Enclosure A

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

Letter L-15-139

AREVA NP Licensing Report No. ANP-3290 Revision 1,
"Reactor Vessel Internals Inspection Plan
for the Davis-Besse Nuclear Power Plant Unit No. 1"

101 pages follow

AREVA

Reactor Vessel Internals Inspection Plan for the Davis-Besse Nuclear Power Plant Unit No. 1

ANP-3290
Revision 1

Licensing Report

March 2015

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	Table 2-1	Changed "resolved" to "fulfilled"
2	Section 4.1.1	Added line: "This deviation was documented in the FENOC Corrective Action Program CR 2014-05971."
3	Section 4.1.3	Added text: "irradiation-assisted stress corrosion cracking (IASCC)"
4	Section 4.2.7.1	Changed "of" to "during"
5	Section 4.2.7.2	Changed "an" to "a"
6	Section 5.8.2	Added text referring to FENOC Procedure NOP-ER-2101, Engineering Program Management.
7	Section 5.9.2	Added text referring to FENOC Procedure NOP-CC-5004, Pressurized Water Reactor Vessel Internals Program.
8	Table 6-1	Revised text for AI#6 to read as follows: "In response to Applicant/Licensee Action Item 6, detailed analyses justifying operation of the inaccessible and non-inspectable component items will be submitted to the NRC within one year of the detection of degradation exceeding the acceptance criteria of the linked MRP-227-A primary component items leading to expansion."
9	Table 6-1	Revised text for AI#7 to read as follows: "In response to Applicant/Licensee Action Item 7, plant-specific analyses will be submitted to the NRC by one year prior to the MRP-227-A inspection of the applicable component items."
10	Table 6-1	Revised text for AI#8 to read as follows: "The response to the remaining open items in Applicant/Licensee Action Item 8 will be submitted to the NRC by April 22, 2017."
11	Section 6.2.2.2	Added reference to "103-3359NP-000"

12	Section 6.2.2.2	Revised text as follows: "MRP-227-A Table 4-7 does not list any Existing Programs Components for B&W Units. Davis-Besse does however have actions in place associated with the original vent valve locking device aging management."
13	Section 6.2.2.2	Revised text to read as follows: "An existing program is in place at each of the B&W-designed units that requires testing and inspection of the vent valve assemblies each refueling outage. The aging management measures provided in these requirements include a provision to visually inspect the valve body and disc seating surfaces. Continuation of the existing vent valve testing and inspection requirements will manage cracking of the vent valve component items that could cause loss of the vent valve function."
14	Section 6.2.4.2	Added reference to FENOC Condition Report 2014-06427
15	Section 6.2.4.2	Revised text as follows: "In response to Applicant/Licensee Action Item 6, detailed analyses justifying operation of the inaccessible and non-inspectable component items will be submitted to the NRC within one year of the detection of degradation exceeding the acceptance criteria of the linked MRP-227-A primary component items leading to expansion."
16	Section 6.2.5.2	Changed "will develop" to "is developing"
17	Section 6.2.5.2	Revised text as follows: "In response to Applicant/Licensee Action Item 7, plant-specific analyses will be submitted to the NRC by one year prior to the MRP-227-A inspection of the applicable component items."
18	Section 6.2.6.2	Revised text as follows: "The response to the remaining open items in Applicant/Licensee Action Item 8 will be submitted to the NRC by April 22, 2017."
19	Section 8.0	Added References 14, 26, 27, 29, 30, and 31 and renumbered applicable existing references.
20	Appendix A, Page A-2	Changed "On-time" to "One-Time"

21	Appendix D, Page D-1	Revised last paragraph to read as follows: “Note the alternative actions to address these deviations have been documented in the FENOC Corrective Action Program (CR 2014-05971) and being processed per FENOC procedure NOP-CC-5003, Processing Deviations to Materials Initiative Guidance.”

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
AMP	Aging Management Program
AMR	Aging Management Review
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock and Wilcox
B&WOG	B&W Owner's Group
CAP	Corrective Action Program
CASS	Cast Austenitic Stainless Steel
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CRGT	Control Rod Guide Tube
CSA	Core Support Assembly
CSS	Core Support Shield
CUF	Cumulative Usage Factor
DB-1	Davis-Besse Nuclear Power Plant Unit No. 1
EPRI	Electric Power Research Institute
FD	Flow Distributor
FENOC	FirstEnergy Nuclear Operating Company
FIV	Flow-Induced Vibration
FMECA	Failure Modes, Effects, and Criticality Analysis
GALL	Generic Aging Lessons Learned (NUREG-1801)
I&E Guidelines	Inspection and Evaluation Guidelines (MRP-227-A)
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IBSP	Internals Bolting Surveillance Program
IE	Irradiation Embrittlement
IGSCC	Intergranular Stress Corrosion Cracking
IMI	Incore Monitoring Instrumentation
ISG	Interim Staff Guidance
ISI	Inservice Inspection
JOBB	Joint Owners' Baffle Bolt (Program)
LCB	Lower Core Barrel
LR	License Renewal
LRA	License Renewal Application
LTS	Lower Thermal Shield

MRP	Materials Reliability Program
MUR	Measurement Uncertainty Recapture
NDE	Non-destructive Examination
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
QA	Quality Assurance
QAPM	Quality Assurance Program Manual
RAIs	Requests for Additional Information
RFO	Refueling Outage
RV	Reactor Vessel
RVI(1)	Reactor Vessel Internals
SCC	Stress Corrosion Cracking
SE	Safety Evaluation
SER	Safety Evaluation Report
SSHT	Surveillance Specimen Holder Tube
TLAA	Time-Limited Aging Analysis
TE	Thermal Embrittlement
TJ	Technical Justification
TS	Technical Specifications
UCB	Upper Core Barrel
USAR	Updated Safety Analysis Report
U.S.	United States
UT	Ultrasonic Testing (Non-destructive Examination Technique)
UTS	Upper Thermal Shield
VT-3	Visual Examination (Non-destructive Examination Technique)

(1) RVI is used in the direct quotes from the NRC to indicate Reactor Vessel Internals. In this document, other forms such as RV internals or reactor vessel internals are used to avoid confusion with Reactor Vessel Integrity (RVI), as is commonly used in other AREVA reports.

1.0 INTRODUCTION

The purpose of this report is to document the Davis-Besse Nuclear Power Plant Unit No. 1 (DB-1) Reactor Vessel (RV) Internals Inspection Plan for submittal to the United States (U.S.) Nuclear Regulatory Commission (NRC). This report provides a description of the DB-1 RV Internals Inspection Plan as it relates to the management of aging effects consistent with previous commitments. The DB-1 RV Internals Inspection Plan is based on MRP-227-A, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines"[1].

The RV Internals Inspection Plan is a part of the DB-1 Reactor Vessel Internals Aging Management Program (AMP). The overall program consists of not only the RV internals inspections described in this report, but includes other DB-1 and Industry Programs and Activities as discussed in Section 4.0 of this Inspection Plan. Section 5.0 of this Inspection Plan compares the DB-1 AMP to the ten (10) element program in Section XI.M16A of the NUREG-1801, Revision 2 "Generic Aging Lessons Learned (GALL)" report[2] as required by Applicant/Licensee Action Item 8 of the NRC Safety Evaluation Report (SER) documented in MRP-227-A.

On May 28, 2013, the NRC staff issued License Renewal Interim Staff Guidance (LR-ISG), LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internals Components for Pressurized Water Reactors"[3]. The LR-ISG directs applicants to follow the guidance in MRP-227-A and provides changes to NUREG-1801, Revision 2 (GALL) and NUREG-1800, Revision 2 "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR)[4] for the aging management of PWR RV internals.

This DB-1 RV Internals Inspection Plan contains a discussion of the background of the Babcock and Wilcox (B&W)-designed unit RV Internals Programs, sponsored by the utilities through the Electric Power Research Institute (EPRI) PWR Materials Reliability Program (MRP) and the PWR Owners Group (PWROG), and submitted to the NRC through the EPRI MRP. The DB-1 RV Internals Inspection Plan also contains a discussion of operating experience (OE) and relevant existing DB-1 programs including time-limited aging analyses (TLAAs).

The DB-1 RV Internals AMP includes this DB-1 RV Internals Inspection Plan and demonstrates that the program adequately manages the effects of aging for RV internals component items and establishes the basis for providing reasonable assurance that the RV internals component items remain functional through the DB-1 license renewal (LR) period of extended operation.

2.0 BACKGROUND

2.1 *DB-1 License Renewal Background*

By letter dated August 27, 2010, FirstEnergy Nuclear Operating Company (FENOC) submitted the License Renewal Application (LRA) for DB-1 in accordance with Title 10, Part 54, of the Code of Federal Regulations (10 CFR)[5]. Through the LRA, FENOC requested the NRC renew the operating license for DB-1 (License Number NPF-3) for a period of 20 years beyond the current expiration of midnight April 22, 2017.

The NRC staff sent FENOC requests for additional information (RAIs) to align the DB-1 PWR Vessel Internals Program with MRP-227-A. By letter dated September 16, 2011, FENOC provided Amendment No. 15 to the DB-1 LRA[6]. LRA Amendment No. 15 revised, in its entirety, the discussion of the PWR RV Internals Program, the Updated Safety Analysis Report (USAR) supplement description of the PWR RV Internals Program, and USAR supplement Commitment No. 15 in the LRA USAR supplement. The revisions to these LRA sections were provided to support the DB-1 response to the NRC staff's requests for additional information (RAIs) concerning the PWR RV Internals Program at DB-1[7]. Further revisions to the LRA were provided by letter dated March 9, 2012, under LRA Amendment No. 24, as part of the DB-1 supplemental response to RAIs concerning the PWR RV Internals Program at DB-1[8]. The NRC staff accepted the responses since DB-1 committed to implement the program in accordance with MRP-227-A; thus, DB-1 is in alignment with LR-ISG-2011-04. The SER related to the LR of DB-1 was issued in September 2013 by the NRC[9].

Section 3.1 of the DB-1 LRA discusses the RV Internals Aging Management Review (AMR) for LR. The component items that are subject to AMR have been identified in accordance with the requirements of 10 CFR 54.4. The AMPs selected to manage aging effects for the RV internals are identified in Section 3.1.2.1.2 and Table 3.1.2-2 of the LRA and include the PWR Reactor Vessel Internals Program, TLAAs and Water Chemistry. A description of the water chemistry program is provided in Appendix B of the LRA; while the TLAAs are described in Section 4 and Appendix A of the LRA. In the

DB-1 LRA, FENOC committed to 1) participate in industry RV internals aging programs (MRP-227-A); 2) implement the applicable results; and 3) submit for NRC approval, not less than 24 months before the extended period, an inspection plan based on industry recommendation.

Section 3.1.3 of the SER related to the LR of DB-1 concludes that FENOC provided sufficient information to demonstrate that the effects of aging for the RV internals component items, within the scope of LR and subject to an AMR, will be adequately managed so that the intended functions will remain consistent with the current licensing basis (CLB) for the period of extended operation, as required by 10 CFR 54.21(a)(3).

Section 4.8 of the SER related to the LR of DB-1 identifies that the TLAAs from the LRA were reviewed. The SER concludes the identified TLAAs associated with RV internals for DB-1 comply with the requirements of 10 CFR 54.21(c).

The three TLAAs for the RV internals in the DB-1 LRA include low cycle fatigue, high cycle fatigue, and neutron embrittlement. Sections 4.3.2 and 4.2.7 of the SER related to the LR of DB-1 identifies that the TLAAs from the LRA were reviewed. The SER concludes that the LRA identified and evaluated the TLAAs (low cycle fatigue, high cycle fatigue, and reduction in fracture toughness) associated with the DB-1 RV internals, in accordance with 10 CFR 54.21(c)(1)(iii). In addition, the SER concludes that each TLAA will be adequately managed for the period of extended operation. See Section 4.2.4 of this report for further discussion of TLAAs.

Section 6 of the SER related to the LR of DB-1 concludes the NRC staff determined that the requirements of 10 CFR 54.29(a) were met by the DB-1 LRA.

Table 2-1 of this report demonstrates how the DB-1 RV internals LR commitments are being fulfilled.

Table 2-1: DB-1 RV Internals LRA Commitments and Resolutions

Commitment Reference Location	Commitment/Action Items	Fulfilled, as described in reference location
DB-1 LRA, Appendix A, License Renewal Commitment 14	Implement the PWR Reactor Vessel Internals Program as described in LRA Section B.2.32 prior to October 22, 2016.	Being fulfilled by the development and submittal of this DB-1 RV Internals Inspection Plan (based on MRP-227-A) to the NRC for review and approval.
DB-1 LRA, Appendix A, License Renewal Commitment 15	In association with the PWR Reactor Vessel Internals Program, 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. NOTE: The inspection plan will be submitted no later than two years after issuance of the renewed operating license or two years prior to the beginning of the period of extended operation (April 22, 2015), whichever is earlier.	Being fulfilled by the development and submittal of this DB-1 RV Internals Inspection Plan (based on MRP-227-A) to the NRC for review and approval.

2.2 *DB-1 RV Internals AMR/Industry Program Background*

The industry work on the aging of the RV internals resulted in the submittal of "PWR Internals Inspection and Evaluation Guidelines," MRP-227, Revision 0 in January 2009 for NRC review[10]. The NRC issued a Safety Evaluation[11] accepting MRP-227 with certain conditions and applicant/licensee action items. MRP-227-A was then completed in December 2011 with the changes requested by the NRC Safety Evaluation. MRP-227-A is considered the industry program for the DB-1 RV internals. Component items recommended for augmented examinations are categorized as "Primary," "Expansion," or "Existing Programs." Component items not recommended for augmented examinations are categorized as "No Additional Measures." The industry program is intended to provide a consistent approach to the aging management of PWR RV internals across the PWR fleet. For additional information about MRP-227-A, see Section 4.1.1 of this report.

An AMR for LR was performed and the RV internals results are documented in Table 3.1.2-2 of the DB-1 LRA.

Initial augmented examinations for aging degradation mechanisms not yet completed are currently scheduled to be completed prior to the end of the fourth American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code Section XI, 10-year In-Service Inspection (ISI) Interval[12].

2.3 *DB-1 RV Internals AMP Intent*

The DB-1 RV Internals AMP, which will include the DB-1 RV Internals Inspection Plan described in this report after review and approval by the NRC, utilizes a combination of prevention/mitigation and condition monitoring. Where applicable, credit is taken for existing programs (e.g., primary water chemistry program and ASME B&PV Code Section XI inspection program). The DB-1 RV Internals Inspection Plan incorporates requirements for augmented inspections provided by industry guidelines in MRP-227-A. Augmented inspections are in addition to the requirements of ASME B&PV Code Section XI; the augmented inspections do not reduce, alter, or otherwise affect current ASME B&PV Code Section XI inservice inspections at DB-1.

Aging degradation mechanisms that impact the RV internals have been identified in MRP-227-A. MRP-227-A provides augmented examination requirements for detection of the effects of aging degradation mechanisms as listed in Table 2-2.

Table 2-2: RV Internals Aging Degradation Mechanisms and Their Aging Effects

Aging Degradation Mechanism	Aging Effect
Stress Corrosion Cracking (SCC)	Cracking
Irradiation-Assisted Stress Corrosion Cracking (IASCC)	Cracking
Wear	Loss of Material
Fatigue	Cracking
Thermal Aging Embrittlement (TE)	Loss of Ductility and Unstable Crack Extension
Irradiation Embrittlement (IE)	Loss of Ductility and Unstable Crack Extension
Void Swelling and Irradiation Growth	Dimension Change, Distortion, and Cracking
Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation Enhanced Creep	Loss of Mechanical Closure Integrity Leading to Cracking

Section 5.0 of this report compares the DB-1 RV Internals AMP (including this Inspection Plan) to the ten (10) elements in AMP XI.M16A of NUREG-1801, Revision 2 as required by Applicant/Licensee Action Item 8 of MRP-227-A. The DB-1 RV Internals AMP incorporates programs and activities that are credited for managing the aging effects produced by the mechanisms listed in Table 2-2. DB-1 RV internals component items within the scope of the LRA and the SER related to the LR of DB-1[9] have been considered in this DB-1 RV Internals Inspection Plan.

Section 6.0 to this Inspection Plan identifies the DB-1 response to each of the Topical Report Conditions and Applicant/Licensee Action Items in the NRC Safety Evaluation documented in MRP-227-A. While disposition to several items are included in this plan, disposition of certain Applicant/Licensee Action Items will subsequently be addressed as listed in Section 6.0 of this report.

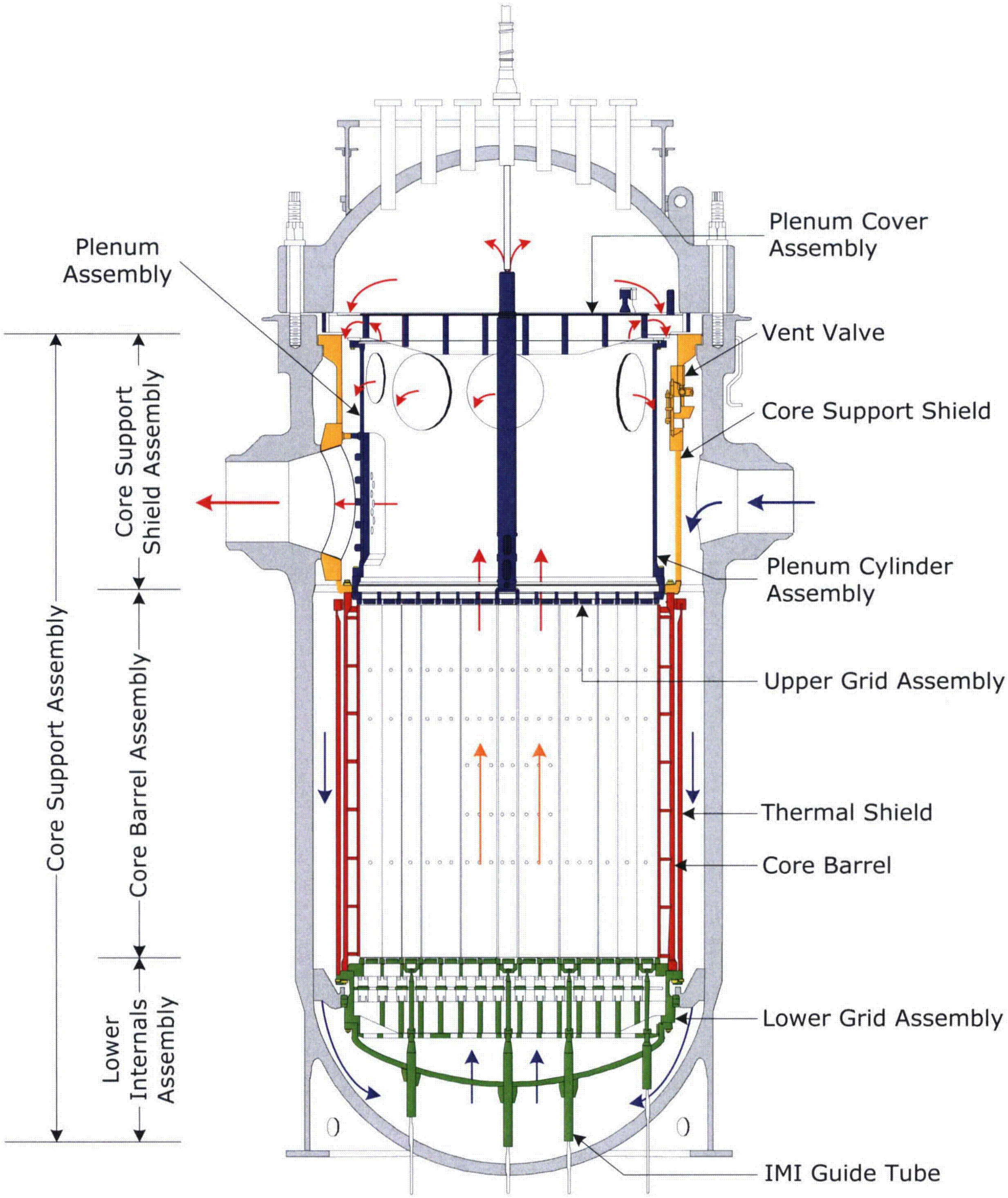
2.4 *DB-1 RV Internals Background*

The reactor vessel internals perform the following safety-related system intended functions, as stated in Section 2.3.1.2 of the DB-1 LRA:

1. Provide support for the core and maintain core in coolable configuration under all operating conditions;
2. Provide shielding to attenuate radiation generated in the core;
3. Control primary coolant distribution to the core as required for design heat removal capability; and
4. Provide support and alignment for control rod drive mechanisms, control rods, and incore detectors.

The DB-1 RV internals consists of two structural subassemblies that are located within the RV: the plenum assembly and the core support assembly (CSA). The general arrangement of the B&W Unit RV internals is shown in Figure 2-1. Note: in the DB-1 LRA the fuel assemblies and the control rod assemblies are noted as not being subject to an AMR due to being short-lived and not requiring aging management.

Figure 2-1: B&W Unit Typical Layout of Reactor Vessel Internals
(Note: Some component items are rotated for clarity)



3.0 PROGRAM OWNER

The DB-1 Technical Services Engineering, Nuclear Engineering Programs Unit is responsible for maintaining and implementing the DB-1 RV Internals Inspection Plan.

4.0 INDUSTRY AND DB-1 PROGRAMS AND ACTIVITIES

The DB-1 RV Internals Inspection Plan is based on the DB-1 LRA, the SER related to the LR of DB-1, and MRP-227-A. The DB-1 RV Internals AMP implements MRP-227-A through this RV Internals Inspection Plan. The DB-1 RV Internals AMP also includes the continuation and utilization of some programs and activities discussed in this section.

4.1 *Industry Programs and Activities*

FENOC participates in industry activities, which support the management of aging of the RV internals. These activities are described in the sections below. These industry programs and activities have helped to define the required examinations and examination techniques for the component items covered by this DB-1 RV Internals Inspection Plan.

4.1.1 MRP-227-A

The MRP-227-A "Pressurized Water Reactor Internals Inspection and Evaluation (I&E) Guidelines" were developed by a team of industry representatives who reviewed available data and industry experience to identify and prioritize I&E requirements for RV internals. MRP-227-A is the result of the industry work and NRC review; the key sequential steps in the process included the following:

- The development of screening criteria, with susceptibility for the eight (8) postulated aging mechanisms relevant to reactor internals and their effects;
- An initial component screening and categorization, using the susceptibility levels and FMECA (failure modes, effects, and criticality analysis) to identify the relative ranking of the component items;
- Engineering assessment and analysis of degradation for component items and assemblies of component items; and

- Aging management strategy development combining the engineering assessment and analysis with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Through this process, the RV internals for all three PWR designs in the U.S. were evaluated, and appropriate requirements for aging management actions specific to each component were provided.

MRP-227-A utilized the screening and ranking process to aid in the identification of required examinations for “Primary” and “Expansion” component items and credits “Existing Programs” when they were deemed adequate.

The basic description of each classification is as follows:

- **“Primary”**

  Those PWR internals that are highly susceptible to the effects of at least one of the eight (8) aging mechanisms were placed in the “Primary” group. The aging management requirements that are needed to ensure functionality of “Primary” component items are described in these I&E guidelines [MRP-227-A]. The “Primary” group also includes component items which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- **“Expansion”**

  Those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight (8) aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the “Expansion” group. The schedule for implementation of aging management requirements for “Expansion” component items will depend on the findings from the examinations of the “Primary” component items at individual plants.

- **"Existing Programs"**

  Those PWR internals that are susceptible to the effects of at least one of the eight (8) aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the "Existing Programs" group. There currently are no B&W unit internals component items in this group.

- **"No Additional Measures"**

  Those PWR internals for which the effects of all eight (8) aging mechanisms are below the screening criteria were placed in the "No Additional Measures" group. Additional component items were placed in the "No Additional Measures" group as a result of FMECA and the engineering assessment and analysis. No further action is required by these guidelines for managing the aging of the "No Additional Measures" component items.

The categorization and analysis processes used in the MRP-227-A approach are not intended to supersede ASME B&PV Code Section XI requirements.

The requirements of MRP-227-A are classified in accordance with NEI 03-08 Guidelines[13]. For the MRP-227-A guidelines there are one (1) "Mandatory", five (5) "Needed", and zero (0) "Good Practice" requirements as follows:

- **"Mandatory"**

  Each commercial U.S. PWR unit shall develop and document a program for management of aging of reactor internal components within thirty-six months following issuance of MRP-227, Rev 0 (that is no later than December 31, 2011).

  FENOC has established procedures to develop and document an AMP for the RV internals. This RV Internals Inspection Plan is part of the AMP for the RV internals for DB-1.

- **"Needed"**

Each commercial U.S. PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A.

The applicable B&W tables are MRP-227-A, Table 4-1 ("Primary"), Table 4-4 ("Expansion"), and Table 5-1 (Examination Acceptance and Expansion Criteria). There are no "Existing Program" component items in MRP-227-A for B&W-designed PWRs. DB-1 has followed the MRP-227-A requirements by the inspection activities already planned or performed as described in this DB-1 RV Internals Inspection Plan. Appendix A is MRP-227-A Table 4-1 modified to reflect only the DB-1 primary component items. Appendix B is MRP-227-A Table 4-4 modified to reflect only DB-1 expansion component items. Appendix C is MRP-227-A Table 5-1 modified to reflect only DB-1 acceptance/expansion criteria.

The Alloy X-750 dowel-to-guide block weld (along with the dowel itself) was removed from the DB-1 internals. Also the Alloy X-750 dowel-to-upper grid fuel assembly support pad weld configuration previously thought not to exist in the DB-1 internals actually does exist. Since these two items are linked in MRP-227-A, a deviation (alternative action) is required for DB-1 that meets the same objective, or level of conservatism exhibited by the original work product. The alternative action includes elevation of the Expansion component item, i.e., Alloy X-750 dowel-to-lower grid fuel assembly support pad weld, to a Primary component item and establishing a new Expansion component item, i.e., Alloy X-750 dowel-to-upper grid fuel assembly support pad weld, specifically for DB-1. Appendix D describes the final alternative action of the new Primary, Expansion, and examination acceptance and expansion criteria tables applicable to DB-1 only. This deviation was documented in the FENOC Corrective Action Program CR 2014-05971[14].

Therefore, implementation of this inspection plan will fulfill this "Needed" requirement for DB-1.

- **"Needed"**

  Examinations specified in these guidelines shall be conducted in accordance with the Inspection Standard [MRP-228].

  Inspection standards developed under MRP-228[15] will be used by FENOC for the augmented examinations described in this DB-1 RV Internals Inspection Plan developed in accordance with MRP-227-A. Implementation of this RV Internals Inspection Plan will fulfill this "Needed" requirement for DB-1.

- **"Needed"**

  Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-227-A guidelines shall be recorded and entered in the plant corrective action program and dispositioned.

  The DB-1 Corrective Action Program (CAP) will be used, as discussed in Section 5.7 of this report. Implementation of this DB-1 RV Internals Inspection Plan will fulfill this "Needed" requirement for DB-1.

- **"Needed"**

  Each commercial U.S. PWR unit shall provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227-A are examined.

  FENOC will provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager for future DB-1 inspection activities within 120 days of the completion of an outage during which PWR RV internals within the scope of MRP-227-A are examined. Implementation of this DB-1 RV Internals Inspection Plan will fulfill this "Needed" requirement for DB-1.

- **"Needed"**

  If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5 [of MRP-227-A], this engineering evaluation shall be conducted in accordance with an NRC-approved evaluation methodology.

  FENOC will disposition all examination results that do not meet the acceptance criteria [in Section 5 of MRP-227-A] in accordance with an NRC approved evaluation methodology. Implementation of this DB-1 RV Internals Inspection Plan will fulfill this "Needed" requirement for DB-1.

4.1.1.1 MRP-227-A Applicability to DB-1

The applicability of MRP-227-A guidelines is based on several general assumptions that were used for the analysis in the development of MRP-227-A. Additional assumptions in the Failure Modes, Effects, and Criticality Analysis (FMECA - MRP-190[16]) and the functionality analysis report (MRP-229, Revision 3[17]) are also included as required by Applicant/Licensee Action Item 1 from the NRC SER documented in MRP-227-A.

MRP-227-A:

The assumptions found in Section 2.4 of MRP-227-A and their applicability to DB-1 are listed below:

- *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.*

  The fuel management program for DB-1 changed from a high to a low leakage core loading pattern prior to 30 years of plant operation. This change was started in DB-1 Cycle 5 (1986) and has been continually implemented through the most recent fuel cycle, DB-1 Cycle 18 (2014). This change is considered to be a preventative action to lessen the effects of aging on the DB-1 RV internals. DB-1 will continue to use a low-leakage core loading pattern.

- *Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.*

  DB-1 operates as a base load unit.

- *No design changes beyond those identified in general industry guidance or recommended by the original vendors.*

  MRP-227-A states that the requirements are applicable to all U.S. PWR operating units as of May 2007 for the three designs (i.e., B&W, Westinghouse, and CE) considered. No modifications have been made to the DB-1 RV internals since May 2007.

MRP-190:

Section 4 of MRP-190 (the FMECA) contains six (6) assumptions and observations. As stated in Section 4 of MRP-190, these assumptions are either bounding or methodological, and do not require plant-specific verification for each of the B&W-designed operating units.

MRP-229, Revision 3:

Section 2.4.1 of MRP-229, Revision 3 (the functionality analysis report) identifies eight (8) bulleted limitations and assumptions; seven (7) of these are programmatic rather than plant-specific. Only one (1) pertains to the unit design and operating history, and that one is discussed below:

- *The effect of power uprates has not been considered by the functionality analysis.*

  DB-1 implemented a measurement uncertainty recapture (MUR) power uprate in 2008. Under an EPRI MRP project, AREVA performed a sensitivity study of the previous functionality assessment that was more representative of B&W unit operation consisting of 60 years of operation and including a power uprate in life. The sensitivity study utilized the same model, geometry, contacts, etc., as the functionality analysis, MRP-229, Revision 3, except the sensitivity study used B&W specific input. The results of this sensitivity study demonstrate that the DB-1 MUR has no effect on the results of the MRP-227-A processes that resulted in the component items listed in Tables 4-1 and 4-4.

Based on the above review, MRP-227-A is applicable to DB-1 and Applicant/Licensee Action Item 1 from the NRC SER documented in MRP-227-A is complete. See Section 6.0 of this report.

4.1.2 Internals Bolting Surveillance Program

Starting in 1981, ultrasonic testing (UT) at several B&W units revealed that multiple RV internals bolts had rejectable UT indications. The failure mechanism was determined to be intergranular stress corrosion cracking (IGSCC). The failed bolts were predominantly fabricated from Alloy A-286, Condition A (ASTM A 453, Grade 660) material[18].

As a result of the noted bolt failures, utilities began replacing bolts where needed. The B&W Owners Group (B&WOG) initiated the Internals Bolting Surveillance Program (IBSP)[19], which was completed by the EPRI PWR MRP, to better assess the IGSCC susceptibility of the replacement bolts. The IBSP exposed replacement bolts to simulated PWR conditions in a laboratory autoclave and an actual PWR environment inside an operating PWR unit. The bolts used in the testing were manufactured from Alloy A-286, Condition A and Alloy X-750, high temperature heat-treatment (HTH) condition materials. The scaled down bolts were tested in two surface conditions, peened and un-peened.

At the completion of the IBSP tests, several peened replacement Alloy A-286 replacement bolts were observed to have developed IGSCC when loaded to a high stress, while the un-peened Alloy A-286 replacement bolts were free from IGSCC when loaded to similarly high stresses for the test duration of 8½ years in the reactor capsules. The Alloy X-750, HTH Condition replacements bolts of both peened and un-peened conditions were free from IGSCC when subjected to the same environmental and loading conditions as the Alloy A-286 bolts for the test duration of 8½ years.

The upper core barrel (UCB) [partial replacement], lower core barrel (LCB) [partial replacement], lower thermal shield (LTS), and surveillance specimen holder tube (SSHT) [partial replacement with 4 bolt locations left vacant] bolts at DB-1 have been replaced with Alloy X-750 HTH bolts. The other locations such as LCB [partial], upper thermal shield (UTS), and flow distributer (FD) bolts are the original Alloy A-286 bolts.

A 2005 evaluation of the IBSP and industry experience resulted in a B&WOG letter and a subsequent PWROG letter, making recommendations for UT examinations of the high strength (Alloy X-750 or Alloy A-286) bolts in the B&W units. These recommendations were made in accordance with NEI 03-08. These UT examination recommendations have been incorporated into MRP-227-A with the “Needed” recommendation being incorporated into the “Primary” category and the “Good Practice” recommendation being incorporated into the “Expansion” category except for the FD bolts which are categorized as “Primary” in MRP-227-A. The MRP-227-A examinations supersede these UT examination recommendations.

4.1.3 Joint Owner’s Baffle Bolt Program

The Joint Owners’ Baffle Bolt (JOBB) Program stemmed from UT examinations of baffle-to-former bolts at several European units. Indications were noted under the bolt head in the head-to-shank fillet radius. The bolt failures were attributed to irradiation-assisted stress corrosion cracking (IASCC). Various tasks, including non-destructive examinations (NDE), irradiation and mechanical testing, corrosion testing, and microstructural evaluation were used to characterize the effect of irradiation on bolting materials under the JOBB program.

The JOBB program is now being managed by EPRI with additional research on RV internals material being performed under EPRI programs. The results of the JOBB program have been incorporated into EPRI MRP reports, and specifically referenced in

Section 2.1 of MRP-227-A. The EPRI MRP provides updates and results to the NRC during meetings; an example of the results presented in these meetings is given in Reference 20.

4.1.4 PWROG and EPRI/MRP

The utilities sponsor activities related to PWR RV internals aging management through both the PWROG and EPRI MRP. FENOC's participation in current PWROG and EPRI MRP activities will continue.

4.2 *DB-1 Programs and Activities*

The DB-1 RV Internals AMP consists of a number of programs and activities that support aging management of the RV internals; these include the ASME B&PV Code Section XI ISI Program, primary water chemistry program, plant technical specifications (vent valve inspection and exercise program), TLAAs, fabrication records searches, UCB, LCB, and FD bolt analyses, past RV internals inspections, core clamping measurements, and a fuel/baffle interaction investigation.

4.2.1 ASME B&PV Code Section XI ISI Program

The ASME B&PV Code Section XI ISI program is an existing program developed under ASME B&PV Code Section XI, Subsection IWB-2500[12]. ASME B&PV Code Section XI, Table IWB-2500-1, Examination Category B-N-3 applies to core support structures (i.e., RV internals). Examination of these B-N-3 RV internals is required once every 10-year ISI interval. The visual examinations (VT-3) are performed with the aid of visual examination tools, in accordance with a code compliant VT-3 procedure.

The next DB-1 ASME B&PV Code Section XI 10-year ISI inspection requiring CSA removal is currently scheduled for the Year 2022 [refueling outage (RFO) 22].

4.2.2 Primary Water Chemistry Program

The DB-1 reactor coolant chemistry program, as implemented by the DB-1 PWR Water Chemistry Program, limits the concentration of oxygen, halogens, and sulfate species in the primary water. The intent of these limits is to prevent the reactor coolant from becoming an environment favorable to stress corrosion cracking (SCC), therefore greatly reducing the probability of SCC.

4.2.3 Plant Technical Specifications

As described in the DB-1 Plant Technical Specifications (TS), Section 5.5.4 "Reactor Vessel Internals Vent Valves Program," vent valve testing and inspections are required to be performed every RFO (i.e., 24 months)[21]. This requirement is fulfilled by the DB-1 Mechanical Maintenance Procedure for Reactor Internals Vent Valve Test and Inspection. The accessible areas of the vent valve are visually inspected, including the locking devices. Additionally, vent valve operation is tested through manual actuation.

4.2.4 Time Limited Aging Analyses

4.2.4.1 Currently Identified TLAAs

As described briefly in Section 2.0 of this report, the DB-1 RV Internals AMP includes three (3) TLAAs that were evaluated and dispositioned in the LRA. The three (3) TLAAs for the RV internals include low cycle fatigue, high cycle fatigue, and neutron embrittlement (i.e., reduction in fracture toughness).

- Low cycle fatigue is managed by the Fatigue Monitoring Program (LRA Section 4.3.2.2.2.1).
- High cycle fatigue (FIV) is projected and the calculated endurance limit used for FIV analyses remains valid through the period of extended operation (LRA Section 4.3.2.2.2.2).
- High cycle fatigue (FIV) is projected and the cumulative usage factors (CUFs) for FIV of select reactor vessel internals (incore instrumentation nozzles and surveillance capsule holder tubes) have been satisfactorily projected for the period of extended operation (LRA Section 4.3.2.2.2.3).

- Reduction of fracture toughness of (stainless steel) RV internals is managed by the PWR RV Internals program for the period of extended operation (LRA Section 4.2.7).

  It should be noted for the neutron embrittlement (reduction of fracture toughness of the RV internals) TLAA, the LRA committed to manage neutron embrittlement by participating in, evaluating, and implementing the results of RV internals industry programs that were then in development. The program (i.e., MRP-227-A) is now approved and is implemented via this inspection plan as part of the DB-1 RV Internals AMP. In the SER related to the LR of DB-1, the NRC staff determined that DB-1's general program description, as revised by LRA Amendment 15, adequately addresses the management of this TLAA for the period of extended operation. However, the effect of irradiation on the mechanical properties and deformation limits of the RV internals that was evaluated for the current term of operation in Appendix E of topical report BAW-10008, Part 1, Revision 1[22] supplemented by DB-1 USAR Appendix 4a will require an update for the period of extended operation. Therefore to fulfill management of this TLAA, a schedule of when this information will be submitted to the NRC will be submitted by April 22, 2017.

4.2.4.2 Consideration of Future TLAAs

Section 6.2.4 of this report describes the resolution to Applicant/Licensee Action Item 6 from the SER in MRP-227-A concerning the justification of acceptability of inaccessible component items. The inaccessible items (core barrel cylinder including vertical and circumferential seam welds, former plates, external baffle-to-baffle bolts and their locking devices, the core barrel-to-former bolts and their locking devices) and the core barrel assembly internal baffle-to-baffle bolts are to be evaluated or a schedule for replacement of the items is to be established; if any of the evaluations meet the definition in 10 CFR 54.3, they may be considered TLAAs.

Section 6.2.5 of this report describes the resolution to Applicant/Licensee Action Item 7 from the SER in MRP-227-A concerning plant-specific analyses to demonstrate RV internals component items fabricated from cast austenitic stainless steel (CASS), martensitic stainless steel, or martensitic precipitation-hardened stainless steel will maintain their functionality during the period of extended operation. These potential analyses must also consider the possible loss of fracture toughness in these items due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. If any of the analyses meet the definition in 10 CFR 54.3, they may be considered TLAAs.

4.2.5 Fabrication Records Searches

Two (2) fabrication records searches for DB-1 RV internals component items listed as "Primary" and "Expansion" in MRP-227, Revision 0 were conducted by AREVA under PWROG Projects. The goals of the records searches were to locate the chemical composition of the CASS items in MRP-227, Revision 0 and to obtain a detailed description of the "Primary" and "Expansion" component items in the MRP-227, Revision 0 including the function, fabrication records, OE, and the anticipated degradation mechanisms. The chemical composition of the CASS items was used to, if possible, screen for susceptibility to thermal aging embrittlement, consistent with the screening criteria used by the EPRI MRP.

In 2014, a search of the original fabrication records was performed to confirm that the core support shield (CSS) upper flange weld was stress relieved, as required by Applicant/Licensee Action Item 4 in the NRC SER documented in MRP-227-A (see Section 6.2.3). The search confirmed that the DB-1 CSS upper flange weld was stress relieved.

4.2.6 UCB, LCB and FD Bolt Analyses

Analyses were performed for the DB-1 UCB, LCB and FD bolts. The stress limits for threaded structural fasteners in Subsection NG of the ASME B&PV Code[23] were used to develop the acceptance criteria for the integrity of the bolted joint. The analytical models are intended to be used to assess inspection results for the UCB, LCB and FD bolts at DB-1. The methodology and acceptance criteria used in the analyses is consistent with the methodology and acceptance criteria of WCAP-17096-NP, Revision 2[24] as modified by the NRC staff's draft safety evaluation of that report.

Several cases representing hypothetical UCB and LCB degraded bolt configurations were analyzed to provide insight to the number and pattern of degraded bolts that can be tolerated without exceeding the analytical criteria. The results indicate a large number of degraded UCB and LCB bolts can be tolerated without exceeding the analytical criteria for the joint if the degraded bolts are not adjacent. If degraded bolts are adjacent to each other, and located in the worst location, the number of degraded UCB and LCB bolts that can be tolerated is lower.

In addition, hypothetical FD bolting patterns were evaluated to provide insight to the number and pattern of degraded bolts that can be tolerated without exceeding the analytical criteria. The results indicate that bolting patterns with up to several consecutive non-functional [i.e., a group of neighboring (adjacent)] bolts are acceptable.

4.2.7 Past RV Internals Inspections

Past inspections of the RV internals include the vent valve testing/inspections and high-strength bolting inspections at DB-1.

4.2.7.1 Vent Valve Inspection

As discussed in Section 4.2.3 of this report, vent valve testing and inspections are required to be performed each RFO (i.e., 24 months). During an early vent valve inspection, it was observed that one (1) of the vent valve jackscrew bushings was backed out from the design position. This observation was made in 2002 and a disposition letter was prepared in 2005. This same observation was made in the 2006 (RFO14) vent valve inspection. The condition was reviewed and concluded to be acceptable. Review of the 2010 (RFO16) visual inspection video of the vent valves showed no evidence of the raised/backed out jackscrew bushing.

A second anomaly identified during the 2010 (RFO16) visual inspection video was discoloration on the valve seat. It is suspected that the discoloration was due to the position of the valve disc during no flow conditions. No evidence of bypass flow across the seating surface was noted, thus it is believed that the valve function has not been compromised.

During the inspection and exercise testing performed in 2012 (RFO17), one (1) vent valve exceeded the stay open force acceptance criteria. The valve was cycled several times, after which the holding force was found to be acceptable. No other notable conditions were observed during the 2012 inspection.

4.2.7.2 High-Strength Bolt UT Examinations at DB-1

In 1990, the UCB bolts were replaced with Alloy X-750 HTH UCB bolts; however several of UCB bolt locations were not replaced due to stuck original UCB bolt shanks. These locations were plugged with the use of three-hole tie plates. In 2010 (RFO16), a UT examination of 100% of the replacement UCB bolts was performed. No recordable indications were detected in any of the replacement UCB bolts. In 2011 (RFO17M), another UT examination of 100% of the replacement UCB bolts was performed, and there were no recordable indications detected in any of the replacement UCB bolts.

Also in 1990, 55% of the original Alloy A-286 LCB bolts at DB-1 were replaced with Alloy X-750 HTH LCB bolts. The remaining original Alloy A-286 LCB bolts at DB-1 are located above the guide blocks. UT examinations were performed at DB-1 on the original Alloy A-286 LCB bolts in 1984 (100%) and 1990 (partial). The 1984 UT examination of the original LCB bolts did not identify any defective bolts. The 1990 partial UT examination of the original LCB bolts did not identify any defective bolts. In 2011 (RFO17M), UT examinations of 100% of all the LCB bolts (i.e., original Alloy A-286 bolts and Alloy X-750 replacement bolts) were performed. No recordable indications were detected in any of the original and replacement LCB bolts.

All FD bolts in the DB-1 RV internals are the original Alloy A-286 bolts. UT examinations were performed at DB-1 on the FD bolts in 1984 (100%) and 1990 (partial). No recordable indications were detected by these examinations.

All UTS bolts in the DB-1 RV internals are the original Alloy A-286 bolts. UT examinations were performed at DB-1 on the UTS bolts in 1984 (100%) and 1990 (partial). No recordable indications were detected by these examinations.

In 1984, all LTS bolts in the DB-1 RV internals were replaced with Alloy X-750 HTH LTS bolts. A partial UT examination was performed in 1990 with no recordable indications detected.

The original Alloy A-286 SSHT bolts at DB-1 were replaced with Alloy X-750 HTH SSHT bolts in 1984 (partial replacement) and 1990 (partial replacement); however, at several of the SSHT bolt locations the bolts were removed without replacement. A partial UT examination was performed on the DB-1 RV internals SSHT bolts in 1990 with no defective bolts identified.

4.2.8 Core Clamping Measurements

Core clamping measurements were performed in 2014 (RFO18). The measurements were intended to satisfy the MRP-227-A examination requirements for a one-time physical measurement of the differential height of top of the plenum rib pads to the reactor vessel seating surface. The measurement was taken with the plenum cover weldments rib pads, plenum cover support flange, and CSS top flange inside the RV, but with the fuel assemblies removed in accordance with Section 4.3.1 of MRP-227-A.

4.2.9 Fuel/Baffle Interaction Investigation

An investigation was conducted on the interaction between the baffle plates and fuel assembly grid straps in similarly-designed B&W units between 2004 and 2010. The results of the most recent investigation identified several apparent and contributing causes. The investigation team recommended high level scoping-feasibility studies to address the proposed preventative actions. FENOC has participated and will continue to participate in this project.

4.3 *Conclusions of Section 4.0*

This section contains a description of the DB-1 Reactor Vessel Internals AMP, including this RV Internals Inspection Plan. The DB-1 RV Internals Inspection Plan is based on the DB-1 LRA, the SER related to the LR of DB-1, and evaluations supporting MRP-227-A. Inspections will consist of the ASME B&PV Code Section XI inspections, vent valve testing and inspections, and the augmented examinations prescribed in MRP-227-A. Changes resulting from the NRC's review of this report will be incorporated as appropriate.

As part of the DB-1 LR Program, FENOC has created two (2) License Renewal Commitments (see Table 2-1).

5.0 DB-1 RV INTERNALS AMP ATTRIBUTE EVALUATION (NUREG-1801, REVISION 2)

The DB-1 RV Internals AMP, which includes this DB-1 RV Internals Inspection Plan, utilizes a combination of prevention and condition monitoring to manage the effects of the eight (8) age-related degradation mechanisms given in Section 2.3 of this report, thereby providing reasonable assurance the RV internals continue to perform their function during the period of extended operation. Where applicable, credit is taken for existing programs and activities (e.g., primary water chemistry program, vent valve testing and inspection, TLAAs, and ASME B&PV Code Section XI inspections). This DB-1 RV Internals Inspection Plan incorporates the industry guidance in MRP-227-A.

This section compares the DB-1 RV Internals AMP, including this Inspection Plan, to the ten (10) elements in AMP XI.M16A from NUREG-1801, Revision 2 as required by Applicant/Licensee Action Item 8 of MRP-227-A. Details of the evaluation results for each of the ten (10) elements are documented in the DB-1 LRA Amendment 15[6] and LRA Amendment 24[8]. The conclusion of this comparison is that the DB-1 Reactor Internals AMP is consistent with the NUREG-1801 XI.M16A program with no exceptions and no enhancements.

Aging Management Program Description (from NUREG-1801, Revision 2, XI.M16A):

> *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 that is submitted to the NRC for review and approval with the application for license renewal.*

5.1 *AMP Element 1 – Scope of Program*

5.1.1 NUREG-1801, XI.M16A Scope of Program

The scope of the program includes all RVI components at Davis-Besse Nuclear Power Plant Unit No. 1, 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 AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

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. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA.*

The guidance in MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227.

5.1.2 DB-1 Scope of Program

A description of the DB-1 RV internals is provided in Section 2.4 of this report. Additional RV internals details are provided in the DB-1 USAR.

The DB-1 RV Internals Program includes the component items identified in MRP-227-A for B&W-designed RV internals applicable to DB-1 (see Appendices A, B, and D of this report).

The DB-1 RV Internals Program includes responses to the applicant/licensee action items in the NRC SER documented in MRP-227-A (see Section 6.0 of this report).

TLAAs are also included in the program to manage the aging of the DB-1 RV internals and are discussed in Section 4.2.4 of this report.

The scope of the DB-1 RV Internals Program includes those component items identified in the results of the DB-1 LRA AMR. Comparison between the DB-1 RV internals LR scope with the results of the industry activity for all DB-1 component items in scope for LR was performed. The only component items that are not included in the MRP-227-A development are the original vent valve miscellaneous locking device parts and the lower grid guide block attachment replacement parts. A review of the component items using the MRP-227-A development methodology was performed and the results indicated the original vent valve miscellaneous locking device parts are categorized as "Category A" or "Existing Programs" and the lower grid guide block attachment replacement parts are categorized as "Category A." The addition of the original vent valve miscellaneous locking device parts as an "Existing Program" is currently under industry review.

Based on the results from fabrication records searches, deviations to the MRP-227-A program exist. These record searches revealed that the original Alloy X-750 dowel-to-guide block weld (along with the dowel itself) for DB-1 was removed due to repositioning and replaced with a stainless steel dowel, dowel cap, and locking weld; in addition, the guide blocks were welded to the lower grid forging. Additionally, the Alloy X-750 dowel-to-upper grid fuel assembly support pad welds were identified in MRP-227-A as not applicable to DB-1. However, a search of the original fabrication records indicates that this welding configuration does exist based on a component transfer from another B&W unit contract to the DB-1 contract. Alternative actions to address these deviations are currently under industry review and will be implemented upon concurrence. The alternative actions include elevation of an Expansion component item (Alloy X-750 dowel-to-lower grid fuel assembly support pad weld) to a Primary component item and establishing a new Expansion component item (Alloy X-750 dowel-to-upper grid fuel assembly support pad weld) specifically for DB-1, as shown in Appendix D of this report.

The limitations and assumptions in MRP-227-A, as well as in the FMECA (MRP-190) and functionality analysis (MRP-229, Revision 3), are applicable to DB-1, and are discussed in detail in Section 4.1.1.1 of this report.

5.1.3 Conclusion

The scope of the DB-1 RV Internals AMP is consistent with the scope of NUREG-1801 program XI.M16A.

5.2 *AMP Element 2 – Preventative Actions*

5.2.1 NUREG-1801, XI.M16A 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 [corrosion], 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 Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."*

5.2.2 DB-1 Preventive Actions

The DB-1 RV Internals AMP credits the PWR Water Chemistry Program for maintaining high water purity (see Section 4.2.2 of this report) and is an existing DB-1 program that is consistent with the ten (10) elements of an effective aging management program described in NUREG-1801, Revision 2, Section XI.M2, "Water Chemistry."

Additionally, DB-1 has implemented low-leakage core loading patterns as a preventative action (see Section 4.1.1.1 of this report).

5.2.3 Conclusion

The preventative actions for the DB-1 RV Internals AMP are consistent with the preventive actions in the NUREG-1801 program XI.M16A.

5.3 *AMP Element 3 – Parameters Monitored or Inspected*

5.3.1 NUREG-1801, XI.M16A Parameters Monitored/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. The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing Programs, such as the requirements for ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse-designed flux thimble tubes in GALL AMP XI.M37, "Flux Thimble Tube Inspection." 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.

5.3.2 DB-1 Parameters Monitored/Inspected

The DB-1 RV Internals Inspection Plan monitors for the detectable effects of the eight (8) aging degradation mechanisms outlined in Section 2.3 of this report. The DB-1 RV Internals AMP credits, and further augments, the ASME Section XI Inservice Inspection Program with the examinations in MRP 227-A Tables 4-1 and 4-4 as applicable to DB-1. See Appendix A and Appendix B of this report.

The DB-1 RV Internals Inspection Plan uses Ultrasonic Testing (UT), Visual Examination (VT-3), and physical measurement to monitor for the detectable effects of the eight (8) degradation mechanisms outlined in Section 2.3 of this report.

5.3.3 Conclusion

The parameters monitored/inspected in the DB-1 RV Internals AMP are consistent with the parameters monitored/inspected in the NUREG-1801 program XI.M16A.

5.4 *AMP Element 4 – Detection of Aging Effects*

5.4.1 NUREG-1801, XI.M16A 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, Existing Requirement 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.

The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): For the DB-1 program, there are no supplemental Primary or Expansion components.

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 include the physical measurements needed for the B&W RV internals core clamping items already identified in MRP-227-A, Table 4-1.

5.4.2 DB-1 Detection of Aging Effects

The methods for detection of aging effects in the DB-1 RV Internals Inspection Plan include UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimension, and visual (VT-3) examinations.

Physical measurements are conducted to detect gross effects of wear (i.e., loss of material). A one-time physical differential height measurement of the plenum rib pad to RV seating surface at DB-1 was performed in 2014 (RFO18) as described in Section 4.2.8 of this report.

VT-3 inspections are used to detect cracking caused by SCC, IASCC, and fatigue and loss of material induced by wear and 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. VT-3 inspections are also used to detect the effects of thermal and irradiation embrittlement through observations of cracking and/or fracture in these items. Where fitting, additional evaluations are being performed to demonstrate why the examination method, schedule, frequency, and coverage are appropriate.

Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by volumetric UT examination for bolting. As described in Section 4.2.7.2 of this report, some UT examinations of MRP-227-A component items have been performed.

The DB-1 RV Internals AMP requires inspections of the "Expansion" component items in MRP-227-A in accordance with the expansion criteria in MRP-227-A.

5.4.3 Conclusion

The detection of aging effects in the DB-1 RV Internals AMP is consistent with the methods identified for detection of aging effects in NUREG-1801 program XI.M16A.

5.5 *AMP Element 5 – Monitoring and Trending*

5.5.1 NUREG-1801, XI.M16A Monitoring and Trending

The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well [as] for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.

5.5.2 DB-1 Monitoring and Trending

The DB-1 RV Internals AMP, including one-time, periodic, and conditional examinations and other aging management methodologies, scheduled in accordance with the ASME B&PV Code, Section XI, *Examination Category B-N-3* and MRP-227-A *examinations*, provide timely detection of aging effects. The DB-1 program includes both the "Primary" component items, and the "Expansion" component items identified in MRP-227-A. The "Expansion" component items will be inspected as required by the results of the examinations of the "Primary" component items.

The DB-1 RV Internals AMP will follow the reporting requirements in MRP-227-A which allow the industry to monitor and trend results, thus driving preemptive industry action through notifications and updating of the MRP-227-A guidelines.

5.5.3 Conclusion

The monitoring and trending in the DB-1 RV Internals AMP is consistent with the monitoring and trending in NUREG-1801 program XI.M16A.

5.6 *AMP Element 6 – Acceptance Criteria*

5.6.1 NUREG-1801, XI.M16A 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.*

5.6.2 DB-1 Acceptance Criteria

The DB-1 RV Internals AMP uses the examination acceptance criteria in Section 5 of MRP-227-A for the "Primary" and "Expansion" component items. This includes examination acceptance criteria for visual examinations, volumetric examinations [as demonstrated in the examination Technical Justification (TJ)], and physical measurements, as well as Criteria for expanding the examinations from the "Primary" component items to include the "Expansion" component items. DB-1 NDE techniques will be qualified to the extent required by MRP-227-A and MRP-228. Component degradation that exceeds the examination acceptance criteria will be evaluated in accordance with WCAP-17096-NP Revision 2, including any additional guidance resulting from the ongoing NRC review of that report.

TJs were developed for the DB-1 RV internals component inspections for VT-3 and UT examinations by AREVA under a PWROG Project. The TJs, where appropriate, include further guidance with respect to examination coverage and relevant conditions.

Relevant conditions requiring corrective action for ASME B&PV Code Section XI Category B-N-3 VT-3 examinations of RV internals component items are detailed in ASME B&PV Code Section XI, IWB-3000.

5.6.3 Conclusion

Acceptance criteria for the DB-1 RV Internals AMP are consistent with the acceptance criteria in NUREG-1801 program XI.M16A.

5.7 *AMP Element 7 – Corrective Actions*

5.7.1 NUREG-1801 XI.M16A Corrective Actions

Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for B&W-designed RVI components in B&W Report No. BAW-2248. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies [now, *AREVA] on behalf of the B&W Owners Group, dated December 9, 1999. Alternative corrective action bases not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation.*

5.7.2 DB-1 Corrective Actions

In accordance with 10 CFR 50, Appendix B[25], FENOC has established a CAP for DB-1. The CAP at DB-1 is implemented through the FENOC Quality Assurance Program Manual (QAPM). Corrective actions are implemented through the FENOC CAP that satisfies the requirements of 10 CFR 50, Appendix B, Criterion XVI. Conditions adverse to quality, an all-inclusive term used in reference to failures, malfunctions, deficiencies, defective items, and nonconformances are identified, reported to management, and corrected. In the case of significant conditions adverse to quality, measures are implemented to ensure that the root cause is determined and that corrective actions are taken to preclude recurrence.

5.7.3 Conclusion

FENOC's corrective actions are consistent with the corrective actions in NUREG-1801 program XI.M16A.

5.8 *AMP Element 8 – Confirmation Process*

5.8.1 NUREG-1801, XI.M16A Confirmation Process

Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls.

5.8.2 DB-1 Confirmation Process

This element is included in the FENOC QAPM, which implements the requirements of 10 CFR 50, Appendix B. The focus of the confirmation process is on the follow-up actions taken to verify effective implementation of corrective actions and preclude repetition of significant conditions adverse to quality. The CAP includes the requirement that measures be taken to preclude repetition of significant conditions adverse to quality. These measures include actions to verify effective implementation of proposed corrective actions. The confirmation process is part of the CAP and, for significant conditions adverse to quality, includes:

- Reviews to assure proposed actions are adequate,
- Tracking and reporting of open corrective actions,
- Root Cause, and
- Reviews of corrective action effectiveness.

Effectiveness reviews are conducted as part of the CAP to ensure that corrective actions have been completed and to identify any repetition of events. The CAP is also monitored for potentially adverse trends. The existence of an adverse trend due to recurring or repetitive adverse conditions will result in the initiation of follow-up actions in the CAP. Program assessments will be performed in accordance with FENOC Procedure NOP-ER-2101, Engineering Program Management[26]. These periodic assessments will evaluate the program's effectiveness against predetermined standards and expectations, and will ensure that programmatic controls are in place to maintain or improve program performance.

5.8.3 Conclusion

FENOC's confirmation process is consistent with the confirmation process in NUREG-1801 program XI.M16A.

5.9 *AMP Element 9 – Administrative Controls*

5.9.1 NUREG-1801 XI.M16A Administrative Controls

The administrative controls for such programs, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation.

5.9.2 DB-1 Administrative Controls

Administrative controls that govern aging management activities are established within the document control procedures that implement: (1) industry standards related to administrative controls and quality assurance (QA) for the operational phase of nuclear power plants, and (2) the requirements of 10 CFR 50, Appendix B, Criterion VI. FENOC Procedure NOP-CC-5004, Pressurized Water Reactor Vessel Internals Program[27], is the governing procedure used to control the program at Davis-Besse.

Plant policies, directives, and procedures are written and controlled to specify and manage various activities, particularly those related to compliance with 10 CFR 50, Appendix B. The phrase "administrative control" refers to the adherence to the policies, directives, and procedures, and includes the formal review and approval process that the plant policies, directives, and procedures undergo as they are issued (and subsequently revised). The individual reports (i.e., the plant policies, directives, and procedures), in conjunction with the plant's QA Program Documents, provide the overall administrative framework to ensure regulatory requirements are met.

5.9.3 Conclusion

FENOC's administrative controls are consistent with the administrative controls in NUREG-1801 program XI.M16A.

5.10 *AMP Element 10 – Operating Experience*

5.10.1 NUREG-1801 XI.M16A Operating Experience

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program or to participate in industry initiatives that perform this function.

The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience.

5.10.2 DB-1 Operating Experience

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR units. In B&W units, the only incidents to date have been in the baffle-to-baffle bolts, RV internals high-strength bolting, and vent valve jackscrews and their locking devices. This operating experience (OE) was considered in the development of the MRP-227-A examination requirements.

FENOC participates in the industry programs for investigating and managing aging effects on RV internals. Through its participation in EPRI MRP activities, FENOC will continue to benefit from the reporting of RV internals inspection information from the industry, and will share its own internals inspection results with the industry, as appropriate. The DB-1 Reactor Internals AMP will report findings in accordance with Section 7 of MRP-227-A.

5.10.3 Conclusion

DB-1's OE is consistent with the OE in NUREG-1801 program XI.M16A.

5.11 *Program Conclusion*

Section 5.0 of this report shows that the DB-1 RV Internals Inspection Plan meets the intent of the ten (10) GALL program elements from Chapter XI, AMP XI.M16A from NUREG-1801; this demonstrates the adequacy of managing the aging effects of the DB-1 RV Internals.

6.0 CONDITIONS/LIMITATIONS AND UNIT SPECIFIC ACTION ITEMS CONTAINED IN MRP-227-A

Seven (7) conditions and limitations are placed on the use of MRP-227-A as well as eight (8) applicant/licensee action items that shall be addressed by applicants/licensees upon implementation of MRP-227-A. The following subsections describe those conditions/limitations and unit-specific actions which are applicable to DB-1.

Table 6-1 of this report gives a summary of the status of the MRP-227-A Topical Report Conditions/Limitations and Applicant/Licensee Action Items for DB-1. Items not applicable to the DB-1 are identified as such.

Table 6-1: Summary of Status of Topical Report Conditions/Limitations and Unit-Specific Actions Contained in MRP-227-A for DB-1

Item	Inspection Plan Location	Status
Topical Report Condition/Limitation 1 (MRP-227-A SER Section 4.1.1)	Not Applicable to DB-1	Not Applicable to DB-1
Topical Report Condition/Limitation 2 (MRP-227-A SER Section 4.1.2)	Not Applicable to DB-1	Not Applicable to DB-1
Topical Report Condition/Limitation 3 (MRP-227-A SER Section 4.1.3)	Section 6.1	Fulfilled for DB-1 by the publication of MRP-227-A
Topical Report Condition/Limitation 4 (MRP-227-A SER Section 4.1.4)	Section 6.1	Fulfilled for DB-1 by the publication of MRP-227-A
Topical Report Condition/Limitation 5 (MRP-227-A SER Section 4.1.5)	Section 6.1	Fulfilled for DB-1 by the publication of MRP-227-A
Topical Report Condition/Limitation 6 (MRP-227-A SER Section 4.1.6)	Section 6.1	Fulfilled for DB-1 by the publication of MRP-227-A
Topical Report Condition/Limitation 7 (MRP-227-A SER Section 4.1.7)	Not Applicable to DB-1	Not Applicable to DB-1
Applicant/Licensee Action Item 1 (MRP-227-A SER Section 4.2.1)	Section 6.2.1	Fulfilled for DB-1 by the publication of this report
Applicant/Licensee Action Item 2 (MRP-227-A SER Section 4.2.2)	Section 6.2.2	Fulfilled for DB-1 by the publication of this report
Applicant/Licensee Action Item 3 (MRP-227-A SER Section 4.2.3)	Not Applicable to DB-1	Not Applicable to DB-1

Item	Inspection Plan Location	Status
Applicant/Licensee Action Item 4 (MRP-227-A SER Section 4.2.4)	Section 6.2.3	Fulfilled for DB-1 by the publication of this report
Applicant/Licensee Action Item 5 (MRP-227-A SER Section 4.2.5)	Not Applicable to DB-1	Not Applicable to DB-1
Applicant/Licensee Action Item 6 (MRP-227-A SER Section 4.2.6)	Section 6.2.4	In response to Applicant/Licensee Action Item 6, detailed analyses justifying operation of the inaccessible and non-inspectable component items will be submitted to the NRC within one year of the detection of degradation exceeding the acceptance criteria of the linked MRP-227-A primary component items leading to expansion.
Applicant/Licensee Action Item 7 (MRP-227-A SER Section 4.2.7)	Section 6.2.5	In response to Applicant/Licensee Action Item 7, plant-specific analyses will be submitted to the NRC by one year prior to the MRP-227-A inspection of the applicable component items.
Applicant/Licensee Action Item 8 (MRP-227-A SER Section 4.2.8)	Section 6.2.6	The response to the remaining open items in Applicant/Licensee Action Item 8 will be submitted to the NRC by April 22, 2017.

6.1 *MRP-227-A Topical Report Conditions/Limitations Applicable to DB-1*

Topical Report Conditions/Limitations applicable to DB-1 are identified in Table 6-1 of this report. These topical report conditions/limitations are fulfilled for DB-1 by the publication of MRP-227-A.

6.2 *MRP-227-A Topical Report Applicant/Licensee Action Items Applicable to DB-1*

6.2.1 Applicant/Licensee Action Item 1

This action item is described in Section 3.2.5.1 and Section 4.2.1 of the SER for MRP-227-A and summarized within this section.

6.2.1.1 Discussion of Requirement

As discussed in Section 3.2.5.1 of the SER, each applicant/licensee is responsible for assessing its unit's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding unit design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RV Internals component items or unit operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227.

6.2.1.2 DB-1 Compliance/Conclusion

DB-1 is bounded by the unit design and operating history assumptions in MRP-227-A, the FMECA (MRP-190) and the B&W design functionality analysis (MRP-229, Revision 3) as addressed in Section 4.1.1.1 of this DB-1 RV Internals Inspection Plan.

Therefore, this application/licensee action item is considered fulfilled for DB-1.

6.2.2 Applicant/Licensee Action Item 2

This action item is described in Section 3.2.5.2 and Section 4.2.2 of the SER for MRP-227-A and summarized within this section.

6.2.2.1 Discussion of Requirement

As discussed in Section 3.2.5.2 of the SER, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RV Internals component items are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1[28], and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RV Internals component items that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RV Internals component items that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by the SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation.

6.2.2.2 DB-1 Compliance/Conclusion

A comparison between Table 4-1 and 4-2 in MRP-189, Revision 1 and the component items designated in the scope of LR for DB-1 has been performed, and two component items were identified as being within the scope of LR and subject to aging management review, but were not included in the relevant MRP-189, Revision 1 tables:

- Original Vent Valve Miscellaneous Locking Device Parts
- Lower Grid Guide Block Attachment Replacement Parts

The sequence steps as described in MRP-189, Revision 1 were used to evaluate the aging mechanisms and resultant aging effects on the component items that could result in degradation leading to significant risk. The results of the evaluation were reviewed for any necessary modifications to the program defined in MRP-227-A. The results of the evaluation categorized the original vent valve miscellaneous locking device parts as "Category A" or "Existing Programs" and the lower grid guide block attachment replacement parts as "Category A."[29] The addition of the original vent valve miscellaneous locking device parts as an "Existing Program" is currently under industry review.

Failure of the locking devices on the original vent valve assemblies due to thermal embrittlement and wear is adequately managed by the ASME B&PV Code Section XI programmatic controls along with the valve testing and inspection requirements each refueling outage (as required by plant Technical Specifications).

MRP-227-A Table 4-7 does not list any Existing Programs Components for B&W Units. Davis-Besse does however have actions in place associated with the original vent valve locking device aging management. AREVA is working with the PWROG and the EPRI MRP to establish interim guidance to account for this discrepancy and incorporate the appropriate information into the next revision of MRP-227. An existing program is in place at each of the B&W-designed units that requires testing and inspection of the vent valve assemblies each refueling outage. The aging management measures provided in these requirements include a provision to visually inspect the valve body and disc seating surfaces. Continuation of the existing vent valve testing and inspection requirements will manage cracking of the vent valve component items that could cause loss of the vent valve function.

Therefore, this application/licensee action item is considered fulfilled for DB-1.

6.2.3 Applicant/Licensee Action Item 4

This action item is described in Section 3.2.5.4 and Section 4.2.4 of the SER for MRP-227-A and summarized within this section.

6.2.3.1 Discussion of Requirement

As discussed in Section 3.2.5.4 of the SER, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the RV in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" examination category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of the SER. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval.

6.2.3.2 DB-1 Compliance/Conclusion

Original fabrication records have confirmed that the DB-1 CSS upper flange weld was stress relieved (See Section 4.2.5 of this report); based on the records, a stress relief was performed on the CSS upper flange weld during original fabrication of the DB-1 RV internals. The CSS upper flange weld does not need to be inspected as a "Primary" component.

Therefore, this application/licensee action item is considered fulfilled for DB-1.

6.2.4 Applicant/Licensee Action Item 6

This action item is described in Section 3.3.6 and Section 4.2.6 of the SER for MRP-227-A and summarized within this section.

6.2.4.1 Discussion of Requirement

As discussed in Section 3.3.6 in the SER, MRP-227 does not propose to inspect the following inaccessible component items: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques. Applicants/licensees shall justify the acceptability of these component items for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the component items. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible component items and, if necessary, provide their plan for the replacement of the component items for NRC review and approval.

6.2.4.2 DB-1 Compliance/Conclusion

FENOC will justify the acceptability of inaccessible and non-inspectable component items (core barrel cylinder including vertical and circumferential seam welds, former plates, external baffle-to-baffle bolts and their locking devices, core barrel-to-former bolts and their locking devices, and internal baffle-to-baffle bolts) for continued operation through the period of extended operation by performing an evaluation or by proposing a schedule for replacement of the component items (documented per FENOC Corrective Action Program CR 2014-06427[30]). In response to Applicant/Licensee Action Item 6, detailed analyses justifying operation of the inaccessible and non-inspectable component items will be submitted to the NRC within one year of the detection of degradation exceeding the acceptance criteria of the linked MRP-227-A primary component items leading to expansion.

6.2.5 Applicant/Licensee Action Item 7

This action item is described in Section 3.3.7 and Section 4.2.7 of the SER for MRP-227-A and summarized within this section.

6.2.5.1 Discussion of Requirement

As discussed in Section 3.3.7 of the SER, the applicants/licensees of B&W reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W Incore Monitoring Instrumentation (IMI) guide tube assembly spiders and CRGT spacer castings will maintain their functionality during the period of extended operation or for additional RV Internals component items that may be fabricated from CASS, martensitic stainless steel, or martensitic precipitation-hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these component items due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to component items that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to component items fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the unit's licensing basis and the need to maintain the functionality of the component items being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.

6.2.5.2 DB-1 Compliance/Conclusion

DB-1 is developing a plant-specific analysis to demonstrate that the IMI guide tube assembly spiders, CRGT spacer castings, and additional RV Internals component items that may be fabricated from CASS, martensitic stainless steel, or martensitic precipitation-hardened stainless steel materials (e.g., CSS vent valve top and bottom retaining rings) will maintain their functionality during the period of extended operation. The analysis will consider the possible loss of fracture toughness in these component items due to thermal embrittlement (TE) and/or IE and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The DB-1 specific analysis will be consistent with DB-1's licensing basis and the need to maintain the functionality of the component items being evaluated under all licensing basis conditions of operation. In response to Applicant/Licensee Action Item 7, plant-specific analyses will be submitted to the NRC by one year prior to the MRP-227-A inspection of the applicable component items.

6.2.6 Applicant/Licensee Action Item 8

This action item is described in Section 3.5.1 and Section 4.2.8 of the SER for MRP-227-A and summarized within this section.

6.2.6.1 Discussion of Requirement

As discussed in Section 3.5.1 in the SER, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SER, as an AMP for the RV internals component items at their facility. This submittal shall include the information identified in Section 3.5.1, Items 1 through 5, of the SER for MRP-227.

[Note: Although DB-1's August 27, 2010, LRA submittal preceded the issuance of the final SE for MRP-227, Revision 0 and NUREG-1801, Revision 2 and only information in Items 1 and 2 shall be identified, the NRC staff determined that it would be appropriate to review the DB-1 program relative to the elements of an acceptable RV Internals AMP described in NUREG-1801, Revision 2 AMP XI.M16A, including the extent to which the program will address information in Items 1 through 5 of the SER for MRP-227.]

6.2.6.2 DB-1 Compliance/Conclusion

As described in Section 3.5.1 of the SER, since DB-1's licensing basis contains a commitment to submit a RV Internals AMP and/or Inspection Program, as discussed in Section 2 of this report, FENOC is making a submittal for NRC review and approval to credit their implementation of MRP-227-A. Items 1 through 5 in Section 3.5.1 of the SER are included in this Inspection Plan:

1. *An AMP for the facility that addresses the ten (10) program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.*
   - The ten (10) program elements as defined in NUREG-1801, Revision 2 are addressed in Section 5.0 of this report. The DB-1 LRA Amendments 15 and 24 included a revision to LRA Section B.2.32 to address the ten (10) elements of an acceptable RV Internals AMP described in GALL Report Revision 2 AMP XI.M16A, and the consistency statement updated to identify the RV Internals Program as a new DB-1 program consistent with the ten (10) elements in GALL Report Revision 2 AMP XI.M16A. The NRC staff determined that the LRA Section B.2.32 revision to address the ten (10) elements in GALL Report Revision 2 AMP XI.M16A is acceptable for satisfying this information item[9].
2. *To ensure the MRP-227 program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and*

approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" examination category components.

- The DB-1 RV Internals Inspection Plan is contained within this report and is consistent with MRP-227-A; it also addresses the DB-1 plant-specific action items (this section). The NEI 03-08 mandatory, needed, and good practice guidelines contained in MRP-227-A were reviewed, and it has been noted that the needed RV internals guidelines implementation requirement is affected for DB-1, thus creating deviations to the MRP-227-A program. The description of these conditions is provided below:

  Based on the results from a records search, the original Alloy X-750 dowel-to-guide block weld (along with the dowel itself) for DB-1 was removed due to repositioning and replaced with a stainless steel dowel, dowel cap, and locking weld; in addition, the guide blocks were welded to the lower grid forging. Additionally, the Alloy X-750 dowel-to-upper grid fuel assembly support pad welds were identified in MRP-227-A as not applicable to DB-1. However, a search of the original fabrication records indicates that this welding configuration does exist based on a component transfer from another B&W unit contract to the DB-1 contract.

  An alternative action, including the basis for determining that the deviations (alternative action) meet the same objective, or level of conservatism exhibited by the original work product, was developed. The alternative action includes elevation of the Expansion component item (Alloy X-750 dowel-to-lower grid fuel assembly support pad weld) to a Primary component item and establishing a new Expansion component item (Alloy X-750 dowel-to-upper grid fuel assembly support pad weld) specifically for DB-1.

These alternative actions are currently under industry review.

3. *The regulation at 10 CFR 54.21(d) requires that an USAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for LR referencing MRP-227, as approved by the NRC, for their RVI component AMP shall ensure that the programs and activities specified as necessary in MRP-227, as approved by the NRC, are summarily described in the USAR supplement.*

   - In DB-1 LRA Amendments 15 and 24, Sections A.1.32 and B.2.32 were revised to state that the DB-1 RV Internals Program will address all plant-specific action items applicable to DB-1 that are established in Section 4.2 of the final MRP-227-A SE in addition to the MRP-227-A specified programs and activities. The NRC staff determined that these revisions to LRA Sections A.1.32 and B.2.32 are appropriate because information item No. 3 from Section 3.5.1 of the final MRP-227-A SE specifies that applicants referencing the staff approved version of MRP-227 as the basis for their RV Internals AMPs shall ensure that the programs and activities, as specified in the staff-approved version of MRP-227, are summarily described in the LRA USAR supplement. The NRC staff also reviewed the USAR supplement for this program, as amended, and concluded that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d)[9].

4. *The regulation at 10 CFR 54.22 requires each applicant for LR to submit any TS changes (and the justification for the changes) that are necessary to manage the effects of aging during the period of extended operation as part of its LR application (LRA). For the plant CLBs that include mandated inspection or analysis requirements for RVI either in the operating license for the facility or in the facility TS, the applicant/licensee shall compare the mandated requirements with the*

recommendations in the NRC-approved version of MRP-227. If the mandated requirements differ from the recommended criteria in MRP-227, as approved by the NRC, the conditions in the applicable license conditions or TS requirements take precedence over the MRP recommendations and shall be complied with.

- In DB-1 LRA Amendments 15 and 24, Sections A.1.32 and B.2.32 were revised to state that the MRP-227-A I&E guidelines require a visual (VT-3) examination of the core support shield (CSS) vent valve retaining rings for every 10-year ISI interval. In addition, DB-1 TS 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 less than or equal to 400 lbs is applied vertically upward. The TS inspection will continue to be performed at the prescribed frequency of 24 months. The MRP-227-A required visual (VT-3) examination will also be performed at the prescribed frequency of once every 10-year ISI interval. The NRC staff determined that these revisions to LRA Sections A.1.32 and B.2.32 are appropriate for addressing information item No. 4 from Section 3.5.1 of the final MRP-227-A SE, because the LRA revisions address the mandated TS requirements in addition to the separate MRP-227-A guidelines for inspection of the CSS vent valves[9].

5. *Pursuant to 10 CFR 54.21(c)(1), the applicant is required to identify all analyses in the CLB for their RVI components that conform to the definition of a TLAA in 10 CFR 54.3 and shall identify these analyses as TLAAs for the application in accordance with the TLAA identification requirement in 10 CFR 54.21(c)(1). MRP-227 does not specifically address the resolution of TLAAs that may apply to applicant/licensee RVI components. Hence, applicants/licensees who implement MRP-227, as approved by the NRC, shall still evaluate the CLB for their facilities to determine if they have plant-specific TLAAs that shall be addressed. If so, the applicant's/licensee's TLAA shall be submitted for NRC review along with the applicant's/licensee's application to implement the NRC-approved version of MRP-227.*

For those cumulative usage factor (CUF) analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. Otherwise, acceptance of these TLAAs shall be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to NUREG-1801, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program". To satisfy the evaluation requirements of ASME Code, Section III, Subsection NG-2160 and NG-3121, the existing fatigue CUF analyses shall include the effects of the reactor coolant system water environment.

- In DB-1 LRA Amendments 15 and 24, Sections A.1.32 and B.2.32 were revised to state that the program includes management of the TLAA for reduction in fracture toughness of the RV internals. The TLAA will be managed in accordance with the implementation of the MRP-227-A guidelines, as amended by the MRP-227-A SE, including all activities associated with the FENOC responses to plant-specific action items identified in Section 4.2 of the SE. In the SER related to the LR of DB-1, the NRC staff determined that DB-1's general program description, as revised by LRA Amendment 15, adequately addresses the management of this TLAA for the period of extended operation[9]. However, the effect of irradiation on the mechanical properties and deformation limits of the RV internals that was evaluated for the current term of operation in Appendix E of Topical Report BAW-10008, Part 1, Revision 1 supplemented by DB-1 USAR Appendix 4A will require an update for the period of extended operation. Therefore, the response to the remaining open items in Applicant/Licensee Action Item 8 will be submitted to the NRC by April 22, 2017.
- The FIV analysis of the RV internals and the incore instrument nozzles were dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and remain valid for the

period of extended operation. In addition, the FIV analysis for the surveillance capsule holder tubes were dispositioned in accordance with 10 CFR 54.21(c)(1)(ii) and has been projected to the end of the period of extended operation.

7.0 SUMMARY AND CONCLUSIONS

This report provides a description of the DB-1 RV Internals Inspection Plan and how it relates to the AMP at DB-1 for the management of aging effects consistent with previous commitments. This DB-1 RV Internals Inspection Plan is based on MRP-227-A, as applicable to DB-1. Section 5.0 of this report demonstrates the DB-1 RV Internals AMP, which includes the DB-1 RV Internals Inspection Plan, is consistent with the ten (10) program elements of NUREG-1801, Revision 2 AMP XI.M16A.

This DB-1 RV Internals Inspection Plan contains a discussion of the background of the B&W-designed unit RV internals programs, including operational experience, TLAAs, and DB-1 programs and activities.

The next removal of the RV internals from the DB-1 RV is scheduled in Year 2022 (RFO22). The appropriate examinations required by ASME B&PV Code Section XI, and the remainder of the initial examinations required by MRP-227-A are currently scheduled to be performed at that time. Any relevant conditions will be documented and dispositioned in FENOC's CAP and reported to the industry.

The DB-1 RV Internals AMP will include this DB-1 RV Internals Inspection Plan and will manage the effects of aging for RV internals. This plan provides reasonable assurance that the RV internals will remain functional through the DB-1 LR period of extended operation.

Section 6.0 to this RV Internals Inspection Plan provides the FENOC responses to the Topical Report Conditions and Applicant/Licensee Action Items found in the NRC SER documented in MRP-227-A.

8.0 REFERENCES

1. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A). EPRI, Palo Alto, CA: 2011, 1022863.

2. NUREG-1801, Revision 2, “Generic Aging Lessons Learned (GALL) Report,” dated December 2010. (NRC Accession No. ML103490041).

3. License Renewal Interim Staff Guidance, LR-ISG-2011-04, “Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors,” dated May 28, 2013. (NRC Accession No. ML12270A251 and ML12270A436).

4. NUREG-1800, Revision 2, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” dated December 2010. (NRC Accession No. ML103490036).

5. Letter Transmitting Davis-Besse Nuclear Power Station, Unit No. 1, Docket No. 50-346, License Number NPF-3, License Renewal Application and Ohio Coastal Management Program Consistency Certification, dated August 27, 2010. (NRC Accession Nos. ML102450565, ML102450567, ML102450563 and ML102450568).

6. Letter Transmitting Davis-Besse Nuclear Power Station, Unit No. 1, Docket No. 50-346, License Number NPF-3, Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640), and License Renewal Application Amendment No. 15, dated September 16, 2011. (NRC Accession No. ML11264A059).

7. NRC Letter, Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640), dated July 11, 2011. (NRC Accession No. ML11174A191).

8. Letter Transmitting 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, dated March 9, 2012. (NRC Accession No. ML12094A383).

9. Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station, Docket Number 50-346, FirstEnergy Nuclear Operating Company, U.S. Nuclear Regulatory Commission, September 2013. (NRC Accession No. ML13248A267).

10. Letter (from Christian B. Larsen) Report Transmittal: Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0), EPRI, Palo Alto, CA, 2008, 1016596, January 12, 2009. (NRC Accession No. ML090160204).

11. Letter from Robert A Nelson (NRC) to Niel Wilmshurst (EPRI), "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. (NRC Accession No. ML11308A770).

12. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, 2007 Edition with 2008 Addenda, American Society of Mechanical Engineers, New York, NY.

13. Nuclear Energy Institute, "Guidelines for the Management of Materials Issues," NEI 03-08, Revision 2, January 2010. (NRC Accession No. ML102880028).

14. FENOC Corrective Action Program: Condition Report (CR) 2014-05971, and Corrective Action (CA) 2014-05971-001 and 2014-05971-002, "PWR Vessel Internals Program (PWRVIP) Incorrect Information Discovered in License Renewal Document and Deviation from NEI 03-08 Document (MRP-227A)," Origination Date: 03/31/2014. [These documents are maintained and controlled within FENOC's Records System and are not retrievable from AREVA Records Management.]

15. Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228). EPRI, Palo Alto, CA: 2009, 1016609.

16. Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190). EPRI, Palo Alto, CA: 2006, 1013233.

17. Materials Reliability Program: Functionality Analysis for Babcock & Wilcox Representative PWR Internals (MRP-229, Rev. 3). EPRI, Palo Alto, CA: 2010, 1022402.

18. J.B. Hall, S. Fyfitch, and K.E. Moore, "Laboratory and Operating Experience with Alloy A-286 and Alloy X-750 RV Internals Bolting Stress Corrosion Cracking," 11th International Conference on Environmental Degradation of Materials in Nuclear Systems, Stevenson, WA, August 10-14, 2003.

19. S. Fyfitch, K. E. Moore, and S. L. Harper, "Development of an In-reactor SCC Program for Internals Bolting," ASME-PVPD-OAC Nuclear Plant Systems/Components Aging Management and Life Extension Conference, San Diego, CA, June 23-27, 1991.

20. "Briefing Material for June 14, 2005 Mtg on Rx Vessel Internals from the EPRI MRP – HT Tang," dated June 14, 2005. (NRC Accession No. ML051710058).

21. NRC Letter, "Davis-Besse Nuclear Power Station, Unit No.1-Issuance of Amendment for the Conversion to the Improved Technical Specifications With Beyond Scope Issues (TAC NOS. MD6319-MD6322, MD6324-MD6333, MD6398-MD6403, MD6644-MD6649, AND MD6684)," dated November 20, 2008. (NRC Accession No. ML082900600).

22. AREVA Inc. Document BAW-10008, Part 1, Rev. 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," June 1970.

23. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG, "Core Support Structures," 1998 Edition through 2000 Addenda.

24. PWR Owners Group: "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (WCAP-17096-NP, Revision 2), Westinghouse Electric Corporation, Pittsburgh, PA: December 2009.

25. U.S. Code of Federal Regulations, Title 10 (10 CFR), "Domestic Licensing of Production and Utilization Facilities," Appendix B to Part 50, "Quality Assurance Criteria for Nuclear Power Plants," Office of the Federal Register, National Archives and Records Administration, 2011.

26. FENOC Procedure NOP-ER-2101, Revision 08, "Engineering Program Management," Effective Date: 07/26/2013. [This document is maintained and controlled within FENOC's Records System and is not retrievable from AREVA Records Management.]

27. FENOC Procedure NOP-CC-5004, Revision 03, "Pressurized Water Reactor Vessel Internals Program," Effective Date: 12/20/2013. [This document is maintained and controlled within FENOC's Records System and is not retrievable from AREVA Records Management.]

28. Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1). EPRI, Palo Alto, CA: 2009, 1018292.

29. AREVA Document, "Davis-Besse Reactor Vessel Internals License Renewal Scope and MRP-189, Revision 1 Comparison," 103-3359NP-000 (ANP-3359NP Revision 0), AREVA Inc., Lynchburg, VA: December 2014.

30. FENOC Corrective Action Program: Condition Report (CR) 2014-06427 and Corrective Action (CA) 2014-06427-001, "PWRVIP, Reactor Vessel Internals Inspection Plan Required Action Item #6 Not Included in Inspection Plan Work Scope," Origination Date: 04/07/2014. [These documents are maintained and controlled within FENOC's Records System and are not retrievable from AREVA Records Management.]

31. FENOC Procedure NOP-CC-5003, Revision 02, "Processing Deviations to Materials Initiative Guidance," Effective Date: 01/17/2011. [This document is maintained and controlled within FENOC's Records System and is not retrievable from AREVA Records Management.]

APPENDIX A
DB-1 PRIMARY COMPONENTS ITEMS

Appendix A contains selected rows from Table 4-1, “B&W Plants Primary Components” from MRP-227-A modified to be DB-1 unit-specific. Only items applicable to DB-1 are included in this Appendix.

Item	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange CSS top flange	Loss of material and associated loss of core clamping pre-load (Wear)	None	One-time physical measurement no later than two refueling outages from the beginning of the license renewal period. Perform subsequent visual (VT-3) examination on the 10-year ISI interval.	Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel. See Figure 4-1 of MRP-227-A.
Control Rod Guide Tube Assembly CRGT spacer castings	Cracking (TE), including the detection of fractured spacers or missing screws	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examination on the 10-year ISI interval.	Accessible surfaces at each of the 4 screw locations (at every 90°) of 100% of the CRGT spacer castings (limited accessibility). See Figure 4-5 of MRP-227-A.
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring (Note 1)	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces (see BAW-2248A, page 4.3 and Table 4-1). See Figure 4-11 of MRP-227-A.

Item	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS bolts and their locking devices SSHT bolts and their locking devices	Volumetric examination (UT) of the bolts within two refueling outages from 1/1/2006 or next 10-year ISI interval, whichever is first. Subsequent examination on 10-year ISI interval unless an evaluation of the baseline results submitted for NRC staff approval justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-7 of MRP-227-A.
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS bolts and their locking devices SSHT bolts and their locking devices	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination on 10-year ISI interval unless an evaluation of the baseline results submitted for NRC staff approval justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-8 of MRP-227-A.

Item	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Barrel Assembly Baffle-to-former bolts	Cracking (IASCC, IE, Overload) (Note 4)	Baffle-to-baffle bolts, Core barrel-to-former bolts	Baseline volumetric examination (UT) no later than two refueling outages from the beginning of the license renewal period with subsequent examination after 10 additional years.	100% of accessible bolts. (Note 3) See Figure 4-2 of MRP-227-A.
Core Barrel Assembly Baffle plates	Cracking (IE), including the detection of readily detectable cracking in the baffle plates	Core barrel cylinder (including vertical and circumferential seam welds), Former plates	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of the accessible surface within 1 inch around each flow and bolt hole. See Figure 4-2 of MRP-227-A.
Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	Cracking (IASCC, IE, Overload), including the detection of missing, non-functional, or removed locking devices or welds	Locking devices including locking welds, for the external baffle-to-baffle bolts and Core barrel-to-former bolts	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible baffle-to-former and internal baffle-to-baffle bolt locking devices. (Note 3) See Figure 4-2 of MRP-227-A.

Item	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS bolts and their locking devices. SSHT bolts and their locking devices	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination on 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-8 of MRP-227-A.
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds(2)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads.	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examination on the 10-year ISI interval.	Accessible surfaces of 100% of the 24 dowel-to-guide block welds. See Figure 4-4 of MRP-227-A.

(2) Based on fabrication records search results, the original Alloy X-750 dowel-to-guide block weld (along with the dowel itself) for DB-1 was removed due to repositioning and replaced with a stainless steel dowel, dowel cap, and locking weld; in addition, the guide blocks were welded to the lower grid forging. This modification was unknown during the preparation of MRP-227. Because these Alloy X-750 dowels and nickel-base locking weld were removed from the guide blocks at DB-1, the requirements for the MRP-227-A Primary item, "Alloy X-750 dowel-to-guide block welds in the lower grid assembly," in Table 4-1 are not applicable to DB-1. (See Appendix D of this document for alternative actions to address this condition.)

Item	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	Cracking (TE/IE), including the detection of fractured or missing spider arms or, Cracking (IE), including separation of spider arms from the lower grid rib section at the weld	Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels, and cap screws are included because of IE of the welds)	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year ISI interval.	100% of top surfaces of 52 spider castings and welds to the adjacent lower grid rib section. See Figures 4-3 and 4-6 of MRP-227-A.

Notes:

1. A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of scratches, pitting, embedded particles, leakage of the seating surfaces, cracking of lock welds and locking cups, jack screws for proper position, and wear. The frequency is defined in DB-1's technical specifications.(3)
2. Examination acceptance criteria and expansion criteria are Appendix C of this report.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Appendix C of this report, must be examined for inspection credit.
4. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking is inspected by UT inspection. The effect of loss of joint tightness on the functionality will be addressed by analysis of the core barrel assembly, which will be performed to address Applicant/Licensee Action Item 6 of the NRC Safety Evaluation for MRP-227-A (See Section 6.2.4 of this report).

(3) AREVA has provided additional information on recommended examination of vent valve locking devices for the B&W units. As part of this information, modified wording regarding inspection of the vent valve miscellaneous locking device parts is as follows:

> "A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of leakage between the valve disc and the valve body (i.e., flow lines across the sealing surface), cracking of lock welds and locking cups, jack screws for proper position, and wear. The frequency is defined in each DB-1's technical specifications."

It is expected that this modified wording will be incorporated in the next revision to MRP-227-A: Table 4-1, Note 1.

APPENDIX B
DB-1 EXPANSION COMPONENTS ITEMS

Appendix B contains selected rows from Table 4-4, “B&W Plants Expansion Components” from MRP-227-A modified to be DB-1 unit-specific. Only items applicable to DB-1 are included in this Appendix.

Item	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Upper thermal shield (UTS) bolts and their locking devices **Core Barrel Assembly** Surveillance specimen holder tube (SSHT) bolts and their locking devices	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB, LCB or FD bolts and their locking devices	Bolt: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 2) See Figure 4-7 of MRP-227-A.
Core Barrel Assembly Core barrel cylinder (including vertical and circumferential seam welds) Former plates	Cracking (IE), including readily detectable cracking	Baffle plates	No examination requirements. Justify by evaluation or by replacement.	Inaccessible See Figure 4-2 of MRP-227-A.
Core Barrel Assembly Baffle-to-baffle bolts Core barrel-to-former bolts	Cracking (IASCC, IE, Overload) (Note 3)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements, Justify by evaluation or by replacement.	An acceptable examination technique currently not available. See Figure 4-2 of MRP-227-A.
			External baffle-to-baffle bolts, core barrel-to-former bolts: No examination requirements. Justify by evaluation or by replacement.	Inaccessible See Figure 4-2 of MRP-227-A.

Item	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	Cracking (IASCC, IE)	Locking devices, including locking welds, of baffle-to-former bolts or internal baffle-to-baffle bolts	No examination requirements. Justify by evaluation or by replacement.	Inaccessible See Figure 4-2 of MRP-227-A.
Lower Grid Assembly Lower grid fuel assembly support pad items: pad, pad-to-rib sections welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels and cap screws are included because of IE of the welds)	Cracking (IE), including the detection of separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads	IMI guide tube spiders and spider-to-lower grid rib section welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of the pads, dowels, and cap screws, and associated welds in 100% of the lower grid fuel assembly support pads. See Figure 4-6 of MRP-227-A.
Lower Grid Assembly Alloy X-750 dowel-to-lower grid fuel assembly support pad welds[4]	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the support pad dowel locking welds. See Figure 4-6 of MRP-227-A.

[4] Based on the proposed alternative action to address the deviation to the NEI 03-08 needed implementation requirement in MRP-227-A, it is proposed that the Expansion component item, "Alloy X-750 dowel-to-lower grid fuel assembly support pad weld," be elevated to a Primary component item and that a new Expansion component item, "Alloy X-750 dowel-to-upper grid fuel assembly support pad weld," be established specifically for DB-1. (See Appendix D of this document for alternative actions to address this condition.)

Item	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Grid Assembly Lower thermal shield (LTS) bolts and their locking devices	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB, LCB and FD bolts and their locking devices	Bolts: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 2) See Figure 4-8 of MRP-227-A.

Notes:

1. Examination acceptance criteria and expansion criteria are Appendix C of this report.
2. A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit.
3. The primary aging degradation mechanisms for loss of joint tightness for these items are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking could be inspected by UT inspection if it were possible for these bolts. Therefore, the effects of loss of joint tightness and/or cracking on the functionality of these bolts relative to the entire core barrel assembly will be addressed by analysis of the core barrel assembly, which will be performed to address Applicant/Licensee Action Item 6 of the NRC Safety Evaluation for MRP-227-A (See Section 6.2.4 of this report).

APPENDIX C
DB-1 EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Appendix C contains selected rows from Table 5-1, "B&W Plants Examination Acceptance and Expansion Criteria" from MRP-227-A. Only items applicable to DB-1 are included in this Appendix.

Item	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange CSS top flange	One-time physical measurement. In addition, a visual (VT-3) examination is conducted for these items. The measured differential height from the top of the plenum rib pads to the vessel seating surface shall average less than 0.004 inches compared to the as-built condition. The specific relevant condition for these items is wear that may lead to a loss of function.	None	N/A	N/A
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring	Visual (VT-3) examination. The specific relevant condition is evidence of damaged or fractured retaining ring material, and missing items.	None	N/A	N/A
Control Rod Guide Tube Assembly CRGT spacer castings	The specific relevant condition for the VT-3 of the CRGT spacer castings is evidence of fractured spacers or missing screws.	None	N/A	N/A

Item	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	1) Volumetric (UT) examination of the UCB bolts. The examination acceptance criteria for the UT of the UCB bolts shall be established as a part of the examination technical justification. 2) Visual (VT-3) examination of the UCB bolt locking devices. The specific relevant condition for the VT-3 of the UCB bolt locking devices is evidence of broken or missing bolt locking devices.	UTS bolts and LTS bolts and their locking devices SSHT bolts and their locking devices	1) Confirmed unacceptable indications exceeding 10% of the UCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: 100% of the accessible UTS bolts and 100% of the accessible LTS bolts and 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the UCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: 100% of the accessible UTS bolt locking devices and 100% of the accessible LTS bolt locking devices and 100% of the accessible SSHT bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the VT-3 of the expansion locking devices is evidence of broken or missing bolt locking devices.

Item	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	1) Volumetric (UT) examination of the LCB bolts. The examination acceptance criteria for the UT of the LCB bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the LCB bolt locking devices. The specific relevant condition for the VT-3 of the LCB bolt locking devices is evidence of broken or missing bolt locking devices.	UTS bolts and LTS bolts and their locking devices SSHT bolts and their locking devices	1) Confirmed unacceptable indications exceeding 10% of the LCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: 100% of the accessible UTS bolts and 100% of the accessible LTS bolts and 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the LCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: 100% of the accessible UTS bolt locking devices and 100% of the accessible LTS bolt locking devices and 100% of the accessible SSHT bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the VT-3 of the expansion locking devices is evidence of broken or missing bolt locking devices.

Item	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Baffle-to-former bolts	Baseline volumetric (UT) examination of the baffle-to-former bolts. The examination acceptance criteria for the UT of the baffle-to-former bolts shall be established as part of the examination technical justification.	Baffle-to-baffle bolts, Core barrel-to-former bolts	Confirmed unacceptable indications in greater than or equal to 5% (or 43) of the baffle-to-former bolts, provided that none of the unacceptable bolts are on former elevations 3, 4, and 5, or greater than 25% of the bolts on a single baffle plate, shall require an evaluation of the internal baffle-to-baffle bolts for the purpose of determining whether to examine or replace the internal baffle-to-baffle bolts. The evaluation may include external baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining whether to replace them.	N/A
Core Barrel Assembly Baffle plates	Visual (VT-3) examination. The specific relevant condition is readily detectable cracking in the baffle plates.	a. Former plates b. Core barrel cylinder (including vertical and circumferential seam welds)	a. and b. Confirmed cracking in multiple (2 or more) locations in the baffle plates shall require expansion, with continued operation of former plates and the core barrel cylinder justified by evaluation or by replacement by the completion of the next refueling outage.	a. and b. N/A

Item	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	Visual (VT-3) examination. The specific relevant condition is missing, non-functional, or removed locking devices including locking welds.	Locking devices, including locking welds, for the external baffle-to-baffle bolts and barrel-to-former bolts	Confirmed relevant conditions in greater than or equal to 1% (or 11) of the baffle-to-former or internal baffle-to-baffle bolt locking devices, including locking welds, shall require an evaluation of the external baffle-to-baffle and core barrel-to-former bolt locking devices for the purpose of determining continued operation or replacement.	N/A
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds[5]	Initial visual (VT-3) examination. The specific relevant condition is separated or missing locking weld, or missing dowel.	Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads[6]	Confirmed evidence of relevant conditions at two or more locations shall require that the VT-3 examination be expanded to include the Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads by the completion of the next refueling outage.	The specific relevant condition for the VT-3 of the expansion dowel locking weld is separated or missing locking weld, or missing dowel.

[5] Based on fabrication records search results, the original Alloy X-750 dowel-to-guide block weld (along with the dowel itself) for DB-1 was removed due to repositioning and replaced with a stainless steel dowel, dowel cap, and locking weld; in addition, the guide blocks were welded to the lower grid forging. This modification was unknown during the preparation of MRP-227. Because these Alloy X-750 dowels and nickel-base locking weld were removed from the guide blocks at DB-1, the requirements for the MRP-227-A Primary item, "Alloy X-750 dowel-to-guide block welds in the lower grid assembly," in Table 4-1 are not applicable to DB-1.

[6] Based on the proposed alternative action to address the deviation to the NEI 03-08 needed implementation requirement in MRP-227-A, it is proposed that the Expansion component item, "Alloy X-750 dowel-to-lower grid fuel assembly support pad weld," be elevated to a Primary component item and that a new Expansion component item, "Alloy X-750 dowel-to-upper grid fuel assembly support pad weld," be established specifically for DB-1. (See Appendix D of this document for alternative actions to address this condition.)

Item	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	1) Volumetric (UT) examination of the FD bolts. The examination acceptance criteria for the UT of the FD bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the FD bolt locking devices. The specific relevant condition for the VT-3 of the FD bolt locking devices is evidence of broken or missing bolt locking devices.	UTS bolts and LTS bolts and their locking devices SSHT bolts and their locking devices	1) Confirmed unacceptable indications exceeding 10% of the FD bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: 100% of the accessible UTS bolts and 100% of the accessible LTS bolts and 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the FD bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: 100% of the accessible UTS bolt locking devices and 100% of the accessible LTS bolt locking devices and 100% of the accessible SSHT bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the VT-3 of the expansion locking devices is evidence of broken or missing bolt locking devices.

Item	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rid section welds	Initial visual (VT-3) examination. The specific relevant conditions for the IMI guide tube spiders are fractured or missing spider arms. The specific relevant conditions for the IMI spider-to-lower grid rib section welds are separated or missing welds.	Lower fuel grid assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds	Confirmed evidence of relevant conditions at two or more IMI guide tube spider locations or IMI guide tube spider-to-lower grid rib section welds shall require that the VT-3 examination be expanded to include lower fuel assembly support pad items by the completion of the next refueling outage.	The specific relevant conditions for the VT-3 of the lower grid fuel assembly support pad items (pads, pad-to-rib section welds, Alloy X-750 dowels, cap screws, and their locking welds) are separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads.

Notes:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).

APPENDIX D
EFFECTS OF DEVIATIONS TO MRP-227-A FOR DB-1

The Alloy X-750 dowel-to-guide block weld with the Alloy 82 weld metal (along with the dowel itself) was removed from the DB-1 internals during fabrication. Also the Alloy X-750 dowel-to-upper grid fuel assembly support pad weld configuration previously thought not to exist in the DB-1 internals actually does exist.

Since these two items are linked in MRP-227-A, a deviation (alternative action) is required for DB-1 that meets the same objective, or level of conservatism exhibited by the original work product.

The alternative action includes elevation of the Expansion component item, Alloy X-750 dowel-to-lower grid fuel assembly support pad weld, to a Primary component item and establishing a new Expansion component item, Alloy X-750 dowel-to-upper grid fuel assembly support pad weld, specifically for DB-1.

The final alternative action includes new Primary, Expansion, and examination acceptance and expansion criteria tables for DB-1 as described in Table D-1, Table D-2, and Table D-3 of this Appendix.

Note the alternative actions to address these deviations have been documented in the FENOC Corrective Action Program (CR 2014-05971) and being processed per FENOC procedure NOP-CC-5003, Processing Deviations to Materials Initiative Guidance[31].

Table D-1: New Primary Component Item for DB-1, Lower Grid Assembly

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Lower Grid Assembly Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	Davis-Besse	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels.	Alloy X-750 dowel-to-upper grid fuel assembly support pad welds	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year ISI interval.	Accessible surfaces of 100% of the dowel locking welds.

Note 2: Examination acceptance criteria and expansion criteria for the B&W component items are in Table D-3.

Table D-2: New Expansion Component Item for DB-1, Upper Grid Assembly

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Grid Assembly Alloy X-750 dowel-to-upper grid fuel assembly support pad welds	Davis-Besse	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the dowel locking welds.

Note 1: Examination acceptance criteria and expansion criteria for the B&W component items are in Table D-3.

Table D-3: New Examination Acceptance and Expansion Criteria for Davis-Besse, Lower Grid Assembly

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Lower Grid Assembly Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	Davis-Besse	Initial visual (VT-3) examination. The specific relevant condition is separated or missing locking weld, or missing dowel.	Alloy X-750 dowel-to-upper grid fuel assembly support pad welds	Confirmed evidence of relevant conditions at two or more locations shall require that the VT-3 examination be expanded to include the Alloy X-750 dowel locking welds to the upper grid fuel assembly support pads by the completion of the next refueling outage.	The specific relevant condition for the VT-3 of the expansion dowel locking weld is separated or missing locking weld, or missing dowel.

Note1: The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).