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November 23, 2011
L-11-354

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 Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640), License Renewal Application Amendment No. 22, and Revised License Renewal Application Boundary Drawings

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 October 11, 2011 (ADAMS Accession No. ML11271A147) and November 8, 2011 (ADAMS Accession No. ML11306A141), and during telephone conferences held on October 5, November 9, and November 14, 2011, the Nuclear Regulatory Commission (NRC) requested supplemental information to complete its review of the License Renewal Application (LRA).

By letter dated October 31, 2011, FENOC responded to three of the four NRC request for additional information (RAI) questions in NRC letter dated October 11, 2011 (ADAMS Accession No. ML11271A147); the response to the fourth RAI question is provided in the Attachment to this letter. The Attachment also provides the FENOC reply to the RAI questions in NRC letter dated November 8, 2011 (ADAMS Accession No. ML11306A141) and supplemental information as discussed in the telephone conferences. The NRC request is shown in bold text followed by the FENOC response.

Enclosure A provides Amendment No. 22 to the DBNPS LRA. Enclosure B provides revised LRA boundary drawings. Enclosure C provides AREVA NP Report No. 32-1172294-002, "Davis Besse 1 SG Flaw Evaluation," dated November 17, 2011. Enclosure D provides AREVA NP Report No. 32-1172523-001, "Davis-Besse 1 SG Flaw Evaluation," dated November 17, 2011.

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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 November 23, 2011.

Sincerely,

Barry S. Allen

Barry S. Allen

Attachment:
Reply to Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), License Renewal Application, Sections 4.7, 3.1, 3.3, 2.3 and B.2.22

Enclosures:

A. Amendment No. 22 to the DBNPS License Renewal Application
B. Revised DBNPS License Renewal Application Boundary Drawings
C. AREVA NP Report No. 32-1172294-002, "Davis Besse 1 SG Flaw Evaluation"
D. AREVA NP Report No. 32-1172523-001, "Davis-Besse 1 SG Flaw Evaluation"

cc: NRC DLR Project Manager
NRC Region III Administrator

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

Section 4.7

Question RAI 4.7.5.2-2 (from NRC letter dated October 11, 2011 (ML11271A147))

Background:

LRA Section 4.7.5.2 addresses the TLAA related to the steam generator 1-2 flaw evaluations. LRA Section 4.7.5.2 states that the subject flaws were analytically evaluated using the ASME Code, Section XI, IWB-3612 acceptance criteria. LRA Section 4.7.5.2 further states that the IWB-3612 analysis of the subject flaws determined that the steam generator shell components containing the flaws would remain acceptable for continued service during the period of extended operation, accounting for flaw growth due to fatigue based on 240 heat-up and cool-down cycles.

By letter dated March 17, 2011 (ADAMS Accession No. ML110680172) the NRC staff submitted a request for additional information (RAI) concerning the plant-specific TLAAs in the Davis-Besse LRA, Sections 4.7.4, 4.7.5.1, and 4.7.5.2. The staff issued RAI 4.7.5.2-1 to request clarification on a number of issues concerning the subject steam generator shell flaws and the ASME Code, Section XI, IWB-3612 analytical evaluations of these flaws.

In RAI 4.7.5.2-1, part (b), the staff requested that the applicant state whether the subject flaws were found to be the result of service-induced degradation or fabrication defects. In RAI 4.7.5.2-1, part (e) the staff requested that the applicant state whether the flaw dimensions have increased since discovery in 1988. The staff also requested that, if the flaw dimensions have increased, the applicant state whether the subject flaws were re-analyzed in accordance with ASME Code, Section XI, IWB-3612 requirements based on the new flaw dimensions. In RAI 4.7.5.2-1, part (g), the staff requested that the applicant provide references for all reports documenting IWB-3612 analytical evaluations of the subject flaws.

Issue:

By letter dated April 15, 2011, the applicant submitted its responses to the staff's RAIs. In its response to RAI 4.5.2.1, part (b), the applicant stated that the subject flaws "were analyzed in accordance with IWB-3612, as required by the ASME [Code], Section XI acceptance standards, and found to be acceptable for continued operation." The staff reviewed the applicant's response to

RAI 4.7.5.2-1, part (b) and noted that the applicant did not state whether the subject flaws were determined to be service-induced or caused by fabrication.

In its response to RAI 4.7.5.2-1, part (e), the applicant stated that "[t]he subject components were reexamined during Cycle 6 (year 1990) and no flaw growth was noted. The subject components, with the exception of the W axis longitudinal seam weld intersection with the shell to lower tubesheet weld, were also reexamined during Cycle 7 (year 1991) and no flaw growth was noted." The staff reviewed the applicant's response to RAI 4.7.5.2-1, part (e), and noted that the RAI response only stated that no flaw growth was noted during the ASME Code, Section IWC-2420(b)-required successive inspections performed in 1990 and the subsequent inspections performed in 1991. The staff noted that the applicant did not state whether any flaw growth was noted for the subject components as a result of any examinations performed on the flawed regions after 1991.

In its response to RAI 4.7.5.2-1, part (g), the applicant stated that the subject flaw evaluations are documented in the following Babcock & Wilcox (B&W) Reports from 1988:

1. Report No. 32-1172294-00, "Davis-Besse 1 SG Flaw Evaluation," dated June 9, 1988
2. Report No. 32-1172294-01, "Davis-Besse 1 SG Flaw Evaluation," dated July 18, 1988
3. Report No. 32-1172523-00, "DB-1 SG Flaw Evaluation," dated July 18, 1988

The above flaw evaluation reports were provided in an enclosure to the April 15, 2011 RAI response. These flaw evaluation reports reference the 1977 Edition of the ASME Code, Section XI, IWB-3612 analytical acceptance standard. The flaw evaluation report summaries state that the subject flaws were found to be acceptable, in accordance with the ASME Code, Section XI, IWB-3612 analytical acceptance standard.

In reviewing the above flaw evaluation reports, the staff determined that the subject flaw evaluations were only performed for normal conditions, and only demonstrated acceptability based on the analytical acceptance criterion for normal (including upset and test) operating conditions, as specified in the ASME Code, Section XI, IWB-3612, paragraph (a). The staff determined that the applicant had not specifically evaluated the subject flaws for emergency and faulted conditions, as required by the 1977 Edition of the ASME Code, Section XI, IWB-3612, paragraph (b).

__Request:__

Based on the above, the staff requests that the applicant provide the following information concerning the subject steam generator flaws and the analytical evaluations performed for these flaws:

(1) Taking into consideration the steam generator shell materials containing the flaws, the secondary side water and steam environment, and the secondary side thermal and pressure stresses to which these shell components are subjected, please state whether any of the surface-breaking indications were believed to have been caused by stress corrosion cracking, or any other service-induced aging effect.

(2) For any inservice examinations performed on the flawed regions of the steam generator shell after 1991, in particular the examinations performed for the steam generator X/Y axis outlet nozzle to shell weld and the lower tubesheet to shell weld during the first and second periods of the third 10-year ISI interval, please state whether these examinations detected any increase in the flaw dimensions, relative to the 1988 flaw dimensions. (The staff notes that any measured increase in flaw dimensions would likely invalidate the analyses performed in the 1988 flaw evaluation reports.)

(3) Please state whether the subject flaws were analyzed for emergency and faulted conditions, as required by the ASME Code, Section XI, IWB-3612, paragraph (b). If the subject flaws were analyzed for emergency and faulted conditions, as required by IWB-3612, paragraph (b), please provide the flaw analyses for these conditions, or explain how the IWB-3612, paragraph (a) analyses, as documented in the 1988 flaw evaluation reports, for normal, upset, and test conditions, would bound the flaw analyses for emergency and faulted conditions. If the subject flaws were not analyzed for emergency and faulted conditions, please provide these analyses, as required by IWB-3612, paragraph (b).

RESPONSE RAI 4.7.5.2-2

(1) In development of LRA Table 3.1.2-4, "Aging Management Review Results – Steam Generators," an aging management review of the steam generator components including the shell was conducted using EPRI Report 1010639, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 4," also known as the "Mechanical Tools."

The steam generator shell material containing the flaws is carbon steel (steel) with an environment of treated water (liquid and steam phases). As stated in Appendix A of the Mechanical Tools, cracking due to stress corrosion cracking

(SCC) is an applicable aging effect for carbon steel exposed to treated water only if there is a potential for microbiologically-influenced corrosion (MIC) contamination, pH less than 10.5 and temperature less than 210°F, and nitrite corrosion inhibitor in use. A review of plant-specific operating experience identified instances of MIC only for open cycle cooling water systems; therefore, MIC is not an age-related concern for treated water environments at Davis-Besse. Therefore, it is concluded that the steam generator flaws were not caused by cracking due to SCC.

In addition, the aging management review addressed cracking due to fatigue as an aging effect requiring further evaluation. As stated in Appendix H of the Mechanical Tools, carbon steel above 220°F is susceptible to cracking due to fatigue. To account for metal fatigue, the ASME Design Code requires calculation of cumulative usage factors (CUF) and the usage factors must be less than 1.0. The steam generators were analyzed for fatigue and the CUFs for the limiting primary and secondary side steam generators locations were calculated based on design transients, and are less than 1.0. As shown in LRA Table 4.3-1, "60-Year Projected Cycles," the accrued cycles as of February 19, 2008, were less than the design cycles (i.e., analyzed cycles). Therefore, it is concluded that the steam generator flaws were not caused by cracking due to fatigue.

Also, the aging management review identified cracking due to flaw growth as an aging effect requiring management for the carbon steel components of the steam generators that are exposed to the treated water environment. Components fabricated in accordance with the ASME Code are presumed to contain material and fabrication flaws whose sizes and character are below the detection threshold of the examination method employed, or less than the acceptance standards. The presence of such flaws, although not detrimental to the structural integrity of the newly constructed component led to the recognition that these flaws might grow in size as a consequence of the loadings imposed on the component during the service lifetime.

Since it was concluded that the steam generator flaws were not caused by cracking due to SCC or by cracking due to fatigue, it is believed that the subject flaws are pre-service flaws that were below the detection threshold of the examination method employed during fabrication of steam generator (SG) 1-2.

(2) The examinations performed for the SG 1-2 lower tubesheet to shell weld in the Cycle 13 refueling outage (first period of the third 10-year Inservice Inspection (ISI) interval) did not detect any flaw growth relative to the 1988 flaw dimensions.

The examinations performed for the SG 1-2 X/Y axis outlet nozzle to shell weld in the Cycle 15 refueling outage (second period of the third 10-year ISI interval) did not detect any flaw growth relative to the 1988 flaw dimensions.

The examinations performed for the SG 1-2 W/X axis outlet nozzle to shell weld in the Cycle 17 mid-cycle outage that commenced on October 1, 2011 (third period of the third 10-year ISI interval) did not detect any flaw growth relative to the 1988 flaw dimensions.

(3) In the evaluations previously provided for NRC review, the steam generator 1-2 flaws were not analyzed for emergency and faulted conditions, as required by the ASME Code, Section XI, IWB-3600(a). As a result, the subject flaw evaluations were revised to address the emergency/faulted conditions as required by subsection IWB-3612 of the ASME Code, Section XI, 1995 Edition with 1996 Addenda (current ASME Code for the current (third) ten year ISI interval for Davis-Besse). The flaw indications were found to be acceptable by the IWB-3612 standard.

The revised flaw evaluations are documented in the AREVA NP Reports listed below, and are provided as Enclosures to this letter as follows:

Enclosure C: 32-1172294-002, "Davis Besse 1 SG Flaw Evaluation," dated November 17, 2011

Enclosure D: 32-1172523-001, "Davis-Besse 1 SG Flaw Evaluation," dated November 17, 2011

<u>Section 3.1</u>

Question RAI 3.1.2.2.16-2 (from NRC letter dated November 8, 2011 (ML11306A141))

<u>Background:</u>

By letter dated October 21, 2011, the applicant responded to RAI 3.1.2.2.16-1, which addressed a need for the aging management of cracking due to primary water stress corrosion cracking (PWSCC) of the steam generator (SG) tube-to-tubesheet welds. In its response, the applicant stated that cracking due to PWSCC will be managed for the steam generator tube-to-tubesheet welds (Alloy 600) by a combination of the PWR Water Chemistry Program and the Steam Generator Tube Integrity Program. The applicant also stated that the Steam Generator Tube Integrity Program will be enhanced to include enhanced visual (EVT-1 or equivalent) examinations to monitor for cracking of the steam generator tube-to tubesheet welds. The applicant further indicated that welds included in the inspection sample will be scheduled for examination in each 10-year period

that occurs during the period of extended operation and unacceptable inspection findings will be evaluated by the Corrective Action Program using criteria in accordance with Section XI of the ASME Code.

In addition, the applicant indicated that a review of Davis-Besse operating experience has not identified any instances of cracking of the steam generator tube-to-tubesheet welds (Alloy 600): therefore, the weld inspection sample size will include 20 percent of the subject weld population or a maximum of 25, whichever is less. The applicant stated that in this case the maximum of 25 applies since the weld population for the two steam generators is greater than 60,000. The applicant also indicated that if the steam generators are replaced in the future with a design such that the tube-to-tubesheet welds are fabricated of Alloy 690-TT material, the examinations will no longer be required.

Issue:

In its review, the staff found a need to clarify whether the "Alloy 690 TT material," which refers to a potential material for future steam generator welds, means Alloy 690 TT tubes with Alloy 690 type weld material (e.g., Alloy 52). The staff also noted that it is not clear whether Section XI of the ASME Code has acceptance criteria for these steam generator tube-to-tubesheet welds. In addition, the staff found a need to further clarify whether the EVT-1 inspection is capable of detecting cracking in the tube-to-tubesheet weld. The staff also requests that the applicant discuss the extent, to which the routine steam generator tube inspections, using bobbin coil or rotating coil examinations, can detect cracking of the tube-to-tubesheet welds.

The staff also found a need for clarification of why a sample size of 25 is adequate to monitor for the cracking of the steam generator tube-to-tubesheet welds, in view of the following considerations: (1) potential variabilities exist in the weld chemistry, environment and stresses in the approximately 60,000 welds, (2) Alloy 600 is susceptible to PWSCC, (3) the applicant's steam generator tubes (Alloy 600) have experienced cracking due to PWSCC, indicating that the degradation mechanism (PWSCC) exists for the steam generator tubes, and (4) the applicant's program has not implemented any inspection intended to detect cracking in the tube to tubesheet welds.

Request:

The applicant indicated that examinations are no longer required if the steam generators are replaced in the future with a design such that the tube to tubesheet welds are fabricated with Alloy 690 TT material. Please, provide information to clarify whether the "Alloy 690 TT material" means Alloy 690 TT tubes with Alloy 690 type tubesheet cladding (e.g., Alloy 52). If not, discuss why

inspections are not necessary to manage cracking due to PWSCC of the replacement steam generator welds.

1. **It is not clear that Section XI of the ASME Code has acceptance criteria for these steam generator tube to tubesheet welds. Please, discuss what acceptance criteria will be used to evaluate the indications found in the inspections.**
2. **Provide information to demonstrate the EVT 1 inspection is capable of detecting cracking in the tube to tubesheet welds. In addition, discuss the extent, to which the routine steam generator tube inspections, using bobbin coil or rotating coil examinations, can detect cracking of the tube to tubesheet welds.**
3. **Provide justification as to why a sample size of only 25 is adequate to monitor for the cracking of the steam generator tube-to-tubesheet welds in view of the following considerations: (1) potential variabilities exist in the weld chemistry, environment and stresses in the approximately 60,000 welds, (2) Alloy 600 tubes are susceptible to PWSCC, (3) the applicant's Alloy 600 tubes have experienced cracking due to PWSCC, indicating that the degradation mechanism (PWSCC) exists for the steam generator tubes, and (4) the applicant's program has not implemented any inspection intended to detect cracking in the tube-to-tubesheet welds.**

RESPONSE RAI 3.1.2.2.16-2

The proposed design of the Davis-Besse replacement steam generators includes steam generator tubes fabricated with Alloy 690TT material, tubesheet cladding fabricated with Alloy 690/52/152 material and autogenous (i.e., no filler material) tube-to-tubesheet welds. NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Revision 2, Section 3.1.2.2.11, "Cracking due to Primary Water Stress Corrosion Cracking," states that, for plants with Alloy 690TT steam generator tubes with Alloy 690 tubesheet cladding, the water chemistry program is sufficient, and no further action or plant-specific aging management program is required. Should the current Davis-Besse steam generators be replaced in the future with a design such that the tubes, tubesheet cladding and tube-to-tubesheet welds are fabricated of Alloy 690 material, the PWR Water Chemistry Program would be sufficient to manage cracking due to primary water stress corrosion cracking (PWSCC).

1. FENOC agrees that Section XI of the ASME Code does not have acceptance criteria for the steam generator tube-to-tubesheet welds.

   As stated in the response to request number 2, below, the inspection method for the existing steam generator tube-to-tubesheet welds is revised to consist of a gross visual inspection of the tube-to-tubesheet welds coupled with eddy-current inspection (i.e., bobbin coil or rotating coil examinations) of the tubes. The acceptance criteria is revised to consist of no indication of cracking or relevant conditions of degradation.

2. In lieu of providing information to demonstrate that the EVT-1 inspection is capable of detecting cracking in the tube-to-tubesheet welds, the inspection method for the existing steam generator tube-to-tubesheet welds is revised to consist of a gross visual inspection of the welds coupled with eddy-current inspection (i.e., bobbin coil or rotating coil examinations) of the tubes. The gross visual inspection of the tube-to-tubesheet welds coupled with eddy-current inspection of the tubes will confirm the structural integrity of the tube-to-tubesheet joint.
3. In lieu of providing justification that the sample size of 25 steam generator tube-to-tubesheet welds is adequate to monitor for PWSCC, the sample size for the existing steam generator tube-to-tubesheet welds is revised as follows.

   Gross visual inspection of the steam generator tube-to-tubesheet welds will be scheduled concurrent with eddy-current inspections of the steam generator tubes. Steam generator tube inspections are scheduled in accordance with Davis-Besse Technical Specification 5.5.8, "Steam Generator (SG) Program." At a minimum, 100% of the tubes are inspected at sequential periods of 60 effective full power months.

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

Section 3.3

Question RAI 3.3.2.14-2 (from NRC letter dated November 8, 2011 (ML11306A141))

Background:

License renewal application (LRA) Table 3.3.2-14, "Aging Management Review Results – Fire Protection," item "Heat Exchanger (tubes) – Fire water storage tank heat exchanger (DB-E52)," originally proposed a one-time inspection to manage the reduction in heat transfer of stainless steel tubes. The Generic Aging Lessons Learned Report (NUREG-1801) states that stainless steel components exposed to steam are susceptible to loss of material and stress corrosion cracking; however, the applicant has not identified these aging effects for this component.

By letter dated July 27, 2011, the staff issued request for additional information (RAI) 3.3.2.14-1 requesting that the applicant justify why loss of material and

stress corrosion cracking are not applicable aging effects for the fire water storage tank heat exchanger tubes exposed to steam.

In its response dated August 26, 2011, the applicant stated that the only license renewal function for the heat exchanger is reduction of heat transfer and the only aging mechanism that is identified as causing the aging effect of reduction of heat transfer is the aging mechanism of fouling. Loss of material and cracking would ultimately affect the pressure boundary function of the tubes. The applicant also stated that if the heat exchanger tubes should leak, fire water would not leak from the tubes; rather, the higher pressure (i.e., approximately 50 psig) steam from the auxiliary steam system on the external surfaces of the tubes would pass through the tubes and mix with fire water (approximately 25 psig), thereby continuing to add heat to the water. Fire water storage tank level would increase due to water entering the system, but level in the tank could be controlled (Le., feed-and-bleed) to prevent the tank from overflowing onto the ground. A breach of the heat exchanger tubes would result in continued heat transfer to fire water, and would not prevent the fire water system from performing its functions.

A teleconference was held on September 13, 2011, to further discuss this issue and determine, with a heat exchanger tube failure, whether the fire water storage tank's design could contain a water/steam environment. The applicant stated that the heat exchanger was not subject to license renewal scope based on the Fire Safety Hazard Analysis. The applicant was asked to fully document the basis for this statement.

In a follow-up response dated October 7,2011, the applicant revised the LRA to delete the fire water storage tank heat exchanger (DB-E52) and fire water storage tank recirculation pump casing (DB-P114). Also License Renewal Boundary Drawing LR-M016A, "Station Fire Protection System," was revised to remove highlighting of the piping and components associated with the fire water storage tank heat exchanger (DB-E52) and fire water storage tank recirculation pump 1-1. The applicant also stated that the fire water storage tank heat exchanger and recirculation pump are not within the scope of license renewal since the subject components do not satisfy the scoping criteria of 10 CFR 54.4(a)(1), (a)(2), or (a)(3). The heat exchanger and the recirculation pump are used to establish initial conditions associated with event assumptions, and perform no fire protection functions. Hence it is the monitoring of the fire water storage tank that is credited with ensuring the appropriate initial conditions and therefore, the heat exchanger and recirculation pump are not in the scope of License Renewal for the Fire Protection regulated event.

However, it is the staff's position that these components are required to maintain temperature in the fire water tank above 35°F. The Fire Hazard Analysis Report (FHAR), Section 8.1.2, Fire Suppression Water System, states that "...the

temperature of the contained water supply is greater than 35°F every 24 hours during October through March" which is verified using surveillance. These components should not be excluded from the fire water system on the basis that they are not required to function to suppress a fire; rather they should be included to support the tank's primary function of maintaining a useable inventory of water at the appropriate temperature to avoid freezing.

A second teleconference was held on November 1, 2011, to discuss that the deletion of these components was not consistent with the current licensing basis (CLB).

Issue:

It is not clear to the staff how the removal of these fire protection system components is consistent with the FHAR associated with the original Davis-Besse fire protection SERs and the plant's CLB.

The staff lacks sufficient information to understand the basis of the applicant's proposal that these components are not included within scope per 10 CFR 54.4(a)(3). The staff believes that these fire protection SSCs are required for compliance to 10 CFR 50.48 and are subject to an AMR as shown in 10 CFR 54.21.

The revised LRA does not appear to demonstrate that the aging effects associated with the fire protection system are adequately managed, so that there is reasonable assurance that the system components will perform their intended functions in accordance with the CLB, during the period of extended operation as required by 10 CFR 54.4(a)(3).

Requests:

Justify how the fire water storage tank will be maintained greater than 35°F at all times without the heat exchanger or provide an appropriate aging management program to manage aging for the original component and their subcomponents inclusive of all applicable aging effects.

If components are excluded and other methods are used for the tank's primary temperature function, then describe the procedure steps that would be used to maintain the fire water storage tank level and temperature.

For example, with the heat exchanger tube failure a "feed and bleed" procedure to prevent tank overflow would be required, or without a heat exchanger for the tank an operational procedure would be needed to create recirculation in the tank and provide flow/heating to the tank and keep temperature greater than 35°F.

When describing these procedures, please document any steps that would require operator action. Please also include a complete list of the SSCs that are part of the procedure including their material, environment, and aging effect that would require age management to operate and support the tank's primary temperature function. Include the aging management program that will be used and list the associated aging management review (AMR) line items documenting their management and associated inspection methods and parameters to be monitored.

Specific to a feed and bleed procedure, please identify the tank temperature upper supply limit to prevent a potential overheating of the tank and net positive suction head issue on the downstream pumps. Please also identify the volume and temperature of feed required to control/maintain the tank's temperature adequately. Also include any isolation steps that would be required to prevent loss of inventory from the tank to the immediate surroundings with a loss of steam pressure to the heat exchanger.

The applicant's most recent response includes the recirculation pump as part of the proposed omitted license renewal scope. Please include any additional AMR line items related to that proposed deletion such as piping components and elements that then would no longer be age managed in the LRA.

Please document the FHAR sections that would support removal of these components while retaining the primary function of adequate fire water supply temperature and maintaining consistency with the plant's CLB.

RESPONSE RAI 3.3.2.14-2

FENOC provides an appropriate aging management program to manage aging for the original component and their subcomponents inclusive of all applicable aging effects. The fire water storage tank heat exchanger, associated recirculation line components (including the recirculation pump) and associated auxiliary steam line components are added to the scope of license renewal for the Fire Protection (i.e., 10 CFR 50.48) regulated event and are subject to an aging management review in accordance with 10 CFR 54.21(a)(1).

The added scoping is shown in two new and four revised License Renewal Boundary Drawings, and in revised LRA Section 2.3.3.3. Also, the aging management review results are provided in revised LRA Table 3.3.2-3, "Aging Management Review Results – Auxiliary Steam and Station Heating Systems," and revised LRA Table 3.3.2-14, "Aging Management Review Results – Fire Protection System."

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

See Enclosure B to this letter for the new and revised LRA Boundary Drawings.

Section 2.3

Supplemental Question RAI 2.3.3.18-4 (from Telecon held on November 9, 2011)

The NRC initiated a telephone conference call with FENOC on November 9, 2011, to discuss the FENOC response to RAI 2.3.3.18-4 submitted under FENOC letter dated October 21, 2011 (ML11298A097), regarding aging management of the letdown coolers.

In its response to RAI 2.3.3.18-4 dated October 21, 2011, the applicant stated that the "letdown coolers are in continuous service and not subject to cyclic loading, eddy current testing of tubes for managing cyclic loading is therefore not applicable." However, the staff noted that in its August 17, 2011, response to an RAI, the applicant stated that the Babcock & Wilcox report regarding the reliability of letdown coolers identified the cause of the recurring tube leaks as fatigue cracking likely initiated by flow-induced vibration.

Issue:

While the staff agrees that the coolers may be in continuous service, there does not appear to be a technical basis for stating that the coolers are not subject to cyclic loading, based on the information previously provided by the applicant. Unless there is some previously undisclosed additional information, fatigue cracking due to flow-induced vibration appears to demonstrate that the letdown cooler tubes are subject to cyclic loading.

After discussions, FENOC agreed that the statement, "The coolers are in continuous service and not subject to cyclic loading...," only considered thermal fatigue (i.e., low-cycle fatigue). Since the coolers are subject to fatigue cracking due to flow-induced vibration (i.e., high-cycle fatigue), FENOC agreed to delete or modify the statement.

In addition, FENOC agreed to provide an enhancement to the Closed Cooling Water Chemistry Program to ensure that component cooling water radiochemistry is sampled on a periodic basis to verify the integrity of the letdown coolers.

FENOC agreed to provide a supplemental response to RAI 2.3.3.18-4 to address the above issues.

SUPPLEMENTAL RESPONSE RAI 2.3.3.18-4

The response to RAI 2.3.3.18-4 submitted under FENOC letter dated October 21, 2011 (ML11298A097), is revised as described below.

> The letdown coolers (DB-E25-1 and 2) and the seal return coolers (DB-E26-1 & 2) in the Makeup and Purification System consist of stainless steel heat exchanger components exposed to treated borated water greater than 60°C (> 140°F). Cracking due to stress corrosion cracking (SCC) in stainless steel heat exchanger components that are exposed to treated borated water greater than 60°C (>140°F) is managed by the PWR Water Chemistry Program. The PWR Water Chemistry Program manages cracking through periodic monitoring and control of contaminants. One-Time Inspection will provide verification of the effectiveness of the PWR Water Chemistry Program to manage cracking. Component Cooling Water radiochemistry will be sampled on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.
>
> The Closed Cooling Water Chemistry Program is revised to include an enhancement to ensure that component cooling water radiochemistry is sampled on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.

LRA Section 3.3.2.2.4.1, Table 3.3.1, Section A.1.8, Table A-1 and Section B.2.8 are revised consistent with this response.

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

<u>**Section B.2.22**</u>

Supplemental Question RAI B.2.22-5 (from Telecon held on November 14, 2011)

The NRC staff initiated a telephone conference with FENOC on October 5, 2011, to discuss the FENOC response to RAI B.2.22-5 submitted in FENOC letter dated September 16, 2011 (ML11264A059), regarding inspection, maintenance, and

repair of the annulus sand pocket accessible and inaccessible areas. The NRC requested additional information about the containment vessel exterior moisture barrier located in the annulus sand pocket area. FENOC stated that additional review was needed to respond to the NRC request for additional information.

The NRC staff initiated a follow-up telephone conference with FENOC on November 14, 2011, to resume discussion of the containment vessel exterior moisture barrier and associated structural components. FENOC agreed to perform visual inspections of the accessible portions of the moisture barrier during refueling outages.

SUPPLEMENTAL RESPONSE RAI B.2.22-5

LRA Table A-1, "Davis-Besse License Renewal Commitments," license renewal future Commitment 36, is revised to include performance of visual inspections, for deterioration, of 100% of the accessible surfaces of the external containment moisture barrier located in the sand pocket area during each refueling outage. Degradation of the moisture barrier will be addressed using the FENOC Corrective Action Program.

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

Enclosure A

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

Letter L-11-354

Amendment No. 22 to the DBNPS License Renewal Application

Page 1 of 31

License Renewal Application Sections Affected

Section 2.3.3.3	Table 3.4.1
Table 2.3.3-3	Section 4.3.3.2
Table 2.3.3-14	Section A.1.8
Section 3.3.2.1.14	Section A.1.38
Section 3.3.2.2.4.1	Table A-1
Table 3.3.1	Section B.2.8
Table 3.3.2-3	Section B.2.38
Table 3.3.2-14	

The 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 *underlined*.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
2.3.3.3	**Pages 2.3-54 and 2.3-55**	**Reason for Scope Determination, 3rd paragraph; and, License Renewal Drawings, 2 new drawings**

In response to RAI 3.3.2.14-2, LRA Section 2.3.3.3, "Auxiliary Steam and Station Heating System," the last paragraph under the subheading "Reason for Scope Determination" is revised, and two new license renewal boundary drawings are added to the list of "License Renewal Drawings," to read as follows:

Reason for Scope Determination

The Auxiliary Steam and Station Heating System is ~~not relied upon to demonstrate compliance with, and does not satisfy the 10 CFR 54.4(a)(3) scoping criteria for, any regulated events.~~ relied upon to demonstrate compliance with, and satisfies the 10 CFR 54.4(a)(3) scoping criteria for, the Fire Protection (10 CFR 50.48) regulated event.

License Renewal Drawings

The following license renewal drawings depict the evaluation boundaries for the system components within the scope of license renewal:

LR-M006F, LR-M010D, LR-M020A, LR-M020B, LR-M020C, LR-M020D, LR-M021, LR-M026B, LR-M027A, LR-M027B, LR-M028C, LR-M028D, LR-M029E

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 2.3.3-3	**Pages 2.3-55 and 2.3-56**	**5 Rows, Intended Function**

In response to RAI 3.3.2.14-2, LRA Table 2.3.3-3, "Auxiliary Steam and Station Heating System Components Subject to Aging Management Review," is revised to include the "pressure boundary" intended function for five component types, and reads as follows:

Component Type	Intended Function (as defined in Table 2.0-1)
Bolting	*Pressure boundary* Structural integrity
Piping	*Pressure boundary* Structural integrity
Trap body	*Pressure boundary* Structural integrity
Tubing	*Pressure boundary* Structural integrity
Valve body	*Pressure boundary* Structural integrity

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 2.3.3-14	**Page 2.3-95**	**3 Rows**

In response to RAI 3.3.2.14-2, the rows associated with the fire water storage tank heat exchanger and the fire water storage tank recirculation pump in LRA Table 2.3.3-14, "Fire Protection System Components Subject to Aging Management Review," are added as follows:

Component Type	**Intended Function (as defined in Table 2.0-1)**
Heat Exchanger (channel, shell, tubes and tubesheet) – Fire water storage tank heat exchanger (DB-E52)	*Pressure boundary*
Heat Exchanger (tubes) – Fire water storage tank heat exchanger (DB-E52)	*Heat transfer*
Pump Casing – Fire water storage tank recirculation pump (DB-P114)	*Pressure boundary*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
3.3.2.1.14	**Page 3.3-19**	**Aging Management Programs, 1 bullet**

In response to RAI 3.3.2.14-2, the Aging Management Program subsection of Section 3.3.2.1.14, "Fire Protection System," is revised to add the PWR Water Chemistry Program as follows:

- *PWR Water Chemistry Program*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
3.3.2.2.4.1	**Page 3.3-40**	**Last sentence**

In response to Supplemental RAI 2.3.3.18-4 related to the letdown coolers, LRA Section 3.3.2.2.4.1, "Stainless Steel PWR Nonregenerative Heat Exchanger Components – Treated Borated Water Greater Than 60°C (> 140°F)," previously revised by FENOC letter dated October 21, 2011 (ML11298A097), is revised to read as follows:

3.3.2.2.4.1 Stainless Steel PWR Nonregenerative Heat Exchanger Components – Treated Borated Water Greater Than 60°C (> 140°F)

Cracking due to stress corrosion cracking and cyclic loading could occur in stainless steel pressurized water reactor (PWR) nonregenerative heat exchanger components exposed to treated borated water greater than 60°C (> 140°F) in the chemical and volume control system. At Davis-Besse, the letdown coolers (DB-E25-1 and 2) and the seal return coolers (DB-E26-1 and 2) in the Makeup and Purification System consist of stainless steel heat exchanger components exposed to treated borated water greater than 60°C (> 140°F). Cracking due to stress corrosion cracking (SCC) in stainless steel heat exchanger components that are exposed to treated borated water greater than 60°C (>140°F) is managed by the PWR Water Chemistry Program. The PWR Water Chemistry Program manages cracking through periodic monitoring and control of contaminants. One-Time Inspection will provide verification of the effectiveness of the PWR Water Chemistry Program to manage cracking. ~~*The coolers are in continuous service and not subject to cyclic loading; therefore, eddy current testing of the tubes to manage cyclic loading is not applicable. Temperature and radioactivity monitoring of shell side water is performed by installed instrumentation.*~~ *The Closed Cooling Water Chemistry Program samples component cooling water radiochemistry on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.3.1	**Page 3.3-51**	**Row 3.3.1-07, "Discussion" column**

In response to Supplemental RAI 2.3.3.18-4, the text in the "Discussion" column of row 3.3.1-07 of LRA Table 3.3.1, "Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of NUREG-1801," previously revised by FENOC letter dated October 21, 2011 (ML11298A097), is revised to read as follows:

Table 3.3.1 Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-07	Stainless steel non-regenerative heat exchanger components exposed to treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking and cyclic loading	Water Chemistry and a plant-specific verification program. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, plant specific	Consistent with NUREG-1801. Cracking due to SCC for stainless steel heat exchanger components in the Auxiliary Systems that are exposed to treated borated water > 60°C (> 140°F) is managed by the PWR Water Chemistry Program. The One-Time Inspection will provide verification of the effectiveness of the PWR Water Chemistry Program to manage cracking. ~~Cracking due to cyclic loading is not applicable since these components are continuously in service and not subject to cyclic~~

Table 3.3.1 Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of NUREG-1801					
Item Number	**Component/Commodity**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
					~~loading.~~ ~~Temperature and radioactivity monitoring of shell side water is performed by installed instrumentation.~~ *The Closed Cooling Water Chemistry Program samples component cooling water radiochemistry on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.* Further evaluation is documented in Section 3.3.2.2.4.1.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.3.1	**Page 3.3-105**	**Row 3.3.1-85, "Discussion" column, 2nd paragraph**

In response to RAI 3.3.2.14-2, the text in the "Discussion" column of row 3.3.1-85 of LRA Table 3.3.1, "Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of NUREG-1801," is revised to read as follows:

Table 3.3.1 Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-85	Gray cast iron piping, piping components, and piping elements exposed to soil, raw water, treated water or closed-cycle cooling water	Loss of material due to selective leaching	Selective Leaching of Materials	No	Consistent with NUREG-1801. *Loss of material due to selective leaching in gray cast iron piping, piping components, and piping elements that are exposed to soil, raw water, treated water and closed cycle cooling water will be detected and characterized by the Selective Leaching Inspection.* This item is also applied to gray cast iron heat exchanger components that are exposed to closed cycle cooling water. This item is also applied to gray cast iron piping, piping components, and piping

Table 3.3.1 Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of NUREG-1801					
Item Number	**Component/Commodity**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
					elements that are exposed to condensation (internal), where the condensation environment is evaluated as equivalent to a raw water environment.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.3.2-3	**Page 3.3-186**	**23 New Rows**

In response to RAI 3.3.2.14-2, 23 new rows are added to LRA Table 3.3.2-3, "Aging Management Review Results – Auxiliary Steam and Station Heating Systems," to read as follows:

Table 3.3.2-3 Aging Management Review Results – Auxiliary Steam and Station Heating Systems

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
--	*Bolting*	*Pressure boundary*	*Steel*	*Air with steam or water leakage (External)*	*Cracking*	*Bolting Integrity*	*VII.I-3*	*3.3.1-41*	*B*
--	*Bolting*	*Pressure boundary*	*Steel*	*Air-indoor uncontrolled (External)*	*Loss of material*	*Bolting Integrity*	*VII.I-4*	*3.3.1-43*	*B*
--	*Bolting*	*Pressure boundary*	*Steel*	*Air-indoor uncontrolled (External)*	*Loss of preload*	*Bolting Integrity*	*VII.I-5*	*3.3.1-45*	*B*
--	*Piping*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Cracking*	*TLAA*	*VIII.B1-10*	*3.4.1-01*	*A 0337*
--	*Piping*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Loss of material*	*Flow-Accelerated Corrosion (FAC)*	*VIII.A-17*	*3.4.1-29*	*A*
--	*Piping*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Loss of material*	*One-Time Inspection*	*VIII.A-16*	*3.4.1-02*	*A 0315*

Table 3.3.2-3 Aging Management Review Results – Auxiliary Steam and Station Heating Systems

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
--	Piping	Pressure boundary	Steel	Steam (Internal)	Loss of material	PWR Water Chemistry	VIII.A-16	3.4.1-02	A
--	Piping	Pressure boundary	Steel	Air-indoor uncontrolled (External)	Loss of material	External Surfaces Monitoring	VII.I-8	3.3.1-58	A
--	Trap Body	Pressure boundary	Gray Cast Iron	Steam (Internal)	Loss of material	Flow-Accelerated Corrosion (FAC)	VIII.A-17	3.4.1-29	A
--	Trap Body	Pressure boundary	Gray Cast Iron	Steam (Internal)	Loss of material	One-Time Inspection	VIII.A-16	3.4.1-02	A 0315
--	Trap Body	Pressure boundary	Gray Cast Iron	Steam (Internal)	Loss of material	PWR Water Chemistry	VIII.A-16	3.4.1-02	A
--	Trap Body	Pressure boundary	Gray Cast Iron	Steam (Internal)	Loss of material	Selective Leaching Inspection	VII.F1-18	3.3.1-85	A 0335
--	Trap Body	Pressure boundary	Gray Cast Iron	Air-indoor uncontrolled (External)	Loss of material	External Surfaces Monitoring	VII.I-8	3.3.1-58	A
--	Tubing	Pressure boundary	Steel	Steam (Internal)	Cracking	TLAA	VIII.B1-10	3.4.1-01	A 0337
--	Tubing	Pressure boundary	Steel	Steam (Internal)	Loss of material	Flow-Accelerated Corrosion (FAC)	VIII.A-17	3.4.1-29	A
--	Tubing	Pressure boundary	Steel	Steam (Internal)	Loss of material	One-Time Inspection	VIII.A-16	3.4.1-02	A 0315

Table 3.3.2-3 Aging Management Review Results – Auxiliary Steam and Station Heating Systems

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
--	*Tubing*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Loss of material*	*PWR Water Chemistry*	*VIII.A-16*	*3.4.1-02*	*A*
--	*Tubing*	*Pressure boundary*	*Steel*	*Air-indoor uncontrolled (External)*	*Loss of material*	*External Surfaces Monitoring*	*VII.I-8*	*3.3.1-58*	*A*
--	*Valve Body*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Cracking*	*TLAA*	*VIII.B1-10*	*3.4.1-01*	*A 0337*
--	*Valve Body*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Loss of material*	*Flow-Accelerated Corrosion (FAC)*	*VIII.A-17*	*3.4.1-29*	*A*
--	*Valve Body*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Loss of material*	*One-Time Inspection*	*VIII.A-16*	*3.4.1-02*	*A 0315*
--	*Valve Body*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Loss of material*	*PWR Water Chemistry*	*VIII.A-16*	*3.4.1-02*	*A*
--	*Valve Body*	*Pressure boundary*	*Steel*	*Air-indoor uncontrolled (External)*	*Loss of material*	*External Surfaces Monitoring*	*VII.I-8*	*3.3.1-58*	*A*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.3.2-14	**Page 3.3-332**	**20 New Rows**

In response to RAI 3.3.2.14-2, 20 new rows are added to LRA Table 3.3.2-14, "Aging Management Review Results – Fire Protection System," to read as follows:

Table 3.3.2-14			Aging Management Review Results – Fire Protection System						
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**
				Fire Protection System					
--	*Heat Exchanger (channel) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Steel*	*Air-indoor uncontrolled (External)*	*Loss of material*	*External Surfaces Monitoring*	*VII.G-5*	3.3.1-59	A
--	*Heat Exchanger (channel) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Steel*	*Raw water (Internal)*	*Loss of material*	*Fire Water*	*VII.G-24*	3.3.1-68	C

Table 3.3.2-14 Aging Management Review Results – Fire Protection System									
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**
--	*Heat Exchanger (shell) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Loss of material*	*One-Time Inspection*	*VIII.B1-8*	*3.4.1-37*	*E 0315*
--	*Heat Exchanger (shell) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Steel*	*Steam (Internal)*	*Loss of material*	*PWR Water Chemistry*	*VIII.B1-8*	*3.4.1-37*	*C*
--	*Heat Exchanger (shell) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Steel*	*Air-indoor uncontrolled (External)*	*Loss of material*	*External Surfaces Monitoring*	*VII.G-5*	*3.3.1-59*	*A*

Table 3.3.2-14 Aging Management Review Results – Fire Protection System									
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**
--	*Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Heat transfer*	*Stainless Steel*	*Raw water (Internal)*	*Reduction in heat transfer*	*Collection, Drainage, and Treatment Components Inspection*	*VII.G-7*	*3.3.1-83*	*E*
--	*Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Heat transfer*	*Stainless Steel*	*Steam (External)*	*Reduction in heat transfer*	*One-Time Inspection*	*N/A*	*N/A*	*G 0315*
--	*Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Heat transfer*	*Stainless Steel*	*Steam (External)*	*Reduction in heat transfer*	*PWR Water Chemistry*	*N/A*	*N/A*	*G*

Table 3.3.2-14 Aging Management Review Results – Fire Protection System									
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**
--	*Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Stainless Steel*	*Raw water (Internal)*	*Cracking*	*Collection, Drainage, and Treatment Components Inspection*	*N/A*	*N/A*	*H*
--	*Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Stainless Steel*	*Raw water (Internal)*	*Loss of material*	*Fire Water*	*VII.G-24*	*3.3.1-68*	*C*
--	*Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Stainless Steel*	*Steam (External)*	*Cracking*	*One-Time Inspection*	*VIII.A-10*	*3.4.1-39*	*C* *0315*

Table 3.3.2-14 Aging Management Review Results – Fire Protection System									
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
--	Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)	Pressure boundary	Stainless Steel	Steam (External)	Cracking	PWR Water Chemistry	VIII.A-10	3.4.1-39	C
--	Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)	Pressure boundary	Stainless Steel	Steam (External)	Loss of material	One-Time Inspection	VIII.A-12	3.4.1-37	C 0315
--	Heat Exchanger (tubes) – Fire Water Storage Tank Heat Exchanger (DB-E52)	Pressure boundary	Stainless Steel	Steam (External)	Loss of material	PWR Water Chemistry	VIII.A-12	3.4.1-37	C

Table 3.3.2-14 Aging Management Review Results – Fire Protection System									
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**
--	*Heat Exchanger (tubesheet) - Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Steel*	*Raw water (Internal)*	*Loss of material*	*Fire Water*	*VII.G-24*	*3.3.1-68*	*C*
--	*Heat Exchanger (tubesheet) - Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Steel*	*Steam (External)*	*Loss of material*	*One-Time Inspection*	*VIII.B1-8*	*3.4.1-37*	*C 0315*
--	*Heat Exchanger (tubesheet) - Fire Water Storage Tank Heat Exchanger (DB-E52)*	*Pressure boundary*	*Steel*	*Steam (External)*	*Loss of material*	*PWR Water Chemistry*	*VIII.B1-8*	*3.4.1-37*	*C*
--	*Pump Casing – Fire Water Storage Tank Recirculation Pump (DB-P114)*	*Pressure boundary*	*Gray Cast Iron*	*Raw water (Internal)*	*Loss of material*	*Fire Water*	*VII.G-24*	*3.3.1-68*	*A*

Table 3.3.2-14 Aging Management Review Results – Fire Protection System									
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**
--	*Pump Casing – Fire Water Storage Tank Recirculation Pump (DB-P114)*	*Pressure boundary*	*Gray Cast Iron*	*Raw water (Internal)*	*Loss of material*	*Selective Leaching Inspection*	*VII.G-14*	*3.3.1-85*	*A*
--	*Pump Casing – Fire Water Storage Tank Recirculation Pump (DB-P114)*	*Pressure boundary*	*Gray Cast Iron*	*Air-indoor uncontrolled (External)*	*Loss of material*	*External Surfaces Monitoring*	*VII.I-8*	*3.3.1-58*	*A*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.4.1	**Page 3.4-37**	**Row 3.4.1-39, "Discussion" column, new paragraph**

In response to RAI 3.3.2.14-2, a new paragraph is added to the "Discussion" column of row 3.4.1-39 of LRA Table 3.4.1, "Summary of Aging Management Programs for Steam and Power Conversion Systems Evaluated in Chapter VIII of NUREG-1801," is revised to read as follows:

Table 3.4.1 Summary of Aging Management Programs for Steam and Power Conversion Systems Evaluated in Chapter VIII of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.4.1-39	Stainless steel piping, piping components, and piping elements exposed to steam	Cracking due to stress corrosion cracking	Water Chemistry	No	Consistent with NUREG-1801. Cracking in stainless steel piping, piping components, and piping elements that are exposed to steam is managed by the PWR Water Chemistry Program. *<u>This item is also applied to stainless steel heat exchanger components that are exposed to steam.</u>* In addition, the One-Time Inspection will provide verification of the effectiveness of the PWR Water Chemistry Program to manage cracking.

<u>Affected LRA Section</u>	**<u>LRA Page No.</u>**	**<u>Affected Paragraph and Sentence</u>**
4.3.3.2	**Page 4.3-23**	**New Bulleted Item**

In response to RAI 3.3.2.14-2, a new bulleted item is added to LRA Section 4.3.3.2, "Non-Class 1 Major Components," as follows:

- <u>*The fire water storage tank heat exchanger is the only non-piping component within the evaluation boundaries of the Fire Protection System that exceeds the fatigue threshold temperature. This heat exchanger was fabricated in accordance with ASME Section VIII Division 1.*</u>

  <u>*No fatigue analysis exists for the fire water storage tank heat exchanger, and therefore, there is no TLAA related to fatigue. This component requires no further fatigue evaluation for the period of extended operation.*</u>

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.1.8	**Page A-11**	**2nd paragraph, new last sentence**

In response to Supplemental RAI 2.3.3.18-4, the second paragraph of LRA Section A.1.8, "Closed Cooling Water Chemistry Program," previously revised by FENOC letter dated August 17, 2011 (ML11231A966), is revised to read as follows:

Also, the Closed Cooling Water Chemistry Program includes corrosion rate measurement at selected locations in the closed cooling water systems. In addition, periodic inspections of opportunity will be conducted when components are opened for maintenance, repair, or surveillance, to ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation. A representative sample of piping and components will be inspected on a 10-year interval, with the first inspection taking place prior to entering the period of extended operation. Systems within the scope of this program are monitored for the presence of microbiological activity in accordance with the EPRI Closed-Cycle Cooling Water guidelines. *Component cooling water radiochemistry is sampled on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.1.38	**Pages A-24 & A-25**	**3rd Paragraph**

In response to RAI 3.1.2.2.16-2, the third paragraph of LRA Section A.1.38, "Steam Generator Tube Integrity Program," previously added in response to RAI 3.1.2.2.16-1 by FENOC letter dated October 21, 2011 (ML11298A097), is revised to read as follows:

In addition, cracking due to PWSCC is managed for the steam generator tube-to-tubesheet welds (Alloy 600) by a combination of the PWR Water Chemistry Program and the Steam Generator Tube Integrity Program. The PWR Water Chemistry Program controls peak levels of various contaminants (e.g., dissolved oxygen, chlorides, fluorides, and sulfates) below the system-specific limits that can accelerate cracking for nickel-alloy components. *The Steam Generator Tube Integrity Program includes~~ enhanced visual (EVT-1 or equivalent) examinations to monitor for cracking of the steam generator tube-to-tubesheet welds. The weld inspection sample size includes 20 percent of the subject weld population or a maximum of 25, whichever is less. In this case the maximum of 25 applies since the weld population for the two steam generators is greater than 60,000. Welds included in the inspection sample are scheduled for examination in each 10-year period that occurs during the period of extended operation. Unacceptable inspection findings shall be evaluated by the Corrective Action Program using criteria in accordance with Section XI of the ASME Code. Should the steam generators be replaced in the future with a design such that the tube-to-tubesheet welds are fabricated of Alloy 690-TT material, the examinations will no longer be required.~~ <u>gross visual inspection of the steam generator tube-to-tubesheet welds coupled with eddy-current inspection (i.e., bobbin coil or rotating coil examinations) of the tubes to monitor for cracking and degradation of the tube-to-tubesheet welds (Alloy 600). Acceptance criteria consist of no indication of cracking or relevant conditions of degradation. Gross visual inspection of the tube-to-tubesheet welds are scheduled concurrent with eddy-current inspection of the steam generator tubes. Steam generator tube inspections are scheduled in accordance with Davis-Besse Technical Specification 5.5.8. Should the steam generators be replaced in the future with a design such that the tubes, tubesheet cladding and tube-to-tubesheet welds are fabricated of Alloy 690 material, only the PWR Water Chemistry Program will manage cracking due to PWSCC of the tube-to-tubesheet welds and the gross visual inspection will no longer be required.</u>*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table A-1	**Page A-69**	**Commitment No. 25**

In response to RAI 3.1.2.2.16-2, LRA Table A-1, "Davis-Besse License Renewal Commitments," license renewal future Commitment No. 25, previously added by FENOC letter dated October 21, 2011 (ML11298A097), is replaced in its entirety to read as follows:

Table A-1

Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
25	*Enhance the Steam Generator Tube Integrity Program to:* • *Include gross visual inspection of the steam generator tube-to-tubesheet welds coupled with eddy-current inspection (i.e., bobbin coil or rotating coil examinations) of the tubes to monitor for cracking and degradation of the tube-to-tubesheet welds (Alloy 600), with acceptance criteria consisting of no indication of cracking or relevant conditions of degradation. Schedule gross visual inspection of the tube-to-tubesheet welds concurrent with eddy-current inspection of the steam generator tubes. Schedule steam generator tube inspections in accordance with Davis-Besse Technical Specification 5.5.8. Should the steam generators be replaced in the future with a design such that the tubes, tubesheet cladding and tube-to-tubesheet welds are fabricated of Alloy 690 material,*	*Prior to April 22, 2017*	*LRA* *and* *FENOC Letter L-11-354*	*A.1.38* *B.2.38* *Response to NRC RAI 3.1.2.2.16-2 from NRC Letter dated November 8, 2011*

Table A-1 Davis-Besse License Renewal Commitments				
Item Number	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
	only the PWR Water Chemistry Program will manage cracking due to PWSCC of the tube-to-tubesheet welds and the gross visual inspection will no longer be required.			

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table A-1	**Page A-69**	**Commitment No. 32**

In response to Supplemental RAI 2.3.3.18-4, LRA Table A-1, "Davis-Besse License Renewal Commitments," license renewal future Commitment No. 32, previously added by FENOC letter dated May 24, 2011 (ML11151A090), is revised to include a new bulleted item, and reads as follows:

Table A-1

Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
32	Enhance the Closed Cooling Water Chemistry program to: • Document the results of periodic inspections of opportunity, performed when components are opened for maintenance, repair, or surveillance. • Ensure that a representative sample of piping and components will be inspected on a 10-year interval, with the first inspection taking place prior to entering the period of extended operation. • *Ensure that component cooling water radiochemistry is sampled on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.*	Prior to April 22, 2017	*FENOC Letters L-11-153 and L-11-354*	*Response to NRC RAI B.2.8-1 from NRC Letter dated April 20, 2011 and Supplemental RAI 2.3.3.18-4 from telecon held with the NRC on November 9, 2011*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table A-1	**Page A-69**	**Commitment 36**

License renewal future Commitment 36, replaced in its entirety based on the response to RAI B.2.22-5 in FENOC letter dated September 16, 2011(ML11264A059), is revised in response to Supplemental RAI B.2.22-5. LRA Table A-1, "Davis-Besse License Renewal Commitments," Commitment 36, third bullet, is revised to include inspections of the containment exterior moisture barrier, and the sixth bullet is revised for clarity, to read as follows:

Table A-1

Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
36	Perform the following actions related to the Containment Vessel sand pocket region each refueling outage: • Perform visual inspection of 100 percent of the accessible areas of the wetted outer surface of the Containment Vessel in the sand pocket region. • Perform visual inspection of accessible dry areas of the outer surface of the Containment Vessel in the sand pocket region and the areas above the grout-to-steel interface up to Elevation 566 feet + 3 inches, - 1 inch. • *Perform visual inspection for deterioration (e.g., missing or damaged grout) of accessible grout <u>and the containment exterior moisture barrier</u> in the sand pocket area.* • Perform opportunistic visual inspections of inaccessible areas	Ongoing	*FENOC Letter<u>s</u> L-11-252 <u>and L-11-354</u>*	*Response to NRC RAI B.2.22-5 from NRC Letter dated July 21, 2011 <u>and Supplemental RAI B.2.22-5 from telecons held with the NRC on October 5 and November 14, 2011</u>*

Table A-1

Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
	of the Containment Vessel in the sand pocket region when such areas are made accessible. • Perform opportunistic visual inspections for deterioration (e.g., missing or damaged grout) of inaccessible grout in the sand pocket region when such areas are made accessible. Inaccessible grout is the grout below the normally-exposed surface of the grout in the sand pocket area. • *Address issues of pitting~~,~~ or microbiologically-influenced corrosion (MIC), and degraded grout, moisture barrier or sealant identified during the inspections using the FENOC Corrective Action Program.* • Sample the water in the sand pocket region when sufficient volumes are available. The number of sampled water volumes will be determined by the number of water volumes observed and the size of those water volumes. Analyze the sample(s) for pH, chlorides, iron and sulfates. Treat or wash (or a combination thereof) the sand pocket area to reduce measured chloride concentrations to less than 250 parts per million (ppm) if the concentration of chlorides in a sample exceeds 250 ppm. Note: Water samples may be taken at different times during each outage. Engineering judgment may be used to determine the priority of the chemical analyses to be performed if sufficient water is not available in a given sample for all analyses.			

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
B.2.8	**Pages B-44 and B-45**	**Program Description subsection, second paragraph, new last sentence; and, Enhancements subsection, new enhancement**

In response to Supplemental RAI 2.3.3.18-4, the second paragraph of LRA Section B.2.8, "Closed Cooling Water Chemistry Program," the "Program Description" subsection, and previously revised by FENOC letter dated August 17, 2011 (ML11231A966), is revised to include a new last sentence. Also, a new enhancement is added to the Enhancements subsection under the existing heading of "Parameters Monitored or Inspected," previously revised by FENOC letter dated May 24, 2011 (ML11151A090). The affected subsections read as follows:

Program Description

Also, the Closed Cooling Water Chemistry Program includes corrosion rate measurement at selected locations in the closed cooling water systems. In addition, periodic inspections of opportunity will be conducted when components are opened for maintenance, repair, or surveillance, to ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation. A representative sample of piping and components will be inspected on a 10-year interval, with the first inspection taking place prior to entering the period of extended operation. Systems within the scope of this program are monitored for the presence of microbiological activity in accordance with the EPRI Closed-Cycle Cooling Water guidelines. *Component cooling water radiochemistry will be sampled on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.*

Enhancements

- ***Parameters Monitored or Inspected***

  The Closed Cooling Water Chemistry program will be enhanced to ensure that component cooling water radiochemistry is sampled on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
B.2.38	**Page B-151**	**Program Description subsection, 4th paragraph; and, Enhancements subsection**

In response to RAI 3.1.2.2.16-2, LRA Section B.2.38, "Steam Generator Tube Integrity Program," the fourth paragraph of subsection "Program Description" and the "Enhancements" subsection, previously added in response to RAI 3.1.2.2.16-1 by FENOC letter dated October 21, 2011 (ML11298A097), are revised to read as follows:

Program Description

In addition, cracking due to PWSCC will be managed for the steam generator tube-to-tubesheet welds (Alloy 600) by a combination of the PWR Water Chemistry Program and the Steam Generator Tube Integrity Program. The PWR Water Chemistry Program controls peak levels of various contaminants (e.g., dissolved oxygen, chlorides, fluorides, and sulfates) below the system-specific limits that can accelerate cracking for nickel-alloy components. *The Steam Generator Tube Integrity Program will include ~~enhanced visual (EVT-1 or equivalent) examinations to monitor for cracking of the steam generator tube-to-tubesheet welds. The weld inspection sample size will include 20 percent of the subject weld population or a maximum of 25, whichever is less. In this case the maximum of 25 applies since the weld population for the two steam generators is greater than 60,000. Welds included in the inspection sample will be scheduled for examination in each 10-year period that occurs during the period of extended operation. Unacceptable inspection findings will be evaluated by the Corrective Action Program using criteria in accordance with Section XI of the ASME Code. Should the steam generators be replaced in the future with a design such that the tube-to-tubesheet welds are fabricated of Alloy 690-TT material, the examinations will no longer be required.~~ <u>gross visual inspection of the steam generator tube-to-tubesheet welds coupled with eddy-current inspection (i.e., bobbin coil or rotating coil examinations) of the tubes to monitor for cracking and degradation of the tube-to-tubesheet welds (Alloy 600). Acceptance criteria will consist of no indication of cracking or relevant conditions of degradation. Gross visual inspection of the tube-to-tubesheet welds will be scheduled concurrent with eddy-current inspection of the steam generator tubes. Steam generator tube inspections are scheduled in accordance with Davis-Besse Technical Specification 5.5.8. Should the steam generators be replaced in the future with a design such that the tubes, tubesheet cladding and tube-to-tubesheet welds are fabricated of Alloy 690 material, only the PWR Water Chemistry Program will manage cracking due to PWSCC of the tube-to-tubesheet welds and the gross visual inspection will no longer be required.</u>*

Enhancements

The following enhancement will be implemented in the identified program elements prior to the period of extended operation.

- **Scope, Parameters Monitored or Inspected, Detection of Aging Effects, Acceptance Criteria**

  The Steam Generator Tube Integrity Program will include ~~enhanced visual (EVT-1 or equivalent) examinations to monitor for cracking of the steam generator tube-to-tubesheet welds (Alloy 600). The weld inspection sample size will include 20 percent of the subject weld population or a maximum of 25, whichever is less. In this case the maximum of 25 applies since the weld population for the two steam generators is greater than 60,000. Welds included in the inspection sample will be scheduled for examination in each 10-year period that occurs during the period of extended operation. Unacceptable inspection findings will be evaluated by the Corrective Action Program using criteria in accordance with Section XI of the ASME Code. Should the steam generators be replaced in the future with a design such that the tube-to-tubesheet welds are fabricated of Alloy 690-TT material, the examinations will no longer be required.~~ gross visual inspection of the steam generator tube-to-tubesheet welds coupled with eddy-current inspection (i.e., bobbin coil or rotating coil examinations) of the tubes to monitor for cracking and degradation of the tube-to-tubesheet welds (Alloy 600). Acceptance criteria will consist of no indication of cracking or relevant conditions of degradation. Gross visual inspection of the tube-to-tubesheet welds will be scheduled concurrent with eddy-current inspection of the steam generator tubes. Steam generator tube inspections are scheduled in accordance with Davis-Besse Technical Specification 5.5.8. Should the steam generators be replaced in the future with a design such that the tubes, tubesheet cladding and tube-to-tubesheet welds are fabricated of Alloy 690 material, only the PWR Water Chemistry Program will manage cracking due to PWSCC of the tube-to-tubesheet welds and the gross visual inspection will no longer be required.

Enclosure B

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

Letter L-11-354

New and Revised DBNPS License Renewal Application Boundary Drawings

6 Pages follow

The following License Renewal Application Boundary Drawings are new and are enclosed:

LR Drawing LR-M006F	**Revision 0**
LR Drawing LR-M020C	**Revision 0**

The following License Renewal Application Boundary Drawings are revised and are enclosed:

LR Drawing LR-M003B	**Revision 2**
LR Drawing LR-M016A	**Revision 3**
LR Drawing LR-M020A	**Revision 2**
LR Drawing LR-M020D	**Revision 2**

The 6 Drawings specifically referenced in Enclosure B have been processed into ADAMS

These drawings can be accessed by the NRC staff within the ADAMS package or by performing a search on the Document/Report Number