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October 21, 2011
L-11-317

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. 20

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 letter dated September 22, 2011 (ADAMS Accession No. ML 11256A149), from Inspector questions during the Region III Inspection Procedure IP-71002 License Renewal Inspection follow-up held the week of August 22, 2011, and by telephone conference calls held on September 16 and October 5, 2011, the Nuclear Regulatory Commission (NRC) requested additional information to complete its review of the License Renewal Application (LRA).

The Attachment provides the FENOC reply to the NRC requests for additional information (RAIs). The NRC request is shown in bold text followed by the FENOC response. The Enclosure provides Amendment No. 20 to the DBNPS LRA.

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NRR

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 October 21, 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 2.3.3, 3.1.2.2, 3.3.2.2, 4.1 and B.2.39

Enclosure:

Amendment No. 20 to the DBNPS License Renewal Application

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

Section 3.1.2.2

Question RAI 3.1.2.2.16-1

Background:

By its letter dated August 17, 2011, the applicant addressed its review results on cracking due to primary water stress corrosion cracking (PWSCC) of steam generator nickel alloy tube-to-tubesheet welds in response to the discussion held in a teleconference call dated July 13, 2011.

In the letter, the applicant stated that upon further review after the conference call with the U.S. Nuclear Regulatory Commission (NRC), it determined that the tube-to-tubesheet welds (Alloy 600 welds) for its steam generators do not have a license renewal intended function and therefore, are not subject to an aging management review. The applicant also stated that the steam generators are Babcock & Wilcox Model 177-FA, once-through design and the tubes and the tubesheets of the steam generators form the pressure boundary between the fluid in the secondary system and the reactor coolant system. The applicant further stated that as provided in Updated Safety Analysis Report (USAR) Section 5.5.2.3, the tubes are expanded (to a partial depth) into the tubesheet and the tubes are seal welded to the tubesheet near the tube ends. In addition, the applicant stated that the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, 1995 Edition with 1996 Addenda, IWA-9000 defines a seal weld as a nonstructural weld intended to prevent leakage, where the strength is provided by a separate means. The applicant stated that the "separate means" in this case is the tube-to-tubesheet expansion joint which forms the pressure boundary and that the tube-to-tubesheet welds are seal welds and therefore, are not part of the pressure boundary.

Issue:

The applicant stated the tube-to-tubesheet welds (Alloy 600 welds) for its steam generators do not have a license renewal intended function and therefore, are not subject to an aging management review. The applicant also stated that the tube-to-tubesheet welds are seal welds and therefore, are not part of the pressure boundary. However, the staff noted that the reactor coolant pressure boundary should provide structural and leak-tight integrity. Furthermore, the applicant's statement that the tube-to-tubesheet welds are intended to prevent leakage indicates that these welds perform the intended function of the reactor coolant

pressure boundary. Therefore, the staff found a need to confirm whether or not the design analysis of the applicant's once-through steam generators, which was used to establish the current licensing basis (CLB), concludes the following: the interference fits between the tubes and the tubesheets are sufficient to ensure the structural and leak-tight integrity of the tube-to-tubesheet joints, without a need for crediting the tube-to-tubesheet welds.

Request:

1) **Confirm whether or not the design analysis, which was used to establish the CLB, concludes that the interference fits are sufficient to ensure the structural and leak-tight integrity of the tube-to-tubesheet joints, without a need for crediting the tube-to-tubesheet welds.**

   If the design analysis concludes that the interference fits are sufficient to ensure the structural and leak-tight integrity, provide the technical basis of the conclusion and list the reference(s) addressing the technical basis.

2) **If the design analysis, which was used to establish the CLB, credits the tube-to-tubesheet welds for ensuring the structural and leak-tight integrity of the tube-to-tubesheet joints, describe how cracking due to PWSCC will be managed for the steam generator tube-to-tubesheet welds.**

RESPONSE RAI 3.1.2.2.16-1

The response titled "Supplemental Response – steam generator aging management review tube-to-tubesheet weld" provided in FENOC letter dated August 17, 2011 (ML11231A966), is replaced in its entirety by the following response.

1) Although the steam generator tube-to-tubesheet weld is classified as a seal weld, FENOC has confirmed that the design analyses used to establish the current licensing basis credits both the interference fit (between the tube and tubesheet) and the tube-to-tubesheet weld for structural and leak-tight integrity. Therefore, the tube-to-tubesheet welds have a License Renewal intended function of "Pressure boundary" and are subject to an aging management review.

   LRA Table 3.1.2-4, "Aging Management Review Results – Steam Generators" is revised to include the aging management review results. In addition, LRA Table 2.3.1-4, "Steam Generator Components Subject to Aging Management Review," is revised to list the tube-to-tubesheet weld with an intended function of "Pressure boundary."

2) 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 be enhanced to include enhanced visual (EVT-1 or equivalent) examinations to monitor for cracking of the steam generator tube-to-tubesheet welds.

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. In this case the maximum of 25 applies since the weld population for the two steam generators is greater than 60,000. The sample size is consistent with other NUREG-1801 programs where the inspection is designed to provide assurance that aging is not occurring. Welds included in the inspection sample will be scheduled for examination in each 10-year period that occurs during the period of extended operation. 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.

LRA Section A.1.38, Section B.2.38, Table A-1 and Table B-2 are revised consistent with this response.

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

Table 2.3.3

Question RAI 2.3.3.18-4

Background:

In its response to RAI 2.3.3.18-3 dated August 17, 2011, the applicant provided the following information:

1) **The letdown coolers performed acceptably from initial startup in 1978 until 1991, when plant personnel detected contamination in the component cooling water (CCW) system, and replaced both letdown coolers in 1993. Then, in 2009, plant personnel identified a small, active reactor coolant leak, and again replaced both letdown coolers in 2010.**

2) A failure analysis had not been performed on the leaking letdown coolers to determine the specific leak location or to verify the failure mechanism because of high radiation dose rates associated with that effort.

SRP-LR Section A.1.2.3.4, "Detection of Aging Effects," states that nuclear power plants are licensed using the principles of redundancy, and diversity, and that degraded components reduce the reliability of the systems, challenge safety systems, and contribute to plant risk. The SRP-LR continues by stating that the effects of aging on a component should be managed to ensure its availability to perform its intended function(s) as designed when called upon, and notes that a program based solely on detecting component failure should not be considered as an effective aging management program for license renewal.

Issue:

Based on the information provided in this recent response, as well as the information provided in response to RAI 2.3.3.18-2 for the same issue, the staff did not consider that the applicant has provided sufficient bases to justify the replacement frequency of every seventh refueling outage (approximately 14 years) for the letdown coolers in the makeup and purification system.

The bases for the staff's position are as follows:

a) The applicant established the replacement frequency based on a qualified life, which was empirically derived using two plant-specific data points of 13 and 16 years, after identifying reactor coolant leakage into the component cooling water system.

b) The applicant has not determined the flaw location, performed flaw sizing, or verified flaw characteristics to allow prediction of flaw stability or growth rate. Without having this information, operation of the letdown cooler with ongoing leakage is risking a failure, which would challenge the pressure relief capability of the component cooling water system and the isolation function of the valves in the makeup and purification system.

c) While past operating experience (although limited) may have shown that the flaw was stable for some period of time, the replacement frequency determination did not appear to consider normal operational pressure transients that the letdown coolers would be expected to experience.

d) The letdown cooler replacement frequency appears to be based on overall calendar time and not actual operational time, considering both refueling and extended outages.

Request:

Provide a letdown cooler replacement frequency that includes adequate margin to initiation of tube leakage and provide the basis for the margin, or propose an aging management program that will adequately manage these components that are within the scope of license renewal.

RESPONSE RAI 2.3.3.18-4

The responses to RAI 3.3.2.2.4-1 and RAI 2.3.3.18-2 submitted in FENOC letter dated June 3, 2011 (ML11159A132), and the response to RAI 2.3.3.18-3 submitted in FENOC letter dated August 17, 2011 (ML11231A966), are 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 Pressurized Water Reactor (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, eddy current testing of tubes for managing cyclic loading is therefore not applicable. The temperature and radioactivity monitoring of shell side water is performed by installed instrumentation.

FENOC withdraws license renewal future Commitment 25 of LRA Table A-1.

LRA Section 2.3.3.18, Table 2.3.3-18, Section 3.3.2.1.18, Section 3.3.2.2.4.1, Table 3.3.1, Table 3.3.2-18 and Table A-1 are revised consistent with this response.

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

Table 3.3.2.2

Question RAI 3.3.2.2.10.4-1

Background:

SRP-LR Table 3.3-1, item 26 states that loss of material due to pitting and crevice corrosion could occur for copper alloy piping, piping components, and piping elements exposed to lubricating oil. The SRP-LR recommends GALL AMP XI.M39, "Lubricating Oil Analysis," to manage the aging effect and further evaluation of a program to verify the effectiveness of the Lubricating Oil Analysis Program, such as XI.M32, "One-Time Inspection," because control of contaminants within the lubricating oil may not have always been adequate to preclude corrosion.

In LRA Tables 3.3.2-14, 3.3.2-18, 3.3.2-30, 3.4.2-1, and 3.4.2-4, the applicant referenced LRA Table 3.3.1, item 3.3.1-26 and generic note I for copper alloy components exposed to lubricating oil and stated that the components have no aging effects requiring management. For these items, the applicant further cited plant-specific notes which state that the components are made of copper alloy with less than 15 percent zinc and are not in contact with a more cathodic metal; therefore, the components have no aging effects requiring management.

Issue:

It is unclear to the staff why the applicant claims that components of copper alloy with less than 15 percent zinc exposed to lubricating oil have no aging effects requiring management. The staff noted that components of copper alloy with less than 15 percent zinc are less susceptible to loss of material than other copper alloys, but that the presence of contaminants (e.g., water) in lubricating oil can create an environment conducive to loss of material, regardless of whether or not the component is in contact with a more cathodic metal.

Request:

Explain why components of copper alloy with less than 15 percent zinc exposed to lubricating oil have no aging effects requiring management or provide an appropriate AMP to manage loss of material.

RESPONSE RAI 3.3.2.2.10.4-1

The LRA is revised to identify loss of material due to pitting and crevice corrosion as an aging effect requiring management for copper alloy components with less than 15 percent zinc exposed to lubricating oil. The Lubricating Oil Analysis Program will be used to manage the aging effect of loss of material. The One-Time Inspection will provide verification of the effectiveness of the Lubricating Oil Analysis program to manage loss of material.

LRA Section 3.3.2.2.10.4, Tables 3.3.1, 3.3.2-1, 3.3.2-14, 3.3.2-18, and 3.3.2-30, Table 3.3.2 Plant-Specific Notes, Section 3.4.2.2.7.3, Tables 3.4.1, 3.4.2-1 and 3.4.2-4, and Table 3.4.2 Plant-Specific Notes are revised consistent with this response.

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

Section 4.1

Supplemental Question RAI 4.1-2

The NRC initiated a telephone conference call with FENOC on September 16, 2011, to discuss the FENOC response to RAI 4.1-2 submitted by FENOC letter dated August 17, 2011 (ML11231A966). In the response, FENOC stated that 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. The NRC staff's concern is that the applicant's basis may be predicated on charpy or thermal aging data that are not up-to-date or conservative when compared to the most recent data for the state of the industry.

It is not clear to the staff whether the assumption that "the lower bound fracture toughness of 139 ksi√in that bounds the saturated fracture toughness of the applicant's materials" remains valid.

To address the NRC staff's concern that the applicant's basis may be predicated on charpy or thermal aging data that are not up-to-date or conservative when compared to the most recent data for the state of the industry, FENOC agreed to compare the thermal aging data used in the ASME Code Case N-481 Evaluation to the most-recent industry data (i.e., NUREG/CR- 4513, Rev. 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," and NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds"), and provide the results in a supplemental response to RAI 4.1-2.

SUPPLEMENTAL RESPONSE RAI 4.1-2

The fracture toughness of the cast austenitic stainless steel is not time dependent as the Davis-Besse ASME Code Case N-481 evaluation [LRA Reference 4.8-18 (SIR-99-040, Revision 1)] used a lower bound fracture toughness value of 139 ksi√in that bounds the saturation fracture toughness of the Davis-Besse material.

The saturation fracture toughness was determined using the methodology outlined in NUREG/CP-0119, Vol. 2, pp. 151-178, "Proceedings of the U.S. Nuclear Regulatory Commission, 19th Water Reactor Safety Information Meeting held at Bethesda, MD, October 28-30, 1991," and considering all available certified material test reports (CMTRs) for the base material and welds of the Davis-Besse reactor coolant pump casings. The saturation fracture toughness value of 139 ksi√in was the minimum calculated for all the CMTRs considered in the evaluation. This minimum saturation fracture toughness value has since been calculated using NUREG/CR- 4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," the most recent publication on this subject. Using the methodology and correlation in this latest NUREG results in the same minimum saturation fracture toughness value for the pump casings.

The fracture toughness for welds considering thermal aging has also been presented in NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds." A conservative J_{1c} fracture toughness value of 40 KJ/m^2 based on the absolute minimum of all available data is provided in this document for aged stainless steel welds; this J_{1c} fracture toughness value translates to 80 ksi√in. This conservative fracture toughness value still bounds the calculated total applied stress intensity factors calculated in Table 4-5 of SIR-99-040, Revision 1, indicating that the conclusions of SIR-99-040, Revision 1, are unchanged even if the methodology outlined in NUREG-CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds," is used for the Davis-Besse pump casing welds.

New LRA Sections 4.7.6 and A.2.7.5, previously submitted by FENOC letter dated August 17, 2011 (ML11231A966), are revised consistent with the above response.

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

Section B.2.39

Supplemental Question RAI B.2.39-9

The NRC staff initiated a telephone conference with FENOC on October 5, 2011, to discuss the FENOC response to RAI B.2.39-9 submitted in FENOC letter dated September 16, 2011 (ML11264A059), regarding operating experience with borated water leakage from the reactor Refueling Canal. The NRC staff requested additional information about three subjects:

1. **Details of the FENOC response to the year 2003 report, "Engineering Assessment Report – Refueling Canal Leakage," recommendations.**

2. **The structural integrity of concrete in containment affected by the borated water leakage, including why it is acceptable to wait until year 2014 to again verify structural integrity of the concrete.**

3. **The rate of the borated water leakage when the Refueling Canal is filled.**

SUPPLEMENTAL RESPONSE RAI B.2.39-9

1. In the year 2003, FENOC initiated two Condition Reports for evaluation of the recommendations included in the 2003 assessment report, "Engineering Assessment Report – Refueling Canal Leakage." For the Refueling Canal leakage, the initial focus of the Condition Report corrective actions was to continue the use of non-destructive examination methods for identification of the leak locations and to repair the leaks. After the corrective actions were assigned, evaluation of industry operating experience had shown that non-destructive examination methods such as vacuum box testing and liquid penetrant examinations were marginally effective. Therefore, resources were used instead on sealing the potential leaks at welds previously identified as suspect in the 2003 assessment report.

   In the year 2005, an epoxy coating was applied to areas suspected of having leaks. In the year 2008, a drawing review identified some grafoil washers that may need to be replaced. A work request was initiated to replace the grafoil washers, but that request has not yet been implemented. In the year 2010, FENOC determined that the epoxy coating was ineffective because the epoxy coating did not stay bonded to the Refueling Canal liner. Therefore, a new plan had to be developed to mitigate the leakage. From March through May of 2010, staged fills of the Refueling Canal were conducted to try to narrow-down the Refueling Canal liner areas that could have sources of leakage.

2. The 2003 assessment report documented that Refueling Canal leakage had negligible impact on the structural integrity of the concrete structures in containment. The leakage was documented at least four years prior to 2003 and occurred only when the Refueling Canal was filled. Since 2003, periodic visual inspections of concrete surface areas affected by the leakage have shown no additional evidence of further damage to rebar or concrete. Therefore, it is acceptable to wait until the year 2014 to perform additional structural integrity verifications in addition to periodic visual inspections. This conclusion is based on plant-specific and industry operating experience with the effects of borated water leakage on concrete and reinforcing steel.

   The plant-specific operating experience included non-destructive testing and petrographic examination of core drills from the east/west tunnel performed in the year 2002 that revealed no significant degradation of the concrete or reinforcing steel. In addition, information about related research and industry operating experience exists that documents the relatively minor effect that borated water leakage, through concrete, has on concrete and carbon steel reinforcing bars. For example, EPRI Report 1019168, "Boric Acid Attack of Concrete and Reinforcing Steel in PWR Fuel Handling Buildings," noted that "reinforcing steel degradation" from boric acid "is minimal." Also, industry operating experience shows that, since the year 1993, another plant has had substantially more refueling canal leakage (i.e., 3-to-7 gallons per minute) than Davis-Besse. As of 2010, with continuing leakage of 3-to-7 gallons per minute when the refueling canal contains water, the industry operating experience states that there had been no known degradation related to the refueling canal leakage.

3. During the current Davis-Besse Cycle 17 mid-cycle outage, the rate of Refueling Canal leakage is estimated at less than 0.2 gallons per minute. This estimate is based on monitoring the volume of Refueling Canal makeup water adjusted by other identified leakage and evaporation losses.

Supplemental Question RAI OIN-380

During the NRC Region III Inspection Procedure (IP) 71002, "License Renewal Inspection," held the week of August 22, 2011, the NRC staff requested that the Davis-Besse License Renewal Structures Monitoring Program include enhancements to the descriptions of the Parameters Monitored or Inspected, the Detection of Aging Effects and the Acceptance Criteria. The FENOC License Renewal Project created Open Item Number (OIN)-380 to track the request, described as follows:

1. **Include the following enhancement to the Parameters Monitored or Inspected program element for the Structures Monitoring Program: Elastomeric**

vibration isolators and structural sealants are monitored for cracking, loss of material, and hardening.

2. **Include the following enhancement to the Detection of Aging Effects program element for the Structures Monitoring Program: Visual inspection of elastomeric vibration isolation elements will be supplemented by feel to detect hardening if the vibration isolation function is suspect.**

3. **Include the following enhancements to the Acceptance Criteria program element for the Structures Monitoring Program: Loose bolts and nuts and cracked high strength bolts are not acceptable unless accepted by engineering evaluation. Structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result in loss of sealing. Elastomeric vibration isolation elements are acceptable if there is no loss of material, cracking, or hardening that could lead to the reduction or loss of isolation function.**

SUPPLEMENTAL RESPONSE RAI OIN-380

LRA Section B.2.39, "Structures Monitoring Program," and Table A-1, "Davis-Besse License Renewal Commitments," license renewal future Commitment 20, are revised to include program procedure enhancements and new license renewal future commitments for:

1. Parameters Monitored or Inspected for the Structures Monitoring Program: Elastomeric vibration isolators and structural sealants are monitored for cracking, loss of material, and hardening.

2. Detection of Aging Effects for the Structures Monitoring Program: Visual inspection of elastomeric vibration isolation elements will be supplemented by feel to detect hardening if the vibration isolation function is suspect.

3. Acceptance Criteria for the Structures Monitoring Program: Loose bolts and nuts and cracked high strength bolts are not acceptable unless accepted by engineering evaluation. Structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result in loss of sealing. Elastomeric vibration isolation elements are acceptable if there is no loss of material, cracking, or hardening that could lead to the reduction or loss of isolation function.

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

Enclosure

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

Letter L-11-317

Amendment No. 20 to the
DBNPS License Renewal Application

Page 1 of 49

License Renewal Application
Sections Affected

Section 2
Table 2.3.1-4
Section 2.3.3.18
Table 2.3.3-18

Section 3
Section 3.1.2.2.16.1
Table 3.1.1
Table 3.1.2-3
Table 3.1.2-4
Section 3.3.2.1.18
Section 3.3.2.2.4.1
Section 3.3.2.2.10.4
Table 3.3.1
Table 3.3.2-1
Table 3.3.2-14
Table 3.3.2-18
Table 3.3.2-30
Table 3.3.2 P-S Notes
Section 3.4.2.2.7.3
Table 3.4.1
Table 3.4.2-1
Table 3.4.2-4
Table 3.4.2 P-S Notes

Section 4
Section 4.7.6

Appendix A
Section A.1.38
Section A.2.7.5
Table A-1

Appendix B
Table B-2
Section B.2.38
Section B.2.39

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
Table 2.3.1-4	**Page 2.3-21**	**New Row**

In response to RAI 3.1.2.2.16-1, LRA Table 2.3.1-4, "Steam Generators Components Subject to Aging Management Review," is revised to include a new row which reads as follows:

Component Type	Intended Function (as defined in Table 2.0-1)
Primary Side; Tube-to-tubesheet Weld	*Pressure boundary*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Section 2.3.3.18	**Page 2.3-105**	**Components Subject to AMR subsection, 2nd bulleted item**

In response to RAI 2.3.3.18-4, LRA Section 2.3.3.18, "Steam Makeup and Purification System," subsection "Components Subject to AMR," the second bulleted item, is deleted as follows:

- ~~*The letdown coolers (DB-E25-1 & 2) are replaced periodically, and are evaluated as short-lived components (consumables). Therefore, the letdown coolers (DB-E25-1 & 2) are not subject to AMR.*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 2.3.3-18	**Page 2.3-106**	**2 New Rows**

In response to RAI 2.3.3.18-4, LRA Table 2.3.3-18, "Makeup and Purification System Components Subject to Aging Management Review," is revised to include two new rows which read as follows:

Component Type	Intended Function (as defined in Table 2.0-1)
Heat Exchanger (channel, shell, tubesheet) – Letdown coolers (DB-E25-1 & 2)	*Pressure boundary*
Heat Exchanger (tubes) – Letdown coolers (DB-E25-1 & 2)	*Heat transfer* *Pressure boundary*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
3.1.2.2.16.1	**Page 3.1-11**	**New [last] sentence**

In response to RAI 3.1.2.2.16-1, a new sentence is added to the end of LRA Section 3.1.2.2.16.1, "Stainless Steel or Nickel-Alloy Steam Generator Components – Reactor Coolant," and the section is revised to read:

3.1.2.2.16.1 Stainless Steel or Nickel-Alloy Steam Generator Components – Reactor Coolant

Cracking due to SCC could occur on the primary coolant side of stainless steel, stainless steel clad, and nickel-alloy clad components. Cracking due to SCC (including PWSCC) on the primary coolant side of Davis-Besse stainless steel, stainless steel clad, and nickel-alloy clad components is managed by the Inservice Inspection Program, Nickel-Alloy Management Program and PWR Water Chemistry Program. *Cracking due to SCC (including PWSCC) in the nickel alloy steam generator tube-to-tubesheet welds is managed by the Steam Generator Tube Integrity Program and PWR Water Chemistry Program.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.1.1	**Page 3.1-26**	**Row 3.1.1-35 "Discussion" column, 2nd paragraph, last sentence**

Text in "Discussion" column of Row 3.1.1-35 is revised based on the response to RAI 3.1.2.2.16-1, and LRA Table 3.1.1, "Summary of Aging Management Programs for Reactor Vessel, Internals, Reactor Coolant System and Reactor Coolant Pressure Boundary, and Steam Generators Evaluated in Chapter IV of NUREG-1801," reads as follows:

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, Reactor Coolant System and Reactor Coolant Pressure Boundary, and Steam Generators Evaluated in Chapter IV of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-35	Steel with stainless steel or nickel alloy cladding primary side components; steam generator upper and lower heads, tubesheets and tube-to-tube sheet welds	Cracking due to stress corrosion cracking and primary water stress corrosion cracking	Inservice Inspection (IWB, IWC, and IWD) and Water Chemistry and for nickel alloy, FSAR supplement commitment to implement applicable plant commitments to (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitment needs to be confirmed	Consistent with NUREG-1801. Cracking due to SCC (including PWSCC) in steel steam generator components with stainless steel or nickel alloy cladding is managed by the Inservice Inspection Program and PWR Water Chemistry Program. ~~Davis-Besse has no nickel-alloy components that refer to this item.~~ <u>*Cracking due to SCC (including PWSCC) in the nickel alloy steam generator tube-to-tubesheet welds is managed by the Steam Generator Tube Integrity*</u>

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, Reactor Coolant System and Reactor Coolant Pressure Boundary, and Steam Generators Evaluated in Chapter IV of NUREG-1801					
Item Number	**Component/Commodity**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
					Program and PWR Water Chemistry Program. Further evaluation is documented in Section 3.1.2.2.16.1.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.1.2-3	**Page 3.1-163**	**8 New Rows**

Eight new rows were added to LRA Table 3.1.2-3, "Aging Management Review Results – Reactor Coolant System and Reactor Coolant Pressure Boundary," and provided in FENOC letter L-11-292 dated October 7, 2011, in response to Supplemental RAI Table 3.1.2-3. However, the table title provided in that response was incorrectly shown as "Aging Management Review Results – Decay Heat Removal and Low Pressure Injection System." The eight new rows added to Table 3.1.2-3 are provided with the corrected table title as follows:

Table 3.1.2-3 Aging Management Review Results – Reactor Coolant System and Reactor Coolant Pressure Boundary

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 <4 inches RV flange leakage line tap weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant (Internal)*	*Cracking - Fatigue*	*TLAA*	*IV.C2-25*	*3.1.1-08*	*A*
--	*Piping <4 inches RV flange leakage line tap weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant (Internal)*	*Cracking - Flaw Growth*	*Inservice Inspection*	*IV.C2-26*	*3.1.1-62*	*C* *0102* *0103*
--	*Piping <4 inches RV flange leakage line tap weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant (Internal)*	*Cracking - PWSCC, SCC/IGA*	*Inservice Inspection*	*IV.C2-13*	*3.1.1-31*	*A*

Table 3.1.2-3	Aging Management Review Results – Reactor Coolant System and Reactor Coolant Pressure Boundary								
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 <4 inches RV flange leakage line tap weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant (Internal)*	*Cracking - PWSCC, SCC/IGA*	*Nickel-Alloy Management*	*IV.C2-13*	*3.1.1-31*	*A 0110*
--	*Piping <4 inches RV flange leakage line tap weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant (Internal)*	*Cracking - PWSCC, SCC/IGA*	*PWR Water Chemistry*	*IV.C2-13*	*3.1.1-31*	*A*
--	*Piping <4 inches RV flange leakage line tap weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant (Internal)*	*Cracking - PWSCC, SCC/IGA*	*Small Bore Class 1 Piping Inspection*	*IV.C2-13*	*3.1.1-31*	*E*
--	*Piping <4 inches RV flange leakage line tap weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant (Internal)*	*Loss of Material*	*PWR Water Chemistry*	*IV.C2-15*	*3.1.1-83*	*A*
--	*Piping <4 inches RV flange leakage line tap weld*	*Pressure boundary*	*Nickel Alloy*	*Air with borated water leakage (External)*	*None*	*None*	*IV.E-3*	*3.1.1-86*	*A 0103*

Affected LRA Section | **LRA Page No.** | **Affected Paragraph and Sentence**

Table 3.1.2-4 | **Page 3.1-185** | **4 New Rows**

In response to RAI 3.1.2.2.16-1, four new rows are added to LRA Table 3.1.2-4, "Aging Management Review Results – Steam Generators," to read as follows:

Table 3.1.2-4 Aging Management Review Results – Steam Generators									
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**
--	*Primary Side; Tube-to-tubesheet Weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant*	*Cracking - Fatigue*	*TLAA*	*IV.D2-15*	*3.1.1-06*	*C 0101*
--	*Primary Side; Tube-to-tubesheet Weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant*	*Cracking - PWSCC, SCC/IGA*	*PWR Water Chemistry*	*IV.D2-4*	*3.1.1-35*	*A 0101*
--	*Primary Side; Tube-to-tubesheet Weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant*	*Cracking - PWSCC, SCC/IGA*	*Steam Generator Tube Integrity*	*IV.D2-4*	*3.1.1-35*	*E 0101*
--	*Primary Side; Tube-to-tubesheet Weld*	*Pressure boundary*	*Nickel Alloy*	*Borated reactor coolant*	*Loss of Material*	*PWR Water Chemistry*	*IV.C2-15*	*3.1.1-83*	*C 0101*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
3.3.2.1.18	**Page 3.3-24**	**"Environments" subsection, new bulleted item**

In response to RAI 2.3.3.18-4 related to the letdown coolers, the "Environments" subsection of LRA Section 3.3.2.1.18, "Makeup and Purification System," is revised to include a new environment in the "Environments" subsection as follows:

- *Closed cycle cooling water > 60°C (> 140°F)*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
3.3.2.2.4.1	**Page 3.3-40**	**Entire section**

In response to 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 June 3, 2011 (ML11159A132), 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 Auxiliary Systems do not contain stainless steel nonregenerative heat exchanger components that are exposed to treated borated water greater than 60°C (> 140°F) and subject to aging management review; therefore, this item is not applicable to Davis-Besse.*~~ *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.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
3.3.2.2.10.4	**Page 3.3-45**	**1st and 2nd sentences**

In response to RAI 3.3.2.2.10.4-1, the first two sentences of LRA Section 3.3.2.2.10.4, "Copper Alloy Piping, Piping Components, and Piping Elements – Lubricating Oil," are revised to read as follows:

3.3.2.2.10.4 Copper Alloy Piping, Piping Components, and Piping Elements – Lubricating Oil

Loss of material due to pitting and crevice corrosion could occur for copper alloy piping, piping components, and piping elements exposed to lubricating oil. Loss of material due to pitting and crevice corrosion for Davis-Besse copper alloy piping, piping components, and piping elements ~~with a zinc content greater than 15%~~ that are exposed to lubricating oil is managed by the Lubricating Oil Analysis Program. Loss of material for copper alloy heat exchanger components ~~with a zinc content greater than 15%~~ that are exposed to lubricating oil is also managed by the Lubricating Oil Analysis Program. The Lubricating Oil Analysis Program manages loss of material through periodic monitoring and control of contaminants, including water. The One-Time Inspection will provide verification of the effectiveness of the Lubricating Oil Analysis Program to manage loss of material.

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 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 June 3, 2011 (ML11159A132), 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	~~*Not applicable.*~~ ~~*The Auxiliary Systems do not contain stainless steel nonregenerative heat exchanger components that are exposed to treated borated water > 60°C (> 140°F) and subject to aging management review.*~~ *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*

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
					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 loading.* *Temperature and radioactivity monitoring of shell side water is performed by installed instrumentation.* 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 3.3-67**	**Row 3.3.1-26, "Discussion" column**

In response to RAI 3.3.2.2.10.4-1, the text in the "Discussion" column of row 3.3.1-26 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
3.3.1-26	Copper alloy piping, piping components, and piping elements exposed to lubricating oil	Loss of material due to pitting and crevice corrosion	Lubricating Oil Analysis and One-Time Inspection	Yes, detection of aging effects is to be evaluated	Consistent with NUREG-1801. *Loss of material in copper alloy piping, piping components, and piping elements exposed to lubricating oil is managed by the Lubricating Oil Analysis Program* ~~*if the zinc content is greater than 15%*~~. The One-Time Inspection will provide verification of the effectiveness of the Lubricating Oil Analysis Program to manage loss of material. *This item is also applied to copper alloy heat exchanger components* ~~*with zinc content greater than 15%*~~ *that are*

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**
					exposed to lubricating oil. ~~*Loss of material due to pitting and crevice corrosion was not identified as an aging effect requiring management for copper alloy piping, piping components, and piping elements with a zinc content less than 15% that are exposed to lubricating oil.*~~ Further evaluation is documented in Section 3.3.2.2.10.4.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.3.2-1	**Page 3.3-140**	**Row 119; and, 1 New Row**

In response to RAI 3.3.2.2.10.4-1, row 119 of LRA Table 3.3.2-1, “Aging Management Review Results – Auxiliary Building HVAC System,” is revised, and a new row is added, to read as follows:

Table 3.3.2-1				**Aging Management Review Results – Auxiliary Building HVAC 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**
119	Tubing	Pressure boundary	Copper Alloy	Lubricating oil (Internal)	~~None~~ *<u>Loss of material</u>*	~~None~~ *<u>Lubricating Oil Analysis</u>*	VII.C1-8	3.3.1-26	~~I~~ ~~0302~~ *<u>A</u>*
--	*<u>Tubing</u>*	*<u>Pressure boundary</u>*	*<u>Copper Alloy</u>*	*<u>Lubricating oil (Internal)</u>*	*<u>Loss of material</u>*	*<u>One-Time Inspection</u>*	*<u>VII.C1-8</u>*	*<u>3.3.1-26</u>*	*<u>A</u>*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.3.2-14	**Page 3.3-337**	**Row 215; and, 1 New Row**

In response to RAI 3.3.2.2.10.4-1, row 215 of LRA Table 3.3.2-14, "Aging Management Review Results – Fire Protection System," previously revised by FENOC letter dated September 16, 2011 (ML11264A059), is revised, and a new row is added, 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
215	Heat Exchanger (tubes) – Gear Housing Oil Cooler	Pressure boundary	Copper Alloy	Lubricating oil (External)	~~None~~ *Loss of material*	~~None~~ *Lubricating Oil Analysis*	VII.G-11	3.3.1-26	~~I~~ ~~0302~~ *C*
--	*Heat Exchanger (tubes) – Gear Housing Oil Cooler*	*Pressure boundary*	*Copper Alloy*	*Lubricating oil (External)*	*Loss of material*	*One-Time Inspection*	*VII.G-11*	*3.3.1-26*	*C*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.3.2-18	**Pages 3.3-368 thru 3.3-397**	**Rows 24, 25, 50, 51, 58 and 59; and, 24 New Rows**

In response to RAI 2.3.3.18-4, rows 24, 25, 50, 51, 58 and 59, previously revised by FENOC letter dated June 3, 2011 (ML11159A132), are revised, and 24 new rows are added to LRA Table 3.3.2-18, "Aging Management Review Results – Makeup and Purification System," to read as follows:

Table 3.3.2-18 Aging Management Review Results – Makeup and Purification 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**
24	Heat Exchanger (channel) – Seal return coolers (DB-E26-1 & 2)	Pressure boundary	Stainless Steel	Treated borated water > 60°C (> 140°F) (Internal)	Cracking	One-Time Inspection	*VII.E1-~~20~~ 5*	*3.3.1-~~90~~ 07*	E 0315
25	Heat Exchanger (channel) – Seal return coolers (DB-E26-1 & 2)	Pressure boundary	Stainless Steel	Treated borated water > 60°C (> 140°F) (Internal)	Cracking	PWR Water Chemistry	*VII.E1-~~20~~ 5*	*3.3.1-~~90~~ 07*	~~C~~ A

Table 3.3.2-18 Aging Management Review Results – Makeup and Purification 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
50	Heat Exchanger (tubes) – Seal return coolers (DB-E26-1 & 2)	Pressure boundary	Stainless Steel	Treated borated water > 60°C (> 140°F) (Internal)	Cracking	One-Time Inspection	*VII.E1-~~20~~* *5*	*3.3.1-~~90~~* *07*	E 0315
51	Heat Exchanger (tubes) – Seal return coolers (DB-E26-1 & 2)	Pressure boundary	Stainless Steel	Treated borated water > 60°C (> 140°F) (Internal)	Cracking	PWR Water Chemistry	*VII.E1-~~20~~* *5*	*3.3.1-~~90~~* *07*	~~C~~ *A*
58	Heat Exchanger (tubesheet) – Seal return coolers (DB-E26-1 & 2)	Pressure boundary	Stainless Steel	Treated borated water > 60°C (> 140°F) (Internal)	Cracking	One-Time Inspection	*VII.E1-~~20~~* *5*	*3.3.1-~~90~~* *07*	E 0315
59	Heat Exchanger (tubesheet) – Seal return coolers (DB-E26-1 & 2)	Pressure boundary	Stainless Steel	Treated borated water > 60°C (> 140°F) (Internal)	Cracking	PWR Water Chemistry	*VII.E1-~~20~~* *5*	*3.3.1-~~90~~* *07*	~~C~~ *A*
--	*Heat Exchanger (channel) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Cracking*	*One-Time Inspection*	*VII.E1-5*	*3.3.1-07*	*E 0315*

Table 3.3.2-18 Aging Management Review Results – Makeup and Purification 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 (channel) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Cracking*	*PWR Water Chemistry*	*VII.E1-5*	*3.3.1-07*	*A*
--	*Heat Exchanger (channel) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Loss of material*	*One-Time Inspection*	*VII.E1-17*	*3.3.1-91*	*E 0315 0329*
--	*Heat Exchanger (channel) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Loss of material*	*PWR Water Chemistry*	*VII.E1-17*	*3.3.1-91*	*C 0329*
--	*Heat Exchanger (channel) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Air with borated water leakage (External)*	*None*	*None*	*VII.J-16*	*3.3.1-99*	*C*
--	*Heat Exchanger (channel) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Air-indoor uncontrolled (External)*	*None*	*None*	*VII.J-15*	*3.3.1-94*	*C*

Table 3.3.2-18 Aging Management Review Results – Makeup and Purification 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) – Letdown coolers (DB-E25-1 & 2)	Pressure boundary	Steel	Closed cycle cooling water > 60C (> 140F) (Internal)	Loss of material	Closed Cooling Water Chemistry	VII.E1-6	3.3.1-48	B
--	Heat Exchanger (shell) – Letdown coolers (DB-E25-1 & 2)	Pressure boundary	Steel	Air with borated water leakage (External)	Loss of material	Boric Acid Corrosion	VII.I-10	3.3.1-89	A
--	Heat Exchanger (shell) – Letdown coolers (DB-E25-1 & 2)	Pressure boundary	Steel	Air-indoor uncontrolled (External)	Loss of material	External Surfaces Monitoring	VII.I-8	3.3.1-58	A
--	Heat Exchanger (tubes) – Letdown coolers (DB-E25-1 & 2)	Heat transfer	Stainless Steel	Treated borated water > 60C (> 140F) (Internal)	Reduction in heat transfer	One-Time Inspection	N/A	N/A	H 0315
--	Heat Exchanger (tubes) – Letdown coolers (DB-E25-1 & 2)	Heat transfer	Stainless Steel	Treated borated water > 60C (> 140F) (Internal)	Reduction in heat transfer	PWR Water Chemistry	N/A	N/A	H

Table 3.3.2-18 Aging Management Review Results – Makeup and Purification 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) – Letdown coolers (DB-E25-1 & 2)*	*Heat transfer*	*Stainless Steel*	*Closed cycle cooling water > 60C (> 140F) (External)*	*Reduction in heat transfer*	*Closed Cooling Water Chemistry*	*VII.E3-5*	*3.3.1-52*	B 0329
--	*Heat Exchanger (tubes) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Cracking*	*One-Time Inspection*	*VII.E1-5*	*3.3.1-07*	E 0315
--	*Heat Exchanger (tubes) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Cracking*	*PWR Water Chemistry*	*VII.E1-5*	*3.3.1-07*	A
--	*Heat Exchanger (tubes) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Loss of material*	*One-Time Inspection*	*VII.E1-17*	*3.3.1-91*	E 0315 0329
--	*Heat Exchanger (tubes) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Loss of material*	*PWR Water Chemistry*	*VII.E1-17*	*3.3.1-91*	C 0329

Table 3.3.2-18 Aging Management Review Results – Makeup and Purification 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) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Closed cycle cooling water > 60C (> 140F) (External)*	*Cracking*	*Closed Cooling Water Chemistry*	*VII.C2-11*	*3.3.1-46*	*D*
--	*Heat Exchanger (tubes) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Closed cycle cooling water > 60C (> 140F) (External)*	*Loss of material*	*Closed Cooling Water Chemistry*	*VII.C2-10*	*3.3.1-50*	*B* *0329*
--	*Heat Exchanger (tubesheet) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Cracking*	*One-Time Inspection*	*VII.E1-5*	*3.3.1-07*	*E* *0315*
--	*Heat Exchanger (tubesheet) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Cracking*	*PWR Water Chemistry*	*VII.E1-5*	*3.3.1-07*	*A*
--	*Heat Exchanger (tubesheet) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Loss of material*	*One-Time Inspection*	*VII.E1-17*	*3.3.1-91*	*E* *0315* *0329*

Table 3.3.2-18 Aging Management Review Results – Makeup and Purification 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) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Treated borated water > 60C (> 140F) (Internal)*	*Loss of material*	*PWR Water Chemistry*	*VII.E1-17*	*3.3.1-91*	*C 0329*
--	*Heat Exchanger (tubesheet) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Closed cycle cooling water > 60C (> 140F) (External)*	*Cracking*	*Closed Cooling Water Chemistry*	*VII.C2-11*	*3.3.1-46*	*D*
--	*Heat Exchanger (tubesheet) – Letdown coolers (DB-E25-1 & 2)*	*Pressure boundary*	*Stainless Steel*	*Closed cycle cooling water > 60C (> 140F) (External)*	*Loss of material*	*Closed Cooling Water Chemistry*	*VII.C2-10*	*3.3.1-50*	*B 0329*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.3.2-18	**Page 3.3-391**	**Row 156; and, 1 New Row**

In response to RAI 3.3.2.2.10.4-1, row 156 of LRA Table 3.3.2-18, "Aging Management Review Results – Makeup and Purification System," is revised, and a new row is added, to read as follows:

Table 3.3.2-18 Aging Management Review Results – Makeup and Purification 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**
156	Valve Body	Pressure boundary	Copper Alloy	Lubricating oil (Internal)	~~None~~ _Loss of material_	~~None~~ _Lubricating Oil Analysis_	VII.E1-12	3.3.1-26	~~I~~ ~~0302~~ _A_
--	_Valve Body_	_Pressure boundary_	_Copper Alloy_	_Lubricating oil (Internal)_	_Loss of material_	_One-Time Inspection_	_VII.E1-12_	_3.3.1-26_	_A_

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.3.2-30	**Page 3.3-527**	**Row 144; and,** **1 New Row**

In response to RAI 3.3.2.2.10.4-1, row 144 of LRA Table 3.3.2-30, "Aging Management Review Results – Station Blackout Diesel Generator System," is revised, and a new row is added, to read as follows:

Table 3.3.2-30 Aging Management Review Results – Station Blackout Diesel Generator 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
144	Valve Body	Pressure boundary	Copper Alloy	Lubricating oil (Internal)	~~None~~ *Loss of material*	~~None~~ *Lubricating Oil Analysis*	~~VII.C1-8~~ *VII.H2-10*	3.3.1-26	~~I~~ ~~0302~~ *A*
--	*Valve Body*	*Pressure boundary*	*Copper Alloy*	*Lubricating oil (Internal)*	*Loss of material*	*One-Time Inspection*	*VII.H2-10*	*3.3.1-26*	*A*

Affected LRA Section | **LRA Page No.** | **Affected Paragraph and Sentence**

Table 3.3.2 Plant-Specific Notes | **Page 3.3-547** | **Row 0302**

In response to RAI 3.3.2.2.10.4-1, row 0302 of Table 3.3.2, "Plant-Specific Notes," is no longer used, and is revised as follows:

Plant-Specific Notes:	
0302	~~*This material is copper alloy < 15% Zn and is not in contact with a more cathodic metal; therefore, there are no aging effects requiring management in the lubricating oil environment.*~~ *Not used.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
3.4.2.2.7.3	**Page 3.4-10**	**2nd sentence; and, New [last] sentence**

In response to RAI 3.3.2.2.10.4-1, the second sentence of LRA Section 3.4.2.2.7.3, "Copper Alloy Piping, Piping Components, Piping Elements – Lubricating Oil," is revised, and a new last sentence is added to better align with LRA Table 3.4.1, "Summary of Aging Management Programs for Steam and Power Conversion Systems Evaluated in Chapter VIII of NUREG-1801," row 3.4.1-18. LRA Section 3.4.2.2.7.3 reads as follows:

3.4.2.2.7.3 Copper Alloy Piping, Piping Components, Piping Elements – Lubricating Oil

Loss of material due to pitting and crevice corrosion could occur for copper alloy piping, piping components, and piping elements exposed to lubricating oil. *At Davis Besse, loss of material due to pitting and crevice corrosion~~, and selective leaching,~~ for copper alloy ~~(copper alloy > 15% Zn)~~ piping, piping components, and piping elements that are exposed to lubricating oil in the Steam and Power Conversion Systems is managed by the Lubricating Oil Analysis Program.* The Lubricating Oil Analysis Program manages loss of material through periodic monitoring and control of contaminants, including water. The One-Time Inspection will provide verification of the effectiveness of the Lubricating Oil Analysis Program to manage loss of material. This item is also applied to copper alloy (copper alloy > 15% Zn) heat exchanger components that are exposed to lubricating oil in the Steam and Power Conversion Systems. <u>*This item is also applied to loss of material due to selective leaching for copper alloy (copper alloy > 15% Zn) components that are exposed to lubricating oil.*</u>

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.4.1	**Page 3.4-24**	**Row 3.4.1-18, "Discussion" column**

In response to RAI 3.3.2.2.10.4-1, the text in the "Discussion" column of row 3.4.1-18 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
3.4.1-18	Copper alloy piping, piping components, and piping elements exposed to lubricating oil	Loss of material due to pitting and crevice corrosion	Lubricating Oil Analysis and One-Time Inspection	Yes, detection of aging effects is to be evaluated	Consistent with NUREG-1801. *Loss of material due to pitting and crevice corrosion in copper alloy ~~(copper alloy > 15% Zn)~~ piping, piping components, and piping elements that are exposed to lubricating oil is managed by the Lubricating Oil Analysis Program.* The One-Time Inspection will provide verification of the effectiveness of the Lubricating Oil Analysis Program to manage loss of material. ~~*Loss of material due to pitting and crevice corrosion was not*~~

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
					~~identified as an aging effect requiring management for copper alloy piping, piping components, and piping elements with a zinc content less than 15% that are exposed to lubricating oil.~~ This item is also applied to copper alloy (copper alloy > 15% Zn) heat exchanger components that are exposed to lubricating oil. This item is also applied to loss of material due to selective leaching for copper alloy (copper alloy > 15% Zn) components that are exposed to lubricating oil. Further evaluation is documented in Section 3.4.2.2.7.3.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.4.2-1	**Page 3.4-46**	**Row 36; and, 1 New Row**

In response to RAI 3.3.2.2.10.4-1, row 36 of LRA Table 3.4.2-1, "Aging Management Review Results – Auxiliary Feedwater System," is revised, and a new row is added, to read as follows:

Table 3.4.2-1 Aging Management Review Results – Auxiliary Feedwater 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**
36	Heat exchanger (tubes) – AFW pump oil coolers	Pressure boundary	Copper Alloy	Lubricating oil (External)	~~None~~ *Loss of material*	~~None~~ *Lubricating Oil Analysis*	VII.C1-8	3.3.1-26	~~I~~ ~~0413~~ C
--	*Heat exchanger (tubes) – AFW pump oil coolers*	*Pressure boundary*	*Copper Alloy*	*Lubricating oil (External)*	*Loss of material*	*One-Time Inspection*	*VII.C1-8*	*3.3.1-26*	C

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.4.2-4	**Page 3.4-85**	**Row 26; and, 1 New Row**

In response to RAI 3.3.2.2.10.4-1, row 26 of LRA Table 3.4.2-4, "Aging Management Review Results – Main Steam System," is revised, and a new row is added, to read as follows:

Table 3.4.2-4 Aging Management Review Results – Main Steam 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
26	Heat exchanger (tubes) – AFW pump turbine bearing lube oil cooler	Pressure boundary	Copper Alloy	Lubricating oil (External)	~~None~~ *Loss of material*	~~None~~ *Lubricating Oil Analysis*	~~VII.C1-8~~ *VIII.G-19*	~~3.3.1-26~~ *3.4.1-18*	~~I~~ ~~0413~~ *C*
--	*Heat exchanger (tubes) – AFW pump turbine bearing lube oil cooler*	*Pressure boundary*	*Copper Alloy*	*Lubricating oil (External)*	*Loss of material*	*One-Time Inspection*	*VIII.G-19*	*3.4.1-18*	*C*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.4.2 Plant-Specific Notes	**Page 3.4-111**	**Row 0413**

In response to RAI 3.3.2.2.10.4-1, row 0413 of Table 3.4.2, "Plant-Specific Notes," is no longer used, and is revised as follows:

Plant-Specific Notes:	
0413	~~*This material is copper alloy < 15% Zn and is not in contact with a more cathodic metal; therefore, there are no aging effects requiring management in the lubricating oil environment.*~~ *Not used.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.7.6	**Page 4.7-6**	**2nd Paragraph, sub-item #1; 2 New Paragraphs (Nos. 3 & 4); and, 5th Paragraph, 1st sentence**

In response to Supplemental RAI 4.1-2, LRA Section 4.7.6, "ASME Code Case N-481 Evaluation," is revised to read as follows:

4.7.6 ASME CODE CASE N-481 EVALUATION

The reactor coolant pumps (RCPs) are the only ASME Code Class 1 pumps installed at Davis-Besse. The pump casings are constructed of cast austenitic stainless steel. The applicable ASME Code for the current Third Ten-Year Inspection Interval for Davis-Besse is ASME Section XI, 1995 Edition, through the 1996 Addenda, as modified by 10 CFR 50.55a or relief granted in accordance with 10 CFR 50.55a. Examination Category B-L-1 of this Code year requires volumetric examination of pump casing welds. ASME Code Case N-481, "Alternative Examination Requirements for Cast Austenitic Pump Casings," provides an alternative to the volumetric examination requirement. This code case allows the replacement of volumetric examinations of primary loop pump casings with fracture mechanics-based integrity evaluation (Item (d) of the code case) supplemented by specific visual examinations. Davis-Besse has invoked the use of Code Case N-481 in place of the volumetric examination requirements of Code Category B-L-1. The NRC has accepted Code Case N-481 for use in inservice inspection programs.

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 documented in Structural Integrity Associates (SIA) report SIR-99-040 [Reference 4.8-18]. This evaluation 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. To determine stability of the postulated flaw, a fracture mechanics evaluation was performed that included a fatigue crack growth analysis to demonstrate that a small initial assumed flaw (10 percent through-wall), corresponding to the acceptance standards of ASME Code, Section XI, Subarticle IWB-3500, would not grow to quarter thickness during plant life. There are two potential time-dependencies in the Code Case N-481 evaluation.

1. *The fracture toughness of the cast austenitic stainless steel is not time dependent as the Davis-Besse ASME Code Case N-481 analysis used a lower bound fracture toughness of 139 ksi$\sqrt{in}$ that bounds the ~~saturated~~ saturation fracture toughness of the Davis-Besse material.*

2. The fatigue crack growth analysis is based on design cycles for a 40 year plant life and therefore, is a TLAA requiring analysis and disposition for license renewal.

With respect to Item No. 1 above, the saturation fracture toughness was determined using the methodology outlined in NUREG/CP-0119, Volume 2, pages 151-178, "Proceedings of the U.S. Nuclear Regulatory Commission, 19^{th} Water Reactor Safety Information Meeting held at Bethesda, MD, October 28-30, 1991," and considering all available certified material test reports (CMTRs) for the base material and welds of the Davis-Besse RCP casings. The saturation fracture toughness value of 139 ksi$\sqrt{in}$ was the minimum calculated for all the CMTRs considered in the evaluation. This minimum saturation fracture toughness value has since been calculated using NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems." Using the methodology and correlation in this NUREG results in the same minimum saturation fracture toughness value for the pump casings.

The fracture toughness for welds considering thermal aging has also been presented in NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds." A conservative J_{1c} fracture toughness value of 40 KJ/m^2 based on the absolute minimum of all available data is provided in this document for aged stainless steel welds; this J_{1c} fracture toughness value translates to 80 ksi$\sqrt{in}$. This conservative fracture toughness value still bounds the calculated total applied stress intensity factors calculated in Table 4-5 of SIR-99-040, Revision 1, indicating that the conclusions of SIR-99-040, Revision 1, are unchanged even if the methodology outlined in NUREG-CR-6428 is used for the Davis-Besse pump casing welds.

With respect to Item No. 2 above, ~~T~~the fatigue crack growth analysis assumed an initial flaw size corresponding to the acceptance standards of ASME Code Section XI and considered all the significant plant transients. This analysis examined the design cycles and determined there were 240 cycles that were significant to flaw growth in the RCPs. Then 2000 cycles were conservatively analyzed, and flaw growth (initial 10 percent assumed through-wall had grown only to 15 percent through-wall) remained well below the quarter thickness postulated flaw. The analyzed cycles of 2000 bound the 60-year projected cycles shown in LRA Table 4.3-1 and therefore, the fatigue crack growth TLAA associated with the ASME Code Case N-481 evaluation will remain valid for the period of extended operation.

Disposition:	10 CFR 54.21(c)(1)(i)	The fatigue crack growth TLAA associated with ASME Code Case N-481 evaluation will remain valid through the period of extended operation.

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

In response to RAI 3.1.2.2.16-1, a new third paragraph is inserted into LRA Section A.1.38, "Steam Generator Tube Integrity Program," 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.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
A.2.7.5	**Page A-50**	**2nd Paragraph, sub-item #1; 2 New Paragraphs (Nos. 3 & 4); and, 5th Paragraph, 1st sentence**

In response to Supplemental RAI 4.1-2, LRA Section A.2.7.5, "ASME Code Case N-481 Evaluation," is revised to read as follows:

A.2.7.5 ASME Code Case N-481 Evaluation

The reactor coolant pumps (RCPs) are the only ASME Code Class 1 pumps installed at Davis-Besse. The pump casings are constructed of cast austenitic stainless steel. The applicable ASME Code for the current Third Ten-Year Inspection Interval for Davis-Besse is ASME Section XI, 1995 Edition, through the 1996 Addenda, as modified by 10 CFR 50.55a or relief granted in accordance with 10 CFR 50.55a. Examination Category B-L-1 of this Code year requires volumetric examination of pump casing welds. ASME Code Case N-481, "Alternative Examination Requirements for Cast Austenitic Pump Casings," provides an alternative to the volumetric examination requirement. This code case allows the replacement of volumetric examinations of primary loop pump casings with fracture mechanics-based integrity evaluation (Item (d) of the code case) supplemented by specific visual examinations. Davis-Besse has invoked the use of Code Case N-481 in place of the volumetric examination requirements of Code Category B-L-1. The NRC has accepted Code Case N-481 for use in inservice inspection programs.

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 documented in Structural Integrity Associates (SIA) report SIR-99-040 [Reference A.2-18]. This evaluation 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. To determine stability of the postulated flaw, a fracture mechanics evaluation was performed that included a fatigue crack growth analysis to demonstrate that a small initial assumed flaw (10 percent through-wall), corresponding to the acceptance standards of ASME Code, Section XI, Subarticle IWB-3500, would not grow to quarter thickness during plant life. There are two potential time-dependencies in the Code Case N-481 evaluation.

1. *The fracture toughness of the cast austenitic stainless steel is not time dependent as the* <u>*Davis-Besse ASME Code Case N-481*</u> *analysis used a lower bound fracture toughness of 139 ksi√in that bounds the* ~~*saturated*~~ <u>*saturation*</u> *fracture toughness of the Davis-Besse material.*

2. The fatigue crack growth analysis is based on design cycles for a 40 year plant life and therefore, is a TLAA requiring analysis and disposition for license renewal.

With respect to Item No. 1 above, the saturation fracture toughness was determined using the methodology outlined in NUREG/CP-0119, Volume 2, pages 151-178, "Proceedings of the U.S. Nuclear Regulatory Commission, 19th Water Reactor Safety Information Meeting held at Bethesda, MD, October 28-30, 1991," and considering all available certified material test reports (CMTRs) for the base material and welds of the Davis-Besse RCP casings. The saturation fracture toughness value of 139 ksi$\sqrt{in}$ was the minimum calculated for all the CMTRs considered in the evaluation. This minimum saturation fracture toughness value has since been calculated using NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems." Using the methodology and correlation in this NUREG results in the same minimum saturation fracture toughness value for the pump casings.

The fracture toughness for welds considering thermal aging has also been presented in NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds." A conservative J_{1c} fracture toughness value of 40 KJ/m^2 based on the absolute minimum of all available data is provided in this document for aged stainless steel welds; this J_{1c} fracture toughness value translates to 80 ksi$\sqrt{in}$. This conservative fracture toughness value still bounds the calculated total applied stress intensity factors calculated in Table 4-5 of SIR-99-040, Revision 1, indicating that the conclusions of SIR-99-040, Revision 1, are unchanged even if the methodology outlined in NUREG-CR-6428 is used for the Davis-Besse pump casing welds.

With respect to Item No. 2 above, ~~T~~the fatigue crack growth analysis assumed an initial flaw size corresponding to the acceptance standards of ASME Code Section XI and considered all the significant plant transients. This analysis examined the design cycles and determined there were 240 cycles that were significant to flaw growth in the RCPs. Then 2000 cycles were conservatively analyzed, and flaw growth (initial 10 percent assumed through-wall had grown only to 15 percent through-wall) remained well below the quarter thickness postulated flaw. The analyzed cycles of 2000 bound the 60-year projected cycles shown in LRA Table 4.3-1 and therefore, the fatigue crack growth TLAA associated with the ASME Code Case N-481 evaluation will 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-65**	**Commitment No. 20**

In response to Supplemental RAI OIN-380 regarding Structures Monitoring Program enhancements, license renewal future Commitment 20 in LRA Table A-1, "Davis-Besse License Renewal Commitments," is revised to include three new bulleted commitments as follows:

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
20	• *Monitor elastomeric vibration isolators and structural sealants for cracking, loss of material, and hardening.* • *Supplement visual inspection of elastomeric vibration isolation elements by feel to detect hardening if the vibration isolation function is suspect.* • *Identify that:* o *loose bolts and nuts and cracked high strength bolts are not acceptable unless accepted by engineering evaluation;* o *structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result in loss of sealing; and,* o *elastomeric vibration isolation elements are acceptable if there is no loss of material, cracking, or hardening that could lead to the reduction or loss of isolation function.*	Prior to April 22, 2017	LRA and FENOC Letters L-11-153, L-11-237<u>,</u> <u>*L-11-292,*</u> <u>*and*</u> <u>*L-11-317*</u>	A.1.39 B.2.39 Responses to NRC RAIs ~~B.2.39-3,~~ B.2.39-4, B.2.39-5, B.2.39-6 and B.2.39-7 from NRC Letter dated April 5, 2011, RAIs B.2.39-11

Table A-1 Davis-Besse License Renewal Commitments				
Item Number	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
				and 3.5.2.3.12-4 from NRC Letter dated July 21, 2011, *Supplemental RAI B.2.39-11 from telecom held with the NRC on September 13, 2011, and* *Supplemental RAI OIN-380 from Region III IP-71002 Inspection*

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

In response to RAI 2.3.3.18-4, license renewal future Commitment No. 25 is no longer needed and is revised to read "Not used," as follows:

Table A-1 Davis-Besse License Renewal Commitments				
Item Number	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
25	~~*FENOC commits to create a preventive maintenance task to periodically replace the letdown coolers (DB-E21-1 & 2) at a set frequency.*~~ *Not used.*	~~*April 22, 2017*~~	~~*LRA*~~	~~*2.3.3.18*~~

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

In response to RAI 3.1.2.2.16-1, new license renewal future Commitment No. 25, revised to read "Not used" in response to RAI 2.3.3.18-4, above, is revised to include a new license renewal future commitment 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 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*	*Prior to April 22, 2017*	*LRA* *and* *FENOC Letter L-11-317*	*A.1.38* *B.2.38* *Response to NRC RAI 3.1.2.2.16-1 from NRC Letter dated September 22, 2011*

Table A-1 Davis-Besse License Renewal Commitments				
Item Number	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
	longer be required. ~~*Not used.*~~			

Affected LRA Section | **LRA Page No.** | **Affected Paragraph and Sentence**

Table B-2 | **Page B-22** | **1 Row**

In response to 3.1.2.2.16-1, the "Steam Generator Tube Integrity Program" row of Table B-2, "Consistency of Davis-Besse Aging Management Programs with NUREG-1801," is revised to read as follows:

Table B-2
Consistency of Davis-Besse Aging Management Programs with NUREG-1801
(continued)

Program Name	New / Existing	Consistent with NUREG-1801	Consistent with NUREG-1801 with Exceptions	Plant-Specific	Enhancement Required
Steam Generator Tube Integrity Program Section B.2.38	Existing	Yes	--	--	*Yes*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
B.2.38	**Page B-151**	**Program Description subsection, new 4th paragraph; NUREG-1801 Consistency subsection, revised sentence; and, Enhancements subsection, new enhancements**

In response to RAI 3.1.2.2.16-1, LRA Section B.2.38, "Steam Generator Tube Integrity Program," a new fourth paragraph is added to subsection "Program Description," the "NUREG-1801 Consistency" subsection is revised, and new enhancements are added to the "Enhancements" subsection, to read as follows:

B.2.38 STEAM GENERATOR TUBE INTEGRITY PROGRAM

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.

NUREG-1801 Consistency

The Steam Generator Tube Integrity Program is an existing Davis-Besse program that, ~~is~~ with enhancement, will be consistent with the 10 elements of an *effective aging management program as described in NUREG-1801, Section XI.M19, "Steam Generator Tube Integrity."*

Enhancements

~~*None.*~~

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.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
B.2.39	**Page B-156**	**Enhancements subsection, 3 new enhancements**

In response to Supplemental RAI OIN-380 regarding Structures Monitoring Program enhancements, LRA Section B.2.39, "Structures Monitoring Program," three new enhancements are added to the "Enhancements" subsection, to read as follows:

- **Parameters Monitored or Inspected**

  The program procedure will be enhanced to require that elastomeric vibration isolators and structural sealants are monitored for cracking, loss of material, and hardening.

- **Detection of Aging Effects**

  The program procedure will be enhanced to require that visual inspection of elastomeric vibration isolators will be supplemented by feel to detect hardening if the vibration isolation function is suspect.

- **Acceptance Criteria**

  The program procedure will be enhanced to state that:

  - *loose bolts and nuts, and cracked high strength bolts are not acceptable unless accepted by engineering evaluation;*
  - *structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result in loss of sealing; and,*
  - *elastomeric vibration isolation elements are acceptable if there is no loss of material, cracking, or hardening that could lead to the reduction or loss of isolation function.*