vessel Internals <u>and Incore Instrument Nozzles</u> Flow Induced Vibration

The reactor vessel internals were analyzed for flow induced vibration. *The classic endurance limit approach to design of components subject to flow-induced vibration was used, except for ~~the incore instrumentation nozzles and~~ the re-designed surveillance capsule holder tubes.* The classic endurance limit approach is based on the observation that a fatigue curve becomes approximately asymptotic to a given value of stress (the endurance limit) for large numbers of cycles. A component can be designed for infinite life by maintaining the actual peak stresses below the endurance limit.

For the Davis-Besse reactor vessel internals, the ASME Code fatigue curve was extended to 1E+12 cycles (the upper bound on the number of cycles for a 40-year design life). The resulting stress value of 20,400 psi was reduced to 18,000 psi as the endurance limit. For 60 years of operation, it follows that 1.5E+12 would bound the expected loading cycles. The extrapolated fatigue curve at 1.5E+12 cycles is approximately 20,200 psi, still above the 18,000 psi that was used as the endurance limit. ~~*As such, the 18,000 psi endurance limit used for the*~~

~~flow induced vibration analyses of the reactor vessel internals remains valid for the period of extended operation.~~ Therefore, the ~~endurance limit for~~ flow induced vibration analysis of the reactor vessel internals and the incore instrument nozzles remains ~~remain~~ valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

~~The effects of fatigue due to flow induced vibration were analyzed for the incore instrument nozzles and re-designed surveillance capsule holder tubes for 40 years of operation. The resulting cumulative usage factors have been projected to remain below the limit of 1.0 for 60 years of operation.~~

~~The flow induced vibration analyses of the incore instrument nozzles and re-designed surveillance capsule holder tubes have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).~~

The re-designed surveillance capsule holder tubes (re-designed holder tubes are installed at Davis-Besse) were analyzed for fatigue due to flow induced vibration. The resulting cumulative usage factor (CUF) is 0.00042. To project the flow induced vibration analysis from 40 years to 60 years of operation, 0.00042 was multiplied by 1.5 resulting in a CUF of 0.00063. The 60-year projected CUF is below the Code design limit of 1.0. Therefore, the surveillance capsule holder tubes flow induced vibration analysis has been satisfactorily projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
A.2.3.2.7	**Page A-40**	**Second, third and fifth paragraphs**

In response to RAI 4.3-8, the second, third and fifth paragraphs of Section A.2.3.2.7 are revised to read:

The auxiliary feedwater thermal sleeve stresses were also analyzed according to the ASME Code for Class I components. *The analysis provided a basis for demonstrating that the AFW thermal sleeve is capable of withstanding ~~300~~ 40,000 cycles of auxiliary feedwater injection transients.*

In addition, the riser flange attachment (auxiliary feedwater nozzle flange) to the steam generator shell was analyzed per ASME Code requirements. However, it was necessary to limit the design life to 875 cycles (heatup/cooldown, boltup/unbolt and AFW initiation) ~~of auxiliary feedwater initiation~~.

The heatup/cooldown, boltup/unbolt and AFW initiation transients ~~*Auxiliary feedwater initiations*~~ *are projected to a maximum of 442 cycles through the period of extended operation.* This projection ~~exceeds the 300 cycles analyzed for the thermal sleeve but~~ is less than the 875 cycles analyzed for the riser flange. *However,* ~~*The*~~ *the number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached.* As such, the effects of aging due to fatigue are managed for the period of extended operation.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.3.2.9 **A.2.3.2.10** **A.2.3.2.11**	**Page A-41**	**New Sections**

In response to RAI 4.3-22, new sections A.2.3.2.9, A.2.3.2.10 and A.2.3.2.11 are added as follows:

A.2.3.2.9 Reactor Vessel

The reactor is designed as a Class A vessel in accordance with the ASME Code, Section III, 1968 Edition through Summer 1968 Addenda. A stress analysis of the entire vessel was conducted under both steady-state and transient operations. The result is a complete evaluation of both primary and secondary stresses and the fatigue life of the entire vessel. The reactor vessel was analyzed for fatigue by the original equipment manufacturer.

The cumulative usage factors for the limiting reactor vessel assembly locations were calculated to be less than 1.0 based on the design transients. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the analyzed numbers of transients are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2.10 Once Through Steam Generators

The primary (tube) and secondary (shell) sides of the once through steam generators are designed to ASME Section III, 1968 Edition through Summer 1968 Addenda. The steam generators were analyzed for fatigue by the original equipment manufacturer. The cumulative usage factors for the limiting primary and secondary side steam generators locations were calculated based on design transients, and are all less than 1.0. In addition, the steam generator remote weld plugs have a limited design life of 33 heatup/cooldown cycles to maintain a fatigue usage of less than 1.0. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2.11 Class 1 Piping

The Davis-Besse reactor coolant system piping, as well as reactor coolant pressure boundary piping in other systems, was designed to American National Standards Institute (ANSI) B31.7 Draft, February 1968 with Errata, June 1968 and also meets the design requirements of ANSI B31.7, 1969 Edition. The B31.7 Piping Code requires evaluation of transient thermal and mechanical load cycles and determination of fatigue usage for Class 1 piping. The reactor head vent and other piping designated as quality group A, B, or C is designed to ASME Section III, 1971 Edition, Class 1, 2 or 3 respectively. Only quality group D piping is designed to ANSI B31.1.

The cumulative usage factors for the Class 1 piping were analyzed based on the design transients, and are all less than 1.0. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.3.2.12	**Page A-41**	**New Section**

In response to RAI 4.3-13, new section A.2.3.2.12 is added as follows:

A.2.3.2.12 High Energy Line Break Postulations

USAR Section 3.6.2.2.1 indicates that the criteria given in Regulatory Guide 1.46 was used in determining the pipe break locations for pipe whip restraint design. This allows the elimination of potential break locations based on cumulative usage factors being less than 0.1, if other stress criteria are also met. The analyzed cycles that were used in the Class 1 HELB break location determinations were compared to the 60-year projected cycles. The comparison determined that the analyzed cycles bound the 60-year projected cycles. Therefore, the Class 1 HELB postulations remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

<u>**Affected LRA Section**</u>	<u>**LRA Page No.**</u>	<u>**Affected Paragraph and Sentence**</u>
Table A-1	**Page A-69**	**Commitment No. 23**

In response to RAI 4.3-18, license renewal future commitment No. 23 in LRA Table A-1, "Davis-Besse License Renewal Commitments," is revised to read (Note: this commitment was previously revised by FENOC Letter dated April 15, 2011; ADAMS Accession No. ML11109A083):

Table A-1

Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
23	In association with the TLAA for effects of environmentally assisted fatigue of the high pressure injection (HPI) nozzle safe end including the associated Alloy 82/182 weld (weld that connects the safe end to the nozzle), and cracking of the HPI/makeup nozzle thermal sleeve, FENOC commits to replace the HPI nozzle safe end including the associated Alloy 82/182 weld, and the thermal sleeve for all four HPI nozzles prior to the period of extended operation. ~~*In addition, FENOC commits to evaluate the environmental effects on the replacement HPI nozzle safe end and the weld that connects the safe end to the nozzle in accordance with NUREG/CR 6260 and the guidance of EPRI Technical Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application," Rev. 1. Any nickel-based alloy locations will be evaluated in accordance with NUREG/CR-6909.*~~ <u>*The Fatigue Monitoring Program will evaluate the environmental effects and manage cumulative fatigue damage for the replacement high pressure injection (HPI) nozzle safe ends and associated welds.*</u>	<u>*Prior to*</u> April 22, 2017	LRA <u>*and FENOC Letters L-11-107 and L-11-203*</u>	A.2.7.4 <u>*Responses to NRC RAIs from NRC Letters dated March 17, 2011 and May 2, 2011*</u>

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table A-1	**Page A-69**	**New Commitment No. 41**

In response to RAI B.2.7-1, as modified per telecon with the NRC held on June 7, 2011, a new license renewal future commitment is added to LRA Table A 1, "Davis-Besse License Renewal Commitments," to read:

Table A-1

Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
41	The EDG Fuel Oil Storage Tanks (DB-T153-1 and DB-T153-2) and the in-scope fuel oil and Service Water buried piping will be cathodically protected in accordance with NACE SP0169-2007 or NACE RP0285-2002.	Prior to April 22, 2017	FENOC Letter L-11-203	Response to NRC RAI B.2.7-1 from NRC Letter dated April 20, 2011, as modified per telecon with the NRC held on June 7, 2011

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table A-1	**Page A-69**	**New Commitment No. 42**

In response to RAI XI.S8-1, a new license renewal future commitment is added to LRA Table A-1, "Davis-Besse License Renewal Commitments," to read:

Table A-1

Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
42	*Implement the Nuclear Safety-Related Coatings Program as described in LRA Section B.2.42.*	*Prior to April 22, 2017*	*FENOC Letter L-11-203*	*Response to NRC RAI from NRC Letter dated April 5, 2011*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table B-1	**Page B-16**	**Row XI.S8**

In response to RAI XI.S8-1, the "Corresponding Davis-Besse AMP" column of row XI.S8 8 of Table B-1 is revised to read:

Table B-1
Correlation of NUREG-1801 and Davis-Besse Aging Management Programs
(continued)

Number	NUREG-1801 Program	Corresponding Davis-Besse AMP
XI.S8	Protective Coating Monitoring and Maintenance Program	Not credited for aging management. Davis-Besse does not credit coatings inside the Containment to manage the effects of aging for structures and components or to ensure that the intended functions of coated structures and components are maintained. Therefore, these coatings do not have an intended function and do not require aging management for license renewal. *However, protective coatings inside the containment vessel are managed by the plant-specific Nuclear Safety-Related Coatings Program.* *See Section B.2.42.*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table B-2	**Page B-22**	**New Row**

In response to RAI XI.S8-1, a new row is added to Table B-2 as follows:

Program Name	New / Existing	Consistent with NUREG-1801	Consistent with NUREG-1801 with Exceptions	Plant-Specific	Enhancement Required
Nuclear Safety-Related Coatings Program Section B.2.42	*Existing*	--	--	*Yes*	--

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
B.2.42	**Page B-166**	**New Section**
Appendix B Table of Contents	**Page B-4**	**New listing – B.2.42**

In response to RAI XI.S8-1, LRA Section B.2.42, "Nuclear Safety-Related Coatings Program," is created to include a new plant-specific aging management program. Although not shown below, LRA Appendix B, "Aging Management Programs," "Table of Contents" on LRA Page B-4 is revised accordingly to include the new section.

New LRA Section B.2.42 reads:

B.2.42 Nuclear Safety-Related Coatings Program

Program Description

The Nuclear Safety-Related Protective Coatings Program is an existing plant-specific condition monitoring program that monitors the performance of Service Level 1 coatings inside containment (e.g., coated structures and components such as steel containment vessel, structural steel, supports, penetrations, and concrete walls and floors) through periodic coating examinations, condition assessments and remedial actions, including repair or testing. The Nuclear Safety-Related Protective Coatings Program defines roles, responsibilities, controls and deliverables for monitoring the condition of coatings in containment. Service Level 1 coatings are subject to the guidance of ASTM International (ASTM) D5163-91, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant," and American National Standards Institute (ANSI) Standard N101.4 (1972), "Quality Assurance for Protective Coatings Applied to Nuclear Facilities". The program follows the guidance of EPRI 1003102, "Guidelines on Nuclear Safety Related Coatings," Revision 1. This program also ensures that the Design Basis Accident (DBA) analysis limits with regard to debris loading from failed coatings will not be exceeded for the Emergency Core Cooling Systems (ECCS) suction strainers. On July 14, 1998 the NRC published Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident because of Construction and Protective Coating Deficiencies and Foreign Material in Containment". The program is implemented as described in the FirstEnergy Nuclear Operating Company (FENOC) response to NRC Generic Letter 98-04, accepted by the NRC. The Nuclear Safety-Related Protective Coatings Program provides reasonable assurance that potentially detrimental aging effects will be adequately detected and mitigated such that Service Level 1 protective coatings are *maintained consistent with the current licensing basis for the period of extended operation.*

NUREG-1801 Consistency

The Nuclear Safety-Related Protective Coatings Program is an existing plant specific program for Davis-Besse. While NUREG-1801 includes a Protective Coating Monitoring and Maintenance Program (XI.S8), the Nuclear Safety-Related Protective Coatings Program is considered plant-specific, and is evaluated against the ten elements described in Appendix A.1, Section A.1.2.3 of NUREG-1800, the Standard Review Plan for License Renewal (SRP-LR).

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- *Scope*

  The Nuclear Safety-Related Protective Coatings Program monitors the performance of Service Level 1 coatings inside containment through periodic coating examinations, condition assessments and remedial actions, including repair or testing. The Nuclear Safety-Related Protective Coatings Program ensures that the Design Basis Accident (DBA) analysis limits with regard to coatings will not be exceeded for the ECCS suction strainers per the response to NRC Generic Letter 98-04. The program consists of periodic visual inspections of the Service Level 1 coatings, looking for any visible defects, such as blistering, cracking, flaking, peeling, delamination, rusting and physical damage. The program was established in accordance with the guidance provided in ASTM D5163, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant".

  The qualification testing of Service Level 1 coatings used for new applications or used as maintenance coatings for repair and replacement activities inside containment is addressed in the FENOC revised response to NRC Generic Letter 98-04 for Davis-Besse. The testing meets the applicable requirements contained in Regulatory Guide (RG) 1.54 Rev. 0, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." Although Davis-Besse was not committed to ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," protective coatings have been evaluated to meet the coatings qualification test criteria per ANSI N101.2.

- *Preventive Actions*

  Protective coatings are not credited for aging management at Davis-Besse. The Nuclear Safety-Related Protective Coatings Program is a condition monitoring program that does not include preventive actions. No actions are taken as part of the Nuclear Safety-Related Protective Coatings Program to prevent aging effects or mitigate age-related degradation.

- *Parameters Monitored or Inspected*

  The Nuclear Safety-Related Protective Coatings Program monitors Service Level 1 coatings in accordance with ASTM D5163, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant", ASTM D 714, "Standard Test method for Evaluating Degree of Blistering of Paints" and SSPC VIS-2, "Standard Method of Evaluating Degree of Rusting on Painted Surfaces".

  Parameters monitored or inspected by the Nuclear Safety-Related Protective Coatings Program include any visible defects, such as blistering, cracking, flaking, peeling, delamination, rusting and physical damage.

  The Nuclear Safety-Related Protective Coatings Program procedure will be revised to clarify that visible defects "rusting and physical damage" are inspection attributes following the guidance of ASTM D5163-08, subparagraph 10.2. The Coating Condition Assessment Inspection Form will be revised to list the same set of degradation parameters for inspection as the governing procedure.

- *Detection of Aging Effects*

  A visual containment inspection is performed for evidence of degraded qualified coatings and identification of unqualified coatings applied to structures and components during each refueling outage in accordance with the guidance in ASTM D5163, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant". The containment inspection includes a visual coating inspection of the accessible areas that are listed in the approved procedure along with location plan maps. Unless conditions warrant a closer review, inspectors are not required to examine portions of the area, structures or components that are inaccessible due to insulation, scaffold or permanent plant SSCs. Conditions that warrant a closer review are evidence of a coating failure where the area of concern is hidden from view by the obstruction. For areas of the Containment Vessel which have visual evidence (identifiable boundary) of repair or touch-up; its location (azimuth and elevation), approximate surface area and average dry film thickness are documented on *the Coating Condition Assessment Inspection Form. Instruments and equipment used for inspection; such as flashlight, acuity card, inspection mirror, camera, telescope, video equipment, magnifying glass, measuring tape, dry film thickness gage, spring micrometer, etc. meet the guidelines of ASTM D5163-08, subparagraph 10.5.*

Coating inspections are performed by coatings inspectors qualified per the requirements of Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel," and ANSI N45.2.6, "Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants". The nuclear safety-related coatings program owner and coating surveillance personnel meet the requirements of EPRI 1003102 Revision 1, "Guidelines on Nuclear Safety Related Coatings".

The Nuclear Safety-Related Protective Coatings Program procedure will be revised to specify the qualifications for inspection personnel, the inspection coordinator and the inspection results evaluator following the guidance of ASTM D5163-08, paragraph 9.

- *Monitoring and Trending*

  The Nuclear Safety-Related Protective Coatings Program incorporates guidance from ASTM D5163, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant". The Nuclear Safety-Related Coatings Program owner develops and manages the Nuclear Safety-Related Protective Coatings Program. The Nuclear Safety-Related Coatings Program owner also maintains the Non-DBA Qualified Protective Coatings Inventory. Inspection results are reviewed and identified degradations are evaluated in accordance with the FENOC Corrective Action Program. Degraded coating that is left in place in an area is documented on the Coating Condition Assessment Inspection form and evaluated by the program owner.

  The Nuclear Safety-Related Protective Coatings Program procedure will be revised to include prioritization of repair areas as either needing repair during the same outage or as postponed to future outages, but under surveillance in the interim period, following the guidance of ASTM D5163-08, subparagraph 11.1.2.

- *Acceptance Criteria*

  The Nuclear Safety-Related Protective Coatings Program characterizes, documents, and tests defective or deficient coatings in accordance with ASTM D5163, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power

Plant". As applicable, coated surfaces are characterized as exhibiting blisters, cracking, flaking, peeling, delamination, abrasion, and holidays. Coating tests are employed for areas where the qualification is in question, representative dry film thickness is obtained for each area on a structure or component which has coating degradation. Evidence of corrosion is further categorized per the guidance of a standard method for evaluating degree of rusting on painted surfaces.

The Coating Surveillance Personnel inspect the containment according to the following degradation definitions:

- *Abrasion – The wearing away of coating material in small shreds as a result of friction.*
- *Blistering - The formation of bubbles in a cured, or nearly cured, coating film after exposure, generally in an aqueous environment.*
- *Cracking - The formation of breaks in a coating film that extend through to the underlying surface.*
- *Delamination - A separation of one coat from another coat within a coating system; or from the substrate.*
- *Flaking - The detachment of small pieces of the coating film.*
- *Holiday - Pinhole, skip, discontinuity, or void in a coating film that exposes the substrate.*
- *Peeling - The separation of one or more coats or layers of a coating system from the substrate.*

Acceptable coatings are free of delamination, blistering, peeling, flaking, cracking and other defects. Coatings not found to be acceptable are documented using the FENOC Corrective Action Program. The protective coating condition assessment and associated Coating Condition Assessment Inspection forms are approved and signed by the Protective Coatings Program Owner or his Designee.

The Nuclear Safety-Related Protective Coatings Program procedure will be revised to improve reporting requirements by following the guidance of ASTM D5163-08, paragraph 11, including a summary report of findings and recommendations for future surveillance or repair, and prioritization of repairs.

- *Corrective Actions*

  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- *Confirmation Process*

  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- *Administrative Controls*

  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- *Operating Experience*

  A review of operating experience indicates that the Nuclear Safety-Related Protective Coatings Program has been effective in monitoring coatings inside containment by identifying degraded conditions, performing evaluations and performing corrective actions ensuring that the DBA analysis limits for debris loading will not be exceeded for the ECCS suction strainers.

  Industry operating experience is documented in NRC Regulatory Guide 1.54 and several NRC Generic Communications including Information Notice 97-13, Generic Letter 98-04, Bulletin 2003-01 and Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors."

  The industry experience cited in these publications deals principally with debris that could block emergency recirculation during a design basis accident.

  In 2003 Davis-Besse provided a revised response to NRC Generic Letter 98-04. During the Cycle 13 refueling outage, FENOC identified via the Corrective Action Program that significant amounts of unqualified coating materials were applied to components inside the containment vessel. FENOC informed the NRC by letter dated September 15, 2003 that incomplete or inaccurate information was provided in the original 1998 Davis-Besse response to Generic Letter 98-04. This issue led to reporting that the containment emergency sump could be significantly challenged by the quantity of failed coating material and other debris present in the Containment *after a postulated Loss of Coolant Accident (LOCA) under Davis-Besse Licensee Event Report (LER) 2002-005. Corrective actions taken for this event were:*

- *The old Containment Emergency Sump Strainer was removed and a new strainer with greater surface area was installed.*
- *Unqualified coatings have been removed from major equipment in Containment and replaced with qualified coatings.*
- *A Nuclear Safety-Related Coatings Program has been developed for coating material controls and application to structures and components located within the Containment.*
- *Where possible, fibrous insulation was removed from Containment. The fibrous insulation and unqualified coatings left in the Containment have been identified and evaluated (in conjunction with other potential debris) for effect on the Emergency Core Cooling System and Containment Spray System. Controls have been established for potential debris sources to ensure requirements are met.*
- *Evaluations were performed in conjunction with the modifications implemented on the containment emergency sump, which examined the Low Pressure Injection System, the High Pressure Injection System, the Containment Spray System, and the Boron Precipitation Control System.*
- *Modifications were implemented for the High Pressure Injection System Pumps.*

In 2004, the NRC concluded that information regarding the reason for the violation based on the FENOC November 11, 1998 response to Generic Letter 98-04, the corrective actions taken, plans to correct the violation and prevent recurrence, and the date when full compliance was achieved, were adequately addressed on the Davis-Besse docket in NRC Inspection Report 50-346/03-19, LERs 2003-002 and 2002-005, and FENOC letters dated February 27, 2004 (ML040620456), November 26, 2003 (ML033370836) and October 24, 2003 (ML040890175). In summary, Davis-Besse had met the requirements of NRC Generic Letter 98-04 and had committed to maintain the Nuclear Safety-Related Protective Coatings Program for coating material controls and coating application to structures and components located within the Containment.

In 2006, the Nuclear Safety-Related Protective Coatings Program documented inspection findings in the Corrective Action Program for the Cycle 14 refueling outage. Inspection findings were:

- *Epoxy topcoat cracking and peeling areas observed on several embedded plates on east and north surfaces of the east Once-Through Steam Generator (OTSG) enclosure (D-ring) walls. Approximately 50 square feet of coating material was cracked or peeling. The coating was applied during initial plant construction.*

- *Upper edge of the west D-ring at edge for the missile shield support shelf had approximately one square foot of peeled coating. The baseplate for a pipe whip restraint located on the east D-ring had approximately two square feet of peeled material.*

- *Approximately one square foot of degraded material was observed on an embedded plate (approximate elevation 625'-0") for west staircase restraint and on two pipe restraint baseplates at an elevation of approximately 650'-0".*

Corrective actions taken were to add the quantity of failed protective coating material to the Non-DBA Qualified Coating Inventory and to plan removal and rework of the failed coating material.

In 2008, NRC Integrated Inspection Report 05000346/2008-03 described the implementation of the Davis-Besse actions documented in the February 28, 2008 response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors." The Davis-Besse resolution of Generic Letter 2004-02 included the installation of a significantly larger strainer within containment. The debris source term was also significantly reduced through removal of nearly all fibrous insulation and completely stripping and recoating the containment dome. Detailed analyses that used bounding limits for debris generation, transport and head loss effect were performed using the NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," and associated NRC Safety Evaluation Report (SER) methods, with permitted deviations. The NRC inspectors reviewed the engineering change packages (ECPs) associated with modifications installed, procedure changes and programmatic controls implemented, and changes for the Updated Safety Analysis Report (USAR) in response to Generic Letter 2004-02. No findings of significance were identified.

In 2008, the Nuclear Safety-Related Protective Coatings Program documented inspection findings in the Corrective Action Program for the

Cycle 15 refueling outage. General coating conditions in Containment remained acceptable. Inspection findings were:

- *Blistering of containment dome coating material in two locations. The degraded material was quantified and added to the Non-DBA Qualified Protective Coatings Inventory.*

- *Peeling of containment vessel top coat material behind the polar crane access ladder between elevations 714' to 722'. The degraded coating material was removed.*

- *Rusting of containment penetrations P3, P4, P5, P6, P7, P9, P10 and P11 was identified and evaluated.*

- *Peeling of epoxy top coat on bottom of northeast, upper OTSG 1-1 support.*

- *Flaking paint on a hot leg platform brace adjacent to the OTSG was quantified and added to the Non-DBA Qualified Protective Coatings Inventory.*

- *Peeled top coat material was found on a lower snubber mounting for OTSG 1-2.*

Several areas of degradation which were noted during this outage had previously been identified and are to be reworked. The degraded material in these areas has been included in the Non-DBA Qualified Protective Coatings Inventory.

In 2011, the Nuclear Safety-Related Protective Coatings Program documented inspection findings in the Corrective Action Program for the Cycle 16 refueling outage. General coating conditions in Containment remained good. Inspection findings were:

- *Blistering of containment vessel coating material in two locations adjacent to the polar crane access ladder at approximately the 660' elevation. The degraded material has been removed.*

- *Peeling coating material on structural steel for the elevation 610'-0" hot leg platform. The degraded material has been removed.*

- *Rusting of containment penetrations identified and previously evaluated. Rework of these penetrations is currently planned to be performed per order during the Cycle 18 refueling outage.*

- *Peeling of epoxy top coat on bottom of northeast, upper OTSG 1-1 support. The degraded material was quantified and added to the Non-DBA Qualified Protective Coatings Inventory.*
- *Flaking paint on hot leg platform brace adjacent to the OTSG. The degraded material was quantified and added to the Non-DBA Qualified Protective Coatings Inventory.*
- *Peeled top coat material was found on a lower snubber mounting for OTSG 1-2. This was quantified and added to the Non-DBA Qualified Protective Coatings Inventory.*

Several areas of degradation which were noted during this outage had been identified previously and are currently planned to be reworked during the Cycle 18 ueling outage. The degraded material in those areas has been included in the Non-DBA Qualified Protective Coatings Inventory.

Enhancements:

None.

Conclusion

The Nuclear Safety-Related Protective Coatings Program is an existing program that has been demonstrated to be capable of monitoring the performance of coatings inside containment. Proper maintenance of protective coatings has ensured that the quantities of unqualified and degraded qualified coatings inside containment are maintained below the acceptance limits. The continued implementation of the Nuclear Safety-Related Protective Coatings Program provides reasonable assurance that the effects of aging will be managed such that the Service Level 1 protective coatings and other coatings in containment are maintained consistent with the current licensing basis for the period of extended operation.

Enclosure B

Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS)

Letter L-11-203

<u>AREVA Report No. 51-9157140-001</u>
<u>"DB-1 Design CUFs and NUREG/CR-6260 Screening for License Renewal,"</u>
<u>dated 6/10/2011</u>

23 pages
(not including this cover sheet)

AREVA

AREVA NP Inc.

Engineering Information Record

Document No.: 51 - 9157140 - 001

DB-1 Design CUFs and NUREG/CR-6260 Screening for License Renewal

AREVA NP INC. NON-PROPRIETARY

Safety Related? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title/Discipline	Signature	P/LP, R/LR, A/A-CRF, A/A-CRI	Date	Pages/Sections Prepared/Reviewed/ Approved or Comments
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Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	All	Original release
001	Page 1 through 3, page 23 Page 1 Entire document Tables 3-1, 3-2, 3-4, 3-5, and 3-9	Updated title page, signature block on pages 2 and 23, Record of Revision Changed 'PROPRIETARY' to 'NON- PROPRIETARY' and deleted 'PROPRIETARY' throughout the document Changed 'NUREG-6260' to 'NUREG/CR-6260' throughout the document Editorial changes per customer comments

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List of Tables

1.0 PURPOSE

The purpose of this document is to screen the RCS pressure boundary items (B&W scope of supply and attached piping within ASME Section XI inspection boundary) for locations where the EAF CUF exceeds 1.0 by using bounding F_{en}s.

2.0 METHODOLOGY

Summary tables of all Class 1 RCS locations with EAF CUF values will be provided. Locations with EAF CUF > 1.0 that are not evaluated as NUREG/CR-6260 locations will be identified. Locations that were evaluated as NUREG/CR-6260 locations will also be provided in a separate table.

Using the environmentally assisted fatigue correction factor, F_{en}, the EAF CUF values are determined as

EAF CUF = Design CUF × F_{en}

The Design CUFs were determined based on the design number of cycles.

The following bounding F_{en} values were calculated based the methods outlined in References 1 through 4,

1.74 for carbon steel (CS)

2.45 for low-alloy steel (LAS)

15.35 for stainless steel with Temperature $T \geq 200°C$ and 2.55 for stainless steel with $T < 200°C$ (SS)

4.52 for nickel-based alloy incore instrument nozzle (NBA)

3.0 SUMMARY OF EAF CUF VALUES

The EAF CUFs (= Design CUF × F_{en}) for all applicable locations are provided in Tables 3-1 through 3-7. The Design CUFs contained in the relevant references are identified in the tables. A summary of locations with EAF CUF > 1.0 is provided in Table 3-8.

Note that some of the values are "exempt" because the locations did not require fatigue analysis in accordance with ASME Code (Paragraph N-415 of Section III of the 1965 Edition, paragraph NB-3222.4 of Section III of the 2007 Edition).

The locations marked with "*" in Tables 3-1 through 3-7 indicate these locations were evaluated as NUREG/CR-6260 locations. These locations are summarized in Table 3-9.

In the following tables:

Table 3-1 EAF CUF Values for Reactor Vessel

Table 3-2 EAF CUF Values for Control Rod Drive Housings

Table 3-3 EAF CUF Values for Reactor Coolant Pump

Table 3-4 EAF CUF Values for Pressurizer

Table 3-5 EAF CUF Values for SG on Primary Side

Table 3-6 EAF CUF Values for SG on Secondary Side

Table 3-7 EAF CUF Values for Reactor Coolant Piping

Table 3-8 Summary of RCS Pressure Boundary Locations with EAF CUF Greater Than 1.0

Table 3-9 EAF Values for NUREG/CR-6260 Locations

Table 3-1 EAF CUF Values for Reactor Vessel and RV Internals Replacement Bolts

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
RV Closure	Head in Shell region	LAS/2.45	0.128	0.314	5 and 6
RV Closure	Vessel in Shell region	LAS/2.45	0.250	0.613	5 and 6
RV Closure	Studs[1]	LAS/2.45[1]	0.726	NA[1]	5 and 6
CRDM Housing	CRDM flanges	SS/15.35	0.169	2.595	5 and 6
CRDM Housing	Vent line flange	SS/15.35	0.127	1.950	5 and 6
CRDM Housing	Blind flanges	SS/15.35	0.000	0.000	5 and 6
CRDM Housing	Motor tube flanges	SS/15.35	0.000	0.000	5 and 6
CRDM Housing	Bolts	LAS/2.45	0.174	0.426	5 and 6
Inlet Nozzle	Nozzle, nozzle to pipe weld, nozzle to shell weld*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
Outlet Nozzle	Nozzle, nozzle to pipe weld, nozzle to shell weld*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
Core Flood Nozzle	Nozzle to safe end	SS/15.35	0.064	0.982	5 and 6
Core Flood Nozzle	Nozzle*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
CF Nozzle Venturi Sleeve	Cylindrical portion	SS/15.35	0.956	14.675	5 and 6
CF Nozzle Venturi Sleeve	weld juncture	SS/15.35	<0.956	14.675	5 and 6
Instrument Nozzles	Nozzle, nozzle to safe end weld*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
Service Structure	Service Structure to head juncture	CS/1.74	0.125	0.218	5 and 6
RV Shell	Nozzle Belt	LAS/2.45	0.024	0.0588	5 and 6
RV Shell	Lower Head*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
Internal Replacement Bolt[2]	Lower core barrel bolts	NBA/4.52	0.46	2.079	7, Appendix C
Internal Replacement Bolt[2]	Upper bore barrel bolts	NBA/4.52	0.0	0.0	7, Appendix C
Internal Replacement Bolt[2]	Lower thermal shield bolts (LTS)	NBA/4.52	0.57	2.576	8
Internal Replacement Bolt[2]	Surveillance Specimen Holder Tube (SSHT)	NBA/4.52	0.02	0.090	9

[1] RV closure studs are not exposed to coolant, therefore not subject to F_{en}.

[2] Not RCS Pressure Boundary

Table 3-2 EAF CUF Values for Control Rod Drive Housings

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
Motor Tube Cap	Motor Tube Cap	SS	Exempt	NA	10 and 11
Motor Tube Extension	Motor Tube Extension	SS	Exempt	NA	10 and 11
Motor Tube Center Section	Motor Tube Center Section	SS	Exempt	NA	10 and 11
Motor Tube Base	Motor Tube Base	SS	Exempt	NA	10 and 11
Motor Tube Base	Lower Flange	SS/15.35	0.00	0.00	10 and 11
High Strength Bolts	Motor Tube Holddown bolts	LAS/2.45	0.13	0.319	10 and 11
Welds	Base to Center Section	--	Exempt	NA	10 and 11
Welds	Center Section to Extension	--	Exempt	NA	10 and 11

Table 3-3 EAF CUF Values for Reactor Coolant Pump

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
Pump Casing	Diffuser Vanes	A-351 Gr. CF8M	Exempt	NA	12 and 13
Pump Casing	Volute/Lower flange	A-351 Gr. CF8M	Exempt	NA	12 and 13
Pump Casing	Upper flange	A-351 Gr. CF8M	Exempt	NA	12 and 13
Pump Casing	Suction Nozzle	A-351 Gr. CF8M	Exempt	NA	12 and 13
Pump Casing	Crotch	A-351 Gr. CF8M	Exempt	NA	12 and 13
Pump Casing	Closure Stud	A-540 Gr. B23 Cl4	Exempt	NA	12 and 13
Pump Cover	Cooling Hole Ligament	A-351 Gr. CF8M SS/15.35	0.56[1]	8.596	12 and 13
Pump Cover	Bearing cavity	A-351 Gr. CF8M SS/15.35	0.964	14.797	12 and 13
Pump Cover	Cap screw seal flange to over	A-540 Gr. B23 Cl5	Exempt	NA	12 and 13
Driver Mount	Lower flange	SA-213 Gr. WCB	Exempt	NA	12 and 13
N9000 Seal Cartridge	Seal Flange	SA-182 Gr. F304	Exempt	NA	12 and 13
N9000 Seal Cartridge	Upper pressure breakdown device	A-351 Gr. CF8M	Exempt	NA	12 and 13
N9000 Seal Cartridge	Upper PBD cap screw	SA-193 Gr. B8	Exempt	NA	12 and 13
Heat Exchanger	Inner coil	SB-167	Exempt	NA	12 and 13
Heat Exchanger	Inner-to-outer coil weld	--	Exempt	NA	12 and 13
Heat Exchanger	Outer coil	SB-407	Exempt	NA	12 and 13

[1] Calculated with an exception to the ASME rules. See the fatigue summary report (Ref. 12) for further details.

Table 3-4 EAF CUF Values for Pressurizer

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
Spray Nozzle	Nozzle, nozzle/head juncture	CS/1.74	0.01	0.0174	14 and 15
Spray Nozzle	Safe end	NBA/4.52	0.01	0.0452	14 and 15
Spray Nozzle	Weld overlay	NBA/4.52	0.01	0.0452	14 and 15
Spray Nozzle	Internal Pipe	SS/15.35	0.33	5.0655	14 and 15
Surge Nozzle	Nozzle, safe end, Safe end-to-elbow weld, weld overlay*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
Pressurizer Support Lugs	Support	CS/1.74	0.01	0.0174	14 and 15
Pressurizer Support Lugs	Shell	CS/1.74	0.01	0.0174	14 and 15
Heater Bundle Closure	Cover plate	LAS/2.45	0.06	0.147	14 and 15
Heater Bundle Closure	Diaphragm plate	SS/15.35	0.60	9.210	14 and 15
Heater Bundle Closure	Seal weld	SS/15.35	0.86	13.201	14 and 15
Heater Bundle Closure	Studs	LAS/2.45	0.25	0.613	14 and 15
Shell	Heater belt transition	CS/1.74	0.13	0.226	14 and 15
Manway closure	Shuds	LAS/2.45	0.35	0.858	14 and 15
3" PZR relief nozzle	Nozzle	CS/1.74	0.04	0.070	14 and 15
3" PZR relief nozzle	Safe	SS/15.35	0.04 max	0.614 max	14 and 15
3" PZR relief nozzle	Upper head	CS/1.74	0.04 max	0.070 max	14 and 15
3" PZR relief nozzle	Weld overlay	NBA/4.52	0.04 max	0.181 max	14 and 15
Other Openings	Vent nozzle	NBA	Exempt	NA	14 and 15
Other Openings	Level Sensing Nozzles (Upper)	NBA	Exempt	NA	14 and 15
Other Openings	Level Sensing Nozzles (Lower)	NBA	Exempt	NA	14 and 15
Other Openings	Level Sensing Nozzles (Lower) Opening	NBA/4.52	0.166	0.750	14 and 15
Other Openings	Thermowell	NBA	Exempt	NA	14 and 15
Other Openings	Thermowell Opening	NBA/4.52	0.166	0.750	14 and 15

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
Other Openings	Sampling Nozzle	NBA	Exempt	NA	14 and 15
Other Openings	Sampling Nozzle Opening	NBA/4.52	0.166	0.750	14 and 15
Other Openings	Manway	CS	Exempt	NA	14 and 15

Table 3-5 EAF CUF Values for SG on Primary Side

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
Shell	Shell	CS/1.74	0.43	0.748	16 and 17
Tubesheet/Shell Interface	Tubesheet	LAS/2.45	0.13	0.319	16 and 17
Tubesheet/Shell Interface	Shell	LAS/2.45	0.11	0.270	16 and 17
Primary Outlet Nozzles	Nozzle	CS/1.74	0.69	1.201	16 and 17
Primary Outlet Nozzles	Nozzle to Shell (lower head) Weld	LAS/2.45	0.12	0.294	16 and 17
Primary Inlet Nozzle	Nozzle to Pipe Weld	CS/1.74	0.17	0.296	16 and 17
Primary Inlet Nozzle	Nozzle to Shell (upper head) Weld	LAS/2.45	0.91	2.230	16 and 17
Ventline Closure	Studs	SA-453 Gr.660	Exempt	NA	16 and 17
Ventline Closure	Nozzle	SA-182 F3165	Exempt	NA	16 and 17
Ventline Closure	Head	SA-533 Gr. B	Exempt	NA	16 and 17
3/8" Dia. Tube Stabilizer	Tube/Stab Weld	NBA/4.52	0.12	0.542	16 and 17
3/8" Dia. Tube Stabilizer	Nail	NBA/4.52	0.07	0.316	16 and 17
1/2" Dia. Tube Stabilizer	Weld Cap	NBA/4.52	0.02	0.090	16 and 17
1/2" Dia. Tube Stabilizer	Stabilizer	NBA/4.52	0.00	0.00	16 and 17
Explosive Tube Plug	Tube	LAS/2.45	0.00	0.00	16 and 17
Explosive Tube Plug	Plug	SB-166	Exempt[1]	NA	16 and 17
Welded U-cup Plug	Plug End	SB-166	Exempt[1]	NA	16 and 17
Welded U-cup Plug	Plug	SB-166	Exempt[1]	NA	16 and 17
Welded U-cup Plug	Plug-to-tube Weld	SB-166	Exempt[1]	NA	16 and 17
Rolled Tube Plug	Plug	SB-166	Exempt[1]	NA	16 and 17
Sleeve Plugs (Large & Small)	Plug	SB-166	Exempt[1]	NA	16 and 17

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
Welded Tube Plug	Plug	NBA/4.52	0.03[1]	0.136	16 and 17
Remote Tube Plug	Plug	NBA/4.52	1.0[1]	4.52	18 through 20

[1] Welded plugs are considered exempt from fatigue analysis per AREVA stress report (Ref. 21) except for the Welded Tube Plug. However Remote welded plugs are not exempt and have a limited (33 cycle) design life (Refs. 18 thr. 20).

Table 3-6 EAF CUF Values for SG on Secondary Side

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
SG Supports	Support Skirt[1]	LAS/2.45	1.0	NA[1]	16 and 17
Steam Outlet Nozzles	Nozzle	CS/1.74	0	0.0	16 and 17
Main Feedwater Nozzles	Nozzle	CS/1.74	0.4	NA[1]	16 and 17
Main Feedwater Nozzles	Nozzle-to-shell	SA-516 Gr 70	Exempt	NA	16 and 17
Main Feedwater Nozzles	Nozzle fillet weld	CS/1.74	0	0	16 and 17
6" Aux Feedwater Nozzle[2]	Nozzle	CS/1.74	0.552[2]	NA[1]	16 and 17
3" Aux Feedwater Nozzle[2]	Studs	LAS/2.45	1.0[2]	NA[1]	16 and 17

[1] These locations are not exposed to primary coolant, therefore not subject to F_{en}

[2] The 6" auxiliary feedwater nozzles are no longer in service. They were replaced with 3" auxiliary feedwater nozzles. The 3" nozzles are limited to 1447 cycles of auxiliary feedwater initiation.

Table 3-7 EAF CUF Values for Reactor Coolant Piping

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
Hot leg piping	RV Outlet (Node 111)	CS/1.74	0.7655	1.332	22 and 23
Hot leg piping	Surge line area (Node 71)	CS/1.74	0.59	1.027	22 and 23
Hot leg piping	Steam Generator connection (Node 139)	CS/1.74	0.4346	0.756	22 and 23
Cold leg, RCP to Rx	Elbow (Node 56)	CS/1.74	0.0503	0.088	22 and 23
Cold leg, RCP to Rx	RCP discharge (Node 69)	CS/1.74	0.043	0.075	22 and 23
Cold leg, RCP to Rx	RCP discharge (Node 143)	CS/1.74	0.043	0.075	22 and 23
Cold leg, SG - RCP	RCP Suction (Node 134)	CS/1.74	0.0783	0.136	22 and 23
Cold leg, SG - RCP	RCP Suction (Node 105)	CS/1.74	0.0778	0.135	22 and 23
Cold leg, SG - RCP	SG discharge (Node 43)	CS/1.74	0.0285	0.050	22 and 23
Surge line	Pipe, elbow including weld*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
Spray line	Spray valve outlet (Node 73)	SS/15.35	0.5184	7.957	22 and 23
Spray line	Spray valve inlet (Node 74)	SS/15.35	0.0951	1.460	22 and 23
Spray line	Near Pressurizer (Node 80)	SS/15.35	0.4124	6.330	22 and 23
Spray line	Cold leg nozzle (Node 1)	SS/15.35	0.0395	0.606	22 and 23
Decay heat nozzle	Node 136	CS/1.74	0.5269	0.917	22 and 23
Decay heat nozzles	nozzle end	CS/1.74	0.89	1.549	22 and 23
Decay heat system piping	Pipe, elbow, weld*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
HPI/MU nozzle	Nozzle, safe end, including weld*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
Hot leg surge nozzle	Nozzle to surge line weld, nozzle taper, and weld overlay*	NUREG/CR-6260 Location. See Table 3-9 for EAF CUF value			
RC drain line	Nozzle to drain line weld	NBA/4.52	0.619	2.798	24
RC letdown line	A sockolet in 2.5" pipe	SS/15.35 (Ref. 25 pg 12)	0.604	9.271	25 (pg 92)
HPI lines	Point 22, 2.5" elbow to elbolet connection	SS/15.35 (Ref. 26 pg 15)	0.981	15.058	26 (pg53)
Core flooding line	Point 27	SS/15.35	0.582	8.934	29 (pg35)

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
Pressurizer safety/relief valve line	Point 180 in 3" pipe stanchion	SS/15.35 (Ref. 30 pg 59)	0.445	6.831	30 (pg92)
RCS Loop 1 cold leg drain line weld overlay repair	Elbow end	SS/15.35	0.839	12.879	31 (Table 23)
RC vent pipe	Reactor head to hot leg vent line	SS/15.35	0.104	1.596	34 (page 24 of 709)

Table 3-8 Summary of RCS Pressure Boundary Locations with EAF CUF Greater Than 1.0

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF = $F_{en} \times$ Des. CUF	Reference
RV Closure	Studs[1]	LAS/2.45	0.726	NA[1]	5 and 6
CRDM Housing	CRDM flanges	SS/15.35	0.169	2.595	5 and 6
CRDM Housing	Vent line flange	SS/15.35	0.127	1.950	5 and 6
CF Nozzle Venturi Sleeve	Cylindrical portion	SS/15.35	0.956	14.675	5 and 6
CF Nozzle Venturi Sleeve	weld juncture	SS/15.35	<0.956	14.675	5 and 6
RV Internal Replacement Bolt	Lower thermal shield bots	NBA/4.52	0.57	2.576	8
Pump Cover	Cooling Hole Ligament	SS/15.35	0.56	8.596	12 and 13
Pump Cover	Bearing cavity	SS/15.35	0.964	14.797	12 and 13
Spray Nozzle	Internal Pipe	SS/15.35	0.33	5.0655	14 and 15
Pressurizer Heater Bundle Closure	Diaphragm plate	SS/15.35	0.6	9.210	14 and 15
Pressurizer Heater Bundle Closure	Seal weld	SS/15.35	0.86	13.201	14 and 15
SG Primary Outlet Nozzles	Nozzle	CS/1.74	0.69	1.201	16 and 17
SG Primary Inlet Nozzle	Nozzle to Shell (upper head) Weld	LAS/2.45	0.91	2.230	16 and 17
SG Primary Remote Tube Plug	Plug	NBA/4.52	1.0	4.52	18 through 20
SG Supports	Support Skirt[1]	LAS/2.45	1.0	NA[1]	16 and 17
3" Aux Feedwater Nozzle	Studs[1]	LAS/2.45	1.0	NA[1]	16 and 17
Hot leg piping	RV Outlet (Node 111)	CS/1.74	0.7655	1.332	22 and 23
Hot leg piping	Surge line area (Node 71)	CS/1.74	0.59	1.027	22 and 23
Spray line	Spray valve outlet (Node 73)	SS/15.35	0.5184	7.957	22 and 23
Spray line	Spray valve inlet (Node 74)	SS/15.35	0.0951	1.460	22 and 23
Spray line	Near Pressurizer (Node 80)	SS/15.35	0.4124	6.330	22 and 23
Decay heat nozzles	nozzle end	CS/1.74	0.89	1.549	22 and 23

[1] These locations are not exposed to primary coolant, therefore not subject to F_{en}.

Assembly	Location	Material/Bounding F_{en}	Design CUF	EAF CUF $= F_{en} \times$ Des. CUF	Reference
RC drain line	Nozzle to drain line weld	NBA/4.52	0.619	2.798	24
RC letdown line	A sockolet in 2.5" pipe	SS/15.35 (Ref. 25 pg 12)	0.604	9.271	25 (pg 92)
HPI lines	Point 22, 2.5" elbow to elbolet connection	SS/15.35 (Ref. 26 pg 15)	0.981	15.058	26 (pg53)
Core flooding line	Point 27	SS/15.35	0.582	8.934	29 (pg35)
Pressurizer safety/relief valve line	Point 180 in 3" pipe stanchion	SS/15.35 (Ref. 30 pg 59)	0.445	6.831	30 (pg92)
RCS Loop 1 cold leg drain line weld overlay repair	Elbow end	SS/15.35	0.839	12.879	31 (Table 23)
RC vent pipe	Reactor head to hot leg vent line	SS/15.35	0.104	1.596	34 (page 24 of 709)

Table 3-9 EAF Values for NUREG/CR-6260 Locations

(Table 4.3-2 from 51-9135330-000 (Ref. 32))

NUREG/CR-6260 generic locations	Davis-Besse plant-specific locations	Material type	Design CUFs	Adjusted CUFs	F_{en}	U_{en}
1 Reactor vessel shell and lower head	Vessel shell and lower head	LAS	0.024	NA[8]	2.45	0.059
	Incore instrument nozzle	NBA	0.770	0.206[5]	4.16	0.857
2 Reactor vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.829	0.146[1]	2.45	0.358
	Reactor vessel outlet nozzle	LAS	0.768	0.335[1]	2.45	0.821
3 Pressurizer surge line	Hot leg surge nozzle inside radius	CS	0.445	NA[8]	1.74	0.774
	Piping adjacent to outboard end of hot leg surge nozzle	SS	0.179	0.07[2]	5.83	0.387
	Piping elbows	SS	0.643	0.239[2]	4.17	0.996
	Piping straights	SS	0.764	0.336[2]	2.52	0.846
	Piping to pressurizer surge nozzle safe end weld	SS	0.51	0.073[2]	8.84	0.644
	Pressurizer surge nozzle inside radius	CS	0.182	NA[8]	1.74	0.317
	Pressurizer surge nozzle safe end	SS	0.108	0.058[1]	15.35	0.892
4 HPI/Makeup nozzle	HPI/Makeup nozzle	CS	0.589	0.348[3]	1.74	0.606
	HPI/Makeup nozzle safe end	SS	0.664	0.550[4]	8.03[6]	4.417[7]
5 Reactor vessel core flood nozzle	Nozzle	LAS	0.0504	NA[8]	2.45	0.123
6 Decay heat Class 1 piping	Decay heat to core flood tee	SS	0.233	NA[8]	2.55	0.595

1. Adjusted CUF obtained by identifying incremental fatigue contribution attributed to the full NSSS design transient cycles for design CUF and reducing those incremental contributions based on the 60-year cycle projections.
2. Adjusted CUF obtained by dividing U_{en} by global F_{en}. Global F_{en} calculated using method from Section 4.2 of MRP-47, Revision 1 (Ref. 33) as described above for the pressurizer surge line.
3. Design CUF reduced from 0.589 to 0.348 by removing conservatisms in the original calculation. Full set of design cycles were used for the calculation.
4. Design CUF reduced from 0.664 to 0.550 by removing conservatisms in the original calculation. Full set of design cycles were used for the calculation.
5. Adjusted CUF obtained by applying the alternating stresses from the original design calculation to the new in-air design curve in NUREG/CR-6909 (Ref. 4) for stainless steel.
6. This is a global F_{en} obtained by dividing U_{en} by the CUF (4.417/ 0.550).
7. 4.417 is >1.0 and is unacceptable for the period of extended operation.
8. Adjusted CUF was not required. Design CUF multiplied by F_{en} resulted in an U_{en} of < 1.0.

4.0 REFERENCES

1. AREVA NP Document 32-9118118-000, "Davis Besse 1 Surge Line Environmentally-Assisted Fatigue Analysis for License Renewal," April 2010

2. AREVA NP Document 32-9119659-001, "DB-1 Hot Leg Surge Nozzle and Pressurizer Surge Nozzle EAF Analysis for License Renewal," June 2010

3. AREVA NP Document 32-9122270-000, "DB-1 High Pressure Injection and Makeup Nozzle EAF Analysis for License Renewal," December 2009

4. NUREG/CR-6909 "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," February 2007

5. AREVA Stress Report Summary, 33-1201205-09, "Stress Report Summary for Reactor Vessel," 26 September 2008

6. *FirstEnergy Calculation C-ME-099.20-003, Rev. 3, "Reactor Vessel Stress Report Summary," 09 April 2009

7. BWNT calculation 32-1219919-02, "BWOG RV Internals Eval," 3 January 1994

8. AREVA calculation 32-114659-01, "Lower Thermal Shield Bolt Stress Analysis," June 1984

9. AREVA calculation 32-1148284-00, "SSHT Inconel Bolt Stress Analysis," July 1984

10. AREVA Stress Report Summary, 33-1240705-04, "Stress Report Summary for Type "C" Control Rod Drive Mechanism," 26 September 2008

11. *FirstEnergy Calculation C-ME-099.20-005, Rev. 2, "Stress Report for Control Rod Drive System," 09 April 2009

12. AREVA Stress Report Summary, 33-1201200-03, "Stress Report Summary for Reactor Coolant Pumps," 26 September 2008

13. *FirstEnergy Calculation C-ME-099.20-006, Rev. 1, "Stress Report Summary for Reactor Coolant Pumps," 09 April 2009

14. AREVA Stress Report Summary, 33-1201210-07, "Stress Report Summary for Pressurizer," 26 September 2008

15. *FirstEnergy Calculation C-ME-099.20-001, Rev. 1, "Pressurizer Stress Report Summary," 09 April 2009

16. AREVA Stress Report Summary, 33-1201163-04, "Stress Report Summary for Once Through Steam Generator," 26 September 2008

17. *FirstEnergy Calculation C-ME-099.20-002, Rev. 2, "Stress Report Summary for Once Through Steam Generator," 09 April 2009

18. Condition Report 02-00275, "Non Compliant Welded SG Tube Plugs," January 28, 2002

19. SAP Notification 600262596, "Update Calculation C-ME-099.20-02 with stress intensities and cumulative usage factors for OTSG rolled plugs"

20. Condition Report 6013113, Rev. 0, "OTSG Remote Welded Plugs and Manually Welded Tapered Welded Plugs," January 30. 2002

21. AREVA Calculation 33-1177631-00, "ASME B&PV Code Calculations for OTSG Rolled Plug," 8 Feb 1990

22. AREVA Stress Report Summary, 33-1201199-08, "Stress Report Summary for Reactor Coolant System Primary Piping," 26 September 2008

23. *FirstEnergy Calculation C-ME-099.20-004, Rev. 1, "RCS Piping Stress Summary Report," 09 April 2009

24. *Teledyne Technical Report TR-3831-10, "ASME Section III Analysis, Class 1, Based on I.E. Bulletin No. 79-14 Program, RC Drain and Purification R.C. Letdown," Book 1 of 2, Rev. A, 11 November 1982 (Davis-Besse File T-001B 1982-11-11 Rev. A)

25. *Teledyne Technical Report E-1495-16, "ASME Section III Analysis, Class 1 R.C. drain – Loop 1-1 and Purification Letdown Piping," Book 1 of 2, June 15, 1973 (Davis-Besse Calculation T-002B R0, 10/03/1988)

26. *Teledyne Technical Report TR-3831-13, "ASME Section III Analysis, Class 1, Based on I.E. Bulletin No. 79-14 Program, High-Pressure Injection Piping Problems T-003-1(b), T-003-2(b) T-004-1(b) and T-004-2(b)," January 7, 1983

27. *Teledyne Technical Report E-1495-14, "Davis Besse Nuclear Power Station ASME Section III Analysis, Class 1 – Decay Heat Removal System Model A, Book 1 of 2," Rev. A, January 12, 1977 (Davis-Besse file T-006A_1977-01-12_0A_NA)

28. *Teledyne Technical Report TR-3831-11, "Davis-Besse Nuclear Power Station Unit 1, ASME Section III Class 1 Analysis based on I.E. Bulletin No. 79-14 Program, Decay Heat Removal, Problem T-006(A)," November 17, 1982 (Davis-Besse file T-006A_1982-11-17_N03_NA

29. *Teledyne Technical Report E-1831-12, "Davis-Besse Nuclear Power Station, Unit 1, ASME Section III Analysis Class 1 Based on I.E. Bulletin No. 79-14 Program Low-Pressure Injection-Core Flooding Piping, Problems T-007(B) and T-008(B)" 8 December 1982

30. *Teledyne Technical Report E-1495-17, Rev. A, Davis Besse Nuclear Power Station, ASME Section III Analysis of Class 1 – Pressurizer Relief Model A, piping, May 25, 1977 (Davis-Besse calculation T-012, Rev. A,25 May 1977)

31. *Calculation Package DB-06Q-304, Rev. 1, "RCS Cold Leg Letdown Line Nozzle Weld Overlay Repair ASME Code Section III Evaluation," 18 May 2006

32. AREVA Document 51-9135330-000, "DB-1 LRA Section 4.3 Environmentally-Assisted Fatigue Input," June 2010

33. EPRI Technical Report 1012017, "Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47, Revision 1), September 2005

34. *Teledyne Technical Report TR-6108, Rev. 4, "Davis-Besse Nuclear Station, Piping Stress Analysis: Reactor Head to Hot Leg Vent," March, 2003

*This reference is not available for entry into the AREVA NP record system. However, it is contained in the customer's records system. These references are acceptable for use as a design input on this contract per AREVA NP procedure 0402-01-039.

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