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March 9, 2012
L-12-015

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), License Renewal Application Amendment No. 24, 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 letter dated December 27, 2011 (ML11333A396), the Nuclear Regulatory Commission (NRC) requested additional information to complete its review of the License Renewal Application (LRA).

The NRC letter contained four requests for additional information (RAIs). The FENOC response to one of the four RAI responses, RAI 3.1.2.2.16-3, was submitted to the NRC by letter dated January 13, 2011 (ML12018A338).

Attachment 1 provides the FENOC responses to two of the four RAIs (B.1.4-2 and B.1.4-3) in the NRC letter. The submittal date for these responses was extended following discussion with Mr. Samuel Cuadrado de Jesus, NRC Project Manager, because additional time was needed for development of the responses due to coordination with multiple site and Fleet departments. Attachment 1 also provides supplemental information on the topics listed below. The NRC request is shown in bold text followed by the FENOC response:

- RAI 2.1-3 regarding abandoned equipment;
- RAI 3.1.2.2-2 regarding Reactor Vessel Internals aging management; and
- LRA Section 4.2, "Reactor Vessel Neutron Embrittlement."

A145
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Enclosure A provides Amendment No. 24 to the DBNPS LRA. Enclosure B provides new and revised License Renewal Boundary Drawings.

The FENOC response to the fourth of four RAIs (B.2.39-13) in the NRC letter is planned to be provided to the NRC by March 30, 2012, due to the need to evaluate and incorporate information from the Davis-Besse Shield Building concrete cracking Root Cause Analysis Report that was recently completed and submitted to the NRC by FENOC letter dated February 27, 2012 (ML120600056).

Attachment 2 identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse) in this document. Any other actions discussed in the submittal represent intended or planned actions by FENOC; they are described only as information and are not Regulatory Commitments. Please notify Mr. Clifford I. Custer, Project Manager – Fleet License Renewal, at (724) 682-7139 of any questions regarding this document or associated Regulatory Commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 9, 2012.

Sincerely,

Barry S. Allen

Barry S. Allen

Attachment:
1. Reply to Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), License Renewal Application, Sections B.1.4, 2.1, 3.1.2 and 4.2
2. Regulatory Commitment List

Enclosures:
A. Amendment No. 24 to the DBNPS License Renewal Application
B. New and Revised DBNPS License Renewal Application Boundary Drawings

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 B.1.4

Question RAI B.1.4-2

Background:

In request for additional information (RAI) B.1.4-1, issued on May 19, 2011, the staff asked the applicant to describe the programmatic activities that will be used to continually identify aging issues, evaluate them, and as necessary, enhance the aging management programs (AMPs) or develop new AMPs for license renewal. In its response dated June 24, 2011, the applicant stated that it currently has a procedurally controlled operating experience review process, as required by NUREG-0737, "Clarification of TMI Action Plan Requirements," Item I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff." The applicant stated that this process provides for the systematic identification and transfer of lessons learned from site and industry experience into fleet and station processes to prevent events and enhance the safety and reliability of its operations.

Issue:

The applicant's response provided a general description of how it considers operating experience on an ongoing basis; however, it does not directly address several areas in RAI B.1.4-1 on which the staff requested information. Further, the applicant's response did not provide specific information on how the operating experience review activities address issues specific to aging. The staff identified the following issues with the applicant's response:

(a) The applicant did not describe the sources of plant-specific operating experience information that it monitors on an ongoing basis. Additional details are needed to determine whether the applicant will consider an adequate scope of information from which to identify potential operating experience related to aging.

(b) It is not clear whether the applicant only reviews certain sources for operating experience information. Additional information is needed to determine whether the applicant's processes would preclude the consideration of relevant operating experience Information, because it is not from a prescribed source.

(c) The staff requested that the applicant indicate which guidance documents require monitoring. The applicant did not indicate whether it considers guidance documents to be a source of operating experience information.

(d) The applicant did not describe its criteria for identifying and categorizing operating experience items as related to aging.

(e) The staff requested that the applicant describe training provided to plant personnel. The applicant did not describe training on aging issues, nor did it indicate whether the training will be provided for those plant personnel responsible for screening, assigning, evaluating, and submitting operating experience items.

(f) The applicant did not describe how evaluations of operating experience related to aging consider the potentially affected plant

- **systems, structures, and components,**
- **materials,**
- **environments,**
- **aging effects,**
- **aging mechanisms, and**
- **AMPs.**

(g) The applicant did not describe how it will consider as operating experience the results of the inspections, tests, analyses, etc., conducted through implementation of the AMPs.

(h) The applicant did not describe the records of operating experience evaluations or how it retains those records.

(i) The applicant stated that operating experience evaluations are prioritized with due dates procedurally specified based on the potential significance of the issue; however, the applicant did not provide details on the evaluation schedules or how it determines the relative significance of issues. It is therefore unclear whether the operating experience evaluations will be completed in a timely manner or whether they will be appropriately prioritized.

(j) The applicant stated that it enters operating experience that potentially represents a condition adverse to quality into the corrective action program; however, the applicant did not explain how a "condition adverse to quality" includes aging. Additional information is needed to determine

whether the corrective action program has a threshold appropriate to capture items related to aging.

(k) The applicant did not describe criteria for considering when AMPs should be modified or new AMPs developed due to operating experience. It also did not describe how it implements these kinds of changes or how it ensures the changes are implemented in a timely manner.

(l) The applicant stated that it shares lessons learned with other utilities to promote industry-wide safety and reliability; however, the applicant did not provide criteria for reporting its plant-specific operating experience on age-related degradation to the industry.

Request:

Provide a response to each item below.

(a) Describe the sources of plant-specific operating experience that are monitored on an ongoing basis to identify potential aging issues.

(b) Indicate whether plant-specific and industry operating experience is only considered from a prescribed list of sources. If only prescribed sources are considered, provide a justification as to why it is unnecessary to consider other sources.

(c) Indicate whether guidance documents are considered as a source of operating experience information. If they are considered as a potential source, provide a plan for considering the content of guidance documents, such as the GALL Report, as operating experience applicable to aging management.

(d) Describe how operating experience issues will be identified and categorized as related to aging.

(e) Describe the training requirements on aging issues for those plant personnel responsible for screening, evaluating, and submitting operating experience items.

(f) Describe how evaluations of operating experience issues related to aging will consider the following:

- **systems, structures, or components**
- **materials**
- **environments**
- **aging effect**

- **aging mechanisms**
- **AMPs**

(g) Describe how the results of the AMP inspections, tests, analyses, etc., will be considered as operating experience.

(h) Describe the operating experience evaluation records with respect to what is considered for aging. Indicate whether these records are maintained in auditable and retrievable form.

(i) Provide details on the operating experience evaluation schedules and justify why they provide for timely evaluations. Also, describe how the relative significance of operating experience items is determined so that the reviews can be prioritized appropriately.

(j) Justify why the corrective action program has an appropriate threshold for capturing issues concerning aging.

(k) Describe the criteria for considering when AMPs should be modified or new AMPs developed due to operating experience. Also, describe the process for implementing changes to the AMPs or for implementing new AMPs; describe how these changes are implemented in a timely manner.

(l) Provide criteria for reporting plant-specific operating experience on age-related degradation to the industry.

If enhancements are necessary, provide an implementation schedule for incorporating them into the existing programmatic operating experience review activities.

RESPONSE RAI B.1.4-2

(a) Describe the sources of plant-specific operating experience that are monitored on an ongoing basis to identify potential aging issues.

The sources of plant-specific operating experience to identify potential aging issues include the following:

- FENOC Corrective Action Program – adverse conditions are documented in the Corrective Action Program, including, when appropriate, the cause and actions necessary to correct and/or prevent recurrence. Adverse conditions as defined in the Corrective Action Program include any event, defect, characteristic, state or activity that prohibits or detracts from safe, efficient nuclear plant operation or a condition that could credibly impact nuclear safety, personnel safety, plant reliability or non-compliance with federal, state,

or local regulations. Adverse conditions that are failures, malfunctions, deficiencies, deviations, defective hardware and non-conformances, or human performance, programmatic, organizational, or management weaknesses that adversely affect Quality (Q), Augmented Quality (AQ), or nuclear safety-related equipment, programs, or processes, are considered conditions adverse to quality. Adverse conditions include conditions adverse to quality, plant reliability issues, any concern that should be trended (e.g., personnel contamination, personnel safety, and unexpected plant equipment failures), and conditions that have significance within a regulatory context.

FENOC does not differentiate whether an adverse condition is aging related when initiating a condition report; adverse conditions resulting from aging related issues are documented in the Corrective Action Program in the same manner as any other adverse condition. The Corrective Action Program, therefore, is considered the primary source of plant-specific operating experience since adverse conditions, including aging-related adverse conditions, are documented in the program database.

- Plant maintenance activities – maintenance activities such as preventive maintenance tasks, inspections, examinations, surveillances, tests or analyses, are sources of plant-specific operating experience, including potential aging issues. Adverse conditions identified during the performance of these types of activities are documented in the Corrective Action Program.
- Plant Operator tours – Operator tours are performed routinely throughout the station and provide the opportunity to identify adverse system, structure or component conditions, including aging issues. Adverse conditions identified during the performance of Plant Operator tours are documented in the Corrective Action Program.
- System Engineer walkdowns – Periodic System Engineer walkdowns performed as part of the System Performance Monitoring Program provide the opportunity to identify adverse system, structure or component conditions, including aging issues. Adverse conditions identified during the course of System Engineer walkdowns are documented in the Corrective Action Program.
- Aging Management Program activities – preventive maintenance tasks, inspections or examinations performed at the direction of aging management programs are sources of plant-specific operating experience, including aging issues. Additionally, trending of program inspection or examination results and effectiveness reviews required by select aging management programs may also provide a source of plant-specific operating experience, including aging issues. Adverse conditions identified during the performance of these types of activities are documented in the Corrective Action Program.

(b) Indicate whether plant-specific and industry operating experience is only considered from a prescribed list of sources. If only prescribed sources are considered, provide a justification as to why it is unnecessary to consider other sources.

FENOC does not consider plant-specific and industry operating experience only from a prescribed list of sources.

For plant-specific operating experience, FENOC considers input from many sources, such as those described in the response to question (a), above. Adverse conditions identified from these sources are documented in the FENOC Corrective Action Program. The Corrective Action Program procedure does not include a prescribed list of sources from which to identify adverse conditions; rather, the procedure provides examples of potential adverse conditions so that plant personnel will identify plant issues from a wide range of sources.

Although the Operating Experience Program does not restrict the use of other sources of operating experience, it does require screening of industry operating experience from a list of prescribed sources at a minimum. The list of prescribed operating experience sources includes:

- Institute of Nuclear Power Operations (INPO) Event Reports;
- INPO Significant Operating Experience Reports (SOERs);
- INPO Operating Experience Reports;
- Nuclear Regulatory Commission (NRC) Information Notices (INs);
- NRC Regulatory Issue Summaries (RISs); and,
- NRC Regulatory changes.

The Operating Experience Program procedure also allows for processing operating experience from additional sources not listed above on a case-by-case basis.

Other programs and processes, such as those provided in the examples identified below, are also available to FENOC for obtaining industry operating experience and aging-related information:

- The Continuous Equipment Performance Improvement process (derived directly from INPO's AP-913, "Equipment Reliability Process Description") requires periodic updates to the process based on evaluation of plant-specific and industry operating experience from sources such as new vendor recommendations, Electric Power Research Institute (EPRI) preventive maintenance templates, the INPO Equipment Performance and Information Exchange (EPIX) system, or aging studies.

- The Materials Degradation Management Program is an industry initiative to adopt Nuclear Energy Institute (NEI) guidance for management of material issues. The Program is applicable to materials of construction within the Reactor Coolant System Pressure Boundary, although it can be applied to materials of construction within other systems. The Materials Degradation Management Program ensures, among other issues, that materials operating experience is shared among utilities.
- Industry owners groups, vendors, and EPRI committees, industry managed groups and users groups share information that can help identify aging concerns.
- Industry benchmarking lessons learned.

(c) Indicate whether guidance documents are considered as a source of operating experience information. If they are considered as a potential source, provide a plan for considering the content of guidance documents, such as the GALL Report, as operating experience applicable to aging management.

FENOC considers NRC Information Notices, Regulatory Issue Summaries, and Regulatory changes as sources of operating experience. However, FENOC does not consider NRC guidance documents as sources of operating experience information.

Guidance documents, such as the NRC NUREG-Series publications, are reports or brochures on regulatory decisions, results of research, results of incident investigations, and other technical and administrative information. These documents are not sources of operating experience; however, the NRC may use NUREGs to compile and report the results of operating experience research from industry sources. For example, new License Renewal Interim Staff Guidance documents or revisions to NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," and NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," provide, in addition to license renewal guidance, the results of research on previously-identified industry operating experience and the associated lessons-learned, and are historical documents by the time they are issued.

FENOC is actively involved with numerous industry users groups and committees, which regularly share information on new industry operating experience. Changes to guidance documents are often identified and discussed at these industry meetings, allowing FENOC to be informed of the issuance of new or revised guidance documents.

While the Operating Experience Program procedure does not include NRC guidance documents on the list of operating experience sources to be reviewed, the procedure does include Regulatory Issue Summaries, which can be used by the NRC to inform

licensees of changes to guidance documents. For example, FENOC was made aware of Revision 2 to NUREG-1800 and 1801 through the Nuclear Energy Institute (NEI) License Renewal Working Groups and through Regulatory Issue Summary 2011-05, "Information on Revision 2 to the Generic Aging Lessons Learned Report for License Renewal of Nuclear Power Plants." Regulatory Issue Summary 2011-05 was screened by FENOC using the Operating Experience Program.

(d) Describe how operating experience issues will be identified and categorized as related to aging.

Plant-specific aging-related operating experience issues documented in the FENOC Corrective Action Program are processed and investigated in the same manner as other adverse conditions. However, the Corrective Action Program Nuclear Operating Procedure is planned to be revised to require that a condition report investigation of an aging-related issue for structures and passive components include consideration of the affected structure or component, material, environment, aging effect, aging mechanism, and aging management program, with feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.

Industry aging-related operating experience items are received and screened no differently than other industry operating experience items received through existing operating experience pipelines. Aging-related operating experience items are not typically received with an "aging" flag or designator; rather, the operating experience items identify a condition or event experienced at another station. For example, corrosion-related or cracking failures may not necessarily be designated as "aging-related."

The Fleet Operating Experience Program Manager and Station Operating Experience Coordinators review INPO's Nuclear Network and incoming correspondence for new external operating experience items. A screening process is performed for the external operating experience items to determine the susceptibility of a similar and unacceptable event or condition occurring at FENOC. If the operating experience item is screened as not applicable (NA) to FENOC, no further action is taken. An operating experience item that is determined to be applicable to FENOC is further screened as follows:

- If the operating experience item represents a potential operability or reportability concern, the operating crew Shift Manager or Shift Engineer is notified and a condition report is written to document the issue in the Corrective Action Program.
- If the operating experience item represents an adverse condition, a condition report is written to document the issue in the Corrective Action Program.

- If the operating experience item is identified for an "Evaluation Required Review," a formal evaluation with supervisory or management oversight is initiated.
- If the operating experience item requires further screening, it is submitted to an Operating Experience Coordinator peer group, which may include subject matter experts as needed, for a more detailed review to determine susceptibility to FENOC stations.
- If the operating experience item is screened as applicable, but may not require a document evaluation, the item is considered "Information Only," and is listed in the Weekly Operating Experience Summary.
    - The Section Operating Experience Coordinator is responsible for reviewing the Weekly Operating Experience Summary and determining whether any of the items require an evaluation, enlisting subject matter experts as necessary to assist in the determination.
    - These Information Only operating experience reports are also communicated and distributed to personnel via the Weekly Operating Experience Summary. No specific evaluation is required for operating experience items identified as Information Only reviews. However, if the reviewer determines that further evaluation is required, the reviewer should initiate an Evaluation Required Review by contacting the Fleet or Station Operating Experience Coordinator.

The Operating Experience Program Nuclear Operating Business Practice is planned to be revised to require that an Evaluation Required Review of aging-related operating experience issues for structures and passive components include consideration of the affected structure or component, material, environment, aging effect, aging mechanism, and aging management program, with feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.

Additionally, the Corrective Action Review Board Nuclear Operating Business Practice is planned to be revised to ensure that the Corrective Action Review Board questions whether aging was considered for condition report investigations that are reviewed by the Board.

See the enhancements descriptions at the end of this response.

(e) Describe the training requirements on aging issues for those plant personnel responsible for screening, evaluating, and submitting operating experience items.

The training requirements for personnel responsible for screening, evaluating and submitting (to the industry) operating experience items consist of completion of position-specific document and procedure reviews, proficiency demonstrations and interviews that provide personnel with the ability to independently perform the activity. Mentoring is used during the training process to provide direction, coaching, and oversight by a recognized technical expert who is qualified in the specific task requirements to ensure job performance requirements are understood and competency is achieved.

A training "needs analysis" is planned to determine and document recommended enhancements to the training requirements for those personnel responsible for screening, evaluating and submitting (to the industry) aging-related operating experience items.

See the enhancements descriptions at the end of this response.

(f) Describe how evaluations of operating experience issues related to aging will consider the following:

- **systems, structures, or components**
- **materials**
- **environments**
- **aging effect**
- **aging mechanisms**
- **AMPs**

For plant-specific aging-related operating experience issues, the Corrective Action Program Nuclear Operating Procedure is planned to be revised to require that a condition report investigation of an aging-related issue for structures and passive components include consideration of the affected structure or component, material, environment, aging effect, aging mechanism, and aging management program, with feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.

For industry aging-related operating experience items, the Operating Experience Program Nuclear Operating Business Practice is planned to be revised to require that an Evaluation Required Review of aging-related operating experience issues for

structures and passive components include consideration of the affected structure or component, material, environment, aging effect, aging mechanism, and aging management program, with feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.

See the enhancements descriptions at the end of this response.

(g) Describe how the results of the AMP inspections, tests, analyses, etc., will be considered as operating experience.

Degraded or non-conforming conditions or results that do not meet the defined acceptance criteria identified during aging management program inspections, tests, analyses or other program activities are considered adverse conditions and are documented in condition reports using the FENOC Corrective Action Program. These results and the associated condition report are considered operating experience for the affected aging management program.

The applicable site or fleet Management Review Board is responsible to review condition reports to identify internal operating experience that has the potential to be shared with the other FENOC sites and the industry. The following attributes are used to make the determination:

- Important to nuclear, public, radiological, and personnel safety
- Important to power generation capability
- Events with important generic implications
- Events for which an investigation was performed and the lessons learned would be beneficial to know about if the event had occurred at another station
- Events required to be reported by INPO Event Report (IER) Level 1 or SOER recommendations

The results of aging management program inspections, tests, analyses or other program activities that meet the defined acceptance criteria are also considered operating experience for the affected aging management program. These results are used as feedback to the aging management program for trending purposes, and for evaluation to determine, based on the component, material, environment and aging effect combinations managed, whether the frequency of future inspections needs to be adjusted or new inspections established.

The Corrective Action Program Nuclear Operating Procedure is planned to be revised to require that a condition report investigation of an aging-related issue for structures and passive components include consideration of the affected structure or component, material, environment, aging effect, aging mechanism, and aging

management program, with feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.

See the enhancements descriptions at the end of this response.

(h) Describe the operating experience evaluation records with respect to what is considered for aging. Indicate whether these records are maintained in auditable and retrievable form.

Operating experience evaluation records are processed using existing plant procedures. Plant-specific operating experience is documented in condition reports and processed using the FENOC Corrective Action Program. Aging issues that would be captured in the Corrective Action Program include adverse conditions such as the following:

- aging-related degradation that adversely affects or threatens plant systems, structures or components;
- adverse aging-related trends identified in system health reports or aging management programs;
- aging management program weaknesses or programs identified as ineffective; or,
- results that do not meet the acceptance criteria from inspections, examinations, tests, analyses, or other activities performed under an aging management program.

Condition reports for adverse conditions and related documents captured in the Corrective Action Program database are quality records and are auditable and retrievable.

In a similar manner, industry operating experience sources are monitored and operating experience, including aging-related operating experience, is entered into the Operating Experience Program database. External operating experience items are exposed to a screening process that provides for three possible outcomes:

- Not applicable (NA) to FENOC, and no further action is taken;
- Applicable to FENOC, but may not require a documented evaluation, and are therefore considered "Information Only;" or,
- Applicable to FENOC and require an evaluation ("Evaluation Required Review").

Operating experience reports that are screened as "Information Only" are unlikely to affect nuclear or personnel safety, plant reliability or availability, or are unlikely to need follow-up actions. Information Only operating experience items are

communicated and distributed to personnel in a weekly operating experience summary. If a reviewer later determines that further evaluation is required, the reviewer should initiate an "Evaluation Required Review" by contacting the Fleet or Station Operating Experience Coordinator.

An Evaluation Required Review is performed for operating experience reports that are determined to require a FENOC or site evaluation, or when the initial screening identifies the need for further evaluation and/or actions to address the operating experience issue. The degree of rigor involved in evaluating operating experience information should be based on the significance of the issue identified. Documents captured in the Operating Experience Program database are retrievable.

(i) Provide details on the operating experience evaluation schedules and justify why they provide for timely evaluations. Also, describe how the relative significance of operating experience items is determined so that the reviews can be prioritized appropriately.

Plant-specific operating experience is documented in condition reports and processed using the FENOC Corrective Action Program. Supervisor reviews of condition reports are expected to be performed as soon as possible, normally within one business day of initiation of the condition report. The Management Review Board, which consists of a collegial review and concurrence by the Managers from a representative cross-section of plant disciplines, reviews new condition reports daily and establishes the due dates for condition report evaluations. Due dates are set based on the safety significance of the issue; the standard due date for condition report evaluations is 30 days. For plant-specific operating experience that should be shared with the industry, the goal is to issue the report to the industry within 50 days of the origination date of the associated condition report.

Industry operating experience is documented in the Operating Experience database and processed using the FENOC Operating Experience Program. If the operating experience issue is screened such that an Evaluation Required Review is assigned and no operability, reportability or adverse conditions are identified, then time is available to properly evaluate the issue. For operating experience Evaluation Required Reviews for INPO Level 1 and 2 Event Reports, INPO is to be informed of the FENOC action plan within 150 days. For Evaluation Required Reviews for other operating experience items, the evaluation due date is initially established as 150 days.

The determinations of the relative significance and the resulting determination of timeliness for the evaluation of operating experience items are subjective, but are based on the plant knowledge and experience of the reviewers. Considerations include items such as the real or potential impact to nuclear safety, generation risk, plant operation, systems, structures or components, programs, potential operability

or reportability concerns, or whether an adverse condition exists. The real or potential impacts may not be obvious, and research may need to be performed to determine whether the station has similar systems, structures or components, materials, or conditions such that the station would be affected.

(j) Justify why the corrective action program has an appropriate threshold for capturing issues concerning aging.

The FENOC Corrective Action Program has an appropriate threshold for capturing issues concerning aging because the program requires that adverse conditions, regardless of their nature (including aging-related degradation), are documented in a condition report in the Corrective Action Program. The training provided to plant personnel coupled with procedural guidance and supervisory and management oversight ensures that adverse conditions, including issues concerning aging, are appropriately captured and evaluated. Periodic regulatory and utility-based assessments and effectiveness reviews of the Corrective Action Program are used to confirm that adverse conditions are appropriately addressed.

(k) Describe the criteria for considering when AMPs should be modified or new AMPs developed due to operating experience. Also, describe the process for implementing changes to the AMPs or for implementing new AMPs; describe how these changes are implemented in a timely manner.

The FENOC Corrective Action Program requires development of actions that are effective at addressing conditions adverse to quality. Examples of effective correction actions that address aging may include modification of existing or development of new aging management programs. Adverse aging conditions may be identified via plant-specific or industry operating experience that indicate one or more of the following:

- a new aging effect is identified;
- the applicable aging effect or aging mechanism for a given combination of system, structure or component, material, environment, aging effect and aging mechanism is not being effectively managed;
- adverse aging-related trends are identified in system health reports or aging management programs;
- aging management program weaknesses are identified or programs are identified as ineffective; or,

- results are obtained that do not meet the acceptance criteria from inspections, examinations, tests, analyses, or other activities performed under an aging management program.

Revision of existing or development of new aging management programs based on operating experience evaluations would be performed through corrective actions using the Corrective Action Program, or by action items identified in the Operating Experience Program database. Assigned program owners would develop revisions to their assigned aging management program implementing procedures based on the results of an evaluation of the operating experience issues. A regulatory applicability review and, as applicable, a 10 CFR 50.59 review are performed to confirm whether the change can be made without prior NRC approval, and whether the change impacts any information in the Davis-Besse Updated Safety Analysis Report. New aging management program implementing procedures would be developed based on the activities involved and the systems, structures or components affected. The revised or new aging management program implementing procedure would undergo an internal FENOC review and approval process. The due dates for corrective actions and Operating Experience Program action items are closely tracked and managed.

For plant-specific aging-related operating experience issues, the Corrective Action Program Nuclear Operating Procedure is planned to be revised to require that a condition report investigation of an aging-related issue for structures and passive components include consideration of the affected structure or component, material, environment, aging effect, aging mechanism, and aging management program, with feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.

For industry aging-related operating experience items, the Operating Experience Program Nuclear Operating Business Practice is planned to be revised to require that an Evaluation Required Review of aging-related operating experience issues for structures and passive components include consideration of the affected structure or component, material, environment, aging effect, aging mechanism, and aging management program, with feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.

See the enhancements descriptions at the end of this response.

(l) Provide criteria for reporting plant-specific operating experience on age-related degradation to the industry.

Noteworthy plant-specific operating experience is shared with the other FENOC sites and the industry. The applicable site or fleet Management Review Board is

responsible to identify internal operating experience that has the potential to be shared with the other FENOC sites and the industry. The following attributes are used to make the determination:

- Important to nuclear, public, radiological, and personnel safety
- Important to power generation capability
- Events with important generic implications
- Events for which an investigation was performed and the lessons learned would be beneficial to know about if the event had occurred at another station
- Events required to be reported by INPO Event Report (IER) Level 1 or SOER recommendations

The Operating Experience Program procedure provides the guidance on sharing internal operating experience.

If enhancements are necessary, provide an implementation schedule for incorporating them into the existing programmatic operating experience review activities.

The following training activities and procedure changes will be completed on or before December 31, 2012:

1. Perform a training needs analysis to determine and document recommended enhancements to the training requirements for those plant personnel responsible for screening, evaluating and submitting (to the industry) aging-related operating experience items. Based on the results of the training needs analysis, identify the appropriate training materials.

2. Revise Nuclear Operating Business Practice NOBP-LP-2100, "FENOC Operating Experience Process," to require that an Evaluation Required Review of aging-related operating experience issues for structures and passive components includes:

    a. consideration of the material, environment, aging effect, aging mechanism, and aging management program for the affected structure or component; and,

    b. a provision for feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.

3. Revise Nuclear Operating Procedure NOP-LP-2001, "Corrective Action Program," to require that a condition report investigation of an aging-related issue for structures and passive components includes:
    a. consideration of the material, environment, aging effect, aging mechanism, and aging management program for the affected structure or component; and,
    b. a provision for feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.
4. Revise Nuclear Operating Business Practice NOBP-LP-2008, "FENOC Corrective Action Review Board," to ensure that the Corrective Action Review Board questions whether aging was considered for condition report investigations that are reviewed by the Board.

See Attachment 2 to this letter for the regulatory commitments.

Question RAI B.1.4-3

Background:

In RAI B.1.4-1, the staff asked the applicant to provide, in accordance with 10 CFR 54.21(d), a USAR supplement a summary description of the programmatic activities for the ongoing review of operating experience, as required by 10 CFR 54.21(d). By letter dated August 17, 2011, the applicant provided this description:

> **Existing FENOC processes require reviews of relevant site and industry operating experience and periodic benchmarking to ensure program enhancements are identified and implemented. Such ongoing reviews identify potential needs for aging management program revisions to ensure their effectiveness throughout the period of extended operation.**

Issue:

As described above in RAI B.1.4-2, the applicant described generally how it intends to consider operating experience on an ongoing basis; however, it did not provide specific information on how its operating experience review activities address issues related to aging. Similarly, the above entry for USAR supplement

also lacks details on how aging is considered in the ongoing operating experience reviews.

Request:

Consistent with the response to RAI B.1.4-2, provide additional details in the USAR supplement on how the ongoing operating experience review activities address issues specific to aging.

RESPONSE RAI B.1.4-3

FENOC replaces the operating experience summary description in License Renewal Application (LRA) Section A.1, "Summary Descriptions of Aging Management Programs and Activities," provided by letter dated August 17, 2011 (ML11231A966), with a new description that is consistent with the FENOC response to RAI B.1.4-2, above.

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

Section 2.1

Supplemental Question RAI 2.1-3 Abandoned Equipment

In a supplemental response to RAI 2.1-3 titled, "SUPPLEMENTAL RESPONSE – Abandoned Equipment," submitted by letter dated September 16, 2011 (ML11264A059), FENOC provided a plan to ensure abandoned equipment is identified, isolated and drained, as follows:

1. **Determine the scope of abandoned equipment – includes review of Piping & Instrumentation Diagrams (P&IDs), plant walkdowns, and review of the Shift Operations Management System (eSOMS) clearance database.**
2. **Determine the status of abandoned equipment – includes review of system status files and the eSOMS database for as-left valve positions, walkdowns to validate valve position status, and ultrasonic testing to confirm that abandoned piping is drained.**
3. **Place abandoned equipment in a configuration that will not impact safety-related equipment – create and implement Operations Evolution Orders to isolate and drain abandoned systems with fluids, and create and**

implement Document Change Requests as necessary to correct the configuration of the plant as shown on plant drawings.

SUPPLEMENTAL RESPONSE RAI 2.1-3 Abandoned Equipment

FENOC completed the actions to ensure abandoned equipment is identified, isolated and drained as provided in the plan submitted by FENOC letter dated September 16, 2011 (ML11264A059). Identification of the scope of abandoned equipment that could impact safety-related equipment was determined through a review of Piping & Instrumentation Diagrams (P&IDs), plant walkdowns, and review of the Shift Operations Management System (eSOMS) clearance database.

The status of the abandoned equipment that could impact safety-related equipment was determined through review of system status files and the eSOMS database for as-left valve positions, walkdowns to validate valve position status, and ultrasonic testing to confirm that abandoned piping is drained.

Abandoned equipment that could impact safety-related equipment was verified to be isolated and drained with the exception of components associated with the Service Water System intake crib air bubbler compressors, and the Miscellaneous Liquid Radwaste System degasifier skid, miscellaneous waste evaporator skid, evaporator storage tank pumps, and primary water transfer pumps. The subject components are added to the scope of license renewal in accordance with 10 CFR 54.4(a)(2), as follows:

- The Lake Erie elevation is such that it provides a nonisolable source of water to the discharge piping associated with the abandoned intake crib air bubbler compressors. The components associated with these discharge lines contain fluid, and are, therefore, added to the scope of license renewal. License renewal boundary drawing LR-M012E is revised to highlight these additional components. LRA Section 2.3.3.26, "Service Water System," is revised to add license renewal boundary drawing LR-M012E to the list of license renewal boundary drawings that depict the evaluation boundaries of the Service Water System. LRA Table 2.3.3-26, "Service Water System Components Subject to Aging Management Review," is revised to identify a structural integrity function for the orifice component type. LRA Table 3.3.2-26, "Aging Management Review Results – Service Water System," is revised to provide the updated aging management review results for these additional components.

- The components associated with the abandoned degasifier skid, miscellaneous waste evaporator skid, evaporator storage tank pumps, and primary water transfer pumps contain fluid or are not sufficiently isolated to remain drained, and are, therefore, added to the scope of license renewal. New License renewal boundary drawings LR-M010D Sheet 2, LR-M036C Sheet 2 and LR-M039B Sheet 2 were created, and license renewal boundary drawings LR-M010D

Sheet 1, LR-M020B, LR-M036C Sheet 1, LR-M037C, LR-M037D, LR-M037E, LR-M037F, LR-M039A, LR-M039B Sheet 1 and LR-M042B are revised to highlight the additional components. LRA Section 2.3.3.21, "Miscellaneous Liquid Radwaste System," is revised to add additional license renewal boundary drawings to the list of boundary drawings that depict the evaluation boundaries of the system. LRA Table 2.3.3-21, "Miscellaneous Liquid Radwaste System Components Subject to Aging Management Review," is revised to include the new component types. LRA Table 3.3.2-21, "Aging Management Review Results – Miscellaneous Liquid Radwaste System," is revised to provide the updated aging management review results for the additional components.

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

See Enclosure B to this letter for the revision to the LRA Boundary Drawings.

Section 3.1.2

Supplemental Question RAI 3.1.2.2-2 Reactor Vessel Internals Aging Management

The NRC initiated a telephone conference call with FENOC on January 24, 2012, to discuss the pressurized water reactor (PWR) Reactor Vessel Internals Program and how FENOC plans to implement MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)." The staff wanted to know FENOC's plans to address any differences between MRP-227 and MRP-227-A.

FENOC responded that a review of MRP-227-A dated December 2011 against the previous version of MRP-227 dated December 2008 was in progress to identify any changes needed to the Davis-Besse PWR Reactor Vessel Internals (RVI) Program or to the aging management review (AMR) results provided in License Renewal Application (LRA) Table 3.1.2-2, "Aging Management Review Results – Reactor Vessel Internals." FENOC noted that, at a minimum, several changes are required to LRA Table 3.1.2-2.

FENOC agreed to provide a response to address the differences between MRP-227-A and the previous version of MRP-227, and provide the necessary updates for the reactor vessel internals AMR results and the PWR Reactor Vessel Internals (RVI) Program.

The staff requested that FENOC be very clear as to what has and what has not changed, and to identify what was reviewed.

SUPPLEMENTAL RESPONSE RAI 3.1.2.2-2 Reactor Vessel Internals Aging Management

Aging Management Review Results for the Reactor Vessel Internals

In response to RAI 3.1.2.2-3 submitted by FENOC letter dated September 16, 2011 (ML11264A059), LRA Table 3.1.2-2, "Aging Management Review Results – Reactor Vessel Internals," was replaced in its entirety. These aging management review (AMR) results were based on Electric Power Research Institute (EPRI) Report 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," dated December 2008. This report has since been updated with the changes proposed by the NRC in the safety evaluation for MRP-227 as well as changes proposed by the EPRI Materials Reliability Program in response to NRC requests for additional information (RAIs) and issued as EPRI Report 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," dated December 2011.

FENOC completed a review of MRP-227-A dated December 2011 and NRC safety evaluation, "Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI Report), Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680)," against the previous version of MRP-227 dated December 2008, and determined that changes are required to the AMR results provided in LRA Table 3.1.2-2. A summary of the required table changes are listed as follows:

- Row 4 – control rod guide tube (CRGT) spacer casting changed from an expansion component with primary component link of core support shield (CSS) cast outlet nozzles, CSS vent valve discs or in-core monitoring instrumentation (IMI) guide tube spiders to a primary component with no expansion components. This revised classification is consistent with Section 3.3.7 of the NRC safety evaluation.

- Row 10 – CSS cast outlet nozzles changed from a primary component with expansion component of CRGT spacer casting to a 'no additional measures' component. This revised classification is consistent with Table 3-1 of MRP-227-A. As provided in note 2 of the table, "Thermal Embrittlement (TE) revised from 'P' to 'A' based on review of material data (ferrite content) from ONS-3 and DB; final grouping accordingly changed from 'Primary' to 'No Additional Measures.'"

- Row 12 – As discussed in Section 3.7 of the safety evaluation, the CSS vent valve disc was determined to be an active component and not subject to aging management. Therefore, row 12 is changed to "Not used."

- Row 13 – As discussed in Section 3.7 of the safety evaluation, the CSS vent valve disc shaft was determined to be an active component and not subject to aging management. Therefore, row 13 is changed to "Not used."

- Row 20 – In addition to the aging effects of cracking due to irradiation-assisted stress corrosion cracking (IASCC) and reduction in fracture toughness, the aging effects of cracking due to fatigue, loss of material and loss of preload were added for the core barrel-to-former (CBF) bolts. These revised aging effects are consistent with Table 3-1 of MRP-227-A.

- Row 21 – In addition to the aging effects of cracking due to IASCC and reduction in fracture toughness, the aging effects of cracking due to fatigue, loss of material and loss of preload were added for the baffle-to-former (FB) bolts. These revised aging effects are consistent with Table 3-1 of MRP-227-A.

- Row 22 – In addition to the aging effects of cracking due to fatigue and reduction in fracture toughness, the aging effects of loss of material and loss of preload were added for the baffle-to-baffle (BB) bolts – internal. These revised aging effects are consistent with Table 3-1 of MRP-227-A.

- Row 23 – In addition to the aging effects of cracking due to fatigue, cracking due to IASCC and reduction in fracture toughness, the aging effects of loss of material and loss of preload were added for the baffle-to-baffle (BB) bolts – external. These revised aging effects are consistent with Table 3-1 of MRP-227-A.

- Row 42 – Since the CRGT spacer casting was changed to a primary component (see Row 4 above), it is deleted as an expansion component for the IMI guide tube spiders. This revised classification is consistent with Table 4-1 of MRP-227-A.

- Row 43 – Since the CRGT spacer casting was changed to a primary component (see Row 4 above), it is deleted as an expansion component for the IMI guide tube spider-to-lower grid rib section welds. This revised classification is consistent with Table 4-1 of MRP-227-A.

- Plant-specific note 0114 – This note addressed the classification of the flow distributor (FD) bolts and their locking devices but is no longer needed for that purpose since Table 4-1 of MRP-227-A now shows the component as a primary component with expansion components of upper thermal shield (UTS) bolts and their locking devices, lower thermal shield (LTS) bolts and their locking devices, and surveillance specimen holder tube (SSHT) bolts and their locking devices. This revised classification is consistent with Section 4.1.3 of the NRC safety evaluation. Plant-specific note 0114 is now used to provide clarification that components assigned to the no additional measures group were determined to be below the screening criteria for the applicable degradation mechanisms, or were classified under this category due to Failure Modes, Effects, and Criticality

Analyses (FMECA) and functionality analysis findings, and therefore, no further action is required by MRP-227-A for managing the aging of these components.

PWR Reactor Vessel Internals Program

In response to RAI B.2.32-1 submitted by FENOC letter dated September 16, 2011 (ML11264A059), the Davis-Besse PWR Reactor Vessel Internals Program was replaced in its entirety. This revised program, provided in LRA Section B.2.32 and the corresponding Updated Safety Analysis Report (USAR) Supplement (LRA Section A.1.32), addressed each of the five plant-specific aging management program information requirements identified in Section 3.5.1 of the NRC safety evaluation, "Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680)," dated June 22, 2011.

FENOC has completed a review of the five plant-specific aging management program information requirements identified in Section 3.5.1 of the NRC safety evaluation, "Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI Report), Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680)," dated December 16, 2011, and determined that no changes are required to the Davis-Besse PWR Reactor Vessel Internals Program relative to addressing each of the five plant-specific aging management program information requirements.

As noted in the above discussion (see "Row 13" bulleted item) that addressed the required changes to LRA Table 3.1.2-2, the CSS vent valve disc shaft was determined to be an active component and not subject to aging management. Therefore, LRA Sections A.1.32 and B.2.32, both titled "PWR Reactor Vessel Internals Program," are revised to delete any discussion of the vent valve disc shaft.

In addition, the 5th paragraph of the "Detection of Aging Effects" program element for LRA Section B.2.32 was provided to address the flow distributor (FD) bolts classification and examination method/coverage/frequency. Since MRP-227-A has incorporated the changes outlined in Section 4.1.3 of the safety evaluation relative to the flow distributor (FD) bolts, the 5th paragraph is no longer needed and is therefore deleted.

Also, LRA Sections A.1.32 and B.2.32 are also revised to incorporate MRP-227-A versus MRP-227, Revision 0, as the program implementation document.

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

Section 4.2

Supplemental Question Section 4.2 Reactor Vessel Neutron Embrittlement

The NRC initiated a telephone conference call with FENOC on February 9, 2012, to discuss time-limited aging analyses (TLAAs) associated with Section 4.2 of the License Renewal Application (LRA). Two of the topics discussed during the telephone conference are as follows.

In LRA Section 4.2.1, the neutron fluence analysis for 52 EFPY (60-years of operation) was dispositioned as not a time-limited aging analysis (TLAA). The staff's position is that neutron fluence is a TLAA. FENOC agreed to change the LRA to disposition the neutron fluence analysis as a TLAA using 10 CFR 54.21(c)(1)(ii).

In LRA Section 4.2.6 including the associated USAR Supplement Section A.2.2.6, discussion was provided to address the future replacement of the reactor vessel closure head. FENOC agreed to change the LRA to acknowledge that the reactor vessel closure head was replaced in the fall of 2011.

SUPPLEMENTAL RESPONSE Section 4.2 Reactor Vessel Neutron Embrittlement

As provided in LRA Section 4.2.1, "Neutron Fluence," a neutron fluence analysis valid for 52 effective full power years (EFPY) has been prepared for the reactor vessel beltline materials to bound the projected value of 50.3 EFPY for 60-years of operation. LRA Section 4.2.1, including the associated USAR Supplement Section A.2.2.1, "Neutron Fluence," and LRA Table 4.1-1, "Time-Limited Aging Analyses," are revised to disposition the neutron fluence analysis as a time-limited aging analysis (TLAA) in accordance with 10 CFR 54.21(c)(1)(ii).

LRA Section 4.2.6, "Intergranular Separation (Underclad Cracking)," including the associated USAR Supplement Section A.2.2.6, "Intergranular Separation – Underclad Cracking," addressed the future replacement of the reactor vessel closure head/head flange schedule for the fall of 2011. Also, the operating experience program element for LRA Section B.2.29, "Nickel-Alloy Reactor Vessel Closure Head Nozzles Program," addressed the future replacement of the head. The reactor vessel closure head/head flange was replaced in the fall of year 2011. Therefore, LRA Sections 4.2.6, A.2.2.6 and B.2.29 are revised to provide this updated status for the reactor vessel closure head.

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

Regulatory Commitment List

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse) in this document. Any other actions discussed in the submittal represent intended or planned actions by FENOC; they are described only as information and are not Regulatory Commitments. Please notify Mr. Clifford I. Custer, Project Manager – Fleet License Renewal, at (724) 682-7139 of any questions regarding this document or associated Regulatory Commitments.

Regulatory Commitment	Due Date
1. Perform a training needs analysis to determine and document recommended enhancements to the training requirements for those plant personnel responsible for screening, evaluating and submitting (to the industry) aging-related operating experience items. Based on the results of the training needs analysis, identify the appropriate training materials.	December 31, 2012
2. Revise Nuclear Operating Business Practice NOBP-LP-2100, "FENOC Operating Experience Process," to require that an Evaluation Required Review of aging-related operating experience issues for structures and passive components includes: a. consideration of the material, environment, aging effect, aging mechanism, and aging management program for the affected structure or component; and, b. a provision for feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.	December 31, 2012

Regulatory Commitment	Due Date
3. Revise Nuclear Operating Procedure NOP-LP-2001, "Corrective Action Program," to require that a condition report investigation of an aging-related issue for structures and passive components includes: a. consideration of the material, environment, aging effect, aging mechanism, and aging management program for the affected structure or component; and, b. a provision for feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.	December 31, 2012
4. Revise Nuclear Operating Business Practice NOBP-LP-2008, "FENOC Corrective Action Review Board," to ensure that the Corrective Action Review Board questions whether aging was considered for condition report investigations that are reviewed by the Board.	December 31, 2012

Enclosure A

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

Letter L-12-015

Amendment No. 24 to the
DBNPS License Renewal Application

License Renewal Application
Sections Affected

Section 2.3.3.21	Section 4.2.1.3
Table 2.3.3-21	Section 4.2.6
Section 2.3.3.26	Section A.1
Table 2.3.3-26	Section A.1.32
Table 3.1.2-2	Section A.2.2.1
Table 3.1 Plant-Specific Notes	Section A.2.2.6
Table 3.3.2-21	Section B.2.29
Table 3.3.2-26	Section B.2.32
Table 4.1-1	

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
2.3.3.21	**Page 2.3-113**	**License Renewal Drawings, 9 additional drawings**

In response to Supplemental RAI 2.1-3 Abandoned Equipment, LRA Section 2.3.3.21, "Miscellaneous Liquid Radwaste System," is revised to add nine additional license renewal boundary drawings to the list of "License Renewal Drawings," to read as follows:

License Renewal Drawings

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

LR-M010D Sheet 1, LR-M010D Sheet 2, LR-M020B, LR-M031A, LR-M033A, LR-M036A, LR-M036C Sheet 1, LR-M036C Sheet 2, LR-M037C, LR-M037D, LR-M037E, LR-M037F, LR-M037G, LR-M039A, LR-M039B Sheet 1, LR-M039B Sheet 2, LR-M042B, LR-M045, LR-M046, LR-M281N13

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 2.3.3-21	**Page 2.3-114**	**12 new rows**

In response to Supplemental RAI 2.1-3 Abandoned Equipment, LRA Table 2.3.3-21, "Miscellaneous Liquid Radwaste System Components Subject to Aging Management Review," is revised to include 12 new rows for new component types, and reads as follows:

Component Type	Intended Function (as defined in Table 2.0-1)
Heat exchanger (shell) – Degasifier Heat Exchangers	*Structural integrity*
Heat exchanger (shell) – Waste Evaporator Heat Exchangers	*Structural integrity*
Pump casing – Anti Foam Tank Pump (DB-P55 WM)	*Structural integrity*
Pump casing – Degasifier Discharge Pump	*Structural integrity*
Pump casing – Evaporator Storage Tank Pumps (DB-P53-1 & 2)	*Structural integrity*
Pump casing – Primary Water Transfer Pumps (DB-P41-1, 2)	*Structural integrity*
Pump casing – Waste Evaporator Bottoms Pump	*Structural integrity*
Pump casing – Waste Evaporator Distillate Pump	*Structural integrity*
Pump casing – Waste Evaporator Vacuum Pump	*Structural integrity*
Tank – Anti Foam Tank (DB-T30)	*Structural integrity*
Tank – Degasifier (DB-S3)	*Structural integrity*
Tank – Waste Evaporator Concentrator (DB-S2)	*Structural integrity*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
2.3.3.26	**Page 2.3-128**	**License Renewal Drawings, 1 additional drawing**

In response to Supplemental RAI 2.1-3 Abandoned Equipment, LRA Section 2.3.3.26, "Service Water System," is revised to add one additional license renewal boundary drawing to the list of "License Renewal Drawings," to read as follows:

License Renewal Drawings

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

LR-M006D, LR-M012E, LR-M036A, LR-M036B, LR-M041A, LR-M041B, LR-M041C

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 2.3.3-26	**Page 2.3-130**	**Intended Function**

In response to Supplemental RAI 2.1-3 Abandoned Equipment, LRA Table 2.3.3-26, "Service Water System Components Subject to Aging Management Review," is revised to include the "structural integrity " intended function for the orifice component type, and reads as follows:

Component Type	Intended Function (as defined in Table 2.0-1)
Orifice	*Pressure boundary* *Structural integrity* *Throttling*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.1.2-2	**Pages 3.1-60 thru 3.1-121**	**27 Rows, various columns**
Table 3.1.2 Plant-Specific Notes	**Page 3.1-187**	**Note 0114**

In response to Supplemental RAI 3.1.2.2-2 Reactor Vessel Internals Aging Management, 27 rows of LRA Table 3.1.2-2, "Aging Management Review Results – Reactor Vessel Internals," and Plant-Specific Note 0114, previously replaced in their entirety by FENOC letter dated September 16, 2011 (ML11264A059), are revised to read as follows:

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
Plenum Cover Assembly									
3	Alloy X-750 Dowels-to-Plenum Cover Bottom Flange Welds (no additional measures component)	Support	Nickel Alloy	Borated Reactor Coolant with Neutron Fluence	Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-236)	None	*A* *0114*

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
Control Rod Guide Tube (CRGT) Assembly									
4	*CRGT Spacer Casting* ~~*(expansion component with primary component link of CSS Cast Outlet Nozzles, CSS Vent Valve Discs or IMI Guide Tube Spiders)*~~ *(primary component with no expansion components)*	Support	Cast Austenitic Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Reduction in fracture toughness	PWR Reactor Vessel Internals	IV.B4-4 (IV.B4.RP-242)	3.1.1-80	A
5	CRGT Rod Guide Tubes (no additional measures component)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Loss of material - wear	PWR Reactor Vessel Internals	None (IV.B4.RP-237)	None	*A 0114*
6	CRGT Rod Guide Sectors (no additional measures component)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Loss of material - wear	PWR Reactor Vessel Internals	None (IV.B4.RP-237)	None	*A 0114*

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
Core Support Shield (CSS) Assembly									
8	Upper Core Barrel (UCB) Bolts (original bolts) and their locking devices (primary component with expansion components of UTS Bolts and their locking devices, LTS Bolts and their locking devices, and SSHT Bolts and their locking devices)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Bolt: Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	IV.B4-20 (IV.B4.RP-248)	3.1.1-37	*A* ~~0114~~
					Locking Device: Loss of material – wear	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-243)	3.1.1-22	C

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
9	Upper Core Barrel (UCB) Bolts (replacement bolts) and their locking devices (primary component with expansion components of UTS Bolts and their locking devices, LTS Bolts and their locking devices, and SSHT Bolts and their locking devices)	Support	Bolt: Nickel Alloy Locking Devices: Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Bolt: Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	IV.B4-20 (IV.B4.RP-248)	3.1.1-37	*A* ~~*0114*~~
					Bolt: Cumulative fatigue damage - fatigue	TLAA	IV.B4-37 (IV.B4.R-53)	3.1.1-05	A
					Locking Device: Loss of material – wear	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-243)	3.1.1-22	C
10	*CSS Cast Outlet Nozzles* ~~*(primary component with expansion components as follows: CRGT Spacer Casting)*~~ *(no additional measures component)*	Support	Cast Austenitic Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Reduction in fracture toughness	PWR Reactor Vessel Internals	IV.B4-21 (IV.B4.RP-253)	3.1.1-80	*A* *0114*
12	~~*CSS Vent Valve Discs*~~ ~~*(primary component with expansion component of CRGT Spacer Casting)*~~ *Not Used*	~~*Support*~~ ~~*Flow Control*~~	~~*Cast Austenitic Stainless Steel*~~	~~*Borated Reactor Coolant with Neutron Fluence*~~	~~*Reduction in fracture toughness*~~	~~*PWR Reactor Vessel Internals*~~	~~*IV.B4-21 (IV.B4.RP-253)*~~	~~*3.1.1-80*~~	~~*A*~~

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
13	~~CSS Vent Valve Disc Shaft~~ ~~(primary component with no expansion components)~~ Not Used	~~Support~~	~~Stainless Steel~~	~~Borated Reactor Coolant with Neutron Fluence~~	~~Reduction in fracture toughness~~	~~PWR Reactor Vessel Internals~~	~~IV.B4-16 (IV.B4.RP-252); (IV.B4.RP-239)~~	~~3.1.1-22~~	~~A~~
Core Barrel Assembly									
15	Alloy X-750 Core Barrel-to-Former Plate Dowel (no additional measures component)	Support	Nickel Alloy	Borated Reactor Coolant with Neutron Fluence	Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-236)	None	A 0114
					Reduction in fracture toughness	PWR Reactor Vessel Internals	None (IV.B4.RP-237)	None	A 0114
16	Alloy X-750 Dowel-to-Core Barrel Cylinder Fillet Welds (no additional measures component)	Support	Nickel Alloy	Borated Reactor Coolant with Neutron Fluence	Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-236)	None	A 0114
17	Thermal Shield Upper Restraint Cap Screws (Not Exposed) (no additional measures component)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Cracking – fatigue Loss of material – wear Loss of preload	PWR Reactor Vessel Internals	None (IV.B4.RP-237)	None	A 0114

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
20	Core Barrel-to-Former (CBF) Bolts (expansion component with primary component link of FB Bolts)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	*Cracking – IASCC, Fatigue*	PWR Reactor Vessel Internals PWR Water Chemistry	*IV.B4-07 (IV.B4.RP-244); (IV.B4.RP-238)* *None (IV.B4.RP-375)*	3.1.1-30	A
					Reduction in fracture toughness *Loss of material – wear* *Loss of preload*	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-243); (IV.B4.RP-239)	3.1.1-22	A

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
21	Baffle-to-Former (FB) Bolts (primary component with expansion components of BB Bolts and CBF Bolts)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	*Cracking – IASCC, Fatigue*	PWR Reactor Vessel Internals PWR Water Chemistry	*IV.B4-07 (IV.B4.RP-241)* *None (IV.B4.RP-375)*	3.1.1-30	A
					Reduction in fracture toughness *Loss of material – wear* *Loss of preload*	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-240)	3.1.1-22	A
22	Baffle-to-Baffle (BB) Bolts – internal (expansion component with primary component link of FB Bolts)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Cracking - fatigue	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-375)	None	A
					Reduction in fracture toughness *Loss of material – wear* *Loss of preload*	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-243)	3.1.1-22	A

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
23	Baffle-to-Baffle (BB) Bolts – external (expansion component with primary component link of FB Bolts)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Cracking - fatigue	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-375)	None	A
					Cracking - IASCC	PWR Reactor Vessel Internals PWR Water Chemistry	IV.B4-07 (IV.B4.RP-244); (IV.B4.RP-238)	3.1.1-30	A
					Reduction in fracture toughness *Loss of material – wear* *Loss of preload*	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-243); (IV.B4.RP-239)	3.1.1-22	A
26	Lower Core Barrel (LCB) Bolts (original) and their locking devices (primary component with expansion components of UTS Bolts and their locking devices, LTS Bolts and their locking devices, and SSHT Bolts and their locking devices)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Bolt: Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	IV.B4-13 (IV.B4.RP-247)	3.1.1-37	*A* ~~*0114*~~
					Locking Device: Loss of material – wear	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-243)	3.1.1-22	C

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
27	Lower Core Barrel (LCB) Bolts (replacement) and their locking devices (primary component with expansion components of UTS Bolts and their locking devices, LTS Bolts and their locking devices, and SSHT Bolts and their locking devices)	Support	Bolt: Nickel Alloy Locking Devices: Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Bolt: Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	IV.B4-13 (IV.B4.RP-247)	3.1.1-37	A ~~0114~~
					Bolt: Cumulative fatigue damage - fatigue	TLAA	IV.B4-37 (IV.B4.R-53)	3.1.1-05	A
					Locking Device: Loss of material – wear	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-243)	3.1.1-22	C
Upper Grid Assembly									
30	Alloy X-750 Dowel-to-Upper Grid Rib Section Bottom Flange Welds (no additional measures component)	Support	Nickel Alloy	Borated Reactor Coolant with Neutron Fluence	Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-236)	None	A 0114
Lower Grid Assembly									
35	Alloy X-750 Dowel-to-Lower Grid Shell Forging Welds (no additional measures component)	Support	Nickel Alloy	Borated Reactor Coolant with Neutron Fluence	Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-236)	None	A 0114

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
36	Alloy X-750 Dowel-to-Lower Grid Rib Section Welds (no additional measures component)	Support	Nickel Alloy	Borated Reactor Coolant with Neutron Fluence	Cracking - SCC, IASCC	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-236)	None	*A* *0114*
					Reduction in fracture toughness	PWR Reactor Vessel Internals	None (IV.B4.RP-237)	None	*A* *0114*
37	Lower Grid Rib-to-Shell Forging Cap Screws (no additional measures component)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Cracking – fatigue Loss of material – wear Loss of preload	PWR Reactor Vessel Internals	None (IV.B4.RP-237)	None	*A* *0114*
38	Lower Grid Support Post Pipe Cap Screws (no additional measures component)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Cracking – fatigue Loss of material – wear Loss of preload	PWR Reactor Vessel Internals	None (IV.B4.RP-237)	None	*A* *0114*

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
Flow Distributor Assembly									
40	Flow Distributor (FD) Bolts and their locking devices (primary component with expansion components of UTS Bolts and their locking devices, LTS Bolts and their locking devices, and SSHT Bolts and their locking devices)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Bolt: Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	IV.B4-25 (IV.B4.RP-256)	3.1.1-37	A ~~0114~~
					Locking Device: Loss of material – wear	PWR Reactor Vessel Internals	IV.B4-01 (IV.B4.RP-243)	3.1.1-22	C
41	Alloy X-750 Dowel-to-Flow Distributor Flange Welds (no additional measures component)	Support	Nickel Alloy	Borated Reactor Coolant with Neutron Fluence	Cracking - SCC	PWR Reactor Vessel Internals PWR Water Chemistry	None (IV.B4.RP-236)	None	A 0114

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals									
Row No.	**Component Type**	**Intended Function(s)**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801 Rev. 1 Item (Rev. 2 Item)**	**Table 1 Item**	**Notes**
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly									
42	*IMI Guide Tube Spiders (primary component with expansion components of ~~CRGT Spacer Casting and~~ Lower Fuel Assembly Support Pads: Pad, Pad-to-Rib Section Weld, Cap Screw and associated Locking Weld, Alloy X-750 Dowel and Alloy X-750 Dowel Locking Weld)*	Support	Cast Austenitic Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Reduction in fracture toughness	PWR Reactor Vessel Internals	IV.B4-28 (IV.B4.RP-258)	3.1.1-80	A

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
43	*IMI Guide Tube Spider-to-Lower Grid Rib Section Welds* *(primary component with expansion components of ~~CRGT Spacer Casting and~~ Lower Fuel Assembly Support Pads: Pad, Pad-to-Rib Section Weld, Cap Screw and associated Locking Weld, Alloy X-750 Dowel and Alloy X-750 Dowel Locking Weld)*	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Reduction in fracture toughness	PWR Reactor Vessel Internals	IV.B4-31 (IV.B4.RP-259)	3.1.1-22	A

Plant-Specific Notes:	
0114	~~*Flow distributor (FD) bolts were reassigned as a primary component per MRP-227, Rev. 0 as amended by the safety evaluation. Due to this change, the FD bolts are not an expansion component for the UCB and LCB bolts.*~~ *Components assigned to the "No Additional Measures" group were determined to be below the screening criteria for the applicable degradation mechanisms, or were classified under this category due to Failure Modes, Effects, and Criticality Analyses (FMECA) and functionality analysis findings, and therefore, no further action is required by MRP-227-A for managing the aging of these components.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.3.2-21	**Page 3.3-411 thru 3.3-417**	**36 New Rows; and, Rows 55, 60, 65, 70, 75, 79, & 84, "Notes" column**

In response to Supplemental RAI 2.1-3 Abandoned Equipment, 36 new rows are added to LRA Table 3.3.2-21, "Aging Management Review Results – Miscellaneous Liquid Radwaste System." Additionally, during the review, it was identified that generic Note "A" was assigned in the "Notes" column for rows 55, 60, 65, 70, 75, 79, and 84, but should have been Note "C". LRA Table 3.3.2-21 is revised to include new rows and correct assigned notes, as follows:

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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) – Degasifier Heat Exchangers*	*Structural integrity*	*Stainless Steel*	*Raw water (Internal)*	*Loss of material*	*Collection, Drainage, and Treatment Components Inspection*	*VII.C1-15*	*3.3.1-79*	*E*
--	*Heat Exchanger (shell) – Degasifier Heat Exchangers*	*Structural integrity*	*Stainless Steel*	*Air with borated water leakage (External)*	*None*	*None*	*VII.J-16*	*3.3.1-99*	*C*

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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) – Degasifier Heat Exchangers*	*Structural integrity*	*Stainless Steel*	*Air-indoor uncontrolled (External)*	*None*	*None*	*VII.J-15*	*3.3.1-94*	*C*
--	*Heat Exchanger (shell) – Waste Evaporator Heat Exchangers*	*Structural integrity*	*Stainless Steel*	*Raw water (Internal)*	*Loss of material*	*Collection, Drainage, and Treatment Components Inspection*	*VII.C1-15*	*3.3.1-79*	*E*
--	*Heat Exchanger (shell) – Waste Evaporator Heat Exchangers*	*Structural integrity*	*Stainless Steel*	*Air with borated water leakage (External)*	*None*	*None*	*VII.J-16*	*3.3.1-99*	*C*

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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) – Waste Evaporator Heat Exchangers	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	C
--	Pump casing – Anti Foam Tank Pump (DB-P55_WM)	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E
--	Pump casing – Anti Foam Tank Pump (DB-P55_WM)	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	A
--	Pump casing – Anti Foam Tank Pump (DB-P55_WM)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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 – Degasifier Discharge Pump	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E
--	Pump casing – Degasifier Discharge Pump	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	A
--	Pump casing – Degasifier Discharge Pump	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A
--	Pump Casing – Evaporator Storage Tank Pumps (DB-P53-1, 2)	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E
--	Pump Casing – Evaporator Storage Tank Pumps (DB-P53-1, 2)	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	A

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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 – Evaporator Storage Tank Pumps (DB-P53-1, 2)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A
--	Pump casing – Primary Water Transfer Pump	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E
--	Pump casing – Primary Water Transfer Pump	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	A
--	Pump casing – Primary Water Transfer Pump	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A
--	Pump casing – Waste Evaporator Bottoms Pump	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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 – Waste Evaporator Bottoms Pump	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	A
--	Pump casing – Waste Evaporator Bottoms Pump	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A
--	Pump casing – Waste Evaporator Distillate Pump	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E
--	Pump casing – Waste Evaporator Distillate Pump	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	A
--	Pump casing – Waste Evaporator Distillate Pump	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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 – Waste Evaporator Vacuum Pump	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E
--	Pump casing – Waste Evaporator Vacuum Pump	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	A
--	Pump casing – Waste Evaporator Vacuum Pump	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A
--	Tank – Anti Foam Tank (DB-T30)	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E
--	Tank – Anti Foam Tank (DB-T30)	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	C

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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
--	Tank – Anti Foam Tank (DB-T30)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	C
--	Tank – Degasifier (DB-S3)	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E
--	Tank – Degasifier (DB-S3)	Structural integrity	Stainless Steel	Air with borated water leakage (External)	None	None	VII.J-16	3.3.1-99	C
--	Tank – Degasifier (DB-S3)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	C
--	Tank – Waste Evaporator Concentrator (DB-S2)	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Collection, Drainage, and Treatment Components Inspection	VII.C1-15	3.3.1-79	E

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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**
--	*Tank – Waste Evaporator Concentrator (DB-S2)*	*Structural integrity*	*Stainless Steel*	*Air with borated water leakage (External)*	*None*	*None*	*VII.J-16*	*3.3.1-99*	*C*
--	*Tank – Waste Evaporator Concentrator (DB-S2)*	*Structural integrity*	*Stainless Steel*	*Air-indoor uncontrolled (External)*	*None*	*None*	*VII.J-15*	*3.3.1-94*	*C*
55	Tank – DWDT 1-1 (DB-T27)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	~~A~~ C
60	Tank – DWDT 1-1 Hold-up Tank (DB-T161)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	~~A~~ C
65	Tank – Miscellaneous Liquid Waste Monitor Tank (DB-T29)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	~~A~~ C

Table 3.3.2-21 Aging Management Review Results – Miscellaneous Liquid Radwaste 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**
70	Tank – Miscellaneous Waste Drain Tank (DB-T26)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	~~A~~ C
75	Tank – Miscellaneous Waste Evaporator Storage Tank (DB-T28)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	~~A~~ C
79	Tank – Radwaste Demineralizer Skid Vessel (1 through 5)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	~~A~~ C
84	Tank – Waste Polishing Demineralizer (DB-T125)	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	~~A~~ C

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 3.3.2-26	**Page 3.3-482**	**9 New Rows**

In response to Supplemental RAI 2.1-3 Abandoned Equipment, nine new rows are added to LRA Table 3.3.2-26, "Aging Management Review Results – Service Water System," to read as follows:

Table 3.3.2-26 Aging Management Review Results – Service Water 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
--	*Orifice*	*Structural integrity*	*Stainless Steel*	*Raw water (Internal)*	*Loss of material*	*Open-Cycle Cooling Water*	*VII.C1-15*	*3.3.1-79*	*B*
--	*Orifice*	*Structural integrity*	*Stainless Steel*	*Air-indoor uncontrolled (External)*	*None*	*None*	*VII.J-15*	*3.3.1-94*	*A*
--	*Orifice*	*Structural integrity*	*Stainless Steel*	*Condensation (External)*	*Loss of material*	*External Surfaces Monitoring*	*VII.F1-1*	*3.3.1-27*	*E*
--	*Piping*	*Structural integrity*	*Stainless Steel*	*Raw water (Internal)*	*Loss of material*	*Open-Cycle Cooling Water*	*VII.C1-15*	*3.3.1-79*	*B*

Table 3.3.2-26 Aging Management Review Results – Service Water 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
--	Piping	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A
--	Piping	Structural integrity	Stainless Steel	Condensation (External)	Loss of material	External Surfaces Monitoring	VII.F1-1	3.3.1-27	E
--	Valve Body	Structural integrity	Stainless Steel	Raw water (Internal)	Loss of material	Open-Cycle Cooling Water	VII.C1-15	3.3.1-79	B
--	Valve Body	Structural integrity	Stainless Steel	Air-indoor uncontrolled (External)	None	None	VII.J-15	3.3.1-94	A
--	Valve Body	Structural integrity	Stainless Steel	Condensation (External)	Loss of material	External Surfaces Monitoring	VII.F1-1	3.3.1-27	E

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
Table 4.1-1	**Page 4.1-3**	**"Neutron Fluence" row**

In response to Supplemental Question Section 4.2 Reactor Vessel Neutron Embrittlement, the "Neutron Fluence" row of LRA Table 4.1-1, "Time-Limited Aging Analyses," is revised as follows:

Table 4.1-1 Time-Limited Aging Analyses

Results of TLAA Evaluation by Category	54.21(c)(1) Paragraph	LRA Section
Reactor Vessel Neutron Fluence		4.2
Neutron Fluence	~~*Not a TLAA*~~ *(ii)*	4.2.1

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.2.1.3	**Page 4.2-3**	**New 4th paragraph; and, "Disposition"**

In response to Supplemental Question Section 4.2 Reactor Vessel Neutron Embrittlement, LRA Section 4.2.1.3, "Beltline Evaluation," is revised to include a new fourth paragraph and a revised "Disposition" statement, to read as follows:

Neutron fluence analysis valid for 52 EFPY have been prepared for the reactor vessel beltline materials to bound the projected value of 50.3 EFPY for 60-years of operation. Therefore, the neutron fluence analysis has been projected to the end of the period of extended operation.

~~***Disposition:*** *Not a TLAA*~~ ~~*Neutron fluence is an assumption used in various neutron embrittlement TLAAs evaluated below.*~~

Disposition: *10 CFR 54.21(c)(1)(ii)* *The neutron fluence analysis has been projected to the end of the period of extended operation.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.2.6	**Page 4.2-14**	**5th paragraph**

In response to Supplemental Question Section 4.2 Reactor Vessel Neutron Embrittlement, the fifth paragraph on page 4.2-14 of LRA Section 4.2.6, "Intergranular Separation (Underclad Cracking)," is revised to read as follows:

~~*As provided in Confirmatory Action Letter, Number 3-10-001, FENOC has voluntarily committed to shutdown the Davis-Besse plant no later than October 1, 2011, and replace the RV closure head. Therefore, the current head (purchased from the Midland Plant and installed during the Cycle 13 refueling outage) is not considered in the underclad cracking evaluation. The replacement RV closure head/head flange, to be installed during the October 2011 outage, was fabricated using SA-508 Class 3 material, which is not susceptible to intergranular separations.*~~ *The reactor vessel closure head/head flange was replaced in the Fall of year 2011. This replacement head was fabricated using SA-508 Class 3 material, which is not susceptible to the subject intergranular separations.* Therefore, this replacement closure head/head flange is not considered in the underclad cracking evaluation.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.1	**Page A-9**	**3rd paragraph**

In response to RAI B.1.4-3, the third paragraph of LRA Section A.1, "Summary Descriptions of Aging Management Programs and Activities," previously added by FENOC letter dated August 17, 2011 (ML11231A966), is replaced in its entirety to read as follows:

~~*Existing FENOC processes require reviews of relevant site and industry operating experience and periodic benchmarking to ensure program enhancements are identified and implemented. Such ongoing reviews identify potential needs for aging management program revisions to ensure their effectiveness throughout the period of extended operation.*~~

Operating Experience

The intent of evaluating and incorporating operating experience lessons-learned, including aging-related lessons-learned, is prevention. The lessons-learned are used to improve plant operation, equipment material condition, and aging management to minimize equipment degradation and prevent loss of equipment intended functions. The FENOC Operating Experience Program processes and procedures for the ongoing review of operating experience include the following attributes:

- *Personnel responsible for screening, evaluating and submitting (to the industry) aging-related operating experience items are qualified for the task.*
- *While the programs and procedures may specify reviews of certain sources of information, such as NRC generic communications and Institute of Nuclear Power Operations reports, they allow for any potential source of relevant plant-specific or industry operating experience information.*
- *The processes are adequate so as to not preclude the consideration of operating experience related to aging management. The processes allow for appropriately gathering information on structures and passive components within the scope of license renewal, their materials, environments, aging effects, and aging mechanisms, and the aging*

management programs credited for managing the effects of aging, including the activities under these programs (e.g., inspection methods, preventive actions or evaluation techniques).

- *Plant-specific operating experience, including aging-related operating experience, is documented in condition reports and processed using the FENOC Corrective Action Program. Condition reports for adverse conditions and related documents captured in the Corrective Action Program database are quality records and are auditable and retrievable.*

- *Industry operating experience, including aging-related operating experience, is entered into the Operating Experience Program database and screened for applicability to FENOC. Documents captured in the Operating Experience Program database are retrievable.*

- *Evaluations of internal and external aging-related operating experience issues associated with structures and passive components include consideration of the affected structure or component, material, environment, aging effect, aging mechanism, and aging management program, with feedback to the affected aging management program owner for consideration of the impact to aging management program effectiveness.*

- *Aging management program owners review data collected by program activities, use the Corrective Action Program to document adverse conditions to ensure they will be addressed and corrected, maintain required records for the program, maintain the program current, and implement revisions as needed based on program results and internal or external operating experience evaluations. Revision of existing or development of new aging management programs based on operating experience evaluations is performed through corrective actions using the Corrective Action Program, or by action items identified in the Operating Experience Program database.*

- *Noteworthy plant-specific aging-related operating experience is shared with the other FENOC sites and the industry. The Operating Experience Program procedure provides guidance on sharing internal operating experience.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.1.32	**Page A-21**	**Paragraphs 1, 2, 3, 7 and 12 (last)**

In response to Supplemental RAI 3.1.2.2-2 Reactor Vessel Internals Aging Management, the 1st, 2nd, 3rd, 7th and 12th paragraphs of LRA Section A.1.32, "PWR Reactor Vessel Internals Program," previously replaced in its entirety by FENOC letter dated September 16, 2011 (ML11264A059), are revised to read as follows (affected sentences highlighted for clarity):

A.1.32 PWR REACTOR VESSEL INTERNALS PROGRAM

The PWR Reactor Vessel Internals Program relies on implementation of the Electric Power Research Institute (EPRI) Report No.~~1016596~~ 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," and EPRI Report No. 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)," to manage the aging effects on the reactor vessel internal (RVI) components.

This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at Davis-Besse, a Babcock & Wilcox (B&W) designed plant. These aging effects include (a) various forms of cracking, including stress corrosion cracking (SCC), which also encompasses primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; and (d) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep. In addition, the program includes management of the time-limited aging analysis (TLAA) identified in License Renewal Application (LRA) Section A.2.2.7 for reduction in fracture toughness of the reactor vessel internals. *This TLAA will be managed in accordance with the implementation of the MRP-227 guidelines~~, as amended by the MRP-227 safety evaluation,~~ including all activities associated with Davis-Besse's responses to plant-specific action items identified in Section 4.2 of the safety evaluation.*

The program applies the guidance in MRP-227~~, Rev. 0, as amended by the safety evaluation~~ for inspecting, evaluating, and, if applicable, dispositioning non-conforming RVI components at Davis-Besse. The program conforms to the definition of a sampling-based condition monitoring program, as defined by the Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms

will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections if the extent of the degradation effects exceeds the expected levels.

The MRP-227 guidance for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process. Through this process, the reactor internals were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions of each group are provided in GALL Chapter IX.B.

The result of this four-step sample selection process is a set of Primary Internals Component locations for each of the three plant designs (Westinghouse, Combustion Engineering and Babcock & Wilcox) that are expected to show the leading indications of the degradation effects, with another set of Expansion Internals Component locations that are specified to expand the sample should the indications be more severe than anticipated. The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs. A fourth set of internals locations are deemed to require no additional measures. As a result, the program typically identifies 5 to 15 percent of the RVI locations as Primary Component locations for inspections, with another 7 to 10 percent of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15 percent of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. This process thus uses appropriate component functionality criteria, age-related degradation susceptibility criteria, and failure consequence criteria to identify the components that will be inspected under the program in a manner that conforms to the sampling criteria for sampling-based condition monitoring programs in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Consequently, the sample selection process is adequate to assure that the intended function(s) of the PWR reactor internal components are maintained during the period of extended operation.

No existing generic industry programs contain the specificity considered sufficient for monitoring the aging effects addressed by the MRP-227 guidelines for B&W plants. Therefore, no components for B&W plants were placed into the Existing Programs group.

MRP-227 I&E guidelines require a visual (VT-3) examination of the core support shield (CSS) vent valve retaining rings~~ and disc shaft~~ for every 10 year Inservice Inspection Interval. In addition, Davis-Besse Technical Specification 5.5.4 requires testing of the CSS vent valves every 24 months to verify by visual inspection that the valve body and valve disc exhibit no abnormal degradation, verify the valve is not stuck in an open position, and verify by manual actuation that the valve is fully open when a force of ≤400 lbs is applied vertically upward.

The technical specification inspection will continue to be performed at the prescribed frequency of 24 months. The MRP-227 required visual (VT-3) examination will also be performed at the prescribed frequency of every 10 year Inservice Inspection Interval.

The program's use of visual examination methods in MRP-227 for detection of relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code, Section XI rules for visual examination. However, the program's adoption of the MRP-227 guidance for visual examinations goes beyond the ASME Code, Section XI visual examination criteria because additional guidance is incorporated into MRP-227 to clarify how the particular visual examination methods will be used to detect relevant conditions and describes in more detail how the visual techniques relate to the specific RVI components and how to detect their applicable age-related degradation effects.

The technical basis for detecting relevant conditions using volumetric ultrasonic testing (UT) inspection techniques can be found in MRP-228, where the review of existing bolting UT examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations. Specifically, the capability of program's UT volumetric methods to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting in B&W and Westinghouse units, has been well demonstrated by operating experience. In addition, the program's adoption of the MRP-227 guidance and process incorporates the UT criteria in MRP-228, which calls for the technical justifications that are needed for volumetric examination method demonstrations, required by the ASME Code, Section V.

The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227. The program thus provides reasonable assurance for the long-term integrity and safe operation of reactor internals in all commercial operating U.S. PWR nuclear power plants.

Age-related degradation in the reactor internals is managed through an integrated program. Specific features of the integrated program are listed in the following ten program elements. Degradation due to changes in material properties (e.g., loss of fracture toughness) was considered in the determination of inspection recommendations and is managed by the requirement to use appropriately degraded properties in the evaluation of identified defects. The integrated program is implemented by the applicant through an inspection plan.

The Davis-Besse PWR Reactor Vessel Internals Program will address all plant-specific action items applicable to Davis-Besse that are established in

Section 4.2 of the safety evaluation for MRP-227. *In addition, a plant-specific inspection plan for ensuring the implementation of MRP-227 program guidelines~~, as amended by the safety evaluation for MRP-227,~~ and Davis-Besse's responses to the plant-specific action items, as identified in Section 4.2 of the safety evaluation~~ for MRP-227~~, will be submitted for NRC review and approval.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.2.1	**Pages A-30 and A-31**	**1st paragraph; and New last paragraph**

In response to Supplemental Question Section 4.2 Reactor Vessel Neutron Embrittlement, the 1st paragraph of LRA Section A.2.2.1, "Neutron Fluence," is deleted, and a new last paragraph is added to the Section on LRA page A-31, as follows:

1st Paragraph

~~*Neutron fluence is not a TLAA, it is a time-limited assumption used in the evaluation of neutron embrittlement TLAAs.*~~

New Last Paragraph

A neutron fluence analysis valid for 52 EFPY has been prepared for the reactor vessel beltline materials to bound the projected value of 50.3 EFPY for 60 years of operation. Therefore, the neutron fluence analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.2.6	**Page A-34**	**3rd paragraph**

In response to Supplemental Question Section 4.2 Reactor Vessel Neutron Embrittlement, the 3rd paragraph of LRA Section A.2.2.6, "Neutron Fluence," is revised to read as follows:

~~*[Proposed text for this section, pending closure of Confirmatory Action Letter CAL No. 3-10-001 commitments related to the replacement of the Davis-Besse closure head in 2011.]*~~ *The reactor vessel closure head/head flange was replaced in the fall of year 2011.* This replacement head was fabricated using SA-508 Class 3 material, which is not susceptible to the subject intergranular separations. Therefore, this replacement closure head/head flange is not considered in the underclad cracking evaluation.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
B.2.29	**Page B-119**	**2nd Paragraph**

In response to Supplemental Question Section 4.2 Reactor Vessel Neutron Embrittlement, the 2nd paragraph of the "Operating Experience" subsection of LRA Section B.2.29, "Nickel-Alloy Reactor Vessel Closure Head Nozzles Program," on LRA page B-119, is revised as follows:

In March 2010, ultrasonic examinations of the control rod drive mechanism nozzles constructed of Alloy 600 material identified flaws on multiple nozzles. Active leakage was identified on one nozzle. The direct cause was Primary Water Stress Corrosion Cracking. The reactor vessel closure head had been in operation approximately six years. An inside diameter temper bead half-nozzle weld repair was utilized. Post-repair inspections were completed with acceptable results. ~~*As provided in Confirmatory Action Letter, Number 3-10-001, Mark A. Satorius (NRC) to Barry S. Allen (FENOC), dated 6-23-2010, FENOC has voluntarily committed to shutdown the Davis-Besse plant no later than October 1, 2011, and replace the reactor pressure vessel head with one manufactured using materials resistant to PWSCC.*~~ *The reactor vessel closure head was replaced with a new head in the fall of year 2011. The CRDM nozzles for the new head were fabricated using Alloy 690 material that is less susceptible to PWSCC.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
B.2.32	**Pages B-129 thru B-133**	**Various paragraphs**

In response to Supplemental RAI 3.1.2.2-2 Reactor Vessel Internals Aging Management, various paragraphs of LRA Section B.2.32, "PWR Reactor Vessel Internals Program," previously replaced in its entirety by FENOC letter dated September 16, 2011 (ML11264A059), are revised to read as follows (affected sentences highlighted for clarity):

B.2.32 PWR REACTOR VESSEL INTERNALS PROGRAM

Program Description

The PWR Reactor Vessel Internals Program relies on implementation of the Electric Power Research Institute (EPRI) Report No. ~~1016596~~ 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," and EPRI Report No. 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)," to manage the aging effects on the reactor vessel internal (RVI) components.

This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at Davis-Besse, a Babcock & Wilcox (B&W) designed plant. These aging effects include (a) various forms of cracking, including stress corrosion cracking (SCC), which also encompasses primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; and (d) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep. In addition, the program includes management of the time-limited aging analysis (TLAA) identified in License Renewal Application (LRA) Section 4.2.7 for reduction in fracture toughness of the reactor vessel internals. *This TLAA will be managed in accordance with the implementation of the MRP-227 guidelines~~, as amended by the MRP-227 safety evaluation,~~ including all activities associated with Davis-Besse's responses to plant-specific action items identified in Section 4.2 of the safety evaluation.*

The program applies the guidance in MRP-227~~, Rev. 0, as amended by the safety evaluation~~ for inspecting, evaluating, and, if applicable, dispositioning non-conforming RVI components at Davis-Besse. The program conforms to the definition of a sampling-based condition monitoring program, as defined by the

Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections if the extent of the degradation effects exceeds the expected levels.

The MRP-227 guidance for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process. Through this process, the reactor internals were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions of each group are provided in GALL Chapter IX.B.

The result of this four-step sample selection process is a set of Primary Internals Component locations for each of the three plant designs (Westinghouse, Combustion Engineering and Babcock & Wilcox) that are expected to show the leading indications of the degradation effects, with another set of Expansion Internals Component locations that are specified to expand the sample should the indications be more severe than anticipated. The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs. A fourth set of internals locations are deemed to require no additional measures. As a result, the program typically identifies 5 to 15 percent of the RVI locations as Primary Component locations for inspections, with another 7 to 10 percent of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15 percent of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. This process thus uses appropriate component functionality criteria, age-related degradation susceptibility criteria, and failure consequence criteria to identify the components that will be inspected under the program in a manner that conforms to the sampling criteria for sampling-based condition monitoring programs in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Consequently, the sample selection process is adequate to assure that the intended function(s) of the PWR reactor internal components are maintained during the period of extended operation.

No existing generic industry programs contain the specificity considered sufficient for monitoring the aging effects addressed by the MRP-227 guidelines for B&W plants. Therefore, no components for B&W plants were placed into the Existing Programs group.

The program's use of visual examination methods in MRP-227 for detection of relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code, Section XI rules for visual examination. However, the program's adoption of the MRP-227

guidance for visual examinations goes beyond the ASME Code, Section XI visual examination criteria because additional guidance is incorporated into MRP-227 to clarify how the particular visual examination methods will be used to detect relevant conditions and describes in more detail how the visual techniques relate to the specific RVI components and how to detect their applicable age-related degradation effects.

The technical basis for detecting relevant conditions using volumetric ultrasonic testing (UT) inspection techniques can be found in MRP-228, where the review of existing bolting UT examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations. Specifically, the capability of program's UT volumetric methods to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting in B&W and Westinghouse units, has been well demonstrated by operating experience. In addition, the program's adoption of the MRP-227 guidance and process incorporates the UT criteria in MRP-228, which calls for the technical justifications that are needed for volumetric examination method demonstrations, required by the ASME Code, Section V.

The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227. The program thus provides reasonable assurance for the long-term integrity and safe operation of reactor internals in all commercial operating U.S. PWR nuclear power plants.

Age-related degradation in the reactor internals is managed through an integrated program. Specific features of the integrated program are listed in the following ten program elements. Degradation due to changes in material properties (e.g., loss of fracture toughness) was considered in the determination of inspection recommendations and is managed by the requirement to use appropriately degraded properties in the evaluation of identified defects. The integrated program is implemented by the applicant through an inspection plan.

The Davis-Besse PWR Reactor Vessel Internals Program will address all plant-specific action items applicable to Davis-Besse that are established in Section 4.2 of the safety evaluation for MRP-227. *In addition, a plant-specific inspection plan for ensuring the implementation of MRP-227 program guidelines~~, as amended by the safety evaluation for MRP-227,~~ and Davis-Besse's responses to the plant-specific action items, as identified in Section 4.2 of the safety evaluation~~ for MRP-227~~, will be submitted for NRC review and approval.*

NUREG-1801 Consistency

The PWR Reactor Vessel Internals Program is a new Davis-Besse program that will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Rev. 2, Section XI.M16A, "PWR Vessel Internals." The results of an evaluation for each element are provided below.

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

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

- Scope
  The scope of the program includes all RVI components at Davis-Besse, which is built to a B&W NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC-endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's aging management program (AMP) that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

In addition, the scope of the program includes management of the time-limited aging analysis (TLAA) identified in LRA Section 4.2.7 for reduction in fracture toughness of the reactor vessel internals. *This TLAA will be managed in accordance with the implementation of the MRP-227 guidelines~~, as amended by the MRP-227 safety evaluation,~~ including all activities associated with Davis-Besse's responses to plant-specific action items identified in the Section 4.2 of the safety evaluation.*

The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAIs as well. Davis-Besse's responses to the plant-specific action items, as identified in Section 4.2 of the safety evaluation for MRP-227, will be submitted for NRC review and approval.

The guidance in Section 2.4 of MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. General assumptions used in the analysis include:

1) 30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation;
2) base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule; and
3) no design changes beyond those identified in general industry guidance or recommended by the original vendors.

Davis-Besse had approximately 13 years of operation with fresh fuel assemblies at peripheral locations. Cycle 15 has implemented a new failure resistant fuel design in high vulnerability locations. The core design for Davis-Besse is within the assumption of MRP-227. Davis-Besse is a base load plant and has incorporated no design changes beyond those identified in general industry guidance or recommended by the original vendors.

- Preventive Actions

  The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the PWR Water Chemistry Program. The PWR Water Chemistry Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M2, "Water Chemistry."

- Parameters Monitored or Inspected

  The program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

  Specifically, the program implements the parameters monitored/inspected criteria for B&W designed Primary Components in Table 4-1 of MRP-227.

Additionally, the program implements the parameters monitored/inspected criteria for B&W designed Expansion Components in Table 4-4 of MRP-227. No existing generic industry programs contain the specificity considered sufficient for monitoring the aging effects addressed by the MRP-227 guidelines for B&W plants. Therefore, no components for B&W plants were placed into the Existing Programs group. No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measures," in accordance with the analyses reported in MRP-227. As part of the Davis-Besse Inservice Inspection Program, a visual VT-3 examination of the reactor vessel removable core support structure is conducted once per Inservice Inspection interval in accordance with ASME Section XI, Table IWB-2500-1, Examination Category B-N-3.

MRP-227 I&E guidelines require a visual (VT-3) examination of the core support shield (CSS) vent valve retaining rings ~~and disc shaft~~ for every 10 year Inservice Inspection Interval. In addition, Davis-Besse Technical Specification 5.5.4 requires testing of the CSS vent valves every 24 months to verify by visual inspection that the valve body and valve disc exhibit no abnormal degradation, verify the valve is not stuck in an open position, and verify by manual actuation that the valve is fully open when a force of ≤400 lbs is applied vertically upward. The technical specification inspection will continue to be performed at the prescribed frequency of 24 months. The MRP-227 required visual (VT-3) examination will also be performed at the prescribed frequency of every 10 year Inservice Inspection Interval.

- Detection of Aging Effects

  The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimension, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface-breaking discontinuities.

  Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied

for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that need to be applied to inspections of Primary Components and Existing Requirement Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for Primary Components and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.

Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for B&W designed Primary Components in Table 4-1 of MRP-227 and for B&W designed Expansion Components in Table 4-4 of MRP-227.

~~*As provided in Section 4.1.3 of the MRP-227 safety evaluation, the flow distributor-to-shell forging bolts (also known as the flow distributor bolts) in B&W designed plants were added to the "Primary" inspection category. The safety evaluation provides that the examination method shall be volumetric examination (UT), the examination coverage for these components shall conform to the criteria as described in Section 3.3.1 of the safety evaluation, and the re-examination frequency shall be on a 10-year interval similar to other "Primary" inspection category components. For B&W designed plants, no other additional components were added to the "Primary" inspection category and no additional components were added to the "Expansion" inspection category.*~~

In addition, in some cases (as defined in MRP-227), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program includes Section 4.3.1 of MRP-227 that describes the physical measurements needed for the B&W internals core clamping items. In addition, Table 4-1 provides the required examination

method and examination coverage and Table 5-1 provides the acceptance criteria for the physical measurements.

- Monitoring and Trending

  The program requires that all inspections shall be documented for future review; defects shall be documented in accordance with the Davis-Besse corrective action program.

  In addition, the program requires that a summary report of all inspections and monitoring, items requiring evaluation, and new repairs shall be submitted to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227 are examined.

  Section 6 of MRP-227 will not be used by FENOC for evaluating examination results that do not meet the acceptance criteria identified in Section 5 of MRP-227. Rather, FENOC plans to use WCAP-17096-NP, Revision 2 as the framework to develop those generic and plant-specific evaluations triggered by findings in the RVI examinations. *As provided in the safety evaluation for MRP-227, ~~Rev. 0,~~ the NRC staff is currently reviewing WCAP-17096-NP, Revision 2.*

- Acceptance Criteria

  Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.

  The guidance in MRP-227 contains three types of examination acceptance criteria:

  - For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;
  - For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or

pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and,

- For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants are given in Table 5-1 of MRP-227.

Section 6 of MRP-227 will not be used by FENOC for evaluating examination results that do not meet the acceptance criteria identified in Section 5 of MRP-227. Rather, FENOC plans to use WCAP-17096-NP, Revision 2 as the framework to develop those generic and plant-specific evaluations triggered by findings in the RVI examinations. *As provided in the safety evaluation for MRP-227, ~~Rev. 0,~~ the NRC staff is currently reviewing WCAP-17096-NP, Revision 2.*

- Corrective Actions

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

- Confirmation Process

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

- Administrative Controls

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

- Operating Experience

  Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. However, a considerable amount of PWR internals aging degradation has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. For this reason, the U.S. PWR owners and operators began a program a decade ago to inspect the baffle-former bolting in order to determine whether similar problems might be expected in U.S. plants. A benefit of this decision was the experience gained with the UT examination techniques used in the inspections. In addition, the industry began substantial laboratory testing projects in order to gather the materials data necessary to support future inspections and evaluations. Another item with existing or suspected material

degradation concerns that has been identified for PWR components is cracking in some high-strength bolting. This condition has been corrected primarily through bolt replacement with less susceptible material and improved control of pre-load.

Stress corrosion cracking (SCC) has occurred in Alloy A-286 internals bolting in B&W units, this included Davis-Besse. The Alloy A-286 bolt failures in B&W PWR internals were subjected to a comprehensive failure analysis that is documented in BAW-1843PA, "The B&W Owners Group Evaluation of Internal Bolting Concerns in 177FA Plants," dated January 1986. BAW-1843PA was reviewed and approved by the NRC. This failure analysis addressed probable cause of the cracking, assessment of likelihood and consequences of joint failure, and replacement bolt design. The recommended replacement bolts were Alloy X-750 HTH bolts that are less susceptible to SCC and have overall excellent material properties.

Davis-Besse has replaced the majority of the Alloy A-286 bolts for the reactor vessel internals (upper core barrel, lower core barrel, lower thermal shield and surveillance specimen holder tubes) with Alloy X-750 HTH bolts. To satisfy a needed action under NEI 03-08 protocol, Davis-Besse performed UT examinations of 100% of all upper core barrel bolts during the cycle 16 refueling outage. This inspection did not identify any unacceptable indications.

As part of the Inservice Inspection Program, a visual (VT-3) examination of the reactor vessel removable core support structure is conducted once per Inservice Inspection interval in accordance with ASME Section XI, Table IWB-2500-1, Examination Category B-N-3. These inspections have not identified any unacceptable indications.

FENOC participates in the industry programs for investigating and managing aging effects on reactor vessel internals. Through its participation in EPRI MRP activities, FENOC will continue to benefit from the reporting of reactor vessel internals inspection information, and will share its own internals inspection results with the industry, as appropriate.

Conclusion

The PWR Reactor Vessel Internals Program provides reasonable assurance that cracking, including stress corrosion cracking (SCC), which also encompasses primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; loss of material induced by wear; loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; and loss of preload due to thermal and

irradiation-enhanced stress relaxation or creep of the subject reactor vessel internals components will be adequately managed so that intended functions of components within the scope of license renewal are maintained consistent with the current licensing basis for the period of extended operation.

Enclosure B

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

Letter L-12-015

New and Revised DBNPS License Renewal Application Boundary Drawings

14 pages follow

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

LR Drawing LR-M010D Sheet 2	**Revision 0**
LR Drawing LR-M036C Sheet 2	**Revision 0**
LR Drawing LR-M039B Sheet 2	**Revision 0**

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

LR Drawing LR-M010D Sheet 1	**Revision 3**
LR Drawing LR-M012E	**Revision 2**
LR Drawing LR-M020B	**Revision 3**
LR Drawing LR-M036C Sheet 1	**Revision 3**
LR Drawing LR-M037C	**Revision 2**
LR Drawing LR-M037D	**Revision 4**
LR Drawing LR-M037E	**Revision 3**
LR Drawing LR-M037F	**Revision 2**
LR Drawing LR-M039A	**Revision 4**
LR Drawing LR-M039B Sheet 1	**Revision 3**
LR Drawing LR-M042B	**Revision 3**

The 14 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

D-01-D14