Enclosure B
L-21-093

Affidavit
(3 pages follow)

A F F I D A V I T

1. My name is Philip A. Opsal. I am Manager, Product Licensing for Framatome Inc. (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the Technical Report "MRP-227-A Applicant/Licensee Action Item 7 Analysis for the Davis-Besse Nuclear Power Station Unit No. 1", Document Number ANP 3438P Revision 1, referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6 (c), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on February 4 , 2021

____

Philip A. Opsal

Enclosure C
L-21-093

MRP-227-A Applicant/Licensee Action Item 7 Analysis for Davis-Besse Nuclear Power Station Unit No. 1 (Non-Proprietary)
(48 pages follow)

MRP-227-A Applicant/Licensee Action Item 7 Analysis for Davis-Besse Nuclear Power Station Unit No. 1

ANP-3438NP
Revision 1

Technical Report

February 2021

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1 (Rev. 0)	All	Initial Issue
2 (Rev. 1)	Throughout	Updated Proprietary bracketing, updated template
3 (Rev. 1)	Throughout	Various minor changes, updated references to correspond to Section 8.0 reference numbering
4 (Rev. 1)	Section 2.1	Updated reference to WCAP-17096-NP-A, Rev. 2
5 (Rev. 1)	Section 4.1.3	Updated with recent operating experience and associated references
6 (Rev. 1)	Section 6.3	Made reference to interim guidance
7 (Rev. 1)	Section 8.0	Updated Reference 3, removed Reference 4, renumbered remaining references, added References 7-11

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Nomenclature

Acronym	Definition
A/LAI	Applicant/Licensee Action Item
AMS	Aerospace Material Specification
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
B&W	Babcock and Wilcox
CASS	Cast Austenitic Stainless Steel
CMTR	Certified Material Test Report
CRA	Control Rod Assembly
CRGT	Control Rod Guide Tube
CSA	Core Support Assembly
CSS	Core Support Shield
DB-1	Davis-Besse Nuclear Power Station Unit No. 1
DBTT	Ductile to Brittle Transition Temperature
EPRI	Electric Power Research Institute
FATT	Fracture Appearance Transition Temperature
FENOC	FirstEnergy Nuclear Operating Company
FIV	Flow-Induced Vibration
GALL	Generic Aging Lessons Learned (NUREG-1801)
IASCC	Irradiated-Assisted Stress Corrosion Cracking
I&E Guidelines	Inspection and Evaluation Guidelines (MRP-227-A)
IE	Irradiation Embrittlement
IMI	Incore Monitoring Instrumentation
ISI	Inservice Inspection
LOCA	Loss of Coolant Accident
MRP	Materials Reliability Program
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply Systems
PH	Precipitation-Hardened
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
PT	Dye Penetrant Testing (NDE Technique)
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RT	Radiographic Testing (NDE Technique)
SCC	Stress Corrosion Cracking

Acronym	Definition
SE	Safety Evaluation
SER	Safety Evaluation Report
SSE	Safe Shutdown Earthquake
TE	Thermal Aging Embrittlement
U.S.	United States
UT	Ultrasonic Testing (NDE Technique)
VT-3	Visual Examination (NDE Technique)

ABSTRACT

The purpose of this document is to summarize the analyses performed for the applicable component items at DB-1 to complete applicant/licensee action item 7 (A/LAI 7) from MRP-227-A (Reference 1). The document includes a discussion of the purpose, the methodology utilized, a summary of the background, evaluation inputs, evaluation, and conclusion for each component item, and an overall conclusion.

1.0 INTRODUCTION

The Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) developed inspection and evaluation (I&E) guidelines in document Reference 1 for managing long-term aging reactor vessel internal components of PWRs. Specifically, the I&E guidelines are applicable to reactor vessel internal structural components; they do not address fuel assemblies, reactivity control assemblies, or welded attachments to the reactor vessel. The I&E guidelines concentrate on eight (8) aging degradation mechanisms and their aging effects, such as loss of fracture toughness. The I&E guidelines define requirements for inspections that will allow owners of pressurized water reactors (PWRs) to demonstrate that the effects of aging degradation are adequately managed for the period of extended operation. These guidelines contain mandatory and needed requirements and an implementation schedule for nuclear units employing Babcock & Wilcox (B&W) nuclear steam supply systems (NSSS) currently operating in the United States (U.S.).

Reference 1 includes a safety evaluation report (SER) prepared by the U.S. Nuclear Regulatory Commission (NRC). The NRC staff determined whether the guidance contained in the report provided reasonable assurance that the I&E guidelines ensured that the reactor vessel internal components will maintain their intended functions during the period of extended operation. From the determination, seven (7) topical report conditions and eight (8) plant-specific applicant/licensee action items (A/LAIs) were contained in the SER to alleviate issues and concerns of the NRC staff. The plant specific A/LAIs address topics related to the implementation of Reference 1 that could not be effectively addressed on a generic basis. The seventh A/LAI (A/LAI 7) addresses the NRC staff concerns regarding thermal aging embrittlement (TE) and irradiation embrittlement (IE). A/LAI 7 reads as follows:

As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components 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 components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components 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 plant's licensing basis and the need to maintain the functionality of the components 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. This is Applicant/Licensee Action Item 7.

In response to A/LAI 7, the following reactor vessel internals component items were identified as requiring further aging management for FirstEnergy Nuclear Operating Company's (FENOC) Davis-Besse Nuclear Power Station Unit No. 1 (DB-1) based on material type:

- Control Rod Guide Tube (CRGT) Spacer Castings (ASTM A351, Grade CF3M)
  - Categorized as "Primary" Component Item: Screened as potentially susceptible to TE, but not IE

- Incore Monitoring Instrumentation (IMI) Guide Tube Spider Castings (ASTM A351, Grade CF8)
  - Categorized as “Primary” Component Item: Screened as potentially susceptible to IE, but not TE
- Vent Valve Retaining Rings (AMS 5658, Type 15-5 PH)
  - Categorized as “Primary” Component Item: Screened as potentially susceptible to TE, but not IE
- Select Original Vent Valve Locking Device Parts: **[ ]** (ASTM A276, Type 431)
  - Categorized as “Existing Programs”

In accordance with MRP-189-Revision 1 (Reference 2) and the response to A/LAI 2, the cast vent valve bodies, core support shield (CSS) outlet nozzle castings, and plenum cylinder reinforcement casting were all screened as not being susceptible to TE or IE and are therefore outside the scope of A/LAI 7 from Reference 1.

This document will address and fulfill A/LAI 7 of Reference 1 for these reactor vessel internals component items; that is, to develop a plant-specific analysis for DB-1 to demonstrate that these reactor vessel internals component items will maintain their functionality during the period of extended operation, considering the loss of fracture toughness due to TE and/or IE (whichever is applicable).

Each of the applicable component items identified above has its own section (i.e., CRGT Spacer Castings – Section 3.0; IMI Guide Tube Spider Castings – Section 4.0; Vent Valve Retaining Rings – Section 5.0; and Select Original Vent Valve Locking Device Parts – Section 6.0).

Information considered by Framatome to be proprietary is marked with brackets: **[ ]**

2.0 METHODOLOGY

The purpose of this section is to provide various potential methodologies and identify the ultimate methodology used to evaluate the applicable DB-1 reactor vessel internals component items fabricated from cast austenitic stainless steel (CASS), martensitic stainless steel, and precipitation-hardenable (PH) stainless steel type materials.

2.1 *WCAP-17096-NP, Revision 2 Methodology Acceptability*

The Pressurized Water Reactor Owners Group (PWROG) document, WCAP-17096-NP-A, Revision 2 (Reference 3) provides a methodology for developing evaluation procedures to assess the functional impacts of degradation in component items with "observed relevant indications." As will be discussed below, the basis for why these component items are not expected to fail [ ] is part of the methodology used herein to justify that these component items will be expected to maintain their functionality through the period of extended operation. Therefore, the WCAP-17096 methodology [ ] then using the WCAP-17096 methodology may be needed (if replacement of affected component items is not preferred).

2.2 *MRP-227-A Safety Evaluation Suggested Methodologies*

As described in A/LAI 7 of Reference 1, to address the NRC staff concerns regarding TE and IE of potentially susceptible materials, applicants/licensees are required to perform a plant-specific analysis or evaluation demonstrating that certain component items will maintain their functionality during the period of extended operation. In accordance with the safety evaluation (SE) included in Reference 1, possible acceptable approaches may include, but are not limited to:

- Functionality analyses for the set of like components or assembly-level functionality analyses, or

- Component level flaw tolerance evaluation justifying that the MRP-227 recommended inspection technique(s) can detect a structurally significant flaw for the component in question, taking into account the reduction in fracture toughness due to IE and TE; or

- For CASS, if the application of applicable screening criteria for the component's material demonstrates that the components are not susceptible to either TE or IE, or the synergistic effects of TE and IE combined, then no other evaluation would be necessary. For assessment of CASS materials, the licensees or applicants for license renewal (LR) may apply the criteria in Reference 4 as the basis for determining whether the CASS materials are susceptible to the TE mechanism.

2.3 *Methodology Used for DB-1*

An analysis was performed for each of the applicable DB-1 reactor vessel internals component items in accordance with the methodology outlined below:

- Identify appropriate inputs for the evaluation, [

 ]

- Utilize available information to determine if failure is likely or unlikely to occur

- Determine effect of failure on functionality

- Conclude whether components are expected to maintain their functionality through the period of extended operation

3.0 CRGT SPACER CASTINGS

This section summarizes the analysis performed of the DB-1 CRGT spacer castings to fulfill A/LAI 7 from Reference 1.

3.1 *Background*

MRP-227-A provides I&E guidelines for the various reactor vessel internals component items including the DB-1 CRGT spacer castings, which are considered a “Primary” item. The I&E guidelines specify applicability, effect and mechanism, expansion link, examination method/frequency, and examination coverage. The aging degradation mechanism, as described in Table 4-1 of Reference 1, is TE; the CRGT spacer castings are not susceptible to IE.

3.1.1 Description of CRGT Spacer Casting

The DB-1 plenum assembly (upper internals) contains 69 vertical CRGT assemblies that are welded to the plenum cover plate and bolted to the upper grid. The outer portion of the CRGT assembly consists of a pipe welded to the CRGT assembly flange at the bottom end. The inside of each CRGT assembly consists of a brazement subassembly with ten (10) parallel horizontal spacer castings to which are brazed twelve (12) perforated vertical rod guide tubes and four (4) pairs of vertical rod tube guide sectors (also called “C” tubes). Therefore, there are a total of 690 CRGT spacer castings within the DB-1 reactor vessel internals. The CRGT spacer castings are manufactured to the requirements of American Society for Testing and Materials (ASTM) Standard Specification A351-65, Grade CF3M.

3.1.2 Function and Consequence of Failure of the CRGT Spacer Casting

The CRGT assemblies provide control rod assembly (CRA) guidance and protect the CRA from the effects of reactor coolant cross-flow. The outer pipe portion of the CRGT assembly provides the structural connection between the upper grid and the plenum cover in the upper internals. There are openings in the lower region of the CRGT pipe to allow some of the fluid entering the CRGT assembly from the core to exit to the plenum region. The balance of the CRGT assembly has [ ]

The function of the CRGT spacer castings is to provide structural support and alignment to the [ ] vertical rod guide tubes and [ ] vertical rod guide sectors within each CRGT assembly. [ ] The CRA is guided by the brazement subassembly over the entire range of the vertical withdrawal path. In addition, the rod guide tube limits reactor coolant cross-flow on the control rods to limit flow-induced vibration (FIV). The CRGT spacer castings do not have a core support function; however, they do have a safety function relative to control rod alignment, insertion and reactivity issues. Degradation of the CRGT spacer castings could result in degradation in the unit shutdown capability by hindering the insertion of the control rods into the core in the normal anticipated time.

3.1.3 Operating Experience

Appendix A of Reference 1 indicates no cracking has been reported in the PWR reactor vessel internals due solely to TE for CASS materials. Additionally, no known failures of CASS materials due to embrittlement have been reported in the industry. Other B&W units have performed visual examinations (VT-3) of the CRGT spacer castings in accordance with the MRP-227-A guidance with no relevant indications reported.

3.2 *Evaluation Inputs*

This subsection describe the quantitative inputs for the CRGT spacer casting evaluation, such as flaw size, degraded material properties, and stresses.

3.2.1 Flaw Size

As indicated by the MRP-227-A process, the CRGT spacer castings are not screened as potentially susceptible to service induced flaws (i.e., stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), or fatigue). Therefore, the following section focuses on the potential for flaws in the 'as-built' condition from the manufacturing process. The non-destructive evaluation (NDE) methods used to examine the component items prior to their in-service time at DB-1 are summarized below.

Review of the available certified material test reports (CMTRs) for the CRGT spacer castings indicated [

] However, it is reasonable to assume based on the NDE information available on the CMTRs that the CRGT spacer castings used inservice at DB-1 had [ ] and therefore

[

]

3.2.2 Degraded Material Properties

[ ] of the CRGT spacer castings at DB-1 exceed the screening criteria for TE; therefore the DB-1 CRGT spacer castings as a group are considered potentially susceptible to TE, even though some CRGT spacer casting heats are not considered susceptible to TE. For the CRGT spacer castings potentially susceptible to TE, the time to reach saturation in the reduction of mechanical and impact properties was investigated. For the susceptible CRGT spacer casting material, [

]

The expected fluence of the CRGT spacer castings is expected to be less than or equal to [ ] using the light water reactor (LWR) conversion factor from page 6 of Reference 5 for E > 1.0 MeV, which is less than the MRP-175 (Reference 6) screening criterion (≥ 1 dpa or 6.7×10^{20} n/cm^2 [E > 1.0 MeV]) for IE of CASS materials. The first or lowest CRGT spacer casting elevation [ ] due to its proximity to the core. Therefore, the CRGT spacer castings are not considered susceptible to and do not need to be further evaluated for IE or the synergistic effects of TE and IE.

3.2.3 Distortion Evaluation

An evaluation was performed to determine the amount of distortion allowed and still permit the control rod spider to freely pass through the brazement subassembly. The loading conditions used to determine the operational displacement and distortion consisted of deadweight, Loss of Coolant Accident (LOCA), Safe Shutdown Earthquake (SSE), and flow loads. The conclusion of the analysis was [

]

3.3 *Evaluation*

The results of the methodology utilized are organized into several conclusions as discussed in the following sections.

3.3.1 Failure is Unlikely

The CRGT spacer casting material is expected to be [

] No known failures of CASS materials due to embrittlement have been reported in the industry. Additionally, other B&W 177-FA plants have performed 100% visual VT-3 examination of the CRGT spacer castings accessible surfaces at each of the four (4) CRGT spacer casting screw locations (per the guidance in Reference 1) that revealed no recordable indications. This is confirmation of the [

]

Due to the [ ] being present in the CASS material [

]

Therefore, since [ ] are expected in the majority of the CRGT spacer casting volume during normal operation, [ ] for both CASS materials in the PWR reactor vessel internals as well as the CRGT spacer castings themselves, and [ ] failure of the CRGT spacer casting material during the period of extended operation is unlikely.

3.3.2 Effect of Failure on Functionality

[ ] However, the postulated occurrence of [ ] is not considered a credible scenario as described in Section 3.2.3 of this report.

The reported stress distribution reinforces the premise that [ ] is not expected. Not only must a [ ] but it must [ ] Analysis also leads to the conclusion that a [ ] In particular the stresses in the [ ]

Therefore there is no rational reason why [ ] This data directly reinforces the premise that the [ ]

Drop-time testing of the CRAs is performed at the beginning of each cycle per the Technical Specification surveillance requirement. Historically, the rod drop-times are somewhat uniform and easily meet the acceptance criteria. If an unusual drop-time is encountered, the utility normally investigates the possible cause. To date, slow trip times have always been attributed to unusual fuel bow or issues with the control rod drive mechanism. Even though the Technical Specification requires that action be taken to evaluate shutdown margin if a control rod is believed to be incapable of being tripped, the safety analysis already considers that the maximum worth rod may not trip on demand. Thus, the safety analysis anticipated that control rod action may not be perfect even before the aging management issue was identified.

The CRGT spacer castings are typically not examined as a part of the inservice inspection (ISI) program, and the only operational indications of a cracked spacer casting would be unsatisfactory rod drop tests or indication of loose parts. However, the nature of a spacer casting failure that would result in either of these indications is not a credible scenario. As noted above, other B&W 177-FA units have performed VT-3 examinations of the CRGT spacer castings per Reference 1 with no recordable indications reported. The redundant features [

] The rod guide tubes and rod guide sectors have [

] In addition, [

] due to the CRGT spacer casting geometry and [ ] and are not a credible risk.

3.4 *Conclusions*

CASS materials are known to be potentially susceptible due to TE after exposure at PWR reactor vessel internals temperatures for long periods of time, especially those containing higher levels of ferrite and molybdenum. Studies show that saturation of reduction of room temperature impact energy, and correspondingly fracture toughness, [

] No known failures of CASS materials due to TE have been reported in the industry.

Additionally, other B&W 177-FA plants have performed 100% visual VT-3 examination of the CRGT spacer castings accessible surfaces at each of the four (4) CRGT spacer casting screw locations per the guidance in Reference 1. These examinations revealed no recordable indications. This supports the [

]

[ ] performed on the original castings and [

] The stress analysis of the spacer castings [

]

An analysis of the brazement subassembly, including [

] the monitoring of control rod drop times is a normal surveillance requirement. It is currently a requirement that abnormal control rod drop times must be investigated. [

]

The function of the brazement subassembly, of which the CRGT spacer castings are a part, is currently monitored by periodically verifying the control rod movement and control rod drop time. In addition, the aging management plan has implemented a new visual examination to verify that critical regions of the CRGT spacer casting have not failed due to random original defects. The combination of the original licensing basis surveillance, that has not been altered, and the additional visual verification of intact CGRT spacer castings demonstrate that, in the unlikely event of degradation, it could be detected.

Based on the discussion above, it is concluded that the CRGT spacer castings will maintain functionality during the period of extended operation.

4.0 INCORE MONITORING INSTRUMENTATION (IMI) GUIDE TUBE SPIDER CASTINGS

This section summarizes the analysis performed of the DB-1 IMI guide tube spider castings to fulfill A/LAI 7 from Reference 1.

4.1 *Background*

Reference 1 provides I&E guidelines for the various reactor vessel internals component items including the IMI guide tube spider castings, which are considered a "Primary" item. The I&E guidelines specify applicability, effect and mechanism, expansion link, examination method/frequency, and examination coverage. The aging degradation mechanism, as described in Table 4-1 of Reference 1, is IE; the IMI guide tube spider castings are not susceptible to TE.

4.1.1 Description of the IMI Guide Tube Spider Casting

The lower grid assembly provides alignment and support for the fuel assemblies, supports the core barrel assembly and flow distributor, and aligns the IMI guide tubes with the fuel assembly instrument tubes. The IMI guide tube spiders are part of the reactor vessel lower internals and their function is to provide lateral restraint for the upper end of the IMI guide tubes [ ] The IMI guide tube spider resembles a four (4) eared butterfly nut and each of the 52 IMI guide tube spiders is [ ] The outer edges of each of the four (4) spider legs is fillet welded to the walls of the lower grid rib section (two fillet welds per spider leg). The relatively [ ] while providing [ ] to accommodate the axial expansion of the guide tube. The IMI guide tube spiders are fabricated from ASTM A351-65, Grade CF8 material.

4.1.2 Function and Consequence of Failure of the IMI Guide Tube Spider Casting

The function of the IMI guide tube spider casting is to provide lateral restraint for the IMI guide tube and the function of the spider fillet welds is to hold the spider casting in place. The IMI guide tube provides the continuous protected guide path for the in-core instrumentation from their entry into the reactor vessel through the reactor vessel instrumentation nozzles to the entrance into the fuel assembly instrument guide tube.

[

] Loss of function of the in-core instrument guide path would be a sufficient misalignment at the fuel assembly instrument tube entrance to prohibit entry of the in-core instrument. In addition, failure of the guide path could result in wear of the in-core instrument sheath due to FIV and therefore would be considered a loss of function.

4.1.3 Operating Experience

Four B&W 177-FA units have performed VT-3 examinations on the IMI guide tube spiders and their welds to the lower grid rib sections with 100% coverage in accordance with the MRP-227-A guidance. According to Reference 7, no relevant indications were noted during examinations at two of the four B&W 177-FA units. At the third B&W unit, one IMI guide tube spider casting had a linear indication coming from the top of the casting at the weld toe and going downward into the casting material, as noted in Reference 8. At the fourth B&W unit, relevant indications were observed at three spider legs adjacent to the weld, but in no case did more than one spider leg have an indication in a particular spider casting (i.e., one deficient spider leg at three separate spiders), as noted in Reference 9. This required Expansion to the lower grid fuel assembly support pad items, however no relevant indications were observed during these examinations.

4.2 *Evaluation Inputs*

This subsection will describe the quantitative inputs for the evaluation, such as flaw size, degraded material properties, and stresses.

4.2.1 Flaw Size

As indicated by the MRP-227-A process, the IMI guide tube spider castings are not screened as potentially susceptible to service induced flaws (i.e., SCC, IASCC, or fatigue). Therefore, the following section will focus on the potential for flaws in the ‘as-built’ condition from the manufacturing process. The NDE [

]

[ ] IMI guide tube spider castings prior to their service in the reactor internals at DB-1. The CMTRs indicate [

] Specific information regarding [

] However, it is reasonable [

]

4.2.2 Degraded Material Properties

[

]

The IMI guide tube spider castings are consider potentially susceptible to IE based on the expected fluence exposure exceeding the screening criteria (≥ 1 dpa or 6.7×10^{20} n/cm^2 [E > 1.0 MeV]) for CASS materials. The IMI guide tube spider castings are expected to experience [

] During the period of extended operation, the increase in fluence is not expected to cause a significant reduction in the fracture toughness of the IMI guide tube spider castings [

]

However, at the [

] Additionally, an examination of the top surfaces of the IMI guide tube spider castings and the axial areas near the welds adjacent to the lower grid rib section, (as required per Table 4-1 of Reference 1) [

] are a reasonable leading indicator for the balance of the spider castings through the period of extended operation.

4.2.3 Flow-Induced Vibration (FIV) Analysis

A FIV analysis was prepared for the IMI guide tube spiders installed in several B&W units including DB-1. Five (5) configurations of the spider and its associated welds were considered.

[ ]

The results of the analysis show that the [ ]

4.2.4 Stress Analysis

The stress analysis prepared for the IMI guide tube spiders installed at DB-1 considered five (5) configurations:

[ ]

If the [ ] stress distribution is calculated for the IMI spider casting

[

]

4.3 *Evaluation*

The results of the methodology utilized are organized into several conclusions as discussed in the following subsections.

4.3.1 Failure is Unlikely

The expected aging effect for the IMI guide tube spider castings based on an aging mechanism of IE is cracking including fractured or missing spider arms. [

]

As described previously, [ ] prior to their service in the reactor vessel internals at DB-1. [ ] However, the likelihood of [ ] Additionally, [ ] is not expected to be a concern for the IMI guide tube spider castings. [ ]

Therefore, failure of the IMI guide tube spider castings at DB-1 during the period of extended operation is unlikely and not expected to occur.

4.3.2 Effect of Failure on Functionality

The function of the IMI guide tube spider castings is to [ ] Failure of the IMI guide tube spider castings is not expected; however, significant degradation of the in-core instrument guide path (potentially due to contributions from degraded spider castings) could result in misalignment at the fuel assembly instrument tube entrance, prohibiting entry of the IMI itself or wear of the IMI sheath due to FIV.

While the spider castings provide [ ] to the IMI guide tubes, [

] In addition to the IMI guide tube

spider castings, there are [

]

[ ] is not expected to affect the function of the spider. This expectation is due to a few reasons: [

]

Based on the discussion above, the failure of [ ] is not expected to impact the function of the IMI guide tube spider casting.

4.4 *Conclusions*

Due to the [ ] due to IE expected during the period of extended operation [ ] the lack of observed failures, and [ ] the IMI guide tube spider casting material is not expected to fail during the period of extended operation. [

]

Therefore, the IMI guide tube spider castings are expected to perform their function for the period of extended operation.

5.0 VENT VALVE RETAINING RINGS

This section summarizes the analysis performed of the DB-1 vent valve retaining rings to fulfill A/LAI 7 from Reference 1.

5.1 *Background*

MRP-227-A provides I&E guidelines for the various reactor vessel internals component items including the vent valve retaining rings, which are considered a “Primary” item. The I&E guidelines specify applicability, effect and mechanism, expansion link, examination method/frequency, and examination coverage. The aging degradation mechanism, as described in Table 4-1 of Reference 1, is TE; the vent valve retaining rings are not susceptible to IE.

5.1.1 Description of the Vent Valve Retaining Rings

DB-1 has four (4) vent valves installed in the core support shield (CSS) cylinder. Each vent valve is mounted in a vent valve mounting ring (also called the vent valve nozzle), which is welded into the CSS cylinder. For all normal operating conditions, the vent valve is closed but in the event of a pipe rupture in the reactor vessel inlet pipe, the valve will open to permit steam generated in the core to flow directly to the break, and will permit the core to be flooded and adequately cooled after emergency core coolant has been supplied to the reactor vessel. Each vent valve assembly includes two (2) retaining rings with varying thicknesses for a total of eight (8) retaining rings within the DB-1 reactor vessel internals. Each retaining ring has integral threaded bosses at both ends to accept jackscrews. They are fabricated from Aerospace Material Specification (AMS) 5658, Type 15-5 PH stainless steel in the H1100 temper condition.

5.1.2 Function and Consequence of Failure of the Vent Valve Retaining Rings

The function of the vent valve retaining rings is to retain the vent valve body in the vent valve mounting ring. The consequence of failure of a retaining ring or portion of a retaining ring is loss of support function for the valve body [ ] Failure of a retaining ring or portion of a retaining ring [ ]

5.1.3 Operating Experience

Several B&W units have performed visual examinations (VT-3) of the vent valve retaining rings and no relevant indications were noted on the inspected retaining rings; however, there are several known instances of more susceptible types of PH stainless steel materials (e.g., Type 17-4 PH) in other component systems failing.

5.2 *Evaluation Inputs*

This subsection describes the quantitative inputs for the evaluation, including inputs such as flaw size, degraded material properties, and stresses.

5.2.1 Flaw Size

As indicated by the MRP-227-A process, the vent valve retaining rings are not screened as potentially susceptible to service-induced flaws (i.e., SCC, IASCC, or fatigue). Therefore, the following section will focus on the potential for [ ] The NDE methods [ ]

The CMTRs for the DB-1 vent valve retaining rings indicate [

] Specific information

regarding any actual observed flaw sizes [

]

5.2.2 Degraded Material Properties

[

]

Measured fracture toughness of 47.3 ksi√in (52 MPa√m) has been reported for Type 17-4 PH stainless steel material tempered at 1112°F (600°C) for 4 hours in the aged condition (752°F [400°C] at 5,000 hours). The thermally-aged vent valve retaining ring material is [

] is a reasonable lower-bound saturated fracture toughness value for the thermally-aged vent valve retaining ring material. The expected fracture toughness of the vent valve retaining ring material is expected to be [

] is a reasonable lower bound due to [

] In comparison, this fracture toughness is [

] after long-term irradiation exposure.

5.2.3 Stresses

A stress analysis was prepared to provide the [

] in the majority of the component items; but the stresses calculated [

] However, such an analysis is not necessary for the evaluation because (as discussed in Section 5.3.1) the available information indicates [

]

5.3 *Evaluation*

The results of the methodology utilized are organized into several conclusions as discussed in the following subsections.

5.3.1 Failure is Unlikely

For all eight (8) of the vent valve retaining rings at DB-1, [

] to the period of extended operation. Additionally, as of the publication of Reference 1, there is no known cracking of vent valve retaining rings. This can be seen as [

]

Furthermore, the expected lower bound fracture toughness value [

] As of the publication of Reference 1, there are no known cracking of austenitic stainless steel component items in PWR reactor vessel internals due solely to IE. Due to the improbability of [

]

Based on the discussion above, failure of the vent valve retaining rings is not expected during the period of extended operation.

5.3.2 Effect of Failure on Functionality

While failure of the vent valve retaining rings is not expected, this subsection describes the outcome, should a vent valve retaining ring fail, on the two (2) functions of the vent valve retaining ring.

One (1) of the functions of the vent valves is to relieve pressure in the interior of the CSA during a cold leg large break LOCA. The retaining rings, if damaged due to TE (cracked, fractured material, surface irregularities, etc.), could eventually lead to [ ] If this happened, this would [ ] Therefore, it is likely that degradation of the vent valve retaining ring material due to TE will not affect the function of the vent valve during a cold leg large break LOCA.

An additional function of the vent valves is to [ ] The event would be detected [ ] For initial operation of the early B&W plants, the NRC had imposed a [ ] for the possibility of one (1) vent valve being in the failed open position. This penalty was removed as operating experience demonstrated that a failed open vent valve was highly improbable as confirmed by the refueling exercise and visual surveillance program.

5.4 *Conclusion*

Due to expected [ ]

In the unlikely event that failure of the vent valve retaining rings does occur, it is not expected to impair the function of the vent valve to relieve pressure in the interior of the CSA during a cold leg large break LOCA. [ ] should failure of the vent valve retaining rings cause [ ] it will be detectable [ ]

Therefore, the vent valve retaining rings are expected to perform their function for the period of extended operation and in the unlikely event of failure, the primary vent valve functions is not expected to be impaired and the secondary vent valve function that could possibly be impaired would be detectable.

6.0 SELECT ORIGINAL VENT VALVE LOCKING DEVICE PARTS

The vent valve locking devices were not included in the MRP-227-A I&E guidelines; however they were identified as “Existing Programs” in the response to A/LAI 2 from Reference 1. Since the original vent valve locking device [ ] are fabricated from martensitic stainless steel and require aging management, these component items are within the scope of A/LAI 7 from Reference 1.

This section summarizes the assessment for the select original vent valve locking device parts: [ ] to fulfill A/LAI 7 from Reference 1.

6.1 *Background*

DB-1 has four (4) vent valves installed in the CSS cylinder. Each vent valve is mounted in a vent valve mounting ring (also called vent valve nozzle) which is welded into the CSS cylinder. Each vent valve consists of a hinged disc, a valve body with sealing surfaces, a split-retaining ring and fasteners (that retain and seal the perimeter of the valve assembly), and an alignment device (to maintain the correct orientation). For all normal operating conditions, the vent valve is closed but in the event of a pipe rupture in the reactor vessel inlet pipe, the valve will open to permit steam generated in the core to flow directly to the break, and will permit the core to be flooded and adequately cooled after emergency core coolant has been supplied to the reactor vessel.

In the original vent valve locking devices, [ ] The spring holds the pressure plate up so that it engages the [ ] and prevents the jackscrew from turning. Other than the “U” cover and cap screws, the component items of the locking device are [ ] The function of the vent valve jackscrew is to maintain the retaining rings position and to prevent the retaining rings from backing out of the mounting ring groove in the CSS nozzle. The function of the top and bottom retaining rings is to retain the vent valve body in the CSS mounting ring.

6.2 *Evaluation*

The following section contains a discussion of the applicable degradation mechanism and effect of failure on functionality for the select original vent valve locking device parts [ ] fabricated from martensitic stainless steel (Type 431).

6.2.1 Degradation Mechanism

The vent valve assembly has an estimated fluence [ ] the criteria set forth for IE in MRP-175. Therefore, the original vent valve locking device [ ]

6.2.2 Function and Consequence of Failure of the Vent Valve Locking Devices

The primary function of the vent valve assembly is to support core cooling during a cold leg large break LOCA. [ ] The purpose of this section is to discuss how the [ ] could affect the primary or [ ] of the vent valve assembly.

The function of the vent valve locking device [ ] is to prevent the jackscrew from turning during reactor operation, and the function of the vent valve jackscrew is to maintain the vent valve retaining rings in position and to prevent the retaining rings from backing out of the mounting ring groove in the CSS nozzle. The failure of the vent valve [ ] locking device could allow the jackscrew to turn, thus causing vent valve retaining rings to back out of the mounting ring groove in the CSS nozzle, and this could affect the bypass flow function.

One (1) of the functions of the vent valves is to relieve pressure in the interior of the CSA during a cold leg large break LOCA. Failure of both locking devices on one (1) vent valve assembly could cause the backing out of its retaining rings from the mounting ring groove which could eventually lead to [ ] If this happened, this [ ]

An additional function of the vent valves is to [ ] As discussed in Section 5.0, the event would be [ ]

If [ ] the associated jackscrew [ ] The [ ] turning of a single jackscrew on a vent valve assembly could result in a [ ] This scenario of [ ] was considered in an evaluation discussed as part of internal correspondence at B&W to [ ] in the late 1970s at other B&W units. This correspondence indicated that the [ ] and that such [ ] would not affect the function [ ]

6.3 *Conclusions*

The potential failure of the martensitic stainless steel [ ] vent valve locking devices evaluated in this section could result in the failure of the vent valve jackscrew locking device, [ ] jackscrew locking devices in a vent valve assembly failed, the primary function of the vent valve assembly (to support core cooling during a cold leg large break LOCA) would be maintained. If [ ] jackscrew locking device in a vent valve assembly fails, [ ] If [ ] jackscrew locking devices in a vent valve assembly fail, [ ] be [ ]

It is noted that the “Existing Program” in place for the original vent valve locking device aging management at DB-1 is not identified as such in Table 4-7 of Reference 1. Framatome (formerly AREVA) provided interim guidance for the examination of the vent valve locking devices for the B&W units in Reference 10, as an NEI 03-08 “Needed” recommendation. The subsequent revision to MRP-227 (MRP-227-Revision 1-A, Reference 11) incorporates these component items as Primary items.

7.0 OVERALL CONCLUSIONS

A/LAI 7 is applicable to CRGT spacer castings, IMI guide tube spider castings, vent valve retaining rings, and the select original vent valve locking device parts for DB-1.

Based on the evaluation of the CRGT spacer castings, IMI guide tube spider castings, vent valve retaining rings summarized in the sections above, failure during the period of extended operation was found to be improbable for each of these component items. In the unlikely event of a failure occurring, the intended function of the component items is expected to be maintained or the failure will be detectable.

For the applicable original vent valve locking device parts, the evaluation concluded that failure of one or both of the original locking devices on a vent valve assembly would not impact the primary function of the vent valve assembly (to support core cooling during a cold leg large break LOCA). If [ ] original jackscrew locking device in a vent valve assembly fails, [ ] locking devices on a vent valve assembly could impact [ ] but this event will be detectable and there would be no safety consequences.

8.0 REFERENCES

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