WCAP-18012-NP
Revision 0

July 2015

# PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at J.M. Farley Nuclear Plant Unit 2

Westinghouse

**WCAP-18012-NP**
**Revision 0**

# PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at J.M. Farley Nuclear Plant Unit 2

**Bradley T. Carpenter***
Reactor Internals Aging Management

**Mary Ann T. Walsh***
Reactor Internals Aging Management

**Karli N. Szweda***
Reactor Internals Aging Management

**July 2015**

Approved: Patricia C. Paesano*, Manager
Reactor Internals Aging Management

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

# TABLE OF CONTENTS

LIST OF TABLES ..... iv

LIST OF FIGURES ..... v

LIST OF ACRONYMS ..... vi

1 PURPOSE ..... 1-1
2 BACKGROUND ..... 2-1
3 PWR VESSEL INTERNALS PROGRAM OWNER ..... 3-1
3.1 SNC – EXECUTIVE ..... 3-1
3.2 SNC – CORPORATE ..... 3-1
3.3 SNC – FNP SITE ..... 3-3
3.4 PWR PRIMARY SYSTEM INTEGRITY PROGRAM TECHNICAL TEAM ..... 3-4
4 DESCRIPTION OF THE FARLEY NUCLEAR PLANT UNIT 2 REACTOR INTERNALS AGING MANAGEMENT PROGRAMS AND INDUSTRY PROGRAMS ..... 4-1
4.1 EXISTING FARLEY UNIT 2 PROGRAMS ..... 4-4
4.1.1 Water Chemistry Control Program ..... 4-4
4.1.2 Inservice Inspection Program ..... 4-4
4.2 SUPPORTING FARLEY UNIT 2 PROGRAMS AND AGING MANAGEMENT SUPPORTIVE PLANT ENHANCEMENTS ..... 4-5
4.2.1 Reactor Internals Aging Management Review Process ..... 4-5
4.2.2 Reactor Vessel Internals Program ..... 4-5
4.2.3 Flux Detector Thimble Inspection Program ..... 4-5
4.2.4 Control Rod Guide Tube Support Pin Replacement Project ..... 4-6
4.2.5 Power Uprating Project ..... 4-6
4.3 INDUSTRY PROGRAMS ..... 4-6
4.3.1 WCAP-14577, Aging Management for Reactor Internals ..... 4-6
4.3.2 MRP-227-A, Reactor Internals Inspection and Evaluation Guidelines ..... 4-7
4.3.3 Ongoing Industry Programs ..... 4-10
4.4 SUMMARY ..... 4-10
5 FARLEY NUCLEAR PLANT REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES ..... 5-1
5.1 GALL REVISION 2 ELEMENT 1: SCOPE OF PROGRAM ..... 5-1
5.2 GALL REVISION 2 ELEMENT 2: PREVENTATIVE ACTIONS ..... 5-3
5.3 GALL REVISION 2 ELEMENT 3: PARAMETERS MONITORED OR INSPECTED5-4
5.4 GALL REVISION 2 ELEMENT 4: DETECTION OF AGING EFFECTS ..... 5-5
5.5 GALL REVISION 2 ELEMENT 5: MONITORING AND TRENDING ..... 5-10
5.6 GALL REVISION 2 ELEMENT 6: ACCEPTANCE CRITERIA ..... 5-11
5.7 GALL REVISION 2 ELEMENT 7: CORRECTIVE ACTIONS ..... 5-13
5.8 GALL REVISION 2 ELEMENT 8: CONFIRMATION PROCESS ..... 5-14
5.9 GALL REVISION 2 ELEMENT 9: ADMINISTRATIVE CONTROLS ..... 5-14
5.10 GALL REVISION 2 ELEMENT 10: OPERATING EXPERIENCE ..... 5-15
6 DEMONSTRATION ..... 6-1
6.1 DEMONSTRATION OF TOPICAL REPORT CONDITIONS COMPLIANCE TO SE ON MRP-227, REVISION 0 ..... 6-3

6.2 DEMONSTRATION OF APPLICANT/LICENSEE ACTION ITEM COMPLIANCE TO SE ON MRP-227, REVISION 0 ........ 6-4
6.2.1 SE Applicant/Licensee Action Item 1: Applicability of FMECA and Functionality Analysis Assumptions ........ 6-4
6.2.2 SE Applicant/Licensee Action Item 2: PWR Vessel Internal Components within the Scope of License Renewal ........ 6-6
6.2.3 SE Applicant/Licensee Action Item 3: Evaluation of the Adequacy of Plant-Specific Existing Programs ........ 6-7
6.2.4 SE Applicant/Licensee Action Item 4: B&W Core Support Structure Upper Flange Stress Relief ........ 6-8
6.2.5 SE Applicant/Licensee Action Item 5: Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components ........ 6-9
6.2.6 SE Applicant/Licensee Action Item 6: Evaluation of Inaccessible B&W Components ........ 6-9
6.2.7 SE Applicant/Licensee Action Item 7: Plant-Specific Evaluation of CASS Materials ........ 6-10
6.2.8 SE Applicant/Licensee Action Item 8: Submittal of Information for Staff Review and Approval ........ 6-13
7 PROGRAM ENHANCEMENT AND IMPLEMENTATION SCHEDULE ........ 7-1
8 IMPLEMENTING DOCUMENTS ........ 8-1
9 REFERENCES ........ 9-1
APPENDIX A ILLUSTRATIONS ........ A-1
APPENDIX B FARLEY UNIT 2 LICENSE RENEWAL AGING MANAGEMENT REVIEW SUMMARY TABLE ........ B-1
APPENDIX C MRP-227-A AUGMENTED INSPECTIONS ........ C-1

## LIST OF TABLES

Table 6-1 Topical Report Condition Compliance to SE on MRP-227 ....... 6-3

Table 6-2 Summary of Joseph M. Farley Unit 2 CASS Components and Their Susceptibility to TE ....... 6-12

Table 7-1 Aging Management Program Enhancement and Inspection Implementation Summary ....... 7-1

Table B-1 LRA Aging Management Review Summary Table 3.1.2-2 Farley Nuclear Plant LRA ....... B-1

Table C-1 MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals ....... C-1

Table C-2 MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals ....... C-7

Table C-3 MRP-227-A Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals ....... C-10

Table C-4 MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals ....... C-12

# LIST OF FIGURES

Figure A-1 Illustration of Typical Westinghouse Internals Assembly ............ A-1
Figure A-2 Typical Westinghouse Control Rod Guide Card ............ A-2
Figure A-3 Typical Lower Section of Control Rod Guide Tube Assembly ............ A-3
Figure A-4 Major Core Barrel Welds ............ A-4
Figure A-5 Bolting Systems used in Westinghouse Core Baffles ............ A-5
Figure A-6 Core Baffle/Barrel Structure ............ A-6
Figure A-7 Bolting in a Typical Westinghouse Baffle-Former Structure ............ A-7
Figure A-8 Vertical Displacement between the Baffle Plates and Bracket at the Bottom of the Baffle-Former-Barrel Assembly ............ A-8
Figure A-9 Schematic Cross-Sections of the Westinghouse Hold Down Springs ............ A-9
Figure A-10 Typical Thermal Shield Flexure ............ A-9
Figure A-11 Lower Core Support Structure ............ A-10
Figure A-12 Lower Core Support Structure – Core Support Plate Cross-Section ............ A-11
Figure A-13 Typical Core Support Column ............ A-11
Figure A-14 Examples of BMI Column Designs ............ A-12

## LIST OF ACRONYMS

| | |
|---|---|
| AMP | Aging Management Program Plan |
| AMR | Aging Management Review |
| ASME | American Society of Mechanical Engineers |
| B&PV | Boiler and Pressure Vessel |
| B&W | Babcock & Wilcox |
| BMI | bottom-mounted instrumentation |
| BWR | boiling water reactor |
| CASS | cast austenitic stainless steel |
| CE | Combustion Engineering |
| CFR | Code of Federal Regulations |
| CLB | current licensing basis |
| CRGT | control rod guide tube |
| ECP | Engineering Change Package |
| EFPY | effective full-power years |
| EPRI | Electric Power Research Institute |
| ET | electromagnetic testing (eddy current) |
| EVT | enhanced visual testing (a visual NDE method that includes EVT-1) |
| FMECA | failure modes, effects, and criticality analysis |
| FNP | Farley Nuclear Plant |
| GALL | Generic Aging Lessons Learned |
| I&E | Inspection and Evaluation |
| IASCC | irradiation-assisted stress corrosion cracking |
| IE | irradiation embrittlement |
| INPO | Institute of Nuclear Power Operations |
| ISI | inservice inspection |
| ISR | irradiation-enhanced stress relaxation |
| LRA | License Renewal Application |
| LRAAI | license renewal applicant action items |
| MRP | Materials Reliability Program |
| NDE | nondestructive examination |
| NEI | Nuclear Energy Institute |
| NOS | Nuclear Oversight Section |
| NRC | U.S. Nuclear Regulatory Commission |
| NSSS | nuclear steam supply system |
| OE | Operating Experience |
| OEM | Original Equipment Manufacturer |
| OER | Operating Experience Report |
| PH | precipitation-hardenable (heat treatment) |
| PWR | pressurized water reactor |
| PWROG | Pressurized Water Reactor Owners Group (formerly WOG) |
| PWSCC | primary water stress corrosion cracking |

## LIST OF ACRONYMS (cont.)

| | |
|---|---|
| QA | quality assurance |
| RCS | reactor coolant system |
| RIS | Regulatory Issue Summary |
| RO | refueling outage |
| RV | reactor vessel |
| RVI | reactor vessel internals |
| SCC | stress corrosion cracking |
| SE | Safety Evaluation |
| SER | Safety Evaluation Report |
| SNC | Southern Nuclear Company |
| SRP | Standard Review Plan |
| SS | stainless steel |
| TE | thermal embrittlement |
| UFSAR | Updated Final Safety Analysis Report |
| UT | ultrasonic testing (a volumetric NDE method) |
| VT | visual testing (a visual NDE method that includes VT-1 and VT-3) |
| WCAP | Westinghouse Commercial Atomic Power |
| WOG | Westinghouse Owners Group |
| XL | Extra-long Westinghouse Fuel |

Trademark Statement:

INCONEL® is a registered trademark of Special Metals, a Precision Castparts Corp. company.

# 1 PURPOSE

The purpose of this report is to document the Joseph M. Farley Nuclear Plant Unit 2, hereafter Farley Nuclear Plant (FNP) Unit 2, Reactor Vessel (RV) Internals (RVI) Aging Management Program Plan (AMP). The purpose of the AMP is to manage the effects of aging on reactor vessel internals through the license renewal period. FNP Unit 2 enters the license renewal period on March 31, 2021. This document provides a description of the program as it relates to the management of aging effects identified in various regulatory and updated industry-generated documents, in addition to the program documented in the Southern Nuclear Company (SNC) Procedure NMP-ES-029-GL02 [1] in support of license renewal program evaluations. This AMP is supported by existing FNP Unit 2 documents and procedures and, as needed by industry experience or directive in the future, will be updated or supported by additional documents to provide clear and concise direction for the effective management of aging degradation in reactor internals components. These actions provide assurance that operations at FNP Unit 2 will continue to be conducted in accordance with the current licensing basis (CLB) for the reactor vessel internals by fulfilling License Renewal commitments [2], U.S. Nuclear Regulatory Commission (NRC) expectations in the Regulatory Issue Summary (RIS) [3], American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI Inservice Inspection (ISI) programs [4] and industry requirements [5]. This AMP fully captures the intent of the additional industry guidance for reactor internals augmented inspections, based on the programs sponsored by U.S. utilities through the Electric Power Research Institute (EPRI)-managed Materials Reliability Program (MRP) and the Pressurized Water Reactor Owners Group (PWROG).

The main objectives for the FNP Unit 2 RVI AMP are to:

- Demonstrate that the effects of aging on the RVI will be adequately managed for the period of extended operation in accordance with the Code of Federal Regulations, Title 10, Part 54 (10 CFR 54) [6].
- Summarize the role of existing FNP Unit 2 AMPs in the RVI AMP.
- Define and implement the industry-defined (EPRI/MRP and PWROG) pressurized water reactor (PWR) RVI requirements and guidance for managing aging of reactor internals.
- Provide an inspection plan summary for the FNP Unit 2 reactor internals.

FNP Unit 2 License Renewal Commitment 6 [2], "FNP Reactor Vessel Internals Program" commits FNP Unit 2 to:

1. *Implement the FNP Reactor Vessel Internals Program prior to entering the period of extended operation;*
2. *Participate in industry initiatives intended to clarify the nature and intent of aging mechanisms potentially affecting the FNP reactor internals;*

3. *Incorporate the results of these initiatives into the RVI Program; and,*

4. *Submit an inspection plan for the RVI Program for NRC review and approval at least 24 months prior to entering the periods of extended operation for the FNP units.*

Augmented inspections, based on required program enhancements resulting from industry programs, will be implemented as part of the FNP Unit 2 ISI Engineering Program [4]. Corrective actions for augmented inspections will be developed using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI, or as determined independently by SNC, or in cooperation with the industry, to be equivalent or more rigorous than currently defined procedures.

This AMP for the FNP Unit 2 reactor internals demonstrates that the program adequately manages the effects of aging for reactor internals components and establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function through the FNP Unit 2 license renewal period of extended operation. This Westinghouse Commercial Atomic Power (WCAP) topical report supports the FNP Unit 2 License Renewal Commitment 6, which includes a submission to the NRC of an inspection plan for the Reactor Vessel Internals Program, as it would be implemented from the participation of FNP Unit 2 in industry initiatives 24 months prior to the augmented inspection. The implementation schedule for this commitment requires submission to the NRC no later than March 31, 2019.

The development and implementation of this program meets the guidelines provided in the RIS [3].

# 2 BACKGROUND

The management of aging degradation effects in reactor internals is required for nuclear plants considering or entering license renewal, as specified in the NRC Standard Review Plan (SRP) [7]. The U.S. nuclear power industry has been actively engaged in recent years in a significant effort to support the industry goal of responding to these requirements. Various programs have been underway within the industry over the past decade to develop guidelines for managing the effects of aging within PWR reactor internals. In 1997, the Westinghouse Owners Group (WOG)-issued WCAP-14577 [8], "License Renewal Evaluation: Aging Management for Reactor Internals," which was reissued as Revision 1-A in 2001 after receiving NRC Staff review and approval. Later, an effort was engaged by the EPRI MRP to address the PWR internals aging management issue for the three currently operating U.S. reactor designs – Westinghouse, Combustion Engineering (CE) and Babcock & Wilcox (B&W).

The MRP first established a framework and strategy for the aging management of PWR internals components using proven and familiar methods for inspection, monitoring, surveillance and communication. Based upon that framework and strategy, and on the accumulated industry research data, the following elements of an Aging Management Program were further developed [8, 9]:

- Screening criteria were developed, considering chemical composition, neutron fluence exposure, temperature history and representative stress levels, for determining the relative susceptibility of PWR internals components to each of eight postulated aging mechanisms (further discussed in Section 4 of this Program).

- PWR internals components were categorized, based on the screening criteria, into categories that ranged from:

  - Components for which the effects from the postulated aging mechanisms are insignificant
  - Components that are moderately susceptible to the aging effects
  - Components that are significantly susceptible to the aging effects

- Functionality assessments were performed based on representative plant designs of PWR internals components and assemblies of components using irradiated and aged material properties, to determine the effects of the degradation mechanisms on component functionality.

Aging management strategies were developed combining the results of functionality assessment with several contributing factors to determine the appropriate aging management methodology, baseline examination timing and the need and timing of subsequent inspections. Items considered included component accessibility, operating experience (OE), existing evaluations and prior examination results.

The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227-A [5], "PWR Internals Inspection and Evaluation Guidelines" (hereafter referred to as the "I&E Guidelines" or simply "MRP-227-A") provides the industry background, listing of reactor internals components requiring inspection, type of Nondestructive Examination (NDE) required for each component, timing for initial inspections, and criteria for evaluating inspection results. MRP-227-A provides a standardized approach to PWR internals aging management for each unique reactor design (Westinghouse, B&W and CE).

- MRP-228 [10], "Inspection Standard for PWR Internals" provides guidance on the qualification/demonstration of the NDE techniques and other criteria pertaining to the actual performance of the inspections.

The PWROG has also developed "Reactor Internals Acceptance Criteria Methodology and Data Requirements" for the MRP-227 inspections, where feasible [11]. This document has been submitted to the NRC for review and approval, and will be updated to incorporate changes from MRP-227-A [5]. Final reports are to be developed and available for industry use in support of planned license renewal inspection commitments. In some cases, individual plants will develop plant-specific acceptance criteria for some internals components where a generic approach is not practical.

The FNP Unit 2 reactor internals are integral with the reactor coolant system (RCS) of a Westinghouse three-loop nuclear steam supply system (NSSS), a typical illustration of which is provided in Figure A-1.

As described in NUREG-1825 [2], subsection 2.3.1.2.1, the FNP Unit 2 RVI consists of the lower core support structure, the upper core support structure and the in-core instrumentation support structures. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and Control Rod Drive Mechanism (CRDM), direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding and provide guides for the in-core instrumentation.

The lower core support structure consists of the core barrel, the core baffle assemblies, the lower core plate, the neutron shield panels, the lower core support forging, the secondary support assembly and associated support columns. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel, and is restrained at its lower end by a radial support system attached to the vessel wall. The upper core support structure consists of the upper support assembly, the upper core plate, support columns and control rod guide tube assemblies. The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the upper closure head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom head.

The reactor vessel internals functions include structural support, flow distribution and radiation shielding.

FNP Unit 2 was granted a license for extended operation by the NRC through the issuance of a Safety Evaluation Report (SER) in NUREG-1825 [2]. In the SER, the NRC concluded that the FNP Unit 2 License Renewal Application (LRA) adequately identified the RVI components that are within the scope of license renewal, as required by 10 CFR 54.4(a), and those subject to an Aging Management Review (AMR), as required by 10 CFR 54.21(a)(1) [6], and is therefore acceptable. A listing of the

FNP Unit 2 RVI components and subcomponents, already reviewed by the NRC in the SER that are subject to AMP requirements, is included in Tables B-1 and B-2.

In accordance with 10 CFR Part 54 [6], frequently referred to as the License Renewal Rule, FNP Unit 2 has developed a program to direct the performance of aging management reviews of mechanical structures and components [27]. The U.S. industry, as noted through the efforts of the MRP and PWROG, has further investigated the components and subcomponents that require aging management to support continued reliable function. As designated by the protocols of Nuclear Energy Institute (NEI) 03-08 [13], "Guidelines for the Management of Materials Issues," each plant will be required to use MRP-227-A and MRP-228 to develop and implement an AMP for reactor internals no later than three years after the initial industry issuance of MRP-227, Revision 0. MRP-227, Revision 0 was issued in December 2008, and plant AMPs must therefore be completed by December 2011, or sooner, if required by plant-specific License Renewal Commitments. According to [3], FNP Unit 2 is a Category B plant that is expected to submit their RVI AMP based on the guidance of MRP-227-A, consistent with their commitments. Per the SER [2], FNP Unit 2 has a commitment to submit their AMP for approval by the NRC no later than March 31, 2019.

The information contained in this AMP fully complies with the requirements and guidance of the referenced documents. The AMP will manage aging effects of the RVI so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

# 3 PWR VESSEL INTERNALS PROGRAM OWNER

The SNC "PWR Reactor Internals Program Strategic Plan" [1], which is a sub-tier document of the PWR Primary System Integrity Program [34], manages the effects of age-related degradation mechanisms of reactor vessel internals. The successful implementation and comprehensive long-term management of the FNP Unit 2 RVI AMP will require the integration of SNC, corporately and at Farley, and interaction with multiple industry organizations including, but not limited to, the ASME, MRP, NRC and PWROG. The responsibilities of the individual SNC corporate and Farley groups are provided in the following paragraphs. SNC will maintain cognizance of industry activities related to PWR internals inspection and aging management, and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

The overall responsibility for the scheduling and conducting of the PWR Primary System Integrity Program, including the RVI AMP, is the PWR Primary System Integrity Program owner in the Corporate Engineering Programs department.

Additional responsibilities and the appropriate responsible personnel, as described in [34], are discussed in the following subsections.

## 3.1 SNC – EXECUTIVE

- The overall responsibility for successful implementation of the PWR Primary System Integrity program (including reactor internals) resides with the Chief Nuclear Officer. As such, that individual establishes expectations for the implementation of the PRW Primary System Integrity Program.

- Approval of any deviations from mandatory or needed elements in industry documents that affect Farley.

## 3.2 SNC – CORPORATE

- The PWR Primary System Integrity Program owner resides in the Corporate Engineering Programs department and has overall responsibility for the development and maintenance of the PWR Primary System Integrity Program and for the following activities:

  - Development of implementing instructions and guidelines, as needed.

  - Development of a ten outage plan for reactor internals material management. This plan provides inspection and mitigation schedule for each unit over the next ten outages.

  - Providing technical expertise and oversight to the SNC fleet and/or serve as the subject matter expert for reactor internals

  - Participate in industry programs for reactor internals aging management and addressing Primary Water Stress Corrosion Cracking (PWSCC) issues.

- Participate in industry assessments; ensuring program is in alignment with industry guidance and implementing best practices.
- Utilize the technical team to drive best practices and provide oversight.
- Ensuring that industry best practices, industry operating experience from Institute of Nuclear Power Operations (INPO), EPRI, Owners groups and others (e.g., NSSS vendors and regulatory requirements) are communicated to the fleet and incorporated into the applicable program documentation.
- Review examination results, operating conditions, material properties and fabrication history for use in projecting future conditions and actions.
- Processing formal transmittals from the MRP.
- Identifying areas for standardization between the sites/projects with respect to the PWR Primary System Integrity Program.
- Documenting and processing deviations from mandatory or needed elements in industry documents.
- Promptly communicating with the industry issue program Chairman or Project Manager emergent issues that could have safety significance, or represent a new degradation type that may have an effect on industry guidance or the existing knowledge base.
- Participating in program self-assessments and benchmarking activities.
- Providing input to MRP industry inspection surveys.
- Drive susceptible components items towards long term resolution (asset management).
- Communicating program performance gaps to management.
- Periodically observe work activities and provide feedback to individuals and lessons learned to fleet.
- Updating Program Notebook.
- In addition to the above, provide oversight to the site programs, as needed.

• The Engineering Integrity Programs group responsibilities include:
  - Updating the reactors internals inspection plan.
  - Provide the results of augmented examinations which require reporting to the regulatory authority to Nuclear Licensing.

- The Fleet Chemistry group is responsible for sharing information obtained from industry participation with the appropriate SNC personnel on primary chemistry as well as chemical mitigation experience.

## 3.3 SNC – FNP SITE

Plant Management responsibilities include:

- Providing sufficient resources and oversight to the PWR Primary System Integrity Program to ensure PWR Primary System materials degradation do not compromise the integrity of the primary system pressure boundary.

- Ensuring that the responsibility for implementing the site elements of the Program has been clearly defined for each department and assigned to the trained and qualified personnel.

Site Engineering Programs Department responsibilities include:

- Designating a Site Program Owner and backup. Site Program Owners responsibilities are described in NMP-ES-009 [14].

- Provide updates to the Reactor Internals ten outage plan and budget estimates to support the overall program.

- Site Implementation of the PWR Primary System Integrity Program and MRP Guidelines.

- Coordination of engineering evaluations and disposition of indications discovered during vessel examinations.

- Maintaining knowledge of significant operating evolutions that might impact the integrity of the Reactor Pressure Vessel (RPV) upper and lower heads.

- Reviewing and responding to industry OE.

- Coordinating vendor support for any specialized equipment needed to complete the required inspections.

- Outage planning for RPV inspections.

- Develop and implement corrective action plans for PWR Primary System Integrity Program issues as requested by the FNP Engineering Programs Manager.

- Performing site assessments in accordance with NMP-GM-003 [15].

- Updating Program Notebook.
- Generate commitment notebooks in accordance with NMP-ES-063-GL02 [16], 1 year prior to the license renewal period.

## 3.4 PWR PRIMARY SYSTEM INTEGRITY PROGRAM TECHNICAL TEAM

- Support management on PWR Primary System Integrity issues, including recommending optimum technical and management practices for nuclear safety, plant availability and equipment reliability.
- Provide a technical forum for the integration of the various elements needed to implement an effective Program.
- Develop long range plans for assessment, inspection, mitigation and repairs, taking into account material condition, associated projections, industry insight and SNC strategic plans.
- Ownership of the strategic plan for inspection, mitigation, repair and chemistry initiatives.
- Ensure timely review of PWR Primary System Integrity issues by meeting at least once per year.
- Evaluate inspection, mitigation, repair and maintenance technologies with respect to the benefit of primary system integrity and cost.
- Establish strategic goals.
- Evaluate degradation mechanisms and operating conditions.
- Be knowledgeable of industry PWR Primary System Integrity issues and address potential impacts to FNP.
- Drive and adopt industry best practices.
- Provide oversight of implementation of Reactor Internals activities.

# 4 DESCRIPTION OF THE FARLEY NUCLEAR PLANT UNIT 2 REACTOR INTERNALS AGING MANAGEMENT PROGRAMS AND INDUSTRY PROGRAMS

The U.S. nuclear industry, through the combined efforts of utilities, vendors and independent consultants, has defined a generic guideline to assist utilities in developing reactor internals plant-specific aging management programs based on inspection and evaluation. The intent of this program is to ensure the long-term integrity and safe operation of the reactor internals components. SNC has developed this AMP in conformance with the 10 Generic Aging Lessons Learned (GALL) [17] attributes and MRP-227-A [5]. The LRA was based on Rev. 0 of the GALL [12], where this AMP is reconciled to Rev. 2 of the GALL [17].

This reactor internals AMP utilizes a combination of prevention, mitigation and condition monitoring. Where applicable, credit is taken for existing programs such as water chemistry [18] and inspections prescribed by the ASME Section XI Inservice Inspection Program [4], as well as mitigation projects such as support pin replacement [20] and baffle bolt replacement [42], combined with augmented inspections or evaluations as recommended by MRP-227-A.

Aging degradation mechanisms that impact internals have been identified and documented in FNP Unit 2 Aging Management Reviews [21]. The overall outcome of the reviews and the additional work performed by the industry, as summarized in MRP-227-A, is to provide appropriate augmented inspections for reactor internals components to provide early detection of the degradation mechanisms of concern. Therefore, this AMP is consistent with the existing FNP Unit 2 AMR methodology and the additional industry work summarized in MRP-227-A. All sources are consistent and address concerns about component degradation resulting from the following eight material aging degradation mechanisms identified as affecting reactor internals:

- Stress Corrosion Cracking (SCC)

  Stress corrosion cracking (SCC) refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment and metallurgical properties. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

- Irradiation-Assisted Stress Corrosion Cracking

  Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly irradiated components. The aging effect is cracking.

- Wear

  Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

- Fatigue

Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where the crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive.

Fatigue crack initiation and growth resistance are governed by a number of material, structural and environmental factors such as stress range, loading frequency, surface condition and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations such as notches, surface defects and structural discontinuities. The aging effect is cracking.

- Thermal Aging Embrittlement

Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS), martensitic stainless steel, and precipitation-hardenable (PH) stainless steel to high inservice temperatures, which can result in an increase in tensile strength, a decrease in ductility and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS, martensitic stainless steel, and PH stainless steel internals. CASS components have a duplex microstructure and are particularly susceptible to this mechanism. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

- Irradiation Embrittlement

Irradiation embrittlement (IE) is also referred to as neutron embrittlement. When exposed to high-energy neutrons, the mechanical properties of stainless steel and nickel-based alloys can be changed. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

- Void Swelling and Irradiation Growth

  Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation-produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5 percent by volume) has been correlated with extremely low fracture toughness values. Also included in this mechanism is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes within in-core instrumentation tubes that are fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may result in cracking under stress.

- Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep

  The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 1000 hours) at PWR internals temperatures.

  Creep (or more precisely, secondary creep) is a slow, time- and temperature-dependent, plastic deformation of materials that can occur at stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant stress. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or irradiation-enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress, and it can also be affected by void swelling should it occur. The aging effect is a loss of mechanical closure integrity (or preload) that can lead to unanticipated loading that, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

The FNP Unit 2 RVI AMP is focused on meeting the requirements of the 10 elements of an aging management program as described in NUREG-1801, GALL Report Section XI.M16A for PWR Vessel Internals. In the FNP Unit 2 RVI AMP, this is demonstrated through application of existing FNP AMR methodology that credits inspections prescribed by the ASME Section XI Inservice Inspection Program, existing FNP programs and additional augmented inspections based on MRP-227-A recommendations. A description of the applicable existing FNP programs and compliance with the elements of the GALL is contained in the following subsections.

## 4.1 EXISTING FARLEY UNIT 2 PROGRAMS

The overall strategy of SNC for managing aging in reactor internals components is supported by the following existing programs [23]:

- Water Chemistry Control Program
- Inservice Inspection Program

These are established programs that support the aging management of RCS components in addition to the RVI components. Although affiliated with and supporting the RVI AMP, they will be managed under the existing programs.

Brief descriptions of the programs are included in the following subsections.

### 4.1.1 Water Chemistry Control Program

The FNP Water Chemistry Program [18] will manage loss of material and cracking within system components and structures, thereby ensuring continued structural integrity, reliability and availability. The program includes monitoring of detrimental species and addition of chemical additives. The program utilizes the EPRI water chemistry guidelines [25] in establishing chemistry control procedures for FNP. These documents are updated as necessary to reflect improved guidance and industry experience. Prior to adopting a later revision, SNC evaluates the acceptability of implementing requirements.

With one exception, the FNP closed cycle cooling water monitoring and chemistry control methods are consistent with those described in NUREG-1801 [17]. The closed cycle cooling water program described in NUREG-1801 [17] places emphasis on thermal-hydraulic performance testing for pumps and heat exchangers. The FNP program deals with performance monitoring as outlined in Section 5 of EPRI TR-107396 [32] regarding chemistry monitoring.

### 4.1.2 Inservice Inspection Program

The FNP Unit 2 ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD, Program [4] is in accordance with ASME Section XI 2001 Edition with the 2003 Addenda [22]. The FNP ISI Program is implemented in accordance with 10 CFR 50.55a, and is subject to its limitations and modifications. The program manages loss of material, cracking, changes in material properties, loss of preload/stress relaxation, loss of fracture toughness and change in strength in concrete. The program inspections include periodic visual, surface and/or volumetric examinations and leakage tests of Class 1, 2 and 3 pressure-retaining components and their integral attachments, including welds, pump casings, valve bodies and pressure-retaining bolting. The program is updated as required by 10 CFR 50.55a.

The FNP Unit 2 ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD, Program is consistent with the collection of acceptable ASME Section XI subprograms described in NUREG-1801 [17].

## 4.2 SUPPORTING FARLEY UNIT 2 PROGRAMS AND AGING MANAGEMENT SUPPORTIVE PLANT ENHANCEMENTS

### 4.2.1 Reactor Internals Aging Management Review Process

A comprehensive review of aging management of reactor internals was performed according to the requirements of the License Renewal Rule [6] as directed by the Plant Farley commodity review procedure [27]. The Plant Farley License Renewal Commodity Group Review Document [21] documents the results of the aging management review performed in support of FNP Unit 2 license renewal for reactor internals. The FNP Unit 2 LRA was approved by the NRC in NUREG-1825 [2]. RVI components specifically noted as requiring aging management, as identified in the NUREG, are summarized in Table B-1 of this AMP.

The AMR supported the LRA as follows:

- Identified applicable aging effects requiring management
- Associated aging management programs to manage those aging effects
- Identified enhancements or modifications to existing programs, new aging management programs or any other actions required to support the conclusions reached in the review

Aging management reviews were performed for each FNP Unit 2 system that contained long-lived, passive components requiring aging management review, in accordance with the Plant Farley commodity review procedure [27]. This review is not repeated here, but the results are fully incorporated into the FNP Unit 2 RVI AMP.

### 4.2.2 Reactor Vessel Internals Program

The FNP Reactor Vessel Internals Program [1] will be implemented prior to entering the period of extended operation to provide an integrated inspection program that addresses the reactor internals. The program will be used during the period of extended operation to manage the effects of crack initiation and growth due to IASCC; loss of fracture toughness due to irradiation embrittlement, thermal embrittlement (TE) or void swelling; or changes in material properties due to void swelling.

### 4.2.3 Flux Detector Thimble Inspection Program

The FNP Flux Detector Thimble Inspection Program [19] will be implemented prior to entering the period of extended operation to formalize examinations already being performed. It will be used to identify loss of material resulting from fretting/wear in the detector thimbles during the period of extended operations. The program is in response to NRC Bulletin 88-09 [24] with the intention to ensure that pressure boundary integrity of the in-core system flux thimble tubes is maintained.

### 4.2.4 Control Rod Guide Tube Support Pin Replacement Project

The control rod guide tube support pins are used to align the bottom of the control rod guide tube assembly into the top of the core plate. In general, SCC prevention is aided by adherence to strict primary water chemistry limits that effectively prevent SCC and greatly reduce the probability of IASCC. The limits imposed by the Water Chemistry Control Program at FNP Unit 2 are consistent with the latest EPRI guidelines as described in Section 4.1.

Since 1990, ultrasonic testing has indicated that SCC has occurred in certain second generation alloy X-750 (Grade 688) support pins in various plants with greater than 55,000 hours of operation. Prior to replacement, numerous support pins at other plants using alloy X-750 material failed during removal or during operation between 110,900 and 149,000 hours of operation.

In response to industry concern for SCC of the alloy X-750 material, SNC replaced all of the upper internals guide tube support pins at FNP Unit 2 (November 1999) with Westinghouse-supplied strain hardened austenitic type 316 stainless steel support pins to mitigate the possibility of continued SCC of these components. Detailed descriptions of the replacement are contained within the Field Change Notice [20].

### 4.2.5 Power Uprating Project

FNP Unit 2 was originally licensed to operate at 2652 MWt core power (2660 MWt thermal); however, most safety analyses calculations had been performed assuming a higher core power. The FNP Unit 2 power uprate project increased the core operating power to 2775 MWt (2785 MWt thermal). Safety analysis assumed 2831 MWt core power for analyses supporting the power uprate project demonstrating margin to the uprated licensed core power output. Information on the power uprate and supporting analyses can be found in the licensing report [43] and NRC safety evaluation (SE) of the associated FNP license amendment [44].

## 4.3 INDUSTRY PROGRAMS

### 4.3.1 WCAP-14577, Aging Management for Reactor Internals

The WOG (now PWROG) topical report WCAP-14577 [8] contains a technical evaluation of aging degradation mechanisms and aging effects for Westinghouse RVI components. The WOG sent the report to the NRC staff to demonstrate that WOG member plant owners that subscribed to the WCAP could adequately manage effects of aging on RVI during the period of extended operation, using approved aging management methodologies of the WCAP to develop plant-specific aging management programs.

The AMR for the FNP Unit 2 internals documented in [21] utilized WCAP-14577 [8] as an input source regarding applicable aging affects and aging management programs. FNP reactor internal components, plant operating and loading conditions, temperature, pressure and water chemistry are consistent with or bounded by those reflected in [8]. Therefore, the NRC approved topical report [8] is applicable to the FNP AMP.

### 4.3.2 MRP-227-A, Reactor Internals Inspection and Evaluation Guidelines

MRP-227-A, as discussed in Section 2, was developed by a team of industry experts including utility representatives, NSSS vendors, independent consultants and international committee representatives who reviewed available data and industry experience on materials aging. The objective of the group was to develop a consistent, systematic approach for identifying and prioritizing inspection and evaluation requirements for reactor internals. The following subsections briefly describe the industry process.

#### 4.3.2.1 MRP-227-A, RVI Component Category

MRP-227-A used a screening and ranking process to aid in the identification of required inspections for specific RVI components. MRP-227-A credited existing component inspections, when they were deemed adequate, as a result of detailed expert panel assessments conducted in conjunction with the development of the industry document. Through the elements of the process, the reactor internals for all currently licensed and operating PWR designs in the United States were evaluated in the MRP program; appropriate inspection, evaluation and implementation requirements for reactor internals were defined.

Based on the completed evaluations, the RVI components are categorized within MRP-227-A as "Primary" components, "Expansion" components, "Existing Programs" components or "No Additional Measures" components, as described below:

- Primary

  Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in the Inspection & Evaluation (I&E) guidelines. The Primary group also includes components that 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 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 components depends on the findings from the examinations of the Primary components at individual plants.

- Existing Programs

  Those PWR internals that are susceptible to the effects of at least one of the eight 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.

- No Additional Measures Programs

  Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of a failure mode, effects, and criticality analysis (FMECA) and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

The categorization and analysis used in the development of MRP-227-A are not intended to supersede any ASME B&PV Code Section XI [22] requirements. Any components that are classified as core support structures, as defined in ASME B&PV Code Section XI IWB-2500, Category B-N-3, have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a.

#### 4.3.2.2 NEI 03-08 Guidance within MRP-227-A

The industry program requirements of MRP-227-A are classified in accordance with the requirements of the NEI 03-08 protocols. The MRP-227 guideline includes Mandatory and Needed elements as follows:

**Mandatory**

There is one Mandatory element:

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

   FNP Unit 2 Applicability: MRP-227, Revision 0 was officially issued by the industry in December 2008. An AMP must be developed within thirty-six months following issuance of MRP-227, Revision 0. To fulfill this requirement and the license renewal commitments provided in Section 1, SNC developed NMP-ES-029-GL02, "PWR Reactor Internals Strategic Plan" [1]. This program was implemented to meet this requirement as documented in [1].

   According to the NRC Regulatory Issue Summary (RIS) [3], FNP Unit 2 qualifies as a Category B plant because they have a renewed license with a commitment to submit an AMP/inspection plan based on MRP-227, but have not yet been required to do so by their commitment. This AMP fulfills the license renewal commitment to submit an implementation schedule for FNP Unit 2 in accordance with MRP-227-A [5] to the NRC no later than March 31, 2019.

**Needed**

There are five Needed elements:

1. *Each commercial U.S. PWR unit shall implement MRP-227-A, 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.*

FNP Unit 2 Applicability: MRP-227-A augmented inspections have been appropriately incorporated into this AMP for the license renewal period. The applicable Westinghouse tables contained in MRP-227-A, Table 4-3 (Primary), Table 4-6 (Expansion), Table 4-9 (Existing) and Table 5-3 (Examination Acceptance and Expansion Criteria) and are attached herein as Tables C-1, C-2, C-3, and C-4 respectively.

2. *Examinations specified in the MRP-227-A guidelines shall be conducted in accordance with Inspection Standard, MRP-228 [10].*

   FNP Unit 2 Applicability: SNC has developed fleet NDE procedure NMP-ES-024-112 [38] to detail the process for implementation of MRP-228 [10] for PWR Internals NDE requirements at Southern Nuclear facilities. The procedure defines a process to ensure that the combinations of equipment, procedures and personnel used to perform examinations of reactor internals at SNC sites meet the implementation requirements of MRP-228.

3. *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.*

   FNP Unit 2 Applicability: FNP Unit 2 will comply with this requirement.

4. *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.*

   FNP Unit 2 Applicability: As discussed in subsection 4.3.3, SNC will participate in future industry efforts and will adhere to industry directives for reporting, response, and follow-up.

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

   FNP Unit 2 Applicability: FNP Unit 2 will evaluate any examination results that do not meet the examination acceptance criteria in Section 5 of MRP-227-A in accordance with an NRC-approved methodology.

#### 4.3.2.3 GALL AMP Development Guidance

It should be noted that Section XI.M16A of NUREG-1801, Revision 2 [17] includes a description of the attributes that make up an acceptable AMP. These attributes are consistent with the FNP Unit 2 Aging Management Review process. Evaluation of the FNP Unit 2 RVI AMP against GALL attribute elements is provided in Section 5 of this AMP.

As part of License Renewal, SNC agreed to participate in the industry programs applicable to FNP for investigating and managing aging effects on reactor internals. The industry efforts have defined the required inspections and examination techniques for those components critical to aging management of RVI. The results of the industry recommended inspections, as published in MRP-227-A, serve as the basis for identifying any augmented inspections that are required to complete the FNP Unit 2 RVI AMP.

#### 4.3.2.4 MRP-227-A Applicability to FNP Unit 2

The applicability of MRP-227-A to FNP Unit 2 requires compliance with the following MRP-227-A assumptions:

- *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.*

  FNP Unit 2 Applicability: According to the SNC RVI Program [1], the SNC fuel management program changed from a high to a low leakage core loading pattern prior to 30 years of operation of FNP Unit 2.

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

  FNP Unit 2 Applicability: FNP Unit 2 operates as a base load unit [1].

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

  FNP Unit 2 Applicability: MRP-227-A states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs considered. SNC has not made any modifications to the Unit 2 internals since May 2007 [1].

Based on the plant-specific applicability, as stated, the MRP-227-A work is representative for FNP Unit 2.

### 4.3.3 Ongoing Industry Programs

The U.S. industry, through both the EPRI/MRP and the PWROG, continues to sponsor activities related to RVI aging management, including planned development of a standard NRC submittal template, development of a plant-specific implementation program template for currently licensed U.S. PWR plants, and development of acceptance criteria and inspection disposition processes. SNC will maintain cognizance of industry activities related to PWR internals inspection and aging management. SNC will also address/implement industry guidance, stemming from those activities, as appropriate under NEI 03-08 practices.

## 4.4 SUMMARY

It should be noted that the SNC FNP Unit 2, the MRP and the PWROG approaches to aging management are based on the GALL approach to aging management strategies. This approach includes a determination of which reactor internals passive components are most susceptible to the aging mechanisms of concern, and then determination of the proper inspection or mitigating program that provides reasonable assurance that the component will continue to perform its intended function through the period of extended operation. The GALL-based approach was used at Farley for the initial basis of the LRA that resulted in the NRC SER in NUREG-1825 [2].

The approach used to develop the FNP Unit 2 AMP is fully compliant with regulatory directives and approved documents. The additional evaluations and analysis completed by the MRP industry group have provided clarification to the level of inspection quality needed to determine the proper examination method and frequencies. The tables provided in MRP-227-A and included as Appendix C of this AMP provide the level of examination required for each of the components evaluated.

It is the Farley position that use of the AMR produced by the LRA methodology, combined with any additional augmented inspections required by the MRP-227-A industry tables provided in Appendix C, provides reasonable assurance that the reactor internals passive components will continue to perform their intended functions through the period of extended operation.

# 5 FARLEY NUCLEAR PLANT REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES

The FNP Unit 2 RVI AMP is credited for aging management of RVI components for the following eight aging degradation mechanisms and their associated effects:

- Stress corrosion cracking
- Irradiation-assisted stress corrosion cracking
- Wear
- Fatigue
- Thermal aging embrittlement
- Irradiation embrittlement
- Void swelling and irradiation growth
- Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep

The attributes of the FNP Unit 2 RVI AMP and compliance with NUREG-1801 (GALL Report), Section XI.M16A, "PWR Vessel Internals" [17] are described in this section. The GALL identifies 10 attributes for successful component aging management. The framework for assessing the effectiveness of the projected program is established by the use of the 10 elements of the GALL.

SNC fully utilized the GALL process contained in NUREG-1801 [17] in performing the aging management review of the reactor internals in the license renewal process. However, SNC made a commitment (see NUREG-1825 [2]) to incorporate the following: (1) implement the new FNP Reactor Vessel Internals Program prior to entering the period of extended operation, (2) continue to participate in industry initiatives intended to clarify the nature and extent of aging mechanisms affecting the FNP reactor internals, (3) incorporate the results of these initiatives into the RVI program and (4) submit an inspection plan for the RVI Program for NRC review and approval at least 24 months prior to entering the periods of extended operation for the FNP units.

This AMP is consistent with that process and includes consideration of the augmented inspections identified in MRP-227-A and fully meets the requirements of the commitment and GALL, Revision 2. Specific details of the FNP Unit 2 reactor internals AMP are summarized in the following subsections.

## 5.1 GALL REVISION 2 ELEMENT 1: SCOPE OF PROGRAM

### GALL Report AMP Element Description

> *"The scope of the program includes all RVI components at the Farley Nuclear Plant Unit 2 Nuclear Plant, which is built to a Westinghouse 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 of 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"* [17].

**FNP Unit 2 Program Scope**

The FNP Unit 2 reactor internals consist of the lower core support structure, the upper core support structure, and the in-core instrumentation support structures. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDMs, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the in-core instrumentation. The lower core support structure consists of the core barrel, the core baffle assemblies, the lower core plate, the neutron shield panels, the lower core support forging, the secondary support assembly and associated support columns. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel and, at its lower end, is restrained by a radial support system attached to the vessel wall. The upper core support structure consists of the upper support assembly, the upper core plate, support columns and control rod guide tube assemblies. The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the upper closure head, and a lower system to convey and support flux thimbles penetrating the vessel through the bottom head. Additional RVI details are discussed in FNP Unit 2 updated final safety analysis report (UFSAR) subsection 4.2.2, Reactor Vessel Internals.

The FNP Unit 2 RVI subcomponents that required aging management review are indicated in the previously submitted Table 2.3.1-2 of the FNP Unit 2 LRA [23]. The components listed in Table 2.3.1-2 are consistent with those in Appendix B of this report.

The FNP Unit 2 Reactor Internals AMR was conducted and documented in [21]. The table summarizing the results of that review was also documented in Table 3.1.2-2 of the FNP Unit 2 LRA [23]. This table is included in Appendix B of this AMP. The table identifies the aging effects that require management for the components requiring AMR. A column in the tables lists the program/activity that is credited to address the component and aging effect during the period of extended operation. The NRC has reviewed and approved the aging management strategy presented in the Appendix B tables as documented in the SER on license renewal [2].

The results of the industry research provided by MRP-227-A, summarized in the tables of Appendix C, provide the basis for the required augmented inspections, inspection techniques to permit detection and characterizing of the aging effects (cracks, loss of material, loss of preload, etc.) of interest, prescribed frequency of inspection and examination acceptance criteria. The information provided in MRP-227-A is rooted in the GALL methodology. The basic assumptions of MRP-227-A, Section 2.4 are met by FNP Unit 2 and are addressed in subsection 4.3.2.4 of this AMP. The Topical Report Conditions and Applicant/Licensee Action Items provided by the NRC in the SE on MRP-227, Revision 0 [5] are met by FNP, and demonstration of compliance is addressed in Section 6.1 for the Topical Report Conditions and in Section 6.2 for the Applicant/Licensee Action Items. The FNP Unit 2 RVI AMP scope is additionally based on previously established and approved GALL Report approaches through application of the MRP-227-A [5] methodologies to determine those components that require aging management.

**Conclusion**

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP SER.

## 5.2 GALL REVISION 2 ELEMENT 2: PREVENTATIVE ACTIONS

**GALL Report AMP Element Description**

> *"The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, 'Water Chemistry'"* [17].

**FNP Unit 2 Preventive Action**

The FNP Unit 2 RVI AMP includes the Primary Water Chemistry Program [18] as an existing program that complies with the requirements of this element. A description and applicability to the FNP Unit 2 RVI AMP is provided in the following subsection.

**FNP Unit 2 Primary Water Chemistry Program**

The FNP Water Chemistry Program [18] will manage loss of material and cracking within system components and structures, thereby ensuring continued structural integrity, reliability and availability. The program includes monitoring of detrimental species and addition of chemical additives. The FNP program utilizes the EPRI PWR Primary Water Chemistry Guidelines [25] in establishing chemistry control procedures for FNP. Prior to adopting later revisions of the EPRI guidelines, SNC evaluates the acceptability of any changes in implementing requirements. The FNP Water Chemistry Program incorporates the best practices of industry organizations, vendors and utilities.

**Conclusion**

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

## 5.3 GALL REVISION 2 ELEMENT 3: PARAMETERS MONITORED OR INSPECTED

**GALL Report AMP Element Description**

*"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 the 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 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 Westinghouse designed Primary Components in Table 4-3 of MRP-227. Additionally, the program implements the parameters monitored/inspected criteria for Westinghouse designed Expansion Components in Table 4-6 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 Measure," in accordance with the analyses reported in MRP-227"* [17].

**FNP Unit 2 Parameters Monitored or Inspected**

The FNP Unit 2 AMP monitors, inspects and/or tests for the effects of the eight aging degradation mechanisms on the intended function of the FNP Unit 2 PWR internals components through inspection and condition monitoring activities in accordance with the augmented requirements defined under industry directives as contained in MRP-227-A and ASME Section XI [22].

This AMP implements the requirements for the Primary Component inspections from Table 4-3 of MRP-227-A (included in Appendix C of this AMP as Table C-1), the Expansion Component inspections from Table 4-6 of MRP-227-A (included in Appendix C of this AMP as Table C-2) and the Existing Component inspections from Table 4-9 of MRP-227-A (included in Appendix C of this AMP as Table C-3). These tables contain requirements to monitor and inspect the RVI through the period of extended operation to address the effects of the eight aging degradation mechanisms.

For license renewal, the ASME Section XI Program [4] includes periodic visual, surface and/or volumetric examinations and leakage tests of Class 1, 2 and 3 pressure-retaining components and their integral attachments, including welds, pump casings valve bodies and pressure-retaining bolting. The requirements of MRP-227-A only augment and do not replace or modify the requirements of ASME Section XI. This program is consistent with the corresponding program described in the GALL Report [17].

Appendices B and C of this AMP provide a detailed listing of the components and subcomponents and the parameters monitored, inspected and/or tested.

**Conclusion**

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

## 5.4 GALL REVISION 2 ELEMENT 4: DETECTION OF AGING EFFECTS

**GALL Report AMP Element Description**

*"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 needed 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 Westinghouse designed Primary Components in Table 4-3 of MRP-227 and for Westinghouse designed Expansion Components in Table 4-6 of MRP-227.*

*The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): for FNP Unit 2, no additional Primary or Expansion components are relevant to the scope of aging management for the RVI.*

*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 that for the hold down spring. The hold down spring at FNP Unit 2 is fabricated from Type 304 SS that requires inspection by physical measurement"* [17].

**FNP Unit 2 Detection of Aging Effects**

Detection of indications required by the ASME Section XI ISI Program [4] is well established and field-proven through the application of the Section XI ISI Program. Those augmented inspections that are taken from the MRP-227-A recommendations will be applied through use of the MRP-228 inspection standard. This AMP implements the augmented inspection requirements of Table 4-3, Table 4-6 and Table 4-9 from MRP-227-A for the Primary, Expansion and Existing Components, respectively. These are included in Appendix C of this AMP for reference. These tables include the inspection frequency and sampling basis. For the Expansion Components of MRP-227-A, this AMP implements the expansion requirements of Table 5-3 of MRP-227-A (included in Appendix C of this AMP as Table C-4).

Inspection can be used to detect physical effects of degradation including cracking, fracture, wear and distortion. The choice of an inspection technique depends on the nature and extent of the expected damage. The recommendations supporting aging management for the reactor internals, as contained in this report, are built around three basic inspection techniques: (1) visual, (2) ultrasonic and (3) physical measurement. The three different visual techniques include VT-3, VT-1 and EVT-1. The assumptions and process used to select the appropriate inspection technique are described in the following subsections. Inspection standards developed by the industry for the application of these techniques for augmented reactor internals inspections are documented in MRP-228 [10]. SNC has developed a fleet NDE procedure [38] which details the SNC process for implementing the techniques per the requirements prescribed in MRP-228.

VT-1 Visual Examinations

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520 [22]. VT-1 visual examination is intended to identify crack-like surface flaws. Unacceptable conditions for a VT-1 examination are:

- Crack-like surface flaws on the welds joining the attachment to the vessel wall that exceed the allowable linear flaw standards of IWB-3510 [22]
- Structural degradation of attachment welds such that the original cross-sectional area is reduced by more than 10 percent

These requirements are defined to ensure the integrity of attachment welds on the ferritic pressure vessel. Although the IWB-3520 criteria do not directly apply to austenitic stainless steel internals, the clear intent is to ensure that the structure will meet minimum flaw tolerance fracture requirements. In the MRP-227-A recommendations, VT-1 examinations have been identified for components requiring close visual examinations with some estimate of the scale of deformation or wear. Note that in MRP-227-A, VT-1 has only been selected to detect distortion as evidenced by small gaps between the upper-to-lower mating surfaces of CE-welded core shrouds assembled in two vertical sections. Therefore, no additional VT-1 inspections over and above those required by ASME Section XI ISI have been specified.

EVT-1 Enhanced Visual Examination for the Detection of Surface Breaking Flaws

In the augmented inspections detailed in the MRP-227-A for reactor internals, the EVT-1 enhanced visual examination has been identified for inspection of components where surface-breaking flaws are a potential concern. Any visual inspection for cracking requires a reasonable expectation that the flaw length and crack mouth opening displacement meet the resolution requirements of the observation technique. The EVT-1 specification augments the VT-1 requirements to provide more rigorous inspection standards for stress corrosion cracking, and has been demonstrated for similar inspections in boiling water reactor (BWR) internals. Enhanced visual examination (i.e., EVT-1) is also conducted in accordance with the requirements described for visual examination (i.e., VT-1) with additional requirements (such as camera scanning speed). Any recommendation for EVT-1 inspection will require additional analysis to establish flaw-tolerance criteria, which must take into account potential embrittlement due to thermal aging or neutron irradiation. The industry, through the PWROG, has developed an approach for acceptance criteria methodologies to support plant-specific augmented examinations. This work is summarized in WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" [11]. The acceptance criteria developed using these methodologies may be created on either a generic or plant-specific basis because both loads and component dimensions may vary from plant-to-plant within a typical PWR design.

VT-3 Examination for General Condition Monitoring

In the augmented inspections detailed in the MRP-227-A for reactor internals, the VT-3 visual examination has been identified for inspection of components where general condition monitoring is required. The VT-3 examination is intended to identify individual components with significant levels of existing degradation. As the VT-3 examination is not intended to detect the early stages of component cracking or other incipient degradation effects, it should not be used when failure of an individual component could threaten either plant safety or operational stability. The VT-3 examination may be appropriate for inspecting highly redundant components (such as baffle-edge bolts), where a single failure does not compromise the function or integrity of the critical assembly.

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520. These criteria are designed to provide general guidelines. The unacceptable conditions for a VT-3 examination are listed below:

- Structural distortion or displacement of parts to the extent that component function may be impaired
- Loose, missing, cracked or fractured parts, bolting or fasteners
- Foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel
- Corrosion or erosion that reduces the nominal section thickness by more than 5 percent
- Wear of mating surfaces that may lead to loss of function
- Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 percent

The VT-3 examination is intended for use in situations where the degradation is readily observable. It is meant to provide an indication of condition, and quantitative acceptance criteria are not generally required. In any particular recommendation for VT-3 visual examination, it should be possible to identify the specific conditions of concern. For instance, the unacceptable conditions for wear indicate wear that might lead to loss of function. Guidelines for wear in a critical-alignment component may be very different from the guidelines for wear in a large structural component.

Surface Examination

In order to further characterize discontinuities on the surface of components, surface examination can supplement either visual (VT-3) or (VT-1/EVT-1) examinations specified in these guidelines. This supplemental examination may thus be used to reject or accept relevant indications. A surface examination is an examination that indicates the presence of surface discontinuities, and the ASME B&PV Code [22] lists magnetic particle, liquid penetrant, eddy current and ultrasonic examination methods as surface examination alternatives. Here, only the electromagnetic testing (ET), also called eddy current surface examination method, is covered.

When selected for use as a supplemental examination to examinations performed in these guidelines, an ET examination is conducted in accordance with the requirements of the inspection standard [10].

ET examination is widely used for heat exchanger tubing inspections. Eddy currents are induced in the inspected object by electromagnetic coils, with disruptions in the eddy current flow caused by surface or near-surface anomalies detected by suitable instrumentation. Industry experience with ET examination is relatively robust, especially in the aerospace and petroleum refinery industries. The experience base for PWR nuclear systems is moderately robust, particularly for examination of steam generator, flux thimble and heat exchanger tubing.

Ultrasonic Testing

Volumetric examinations in the form of ultrasonic testing (UT) techniques can be used to identify and determine the length and depth of a crack in a component. Although access to the surface of the component is required to apply the ultrasonic signals, the flaw may exist in the bulk of the material. In this proposed strategy, UT inspections have been recommended exclusively for detection of flaws in bolts. For the bolt inspections, any bolt with a detected flaw should be assumed to have failed. The size of the flaw in the bolt is not critical because crack growth rates are generally high, and it is assumed that the observed flaw will result in failure prior to the next inspection opportunity. It has generally been observed through examination performance demonstrations that UT can reliably (90 percent or greater reliability) detect flaws that reduce the cross-sectional area of a bolt by 35 percent.

Failure of a single bolt does not compromise the function of the entire assembly. Bolting systems in the reactor internals are highly redundant. For any system of bolts, it is possible to demonstrate multiple acceptable bolting patterns. The evaluation program must demonstrate that the remaining bolts meet the requirements for an acceptable bolting pattern for continued operation. The evaluation procedures must also demonstrate that the pattern of remaining bolts contains sufficient margin such that continuation of the bolt failure rate will not result in failure of the system to meet the requirements for an acceptable bolting pattern before the next inspection.

Establishment of the acceptable bolting pattern for any system of bolts requires analysis to demonstrate that the system will maintain reliability and integrity in continuing to perform the intended function of the component. This analysis is highly plant-specific. Therefore, any recommendation for UT inspection of bolts assumes that the plant owner will work with the designer to establish acceptable bolting patterns prior to the inspection to support continued operation. For Westinghouse-designed plants, acceptable bolting patterns for baffle-former and barrel-former bolts are available through the PWROG (e.g., [41]).

SNC has been a full participant in the development of the PWROG documents and has full access and use.

Physical Measurement Examination

Continued functionality can be confirmed by physical measurements to evaluate the impact caused by various degradation mechanisms, such as wear or loss of functionality as a result of loss of preload or material deformation. For FNP Unit 2, direct physical measurements are required only for the hold down spring.

**Conclusion**

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

## 5.5 GALL REVISION 2 ELEMENT 5: MONITORING AND TRENDING

### GALL Report AMP Element Description

> *"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 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"* [17].

### FNP Unit 2 Monitoring and Trending

Operating experience with PWR reactor internals has been generally proactive. Flux thimble wear and control rod guide tube split pin cracking issues were identified by the industry and continue to be actively managed. The extremely low frequency of failure in reactor internals makes monitoring and trending based on OE somewhat impractical. The majority of the materials aging degradation models used to develop the MRP-227-A guidelines are based on test data from reactor internals components removed from service. The data are used to identify trends in materials degradation and forecast potential component degradation. The industry continues to share both material test data and OE through the auspices of the MRP and PWROG. SNC has in the past and will continue to maintain cognizance of industry activities and shared information related to PWR internals inspection and aging management.

Inspections credited in Appendix B are based on utilizing the FNP Unit 2 10-year ISI program and the augmented inspections derived from MRP-227-A as documented in Appendix C. The MRP-227-A inspections only augment and do not replace the existing ASME Section XI ISI requirements. These inspections, where practical, are scheduled to be conducted in conjunction with typical 10-year ISI examinations.

Tables C-1, C-2 and C-3 identify the augmented Primary and Expansion inspection and monitoring recommendations, and the Existing programs credited for inspection and aging management. As discussed in MRP-227-A, inspection of the "Primary" components provides reasonable assurance for demonstrating component current capacity to perform the intended functions. Table C-4 in Appendix C identifies the MRP-227-A expansion criteria from the Primary components. If these expansion criteria are met for a component, the associated Expansion component is to be inspected to manage the aging degradation.

Reporting requirements are included as part of the MRP-227-A guidelines. Consistent reporting of inspection results across all PWR designs will enable the industry to monitor reactor internals degradation on an ongoing industry basis as the period of extended operation moves forward. Reporting of examination results will allow the industry to monitor and trend results and take appropriate preemptive action through update of the MRP guidelines.

**Conclusion**

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

## 5.6 GALL REVISION 2 ELEMENT 6: ACCEPTANCE CRITERIA

**GALL Report AMP Element Description**

> *"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. The acceptance criterion for physical measurements performed on the height limits of the Westinghouse-designed hold down springs are required for 304 SS hold down springs. FNP Unit 2 has a 304 SS hold down spring; therefore, FNP Unit 2 is required to produce acceptance criteria for the physical measurements on the hold down spring"* [17].

**FNP Unit 2 Acceptance Criteria**

Those recordable indications that are the result of inspections required by the existing FNP Unit 2 ISI program scope are evaluated in accordance with the applicable requirements of the ASME Code through the existing Corrective Action Program [26].

Inspection acceptance and expansion criteria are provided in Table C-4 of this document. These criteria will be reviewed periodically as the industry continues to develop and refine the information, and will be included in updates to FNP Unit 2 procedures to enable the examiner to identify examination acceptance criteria considering state-of-the-art information and techniques. SNC has a commitment to develop acceptance criteria for the hold down spring physical measurements that will be consistent with the licensing basis for FNP Unit 2 [5].

Augmented inspections, as defined by the MRP-227-A requirements included in this AMP as Table C-1, Table C-2 and Table C-3, that result in recordable relevant conditions will be entered into the plant Corrective Action Program and addressed by appropriate actions that may include enhanced inspection, repair, replacement, mitigation actions or analytical evaluations. An example of an analytical evaluation is using an acceptable bolting WCAP approach, such as those commonly used to support continued component or assembly functionality. Additional analysis to establish acceptable bolting pattern evaluation criteria for the baffle-former bolt assembly, as contained in various industry documents [41], is also considered in determining the acceptance of inspection results to support continued component or assembly functionality.

The industry, through various cooperative efforts, is working to construct a consensus set of tools in line with accepted and proven methodologies to support this element. One of these tools is the PWROG document WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" [11], which details acceptance criteria methodology for the MRP-227 Primary and Expansion components. Status is monitored through direct SNC cognizance of industry (including PWROG) activities related to PWR internals inspection and aging management.

**Conclusion**

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

## 5.7 GALL REVISION 2 ELEMENT 7: CORRECTIVE ACTIONS

### GALL Report AMP Element Description

> *"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 Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, or for B&W-designed RVI components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies 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"* [17].

### FNP Unit 2 Corrective Action

The existing FNP procedure for corrective actions, the "Corrective Action Program" [26] and the ASME Section XI ISI program [4], will be credited for this element. These procedures establish the FNP Unit 2 repair and replacement requirements of ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" [22]. These requirements include the identification of a repair cycle, repair plan, and verification of acceptability for replacements. FNP Unit 2 is committed to performing corrective actions for augmented inspections using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI [22] and MRP-227-A, Section 6 [5].

**Conclusion**

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

## 5.8 GALL REVISION 2 ELEMENT 8: CONFIRMATION PROCESS

**GALL Report AMP Element Description**

> *"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"* [17].

**FNP Unit 2 Confirmation Process**

FNP Unit 2 has an established 10 CFR Part 50, Appendix B Program [28] that addresses the elements of corrective actions, confirmation process and administrative controls. The FNP Unit 2 Program includes non-safety-related structures, systems and components. Quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

**Conclusion**

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

## 5.9 GALL REVISION 2 ELEMENT 9: ADMINISTRATIVE CONTROLS

**GALL Report AMP Element Description**

> *"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"* [17].

**FNP Unit 2 Administrative Controls**

FNP Unit 2 has an established 10 CFR Part 50, Appendix B Program [28] that addresses the elements of corrective actions, confirmation process and administrative controls. QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

**Conclusion**

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

## 5.10 GALL REVISION 2 ELEMENT 10: OPERATING EXPERIENCE

**GALL Report AMP Element Description**

> *"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"* [17].

**FNP Unit 2 Operating Experience**

Extensive industry and FNP Unit 2 OE has been reviewed during the development of the RVI AMP. The experience reviewed includes NRC Information Notices 84-18, "Stress Corrosion Cracking in PWR Systems" [29] and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants" [30]. Most of the industry OE reviewed has involved cracking of austenitic stainless steel baffle-former bolts or SCC of high-strength internals bolting. SCC of control rod guide tube support pins has also been reported.

Early plant OE related to hot functional testing and reactor internals is documented in plant historical records. Inspections performed as part of the 10-year ISI program have been conducted as designated by existing commitments, and would be expected to discover overall general internals structure degradation. To date, very little degradation has been observed industry-wide.

Industry OE is routinely reviewed by SNC engineers using Institute of Nuclear Power Operations (INPO) OE, the Nuclear Network, and other information sources as directed under the applicable procedure [31], for the determination of additional actions and lessons learned.

A review of industry and plant-specific experience with RVI reveals that the U.S. industry, including SNC and FNP Unit 2, has responded proactively to industry issues relative to reactor internals degradation. Three examples that demonstrate this proactive response are the replacement of the Unit 2 control rod guide tube split pins in 1999, the replacement of baffle bolts in 1999, and the upflow conversion of reactor internals in 2002, which are briefly described in the following paragraphs.

FNP Unit 2 Control Rod Guide Tubes Support Pins

In response to the industry concern for SCC of the alloy X-750 material, SNC replaced all of the upper internals guide tube support pins at FNP Unit 2 (November 1999) with Westinghouse-supplied, cold worked Type 316 SS support pins to mitigate the possibility of continued SCC of these components. Detailed descriptions of the replacement are contained in the Field Change Notice [20], and documents referenced within, as well as the plant records [47].

FNP Unit 2 Baffle Bolts

During the Fall 1999 Outage, a proactive decision was made to replace a portion of the 1088 baffle former bolts in response to indications of cracking in 316 Type SS baffle-former bolts observed in a number of plants outside of the U.S. Detailed descriptions of the replacement are contained in the Field Change Notice [42], and documents referenced within, as well as the plant records [46].

A key element of the MRP-227-A guideline is the reporting of age-related degradation of RVI components. SNC, through its participation in PWROG and EPRI-MRP activities, will continue to benefit from reporting of inspection information and will share its own OE with the industry through the reporting requirements of Section 7 of MRP-227-A. The collected information from MRP-227-A augmented inspections will benefit the industry in its continued response to RVI aging degradation.

FNP Unit 2 Upflow Conversion

In response to the fuel rod failures, resulting from flow-induced vibration initiated by reactor coolant crossflow jetting through joints between baffle plates, several plants with Westinghouse-designed reactor internals were field modified to reverse the secondary coolant flow pattern in the baffle/barrel region in order to reduce the jet-driving differential pressure. The original baffle/barrel region coolant flow pattern is known as "downflow" while the modified flow pattern is described as "upflow." Farley Unit 2 reactor internals have been modified for upflow conversion [45].

**Conclusion**

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16A [17] and Commitment 6 in the FNP Unit 2 SER.

# 6 DEMONSTRATION

FNP Unit 2 has demonstrated a long-term commitment to aging management of reactor internals. This AMP is based on an established history of programs to identify and monitor potential aging degradation in the reactor internals. Programs and activities undertaken in the course of fulfilling that commitment include:

- The examinations required by ASME Section XI for the FNP Unit 2 reactor vessel internals have been performed during each 10-year interval since plant operations commenced.

- As documented in FNP operational procedures, reports are continuously reviewed by FNP personnel for applicable issues that indicate operating procedures or programs require updates based on new OE.

- Review of Nuclear Oversight Section (NOS) audit reports, NRC inspection reports and INPO evaluations indicate no unacceptable issues related to RVI inspections.

- The Water Chemistry Control Program at FNP has been effective in maintaining oxygen, halogens and sulfate at levels sufficiently low to prevent SCC, therefore maintaining structural integrity of the reactor vessel internals.

- Replacement control rod guide tube support pins for FNP Unit 2 in 1999 were fabricated from strain-hardened, austenitic type 316 stainless steel materials to increase resistance to SCC (versus original pins) [20].

- Replaced a portion of the 1,088 baffle former bolts during 1999 outage in response to indications of cracking in Type 316 SS baffle-former bolts observed in a number of plants outside of the U.S. [42].

- Completed core power uprate for FNP Unit 2 in 1998 from 2652 MWt to 2775 MWt.

- Completed conversion of reactor internals coolant flow from “downflow” to “upflow” for Unit 2 in 2002.

- SNC has actively participated in past and ongoing EPRI and PWROG RVI activities. SNC will continue to maintain cognizance of industry activities related to PWR internals inspection and aging management, and will address/implement industry guidance stemming from those activities as appropriate under NEI 03-08 practices.

This AMP fulfills the approved license renewal methodology requirement to identify the most susceptible components, and to inspect those components with an indication detection level commensurate with the expected degradation mechanism indication. Augmented inspections, derived from the information contained in MRP-227-A (the industry I&E Guidelines), have been utilized in this AMP to build on existing plant programs. This approach is expected to encourage detection of a degradation mechanism at its first appearance, which is consistent with the ASME approach to inspections. This approach provides reasonable assurance that the internals components will continue to perform their intended function through the period of extended operation.

Typical ASME Section XI examinations identified in the AMP are to be performed in the outage prior to entering the period of extended operation (Spring 2019, RO-26). The previous ISI for FNP Unit 2 was performed in Spring 2010 (RO-20). The augmented inspections discussed in compliance with MRP-227-A requirements have been integrated in the implementation schedule, which is shown in Section 7. Integration of the required inspections will be tracked to completion. As discussed, the industry MRP-227-A guidelines also provide for updates as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

The augmented inspections described in this document, as summarized in Appendix C, combined with the ASME Section XI ISI program inspections, existing FNP programs and use of Operating Experience Reports (OERs), provide reasonable assurance that the reactor internals will continue to perform their intended functions through the period of extended operation.

Table 6-1 lists the seven topical report conditions and Section 6.2 lists the eight applicant action items that came out of the NRC review of MRP-227, as listed in [5], as well as their compliance within this AMP.

## 6.1 DEMONSTRATION OF TOPICAL REPORT CONDITIONS COMPLIANCE TO SE ON MRP-227, REVISION 0

**Table 6-1. Topical Report Condition Compliance to SE on MRP-227**

| Topical Condition | Applicable/ Not Applicable | Compliance in AMP |
|---|---|---|
| 1. High consequence components in the "No Additional Measures" Inspection Category | Applicable | The upper core plate and the lower support forging or casting components are added to Table C-2 as "Expansion Components" linked to the "Primary Component," the control rod guide tube (CRGT) lower flange weld. |
| 2. Inspection of components subject to irradiation-assisted stress corrosion cracking | Applicable | The upper and lower core barrel cylinder girth welds and the lower core barrel flange weld are moved from Table C-2 "Expansion Components" to Table C-1 "Primary Components." |
| 3. Inspection of high consequence components subject to multiple degradation mechanisms | Not Applicable | Not applicable for FNP Unit 2. |
| 4. Imposition of minimum examination coverage criteria for "Expansion" inspection category components | Applicable | Notes 2 through 4 were added to Table C-1, as well as Note 2 to Table C-2 to reflect this condition. |
| 5. Examination frequencies for baffle-former bolts and core shroud bolts | Applicable | In Table C-1 for the baffle-former bolts, the inspection frequency was changed from 10 to 15 additional effective full-power years (EFPY) to subsequent examination on a ten-year interval. |
| 6. Periodicity of the re-examination of "Expansion" inspection category components | Applicable | "Re-inspection every 10 years following initial inspection" was added to every component under the Examination Method/Frequency column in Table C-2. |
| 7. Updating of MRP-227, Revision 0, Appendix A | Applicable | Section 5 is updated to reflect XI.M16A from GALL Revision 2 [17]. |

## 6.2 DEMONSTRATION OF APPLICANT/LICENSEE ACTION ITEM COMPLIANCE TO SE ON MRP-227, REVISION 0

### 6.2.1 SE Applicant/Licensee Action Item 1: Applicability of FMECA and Functionality Analysis Assumptions

*"As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant'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 plant 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 RVI components or plant 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. This is Applicant/Licensee Action Item 1"* [5].

**FNP Unit 2 Compliance**

The process used to verify that the RVI components at FNP Unit 2 are reasonably represented by the generic industry program assumptions (with regard to neutron fluence, temperature, stress values and materials used in the development of MRP-227-A [5]) is:

1. Identification of typical Westinghouse-designed PWR RVI components (MRP-191, Table 4-4 [9]).

2. Identification of FNP Unit 2 RVI components.

3. Comparison of the typical Westinghouse-designed PWR RVI components to the FNP Unit 2 RVI components identified in [23]:

    a. Confirmation that no additional items were identified by this comparison (primarily supports A/LAI 2).

    b. Confirmation that the materials for FNP Unit 2 are consistent with those materials identified in MRP-191, Table 4-4 [9].

    c. Confirmation that the FNP Unit 2 internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.

4. Confirmation that the FNP Unit 2 operating history is consistent with the assumptions in MRP-227-A [5] regarding core loading patterns.

5. Confirmation that FNP Unit 2 materials operated at temperatures within the original design basis parameters.

6. Determination of stress values based on design basis documents.

7. Confirmation that any changes to the FNP Unit 2 RVI components do not impact the application of the MRP-227-A [5] generic aging management strategy.

The FNP Unit 2 RVI components are reasonably represented by the design and operating history assumptions regarding neutron fluence, temperature, materials and stress values in the MRP-191 [9] generic FMECA and in the MRP-232 [33] functionality analysis based on the following:

1. FNP Unit 2 operating history is consistent with the assumptions in MRP-227-A [5] with regard to neutron fluence and fuel management.

    a. The FMECA and functionality analysis for MRP-227-A [5] were based on the assumption of 30 years of operation with high-leakage core loading patterns followed by 30 years of low-leakage core fuel management strategy. As stated in [1], FNP Unit 2 fuel management program changed from a high to a low-leakage core loading pattern prior to 30 years of operation. By operating with a low-leakage core design prior to 30 years, FNP Unit 2 meets the fluence and fuel management assumptions in MRP-191 [9] and requirements for MRP-227-A [5] application.

    b. As stated in [1], FNP Unit 2 has always operated as a base load unit. Therefore, FNP Unit 2 satisfies the assumptions in MRP documents regarding operational parameters affecting fluence.

2. The FNP Unit 2 reactor coolant system operates between $T_{cold}$ and $T_{hot}$ [35, Table 5.1-1]. $T_{cold}$ is no lower than 530.6°F and $T_{hot}$ is no higher than 613.3°F [35, Table 5.1-1]. The design temperature for the vessel is 650°F [35, Table 5.4-1]. Therefore, FNP Unit 2 operating history is within original design basis parameters and is consistent with the assumptions used to develop the MRP-227-A [5] aging management strategy with regard to temperature operational parameters.

3. The FNP Unit 2 RVI components and materials are comparable to the typical Westinghouse-designed PWR RVI components (MRP-191, Table 4-4 [9]).

    a. The components required to be in the FNP Unit 2 program [23] are consistent with those contained in MRP-191 [9]. No additional components are identified for FNP Unit 2.

    b. FNP Unit 2 RVI component materials are consistent with, or equivalent to, those materials identified in MRP-191, Table 4-4 [9] for Westinghouse-designed plants. The exceptions are the upper instrumentation conduit and supports – brackets, clamps, terminal blocks and conduit straps, which are identified as having CF8 material. Several additional components have slightly different materials than those specified in MRP-191; however, they have been determined to have no effect on the recommended MRP aging management inspection sampling strategy. These are dispositioned in the response to A/LAI 2.

    c. FNP Unit 2 internals are the same, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication.

4. FNP Station is a two-unit site with Westinghouse three-loop pressurized water reactors. A power uprate, in which the rated thermal power was increased from 2652 to the present 2775 megawatts thermal, has been implemented since initial commercial operation [43].

   The guide tube assembly split pins were replaced in Unit 2 [20].
   The vessel internals designs were converted from downflow to upflow [43].
   A portion of the Unit 2 baffle former bolts were replaced during the fall 1999 outage [42].

   SNC has not made any other modifications to the Unit 2 reactor internals components since May 2007 [1]. Therefore, modifications to the FNP Unit 2 RVI made over the lifetime of the plant are those specifically directed by the Original Equipment Manufacturer (OEM). The OEM has developed or evaluated design changes and satisfied assumptions for A/LAI 1.

   The design has been maintained over the lifetime of the plant as specified by the OEM, operational parameters are compliant with MRP-227-A [5] requirements with regard to fluence and temperature, and the components are consistent with those considered in MRP-191 [9]. The materials for the components are consistent with those considered in MRP-191 [9]. Therefore, the FNP Unit 2 RVI stress values are represented by the assumptions in MRP-191 [9], MRP-227-A [5] and MRP-232 [33], confirming the applicability of the generic FMECA.

**Conclusion**

The assumptions regarding plant design and operating history made in the FMECA and functionality analyses for the Westinghouse design apply to FNP Unit 2. The FNP Unit 2 complies with A/LAI 1 of the NRC SE regarding MRP-227, Revision 0. Therefore, the requirement is met for application of MRP-227-A [5] as a strategy for managing age-related material degradation in the RVI components.

### 6.2.2 SE Applicant/Licensee Action Item 2: PWR Vessel Internal Components within the Scope of License Renewal

*"As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components 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, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components 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 RVI components 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 this 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. This issue is Applicant/Licensee Action Item 2"* [5].

**FNP Unit 2 Compliance**

This A/LAI requires comparison of the FNP Unit 2 RVI components that are within the scope of license renewal for FNP Unit 2 to those components contained in MRP-191, Table 4-4 [9]. A detailed tabulation of the FNP Unit 2 RVI components [23] was completed and compared to the typical Westinghouse PWR components in MRP-191 [5]. All components required to be included in the FNP Unit 2 program are consistent with those contained in MRP-191 [9].

Several components have different materials than that specified in MRP-191 [9] assessment.

The upper instrumentation conduit and supports – thermocouple straps are CF8. Using the FMECA process, the use of CASS materials for the component: upper instrumentation conduit and supports – brackets, clamps, terminal blocks, and conduit straps was evaluated. The FMECA concluded that the components could be classified as "No Additional Measures" based on a consideration of the likelihood of failure and the likelihood of damage. There is no change to the FNP Unit 2 MRP-227-A inspection requirements as a result of the inclusion of CF8 for these components (brackets, clamps, terminal blocks and conduit straps).

Several additional components have slightly different materials (i.e., different grades of austenitic stainless steel) than those specified in MRP-191; however, they have been determined to have no effect on the recommended MRP aging management inspection sampling strategy.

The material differences have been assessed, and no modifications to the program details in MRP-227-A [5] are needed. This assessment supports the requirement that the NRC AMP shall provide assurance that the effects of aging on the FNP Unit 2 RVI components within the scope of license renewal, but not included in the generic Westinghouse-designed PWR RVI components from MRP-191, Table 4-4 [9], will be managed for the period of extended operation.

The generic scoping and screening of the RVI, as summarized in MRP-191 [9] and MRP-232 [33], to support the inspection sampling approach for aging management of the RVI specified in MRP-227-A [5] are applicable to FNP Unit 2 with no modifications for the FNP Unit 2 components.

**Conclusion**

FNP Unit 2 complies with A/LAI 2 of the NRC SE on MRP-227, Revision 0; therefore, it meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internal components.

### 6.2.3 SE Applicant/Licensee Action Item 3: Evaluation of the Adequacy of Plant-Specific Existing Programs

> *"As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being*

*relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). This is Applicant/Licensee Action Item 3"* [5].

**FNP Unit 2 Compliance**

FNP Unit 2 is compliant with the requirements in MRP-227-A, Table 4-9, as shown in Table C-3 of this document. This is detailed in the plant-specific FNP program documents for ASME Section XI [4] and the plant-specific flux thimble program [19].

In response to the industry concern, the control rod guide tube support pins fabricated from INCONEL® Alloy X-750 were replaced at FNP Unit 2 during the Fall 1999 outage; the replacement support pins utilized improved materials (strain-hardened austenitic stainless steel) that support the proactive management of aging in reactor internals components. Detailed descriptions of the replacement are retained in the plant records [47].

**Conclusion**

FNP Unit 2 complies with Applicant/Licensee Action Item 3 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

### 6.2.4 SE Applicant/Licensee Action Item 4: B&W Core Support Structure Upper Flange Stress Relief

*"As discussed in Section 3.2.5.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel 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" inspection 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 this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval. This is Applicant/Licensee Action Item 4"* [5].

**FNP Unit 2 Compliance**

This Applicant/Licensee Action Item is not applicable to FNP Unit 2 since it only applies to B&W plants.

**Conclusion**

Applicant/Licensee Action Item 4 of the NRC SE on MRP-227, Revision 0 is not applicable to FNP Unit 2.

### 6.2.5 SE Applicant/Licensee Action Item 5: Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components

*"As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227. This is Applicant/Licensee Action Item 5"* [5].

**FNP Unit 2 Compliance**

See Table 7-1. FNP Unit 2 utilizes a Type 304 SS hold down spring; therefore, SNC is planning to perform inspections/physical measurements on the FNP Unit 2 hold down spring according to MRP-227-A. SNC has an internal corrective action program tracking item to obtain the acceptance criteria for the hold down spring in advance of the outage in which measurements will be taken.

**Conclusion**

FNP Unit 2 complies with Applicant/Licensee Action Item 5 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

### 6.2.6 SE Applicant/Licensee Action Item 6: Evaluation of Inaccessible B&W Components

*"As addressed in Section 3.3.6 in this SE, MRP-227 does not propose to inspect the following inaccessible components: 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 components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. 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 components and, if necessary, provide their plan for the replacement of the components for NRC review and approval. This is Applicant/Licensee Action Item 6"* [5].

**FNP Unit 2 Compliance**

This Applicant/Licensee Action Item is not applicable to FNP Unit 2 since it only applies to B&W plants.

**Conclusion**

Applicant/Licensee Action Item 6 of the NRC SE on MRP-227, Revision 0 is not applicable to FNP Unit 2.

### 6.2.7 SE Applicant/Licensee Action Item 7: Plant-Specific Evaluation of CASS Materials

*"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" [5].*

**FNP Unit 2 Compliance**

The NRC final SE on MRP-227, subsection 3.3.7 [5] states that, for assessment of CASS materials, the applicant/licensee for license renewal may apply the criteria in the NRC letter of May 19, 2000, "License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components" [36] as the basis for determining whether the CASS materials are susceptible to the thermal aging mechanism. If the application of the applicable screening criteria for the components material demonstrates that the components are not susceptible to either TE or IE, or to the synergistic effects of TE and IE combined, then no other evaluation would be necessary.

The FNP Unit 2, the mixing devices, upper instrumentation conduit and supports (stops and gussets), upper support column assemblies – bases (mixer and orifice base) and bottom-mounted instrumentation (BMI) column assemblies – cruciform (standard and special) are CASS.

For each of the CASS components, the elemental percentages from the chemical data retrieved from CMTRs for the CASS component are input into Hull's formula (per guidance of NUREG/CR-4513 [37]) to calculate the delta ferrite content of the CASS material. The CMTRs do not list the element percentage for nitrogen; thus, per the guidance of NUREG/CR-4513, nitrogen is assumed to be 0.04 percent [37]. The CMTRs do not list an elemental percentage for molybdenum. A-351, Grade CF8 did not have a requirement for percent molybdenum in 1974. The 2013 Edition of the ASME Code has SA-351, Grade CF8 chemistry requirements that specify a maximum of 0.5 percent molybdenum; thus, this maximum value is input into Hull's formula. Where CMTRs were not located, a conservative combination of ASME A351, Grade CF8 chemical requirements was input into Hull's formula. The results of the TE evaluation for the FNP Unit 2 CASS components are summarized in Table 6-2.

Based on the criteria of the NRC letter dated May 19, 2000 [36],

- The upper instrumentation conduit and supports (gussets) are shown as not susceptible to TE; however, the upper instrumentation conduit and supports (stops) are considered as potentially susceptible to TE.
- The mixing devices are not susceptible to TE.
- The upper support column – bases (mixing style) are not susceptible to TE.
- The upper support column – bases (orifice style) 12 of 13 are not susceptible to TE; one is considered as potentially susceptible to TE.
- Six of the BMI column cruciforms (standard) are not susceptible to TE. The BMI column cruciforms (special) are considered as potentially susceptible to TE.

All the above components were considered in MRP-191 and were screened for susceptibility to material degradation, including consideration of TE and IE. With the exception of the upper instrumentation conduit and supports (stops, gussets, clamps and support blocks), the above components were screened as CASS and considered for TE in MRP-191. The assessment of the upper instrumentation conduit and supports (stops, gussets, clamps and support blocks), taking into consideration their potential susceptibility to TE and their impact on the FNP aging management strategy, is discussed in the response to A/LAI 2.

No martensitic SS or martensitic precipitation hardening (PH)-SS components were identified for the FNP Unit 2 reactor vessel internals.

**Conclusion**

It is concluded that continued application of the MRP-227-A [5] strategy will meet the requirement for managing age-related degradation of the FNP Unit 2 CASS reactor vessel internals components.

**Table 6-2. Summary of Joseph M. Farley Unit 2 CASS Components and Their Susceptibility to TE**

| CASS Component MRP-191 [9] Name | Material | Molybdenum Content (Percent) | Casting Method | Ferrite Content (Percent) | Susceptibility to TE (Based on NRC Letter [36]) |
|---|---|---|---|---|---|
| Mixing Devices | ASTM A351, Grade CF8 | 0.5 Maximum | Static | $\leq 20\%^{(1)}$ | Not susceptible(1) |
| Upper Support Column – Upper Instrumentation Conduit and Supports (Stops on Mixing Devices) | ASTM A351, Grade CF8 | 0.5 Maximum | Static | Possible $> 20\%^{(2)}$ | Potentially Susceptible(2) |
| Upper Support Column – Upper Instrumentation Conduit and Supports (Gussets on USC) | ASTM A351, Grade CF8 | 0.5 Maximum | Static | $\leq 20\%^{(1)}$ | Not susceptible(1) |
| Upper Support Column Assemblies, Column bases | ASTM A351, Grade CF8 | 0.5 Maximum | Static | 12 of 13 $\leq 20\%^{(1)}$ 1 of 13 Possible $> 20\%^{(2)}$ | 12 of 13 Not Susceptible(1) 1 of 13 Potentially Susceptible(2) |
| Upper Support Column Assemblies, mixer bases | ASTM A351, Grade CF8 | 0.5 Maximum | Static | $\leq 20\%^{(1)}$ | Not susceptible(1) |
| Bottom-Mounted Instrumentation (BMI) Column Assemblies, column cruciform (standard cruciform) | ASTM A351, Grade CF8 | 0.5 Maximum | Static | 6 of 24 $\leq 20\%^{(1)}$ Remaining Possible $> 20\%^{(2)}$ | 6 of 24 Not Susceptible(1) Remaining Potentially Susceptible(2) |
| Bottom-Mounted Instrumentation (BMI) Column Assemblies, column cruciform (special cruciform) | ASTM A351, Grade CF8 | 0.5 Maximum | Static | Possible $> 20\%^{(2)}$ | Potentially Susceptible(2) |

Notes:
1. Conclusion is based on CMTR chemistry data.
2. Where CMTR not located, conservative combination of ASME A351, Grade CF8 chemical requirements input into Hull's formula shows ferrite content can exceed 20 percent.

### 6.2.8 SE Applicant/Licensee Action Item 8: Submittal of Information for Staff Review and Approval

*"As addressed in Section 3.5.1 in this SE, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE. This is Applicant/Licensee Action Item 8"* [5].

**FNP Unit 2 Compliance**

FNP Unit 2, per the RIS [3], is considered a Category B plant that is expected to submit their RVI AMP based on the guidance of MRP-227-A, consistent with their commitments. Per the SER [2], FNP Unit 2 has a commitment to submit their AMP for approval by the NRC no later than March 31, 2019.

**Conclusion**

FNP Unit 2 complies with Applicant/Licensee Action Item 8 of the NRC SE on MRP-227, Revision 0, and therefore meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

# 7 PROGRAM ENHANCEMENT AND IMPLEMENTATION SCHEDULE

The requirements of MRP-227-A are based on an 18-month refueling cycle and consider both EFPY and cumulative operation. The information contained in Table 7-1 is based on inspection information requirements from MRP-227-A, and includes a description of the latest scope of inspection pertaining to the reactor internals AMP. Should a change occur in plant operational practices or operating experience result in changes to the projections, appropriate updates will be performed on affected plant documentation in accordance with approved procedures.

**Table 7-1. Aging Management Program Enhancement and Inspection Implementation Summary**

| Refueling Outage No. | Project Outage/Year | Estimated EFPY(2) | AMP-Related Scope(1) | Inspection Method and Criteria | Comments |
|---|---|---|---|---|---|
| 26 | Spring 2019 | 32.91 | ASME Code Section XI 10-Year ISI(4) | ASME Code Section XI | |
| | | | Initial MRP-227-A augmented inspections of the upper and lower core barrel flange welds, and the upper and lower core barrel cylinder girth welds. | MRP-227-A visual (EVT-1) inspection in accordance with MRP-228 specifications. | The initial inspection window for these components is no later than two refueling outages from the beginning of extended operation. While the inspections are planned for RO-27, FNP has the option to perform these inspections until RO-29. |
| | | | Initial MRP-227-A augmented inspections of guide plates (cards). | Inspect and measure in accordance with WCAP-17451 requirements. | The initial inspection window for the guide plates (cards) is no later than two refueling outages from the beginning of extended operation. FNP has the option to perform these inspections until RO-29. |

**Table 7-1. Aging Management Program Enhancement and Inspection Implementation Summary (cont.)**

| Refueling Outage No. | Project Outage/Year | Estimated EFPY(2) | AMP-Related Scope(1) | Inspection Method and Criteria | Comments |
|---|---|---|---|---|---|
| 26 (cont.) | Spring 2019 | 32.91 | Initial MRP-227-A augmented inspections of control rod guide tube lower flange welds. | MRP-227-A inspections in accordance with MRP-228 specifications. | The initial inspection window for the control rod guide tube lower flange welds is no later than two refueling outages from the beginning of extended operation. FNP has the option to perform these inspections until RO-29. |
| 27 | Fall 2020 | 34.31 | Not Applicable | Not Applicable | Extended period of operation begins at midnight on March 31, 2021. |
| 28 | Spring 2022 | 35.70 | Initial MRP-227-A augmented inspections for baffle-edge bolts and the baffle-former assembly completed before or during this outage.<br><br>Initial MRP-227-A augmented inspections of hold down spring. | MRP-227-A inspections in accordance with MRP-228 specifications.<br><br>Direct measurement of hold down spring. | The initial inspection window for baffle-edge bolts and the baffle-former assembly is between 20 and 40 EFPY. While the inspections are planned for RO-28, FNP has the option to perform these inspections until RO-31.<br><br>The initial inspection window for the hold down spring is within three cycles of the beginning of license renewal period. While the inspection is planned for RO-28, FNP has the option to perform this inspection until RO-30. |
| 29 | Fall 2023 | 37.10 | Not Applicable | Not Applicable | Not Applicable |
| 30 | Spring 2025 | 38.49 | Not Applicable | Not Applicable | Not Applicable |

**Table 7-1. Aging Management Program Enhancement and Inspection Implementation Summary (cont.)**

| Refueling Outage No. | Project Outage/Year | Estimated EFPY(2) | AMP-Related Scope(1) | Inspection Method and Criteria | Comments |
|---|---|---|---|---|---|
| 31 | Fall 2026 | 39.89 | Initial MRP-227-A augmented inspections for baffle-former bolts(3) completed before or during this outage. | MRP-227-A inspections in accordance with MRP-228 specifications. | The initial inspection window for the baffle-former bolts is between 25 and 35 EFPY. The replacement baffle bolts will be at approximately 25 EFPY at the time of inspection. A technical justification will document the acceptability of performing the inspection of the original bolts aged beyond 35 EFPY. |
| 32 | Spring 2028 | 41.28 | ASME Code Section XI 10-Year ISI(4) | ASME Code Section XI | |
| | | | Subsequent MRP-227-A augmented inspections of the upper and lower core barrel flange welds, and the upper and lower core barrel cylinder girth welds. | MRP-227-A visual (EVT-1) inspection in accordance with MRP-228 specifications. | The subsequent inspection window for these components is ten years after the initial inspection. |
| | | | Subsequent MRP-227-A augmented inspections of guide plates (cards). | Inspect and measure in accordance with WCAP-17451 requirements. | The subsequent inspection window for these components is ten years after the initial inspection. |
| | | | Subsequent MRP-227-A augmented inspections of control rod guide tube lower flange welds. | MRP-227-A inspections in accordance with MRP-228 specifications. | The subsequent inspection window for these components is ten years after the initial inspection. |
| 33 | Fall 2029 | 42.68 | Not Applicable | Not Applicable | Not Applicable |

**Table 7-1. Aging Management Program Enhancement and Inspection Implementation Summary (cont.)**

| Refueling Outage No. | Project Outage/Year | Estimated EFPY(2) | AMP-Related Scope(1) | Inspection Method and Criteria | Comments |
|---|---|---|---|---|---|
| 34 | Spring 2031 | 44.07 | Subsequent MRP-227-A augmented inspections for baffle-edge bolts and the baffle-former assembly completed before or during this outage. | MRP-227-A inspections in accordance with MRP-228 specifications. | The subsequent inspection window for these components is 10 years after the initial inspection. |
| 35 | Fall 2032 | 45.47 | Not Applicable | Not Applicable | Not Applicable |
| 36 | Spring 2034 | 46.86 | Not Applicable | Not Applicable | Not Applicable |
| 37 | Fall 2035 | 48.26 | Subsequent MRP-227-A augmented inspections for baffle-former bolts completed before or during this outage. | MRP-227-A inspections in accordance with MRP-228 specifications. | The subsequent inspection window for these components is 10 years after the initial inspection. |
| 38 | Spring 2037 | 49.65 | ASME Code Section XI 10-Year ISI(4) | ASME Code Section XI | |
| 39 | Fall 2038 | 51.05 | Not Applicable | Not Applicable | Not Applicable |
| 40 | Spring 2040 | 52.44 | Not Applicable | Not Applicable | Not Applicable |
| N/A | N/A | N/A | Not Applicable | Not Applicable | Renewed Operating License expires March 31, 2041 |

**Table 7-1. Aging Management Program Enhancement and Inspection Implementation Summary (cont.)**

| Refueling Outage No. | Project Outage/Year | Estimated EFPY(2) | AMP-Related Scope(1) | Inspection Method and Criteria | Comments |
|---|---|---|---|---|---|

**Notes:**

1. Future refueling outage plans are subject to change due to considerations to coordinate and optimize outage refueling activities.
2. From the EFPY estimates provided in [1] each calendar year is the equivalent of 0.93 EFPY. FNP Unit 2 is at 25 EFPY during the Fall of 2010.
3. A portion of the baffle-former bolts were replaced during the Fall 1999 Outage. Therefore, at the time of the Fall 2026 outage the original baffle-former bolts will be at approximately 40 EFPY while the replacement baffle-former bolts will be at approximately 25 EFPY. A technical justification will document the acceptability of performing the MRP-227-A inspection during this outage with the original bolts aged beyond 35 EFPY.
4. ASME Section XI rules are followed for the In-Service Inspections, which allows for adjustment from the 10-year subsequent inspection requirement in order to align with a scheduled plant outage. The subsequent ASME Section XI inspection dates provided in this table could be adjusted as a result, but will comply with the Code.

# 8 IMPLEMENTING DOCUMENTS

As noted within this AMP document, the FNP Unit 2 PWR Vessel Internals Program is documented in [1]. The FNP Unit 2 AMP also references the Water Chemistry Program and the ASME Section XI Inservice Inspection, subsections IWB, IWC and IWD Program. MRP-227-A augmented examinations (Appendix C), recommended as a result of industry programs, will be included in the existing ASME Section XI program. SNC has also developed a fleet NDE procedure NMP-ES-024-112 [38] "Materials Reliability Program (MRP) MRP-228 Implementation PWR RPV Internals Inspections" to establish a process for implementing the requirements of MRP-228.

SNC documents associated with the existing FNP programs and considered to be implementing documents of the PWR Vessel Internals Program are:

- NMP-CH-100-GL01, "Farley Primary Water Chemistry Strategic Plan" [18]
- FNP-0-SYP-22.0, "Flux Thimble Tube Examination Program" [19]
- NMP-ES-018, "ASME Section XI ISI Program" [4]
- NMP-ES-029, "PWR Primary System Integrity" [34]
- NMP-ES-024-112, "Materials Reliability Program (MRP) MRP-228 Implementation PWR RPV Internals Inspections" [38]

The RVI AMP relies on the Water Chemistry Program for maintaining high water purity to reduce susceptibility to cracking due to SCC. Additional procedures may be updated or created as OE for augmented examinations is accumulated.

Based on this information, the AMP for FNP Unit 2 RVI provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

# 9 REFERENCES


1. Southern Nuclear Company Procedure, NMP-ES-029-GL02, "PWR Reactor Vessel Internals Program Strategic Plan," Version 3.0.

2. U.S. Nuclear Regulatory Commission, NUREG-1825, "Safety Evaluation Report Related to the License Renewal of Joseph M. Farley, Units 1 and 2," Docket Nos. 50-348 and 50-364, Southern Nuclear Operating Company, Inc., May 2005.

3. U.S. Nuclear Regulatory Commission Document, ML111990086, "NRC Regulatory Issue Summary 2011-07 License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," July 21, 2011.

4. Southern Nuclear Company Procedure, NMP-ES-018, "SNC Inservice Inspection Engineering Program."

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

6. U.S. Nuclear Regulatory Commission, Code of Federal Regulations, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Washington, D.C., Federal Register, Volume 77, No. 39907, dated May 8, 1995 and last updated on July 6, 2012.

7. U.S. Nuclear Regulatory Commission Document, NUREG-1800, Rev. 2, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (SRP-LR)," December 2010.

8. Westinghouse Report, WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," March 2001.

9. *Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006. 1013234.

10. *Materials Reliability Program: Inspection Standard for PWR Internals – 2012 Update (MRP-228, Rev. 1).* EPRI, Palo Alto, CA: 2012. 1025147.

11. Westinghouse Report, WCAP-17096-NP, Rev. 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009.

12. U.S. Nuclear Regulatory Commission Document, NUREG-1801, Rev. 0, "Generic Aging Lessons Learned (GALL) Report," July 2001.

13. Nuclear Energy Institute Document, NEI 03-08, Rev. 2, "Guideline for the Management of Materials Issues," Washington, D.C., January 2010.

14. Southern Nuclear Company Procedure, NMP-ES-009, "Engineering Programs."

15. Southern Nuclear Company Procedure, NMP-GM-003, "Self Assessments and Benchmark Procedure."

16. Southern Nuclear Company Procedure, NMP-ES-063-GL02, "Farley License Renewal Program Manual," Version 1.2, June 2015.

17. U.S. Nuclear Regulatory Commission Document, NUREG-1801, Rev. 2, "Generic Aging Lessons Learned (GALL) Report," December 2010.

18. Southern Nuclear Company Procedure, NMP-CH-100-GL01, "Farley Primary Water Chemistry Strategic Plan."

19. Farley Nuclear Plant Procedure, FNP-0-SYP-22.0, "Flux Detector Thimble Inspection Program," Version 1.0, April 10, 2006. (Implemented via Preventative Maintenance Activity N2C56007)

20. Farley Unit 2 Field Change Notice, APRO-40576, "Control Rod Guide Tube Support Pin," November 1999.

21. Farley Nuclear Plant Procedure, CGR-RVI-101, "Plant Farley License Renewal Commodity Group Review: Reactor Vessel Internals," Rev. 0, November 15, 2005.

22. ASME Boiler and Pressure Vessel Code Section XI, 2001 Edition through 2003 Addenda.

23. SNC Report, "Joseph M. Farley License Renewal Application," September 2003 (ADAMS Accession Nos. ML032721356, ML032721360).

24. U.S. NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 26, 1988.

25. *Pressurized Water Reactor Primary Water Chemistry Guidelines, Revision 7*, EPRI, Palo Alto, CA: 2014. 3002000505.

26. Southern Nuclear Company Procedure, "Corrective Action Program," NMP-GM-002.

27. Farley Nuclear Plant Procedure, LR-2-12, "Plant Farley Commodity Review Procedure," Version 1.0.

28. Southern Nuclear Company, Inc., "Quality Assurance Topical Report," Version 14.0, June 2, 2015.

29. U.S. Nuclear Regulatory Commission Information Notice 84-18, "Stress Corrosion Cracking in Pressurized Water Reactor Systems," March 7, 1984.

30. U.S. Nuclear Regulatory Commission Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants," March 25, 1998.

31. Southern Nuclear Company Procedure, "Operating Experience Program," NMP-GM-008.

32. *Closed Cooling Water Chemistry Guideline, Revision 0*, EPRI, Palo Alto, CA: 2013. 3002000590.

33. *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals Components (MRP-232, Rev. 1).* EPRI, Palo Alto, CA: 2012. 1021029.

34. Southern Nuclear Company Procedure, NMP-ES-029, "PWR Primary System Integrity Program," Version 8.1, March 6, 2014.

35. Updated Final Safety Analysis Report (UFSAR) FNP-FSAR-4, Rev. 26, December 2014.

36. U.S. Nuclear Regulatory Commission Letter, "License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," May 19, 2000. (NRC ADAMS Accession No. ML003717179)

37. U.S. Nuclear Regulatory Commission, NUREG/CR-4513, Rev. 1, "Estimation of Fracture Toughness of Cast Stainless Steel During Thermal Aging in LWR Systems," August 1994. (NRC ADAMS Accession No. ML052360554)

38. Southern Nuclear Company Procedure, NMP-ES-024-112, "Materials Reliability Program (MRP) MRP-228 Implementation PWR RPV Internals Inspections."

39. Not Used.

40. Not Used.

41. Westinghouse Report, WCAP-15664, Rev. 0, "Determination of Acceptable Baffle-Barrel-Bolting for Three-Loop Westinghouse 15x15 Downflow and 17x17 Standard Upflow Domestic Plants," December 2001. (Westinghouse Proprietary Class 2)

42. Farley Unit 2 Field Change Notice, APRO-40577, "Replacement Baffle Bolts," March 2000.

43. Westinghouse Report, WCAP-14723, "Farley Nuclear Power Plant Units 1 and 2 Power Uprate Project NSSS Licensing Report," January 1997.

44. NRC SER on Farley Units 1 and 2 License Amendment No. 129, April 29, 1998. (ADAMS Accession No. ML012140259)

45. Westinghouse Letter, ALA-02-88, Rev. 0, "Transmittal of Reactor Internals Upflow Conversion Final Field Service Report – 2R15," November 11, 2002.

46. Farley Nuclear Plant Design Change Package (DCP) No. 98-2-9345, "Baffle Bolt Replacement (Unit 2)."

47. Farley Nuclear Plant Design Change Package (DCP) No. 97-2-9190 (Unit 2 Split Pin Replacement).

# APPENDIX A
# ILLUSTRATIONS

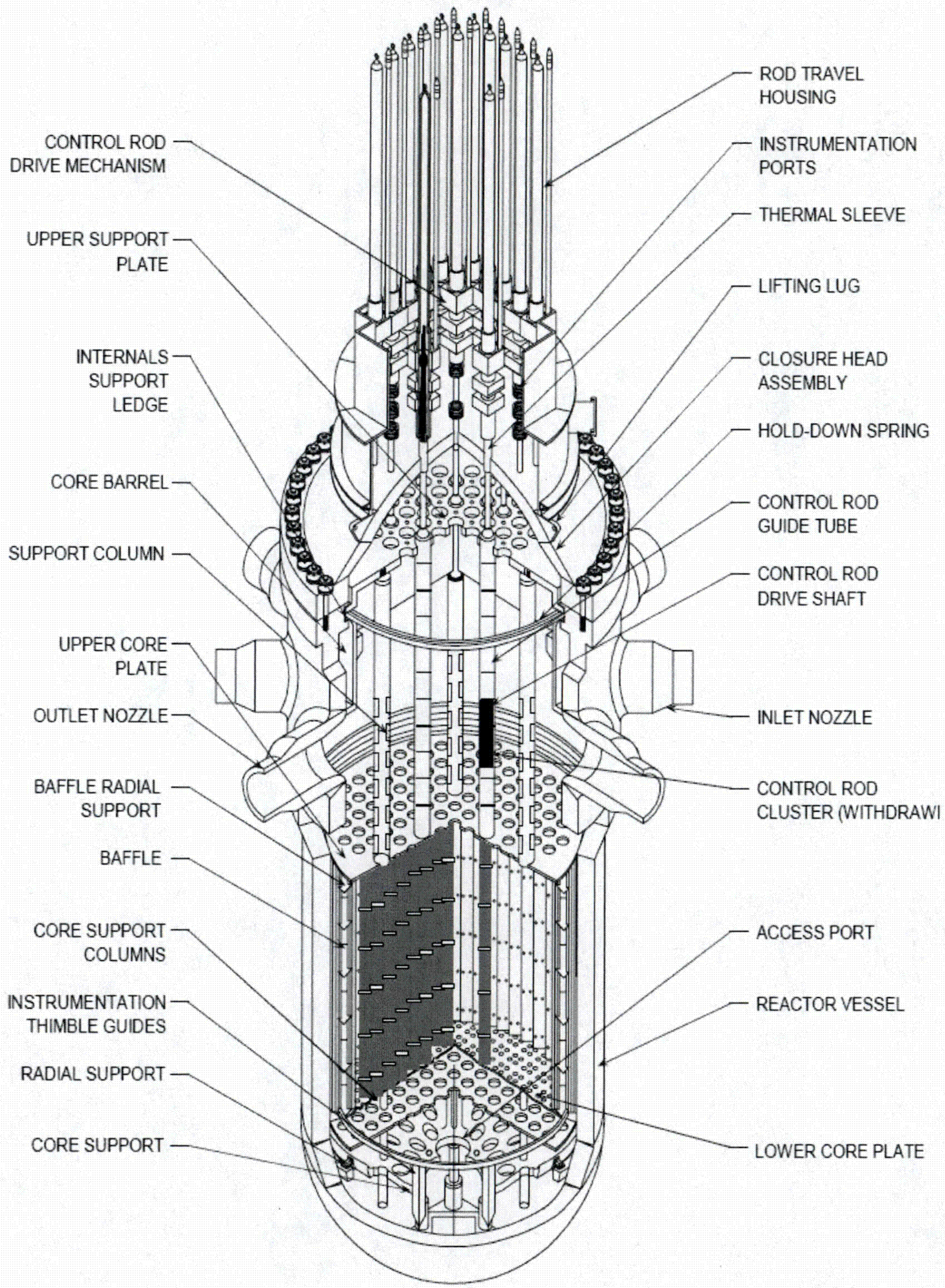


**Figure A-1. Illustration of Typical Westinghouse Internals Assembly**

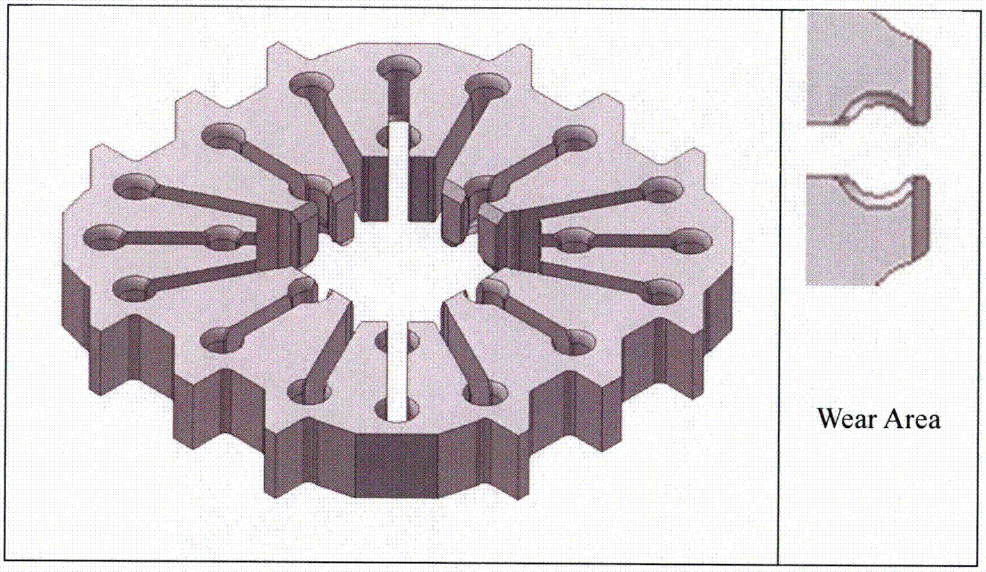


**Figure A-2. Typical Westinghouse Control Rod Guide Card**

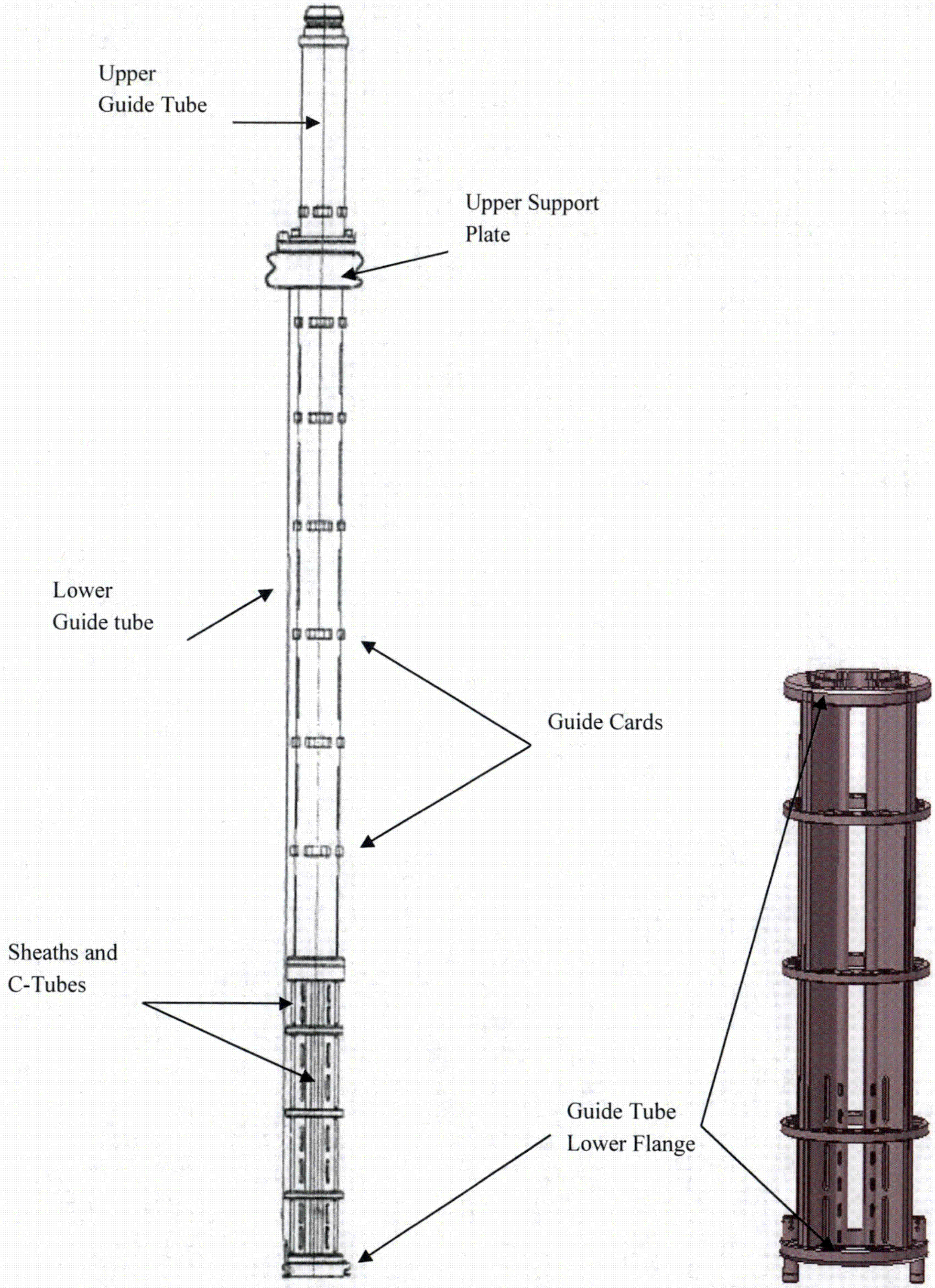


**Figure A-3. Typical Lower Section of Control Rod Guide Tube Assembly**

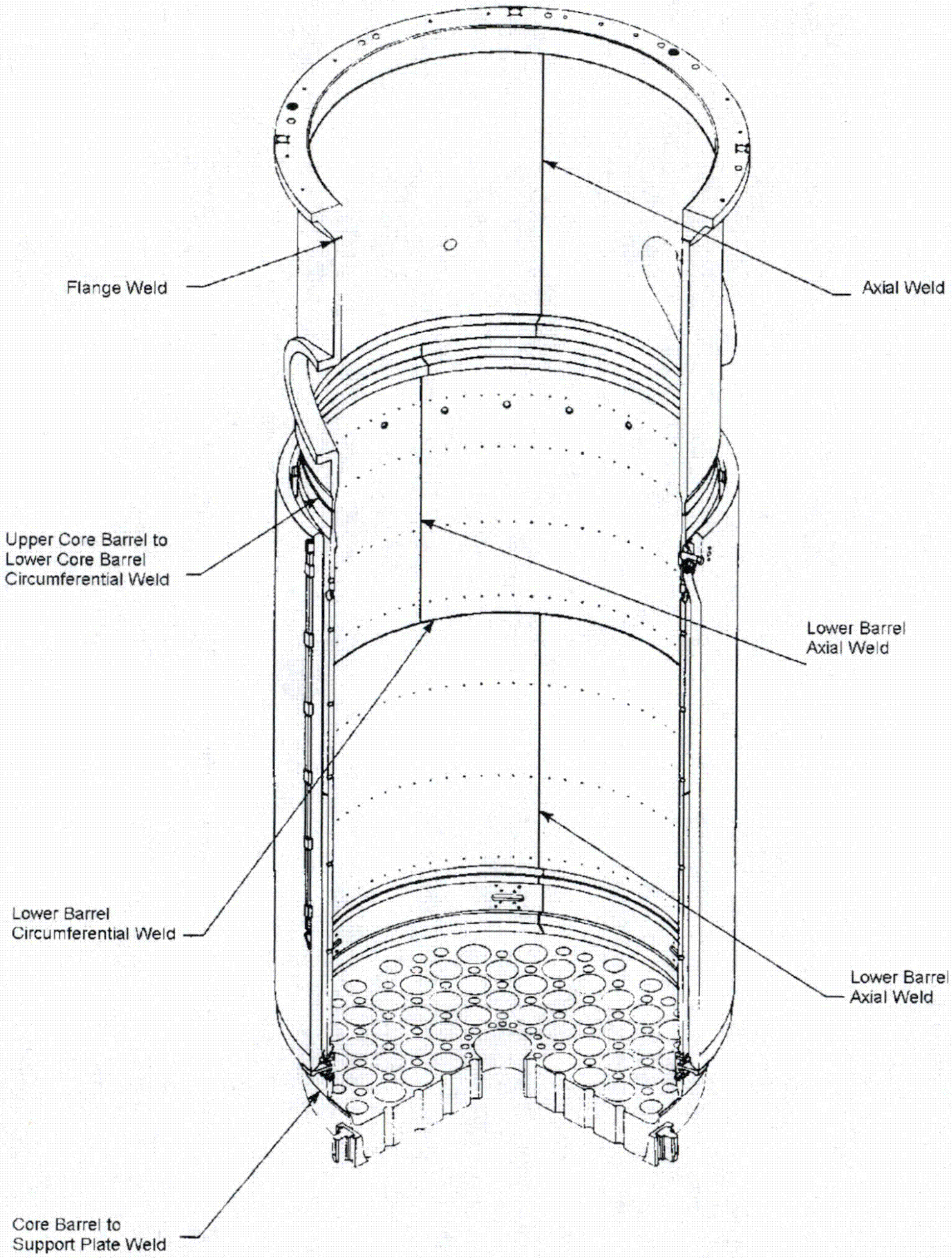


**Figure A-4. Major Core Barrel Welds**

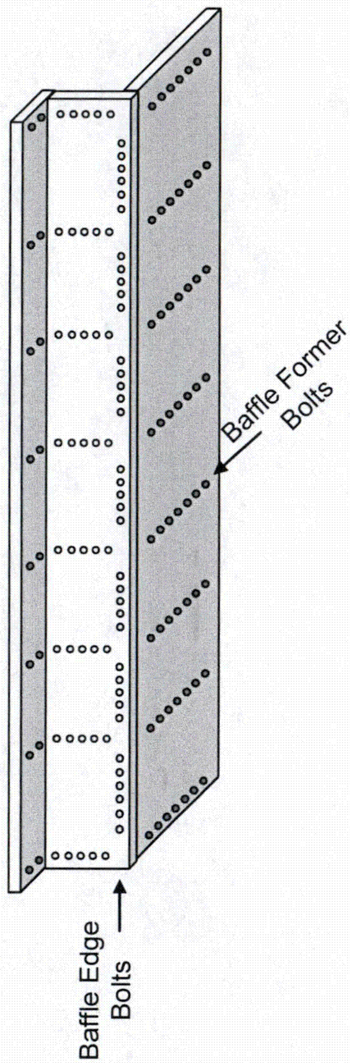


**Figure A-5. Bolting Systems used in Westinghouse Core Baffles**

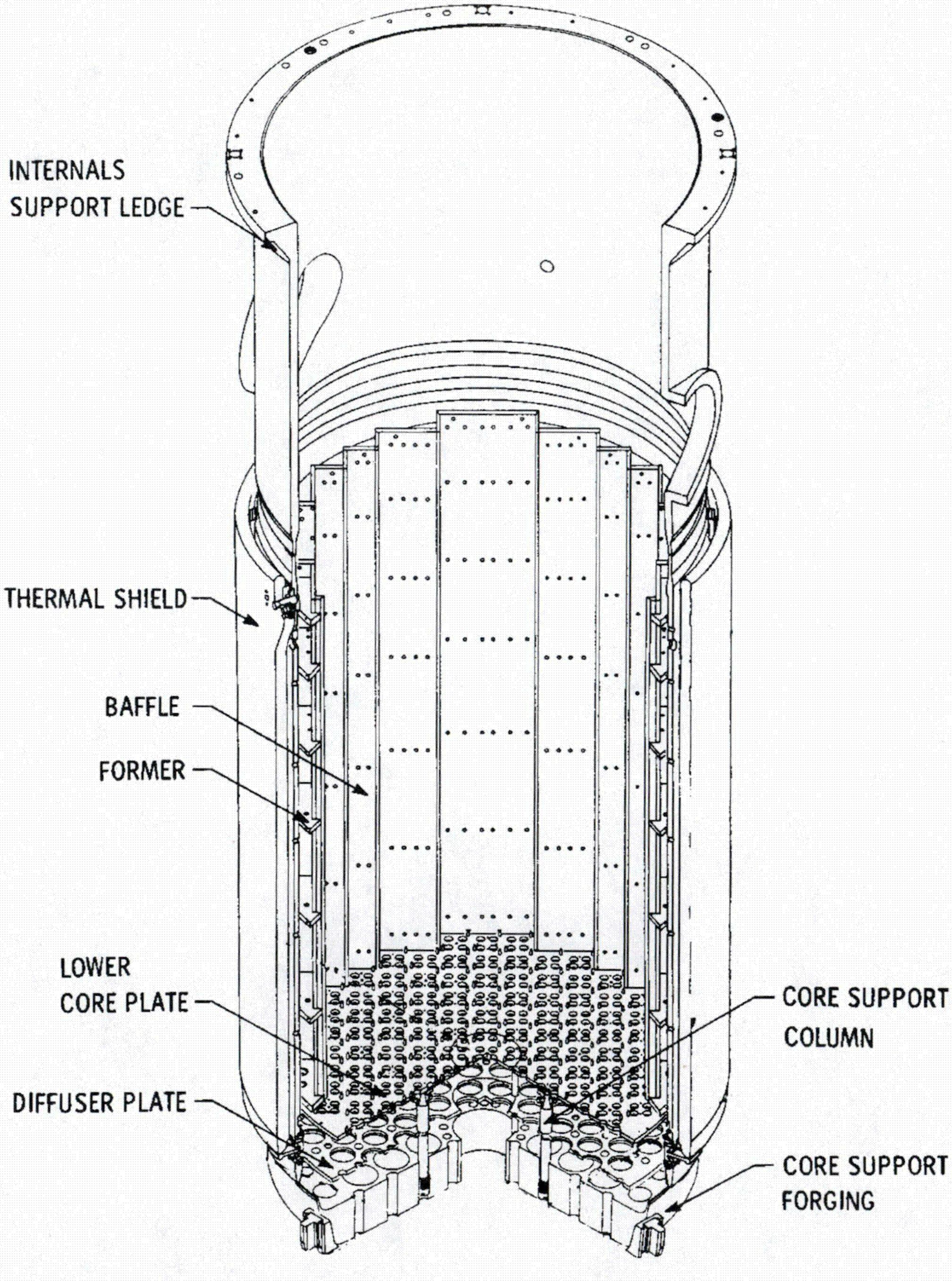


**Figure A-6. Core Baffle/Barrel Structure**

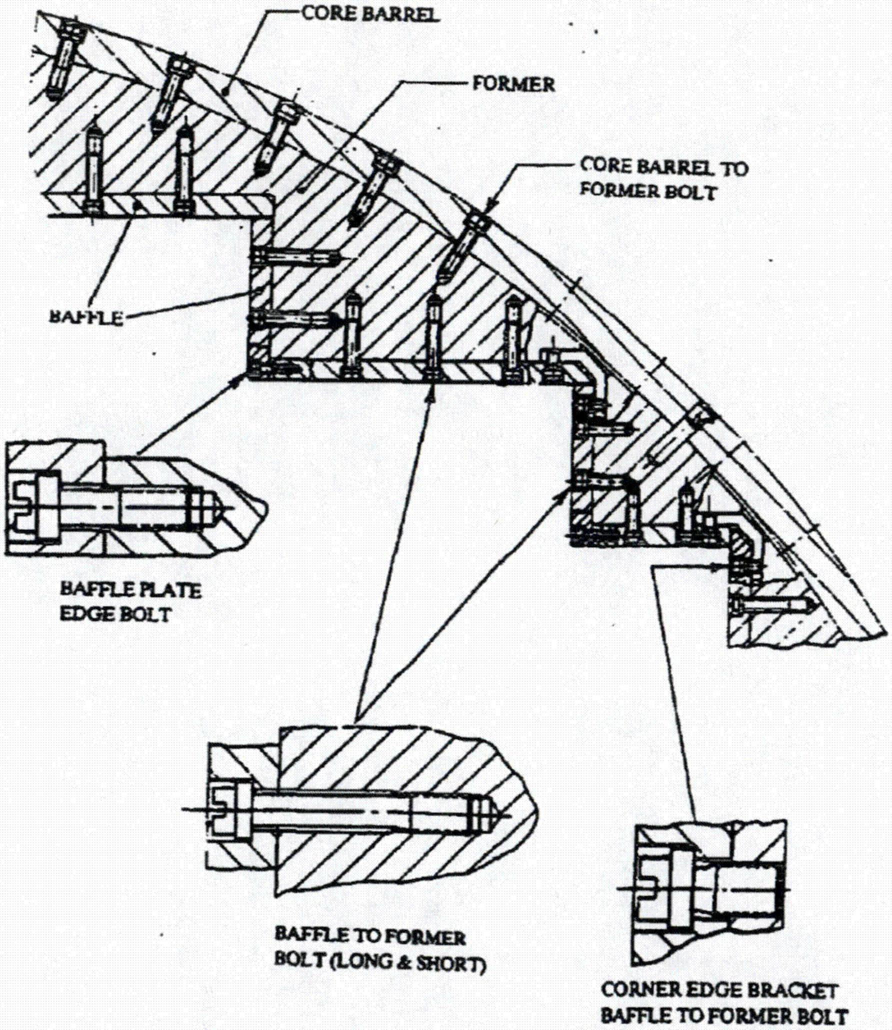


**Figure A-7. Bolting in a Typical Westinghouse Baffle-Former Structure**

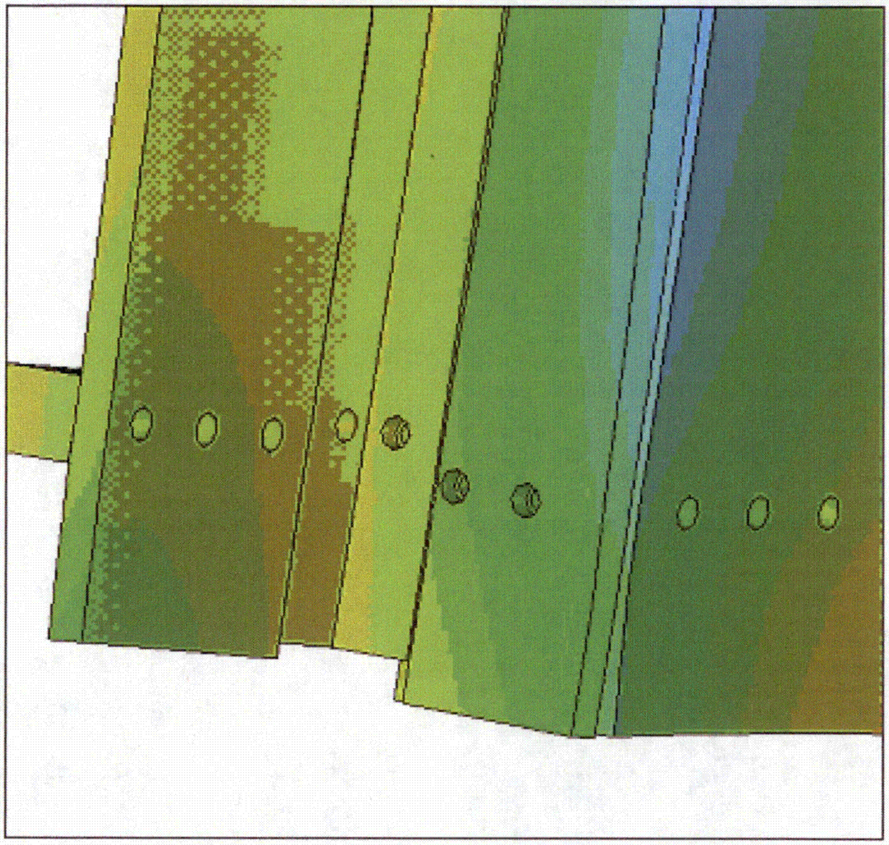

Figure A-8. Vertical Displacement between the Baffle Plates and Bracket at the Bottom of the Baffle-Former-Barrel Assembly

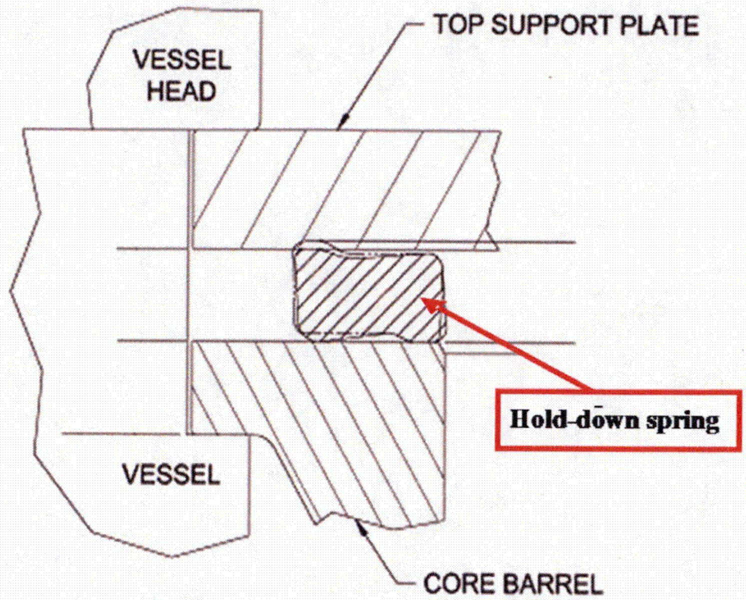


**Figure A-9. Schematic Cross-Sections of the Westinghouse Hold Down Springs**

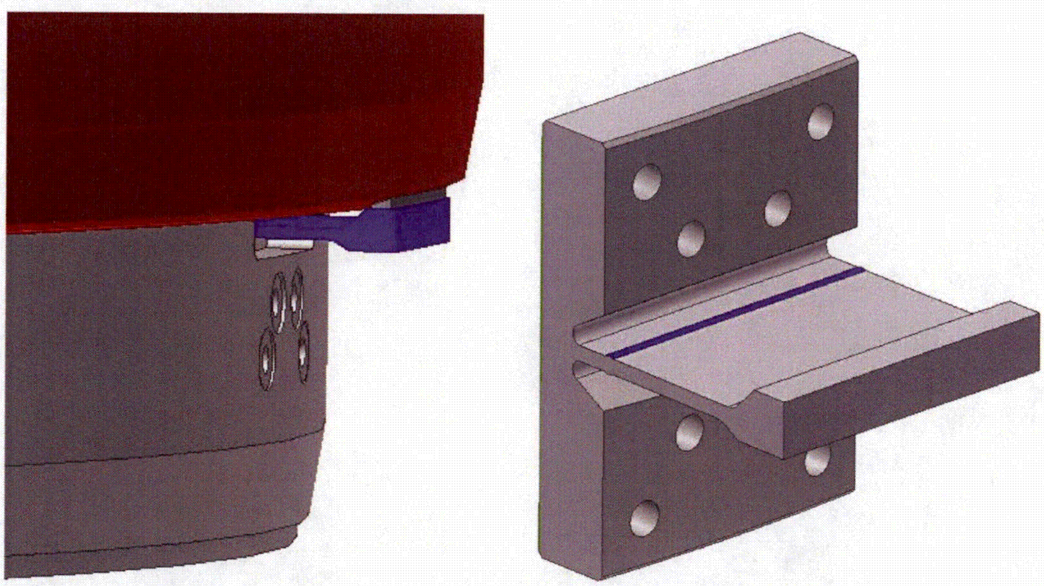

**Figure A-10. Typical Thermal Shield Flexure**

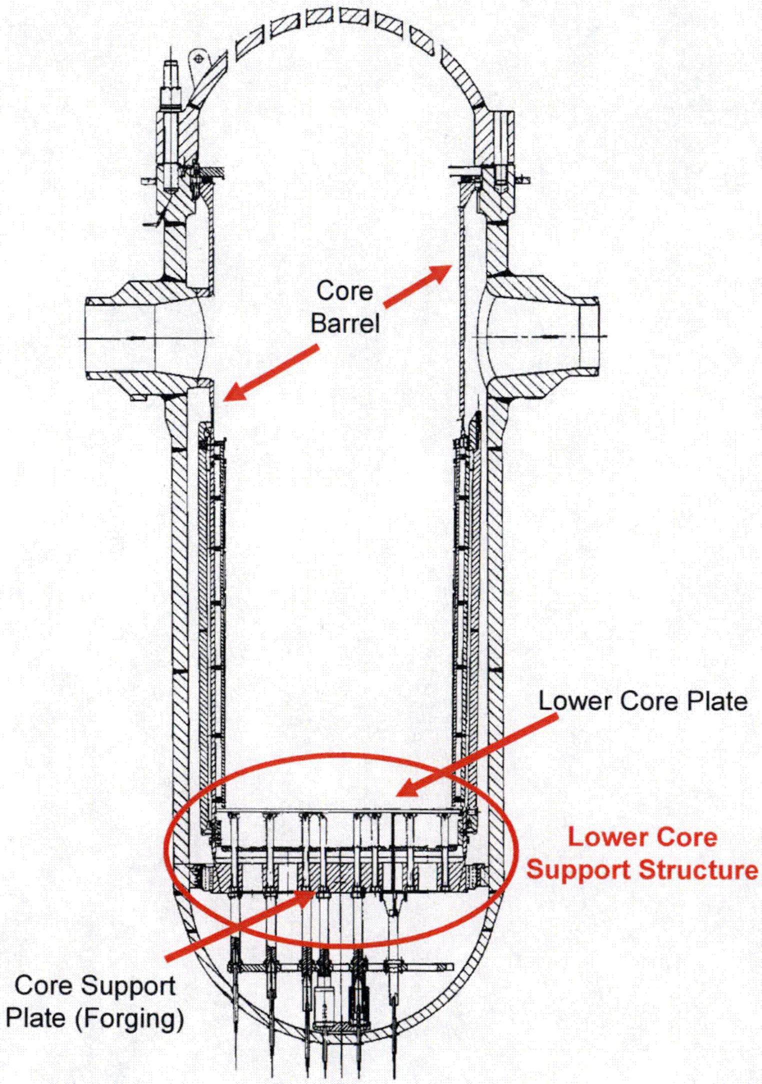


**Figure A-11. Lower Core Support Structure**

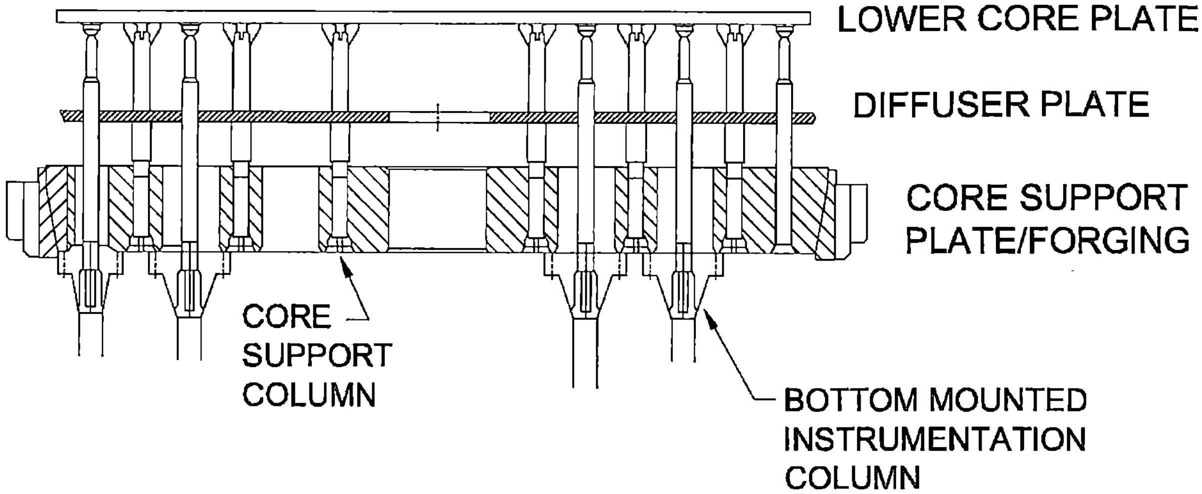


**Figure A-12. Lower Core Support Structure – Core Support Plate Cross-Section**

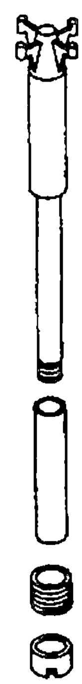

**Figure A-13. Typical Core Support Column**

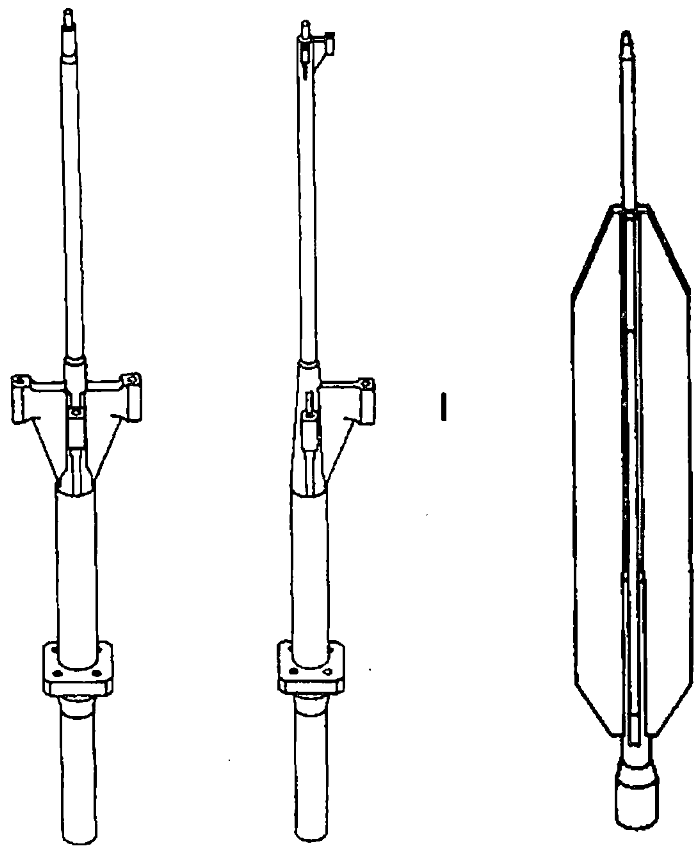

**Figure A-14. Examples of BMI Column Designs**

# APPENDIX B
# FARLEY UNIT 2 LICENSE RENEWAL AGING MANAGEMENT REVIEW SUMMARY TABLE

The content in Table B-1 of Appendix B is extracted from Table 3.1.2-2 of the license renewal application approved by the NRC.

**Table B-1. LRA Aging Management Review Summary Table 3.1.2-2**
**Farley Nuclear Plant LRA**

| Component Type | Aging Effect Requiring Management | Aging Management Program[1] |
|---|---|---|
| 1. Baffle and Former Plates | Change in Material Properties | Reactor Vessel Internals Program (B.5.1) |
| 2. Baffle and Former Plates | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 3. Baffle and Former Plates | Cracking | Water Chemistry Control Program (B.3.2) |
| 4. Baffle and Former Plates | Loss of Fracture Toughness | Reactor Vessel Internals Program (B.5.1) |
| 5. Baffle and Former Plates | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 6. Baffle Bolts | Change in Material Properties | Reactor Vessel Internals Program (B.5.1) |
| 7. Baffle Bolts | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 8. Baffle Bolts | Cracking | Water Chemistry Control Program (B.3.2) |
| 9. Baffle Bolts | Loss of Fracture Toughness | Reactor Vessel Internals Program (B.5.1) |
| 10. Baffle Bolts | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 11. Baffle Bolts | Loss of Preload/Stress Relaxation | Reactor Vessel Internals Program (B.5.1) |
| 12. Baffle Bolts | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 13. BMI Column Cruciforms | Cracking | Water Chemistry Control Program (B.3.2) |
| 14. BMI Column Cruciforms | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 15. BMI Column Cruciforms | Loss of Fracture Toughness | Reactor Vessel Internals Program (B.5.1) |

**Table B-2. LRA Aging Management Review Summary Table 3.1.2-2
Farley Nuclear Plant LRA (cont.)**

| Component Type | Aging Effect Requiring Management | Aging Management Program[1] |
|---|---|---|
| 16. BMI Column Cruciforms | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 17. BMI Columns (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 18. BMI Columns (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 19. BMI Columns (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 20. BMI Columns (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 21. Clevis Inserts and Fasteners | Cracking | Water Chemistry Control Program (B.3.2) |
| 22. Clevis Inserts and Fasteners | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 23. Clevis Inserts and Fasteners | Loss of Material | Inservice Inspection Program (B.3.1) |
| 24. Clevis Inserts and Fasteners | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 25. Clevis Inserts and Fasteners | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 26. Control Rod Guide Tube Assemblies (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 27. Control Rod Guide Tube Assemblies (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 28. Control Rod Guide Tube Assemblies (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 29. Control Rod Guide Tube Assemblies (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 30. Core Barrel and Core Barrel Flange | Cracking | Water Chemistry Control Program (B.3.2) |
| 31. Core Barrel and Core Barrel Flange | Cracking | Reactor Vessel Internals Program (B.5.1) |

**Table B-2. LRA Aging Management Review Summary Table 3.1.2-2 Farley Nuclear Plant LRA (cont.)**

| Component Type | Aging Effect Requiring Management | Aging Management Program(1) |
|---|---|---|
| 32. Core Barrel and Core Barrel Flange | Loss of Fracture Toughness | Reactor Vessel Internals Program (B.5.1) |
| 33. Core Barrel and Core Barrel Flange | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 34. Core Barrel Outlet Nozzles | Cracking | Water Chemistry Control Program (B.3.2) |
| 35. Core Barrel Outlet Nozzles | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 36. Core Barrel Outlet Nozzles | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 37. CRGT Support Pins | Cracking | Water Chemistry Control Program (B.3.2) |
| 38. CRGT Support Pins | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 39. CRGT Support Pins | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 40. CRGT Support Pins | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 41. Flux Thimble Tubes | Cracking | Water Chemistry Control Program (B.3.2) |
| 42. Flux Thimble Tubes | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 43. Flux Thimble Tubes | Loss of Material | Flux Detector Thimble Inspection Program (B.5.2) |
| 44. Flux Thimble Tubes | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 45. Head/RPV Alignment Pins (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 46. Head/RPV Alignment Pins (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 47. Head/RPV Alignment Pins (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 48. Head/RPV Alignment Pins (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 49. Head Cooling Spray Nozzles | Cracking | Water Chemistry Control Program (B.3.2) |
| 50. Head Cooling Spray Nozzles | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 51. Head Cooling Spray Nozzles | Loss of Material | Water Chemistry Control Program (B.3.2) |

**Table B-2. LRA Aging Management Review Summary Table 3.1.2-2**
**Farley Nuclear Plant LRA (cont.)**

| Component Type | Aging Effect Requiring Management | Aging Management Program(1) |
|---|---|---|
| 52. HJTC Probe Holder Extension, and Probe Holder Shroud Assemblies (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 53. HJTC Probe Holder Extension, and Probe Holder Shroud Assemblies (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 54. HJTC Probe Holder Extension, and Probe Holder Shroud Assemblies (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 55. HJTC Probe Holder Extension, and Probe Holder Shroud Assemblies (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 56. Internals Holddown Spring | Cracking | Water Chemistry Control Program (B.3.2) |
| 57. Internals Holddown Spring | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 58. Internals Holddown Spring | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 59. Internals Holddown Spring | Loss of Material | Inservice Inspection Program (B.3.1) |
| 60. Internals Holddown Spring | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 61. Lower Core Plate and Fuel Alignment Pins (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 62. Lower Core Plate and Fuel Alignment Pins (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 63. Lower Core Plate and Fuel Alignment Pins (with associated fasteners) | Loss of Fracture Toughness | Reactor Vessel Internals Program (B.5.1) |
| 64. Lower Core Plate and Fuel Alignment Pins (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |

**Table B-2. LRA Aging Management Review Summary Table 3.1.2-2 Farley Nuclear Plant LRA (cont.)**

| Component Type | Aging Effect Requiring Management | Aging Management Program[1] |
|---|---|---|
| 65. Lower Core Plate and Fuel Alignment Pins (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 66. Lower Support Columns (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 67. Lower Support Columns (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 68. Lower Support Columns (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 69. Lower Support Columns (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 70. Lower Support Forging | Cracking | Water Chemistry Control Program (B.3.2) |
| 71. Lower Support Forging | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 72. Lower Support Forging | Loss of Fracture Toughness | Reactor Vessel Internals Program (B.5.1) |
| 73. Lower Support Forging | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 74. Neutron Panels (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 75. Neutron Panels (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 76. Neutron Panels (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 77. Neutron Panels (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 78. Radial Support Keys and Fasteners | Cracking | Water Chemistry Control Program (B.3.2) |
| 79. Radial Support Keys and Fasteners | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 80. Radial Support Keys and Fasteners | Loss of Material | Inservice Inspection Program (B.3.1) |
| 81. Radial Support Keys and Fasteners | Loss of Material | Water Chemistry Control Program (B.3.2) |

**Table B-2. LRA Aging Management Review Summary Table 3.1.2-2
Farley Nuclear Plant LRA (cont.)**

| Component Type | Aging Effect Requiring Management | Aging Management Program(1) |
|---|---|---|
| 82. Radial Support Keys and Fasteners | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 83. Secondary Core Support Assembly (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 84. Secondary Core Support Assembly (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 85. Secondary Core Support Assembly (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 86. Secondary Core Support Assembly (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 87. Upper Core Plate Alignment Pins (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 88. Upper Core Plate Alignment Pins (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 89. Upper Core Plate Alignment Pins (with associated fasteners) | Loss of Material | Inservice Inspection Program (B.3.1) |
| 90. Upper Core Plate Alignment Pins (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 91. Upper Core Plate Alignment Pins (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 92. Upper Core Plate and Fuel Alignment Pins (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 93. Upper Core Plate and Fuel Alignment Pins (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 94. Upper Core Plate and Fuel Alignment Pins (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |

**Table B-2. LRA Aging Management Review Summary Table 3.1.2-2 Farley Nuclear Plant LRA (cont.)**

| Component Type | Aging Effect Requiring Management | Aging Management Program[1] |
|---|---|---|
| 95. Upper Core Plate and Fuel Alignment Pins (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 96. Upper Instrumentation Conduit and Supports (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 97. Upper Instrumentation Conduit and Supports (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 98. Upper Instrumentation Conduit and Supports (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 99. Upper Instrumentation Conduit and Supports (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 100. Upper Support Assembly (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 101. Upper Support Assembly (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 102. Upper Support Assembly (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 103. Upper Support Assembly (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |
| 104. Upper Support Column Bases | Cracking | Water Chemistry Control Program (B.3.2) |
| 105. Upper Support Column Bases | Cracking | Reactor Vessel Internals Program (B.5.1) |
| 106. Upper Support Column Bases | Loss of Fracture Toughness | Reactor Vessel Internals Program (B.5.1) |
| 107. Upper Support Column Bases | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 108. Upper Support Columns (with associated fasteners) | Cracking | Water Chemistry Control Program (B.3.2) |
| 109. Upper Support Columns (with associated fasteners) | Cracking | Reactor Vessel Internals Program (B.5.1) |

**Table B-2. LRA Aging Management Review Summary Table 3.1.2-2 Farley Nuclear Plant LRA (cont.)**

| Component Type | Aging Effect Requiring Management | Aging Management Program[1] |
|---|---|---|
| 110. Upper Support Columns (with associated fasteners) | Loss of Material | Water Chemistry Control Program (B.3.2) |
| 111. Upper Support Columns (with associated fasteners) | Loss of Preload/Stress Relaxation | Inservice Inspection Program (B.3.1) |

Notes:

1. Information in parentheses are the Appendix B section numbers in the Farley LRA.

# APPENDIX C
# MRP-227-A AUGMENTED INSPECTIONS

**Table C-1. MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals**

| Item | Applicability | Effect (Mechanism) | Expansion Link(1) | Examination Method/Frequency(1) | Examination Coverage |
|---|---|---|---|---|---|
| **Control Rod Guide Tube Assembly**<br>Guide plates (cards) | All plants | Loss of Material (Wear) | None | Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval. | 20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined.<br>See Figure A-2. |
| **Control Rod Guide Tube Assembly**<br>Lower flange welds | All plants | Cracking (SCC, Fatigue)<br>Aging Management (IE and TE) | Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast), Upper core plate, Lower support forging/casting | Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval. | 100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies(2).<br>See Figure A-3. |
| **Core Barrel Assembly**<br>Upper core barrel flange weld | All plants | Cracking (SCC) | Lower support column bodies (non-cast)<br>Core barrel outlet nozzle welds | Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval. | 100% of one side of the accessible surfaces of the selected weld and adjacent base metal(4).<br>See Figure A-4. |

**Table C-1. MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Effect (Mechanism) | Expansion Link(1) | Examination Method/Frequency(1) | Examination Coverage |
|---|---|---|---|---|---|
| **Core Barrel Assembly**<br>Upper and lower core barrel cylinder girth welds | All plants | Cracking (SCC, IASCC, Fatigue) | Upper and lower core barrel cylinder axial welds | Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval. | 100% of one side of the accessible surfaces of the selected weld and adjacent base metal(4).<br>See Figure A-4. |
| **Core Barrel Assembly**<br>Lower core barrel flange weld(5) | All plants | Cracking (SCC, Fatigue) | None | Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examinations on a ten-year interval. | 100% of one side of the accessible surfaces of the selected weld and adjacent base metal(4). |

**Table C-1. MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Effect (Mechanism) | Expansion Link(1) | Examination Method/Frequency(1) | Examination Coverage |
|---|---|---|---|---|---|
| **Baffle-Former Assembly**<br>Baffle-edge bolts | All plants with baffle-edge bolts<br>Note:<br>FNP Unit 2 has baffle-edge bolts. | Cracking (IASCC, Fatigue) that results in<br>• Lost or broken locking devices<br>• Failed or missing bolts<br>• Protrusion of bolt heads<br>Aging Management (IE and ISR)(6) | None | Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval. | Bolts and locking devices on high-fluence seams. 100% of components accessible from core side(3).<br>See Figures A-5, A-6 and A-7. |
| **Baffle-Former Assembly**<br>Baffle-former bolts | All plants | Cracking (IASCC, Fatigue)<br>Aging Management (IE and ISR)(6) | Lower support column bolts, Barrel-former bolts | Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.<br>Note: Farley Unit 2 will perform a baseline examination of the replacement baffle-former bolts within this EFPY range. | 100% of accessible bolts(3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs.<br>See Figures A-5 and A-6. |

**Table C-1. MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Effect (Mechanism) | Expansion Link(1) | Examination Method/Frequency(1) | Examination Coverage |
|---|---|---|---|---|---|
| **Baffle-Former Assembly**<br>Assembly<br>(Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates) | All plants | Distortion (Void Swelling), or Cracking (IASCC) that results in:<br>• Abnormal interaction with fuel assemblies<br>• Gaps along high fluence baffle joint<br>• Vertical displacement of baffle plates near high fluence joint<br>• Broken or damaged edge bolt locking systems along high fluence baffle joints | None | Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval. | Core side surface, as indicated.<br>See Figure A-8. |

**Table C-1. MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Effect (Mechanism) | Expansion Link(1) | Examination Method/Frequency(1) | Examination Coverage |
|---|---|---|---|---|---|
| **Alignment and Interfacing Components**<br>Internals hold down spring | All plants with 304 stainless steel hold down springs<br>Note:<br>FNP Unit 2 hold down spring is 304 SS. | Distortion (Loss of Load)<br>Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms. | None | Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years. | Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty.<br>See Figure A-9. |

**Table C-1. MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Effect (Mechanism) | Expansion Link(1) | Examination Method/Frequency(1) | Examination Coverage |
|---|---|---|---|---|---|
| **Thermal Shield Assembly**<br>Thermal shield flexures | All plants with thermal shields<br>Note:<br>FNP Unit 2 RVI does not have a thermal shield. | Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation. | None | Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval. | 100% of thermal shield flexures.<br>See Figures A-6 and A-10. |

Notes:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table C-4.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table C-4, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table C-4, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.

**Table C-2. MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals**

| Item | Applicability | Effect (Mechanism) | Primary Link(1) | Examination Method/Frequency(1) | Examination Coverage |
|---|---|---|---|---|---|
| **Upper Internals Assembly**<br>Upper Core Plate | All plants | Cracking (Fatigue, Wear) | CRGT lower flange weld | Enhanced visual (EVT-1) examination.<br>Re-inspection every 10 years following initial inspection. | 100% of accessible surfaces(2). |
| **Lower Internals Assembly**<br>Lower support forging or castings | All plants<br>Note:<br>FNP Unit 2 has a lower support forging | Cracking<br>Aging Management (TE in Casting) | CRGT lower flange weld | Enhanced visual (EVT-1) examination.<br>Re-inspection every 10 years following initial inspection. | 100% of accessible surfaces(2).<br>See Figure A-12. |
| **Core Barrel Assembly**<br>Barrel-former bolts | All plants | Cracking (IASCC, Fatigue)<br>Aging Management (IE, Void Swelling and ISR) | Baffle-former bolts | Volumetric (UT) examination.<br>Re-inspection every 10 years following initial inspection. | 100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads(2).<br>See Figure A-7. |
| **Lower Support Assembly**<br>Lower support column bolts | All plants | Cracking (IASCC, Fatigue)<br>Aging Management (IE and ISR) | Baffle-former bolts | Volumetric (UT) examination.<br>Re-inspection every 10 years following initial inspection. | 100% of accessible bolts or as supported by plant-specific justification(2).<br>See Figures A-11, A-12 and A-13. |

**Table C-2. MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Effect (Mechanism) | Primary Link[(1)] | Examination Method/Frequency[(1)] | Examination Coverage |
|---|---|---|---|---|---|
| **Core Barrel Assembly**<br>Core barrel outlet nozzle welds | All plants | Cracking (SCC, Fatigue)<br>Aging Management (IE of lower sections) | Upper core barrel flange weld | Enhanced visual (EVT-1) examination.<br>Re-inspection every 10 years following initial inspection. | 100% of one side of the accessible surfaces of the selected weld and adjacent base metal[(2)].<br>See Figure A-4. |
| **Core Barrel Assembly**<br>Upper and lower core barrel cylinder axial welds | All plants | Cracking (SCC, IASCC)<br>Aging Management (IE) | Upper and lower core barrel cylinder girth welds | Enhanced visual (EVT-1) examination.<br>Re-inspection every 10 years following initial inspection. | 100% of one side of the accessible surfaces of the selected weld and adjacent base metal[(2)].<br>See Figure A-4. |
| **Lower Support Assembly**<br>Lower support column bodies (non cast) | All plants | Cracking (IASCC)<br>Aging Management (IE) | Upper core barrel flange weld | Enhanced visual (EVT-1) examination.<br>Re-inspection every 10 years following initial inspection. | 100% of accessible surfaces[(2)].<br>See Figures A-11, A-12 and A-13. |
| **Lower Support Assembly**<br>Lower support column bodies (cast) | All plants<br>Note:<br>FNP Unit 2 lower support column bodies are non-cast. | Cracking (IASCC) including the detection of fractured support columns<br>Aging Management (IE) | Control rod guide tube (CRGT) lower flanges | Visual (EVT-1) examination.<br>Re-inspection every 10 years following initial inspection. | 100% of accessible support columns[(2)].<br>See Figures A-11, A-12 and A-13. |

**Table C-2. MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Effect (Mechanism) | Primary Link(1) | Examination Method/Frequency(1) | Examination Coverage |
|---|---|---|---|---|---|
| **Bottom Mounted Instrumentation System**<br>Bottom-mounted instrumentation (BMI) column bodies | All plants | Cracking (Fatigue) including the detection of completely fractured column bodies<br>Aging Management (IE) | Control rod guide tube (CRGT) lower flanges | Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles.<br>Re-inspection every 10 years following initial inspection.<br>Flux thimble insertion/withdrawal to be monitored at each inspection interval. | 100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal.<br>See Figures A-12 and A-14. |

Notes:
1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table C-4.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).

**Table C-3. MRP-227-A Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals**

| Item | Applicability | Effect (Mechanism) | Reference | Examination Method | Examination Coverage |
|---|---|---|---|---|---|
| **Core Barrel Assembly**<br>Core barrel flange | All plants | Loss of material (Wear) | ASME Code Section XI | Visual (VT-3) examination to determine general condition for excessive wear. | All accessible surfaces at specified frequency. |
| **Upper Internals Assembly**<br>Upper support ring or skirt | All plants | Cracking (SCC, Fatigue) | ASME Code Section XI | Visual (VT-3) examination. | All accessible surfaces at specified frequency. |
| **Lower Internals Assembly**<br>Lower core plate<br>XL lower core plate(1) | All plants | Cracking (IASCC, Fatigue)<br>Aging Management (IE) | ASME Code Section XI | Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity. | All accessible surfaces at specified frequency. |
| **Lower Internals Assembly**<br>Lower core plate<br>XL lower core plate(1) | All plants | Loss of material (Wear) | ASME Code Section XI | Visual (VT-3) examination. | All accessible surfaces at specified frequency. |
| **Bottom-Mounted Instrumentation System**<br>Flux thimble tubes | All plants | Loss of material (Wear) | NUREG-1801, Rev. 1 | Surface (ET) examination. | Eddy current surface examination, as defined in plant response to IEB 88-09. |

**Table C-3. MRP-227-A Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Effect (Mechanism) | Reference | Examination Method | Examination Coverage |
|---|---|---|---|---|---|
| **Alignment and Interfacing Components**<br>Clevis insert bolts | All plants | Loss of material (Wear)(2) | ASME Code Section XI | Visual (VT-3) examination. | All accessible surfaces at specified frequency. |
| **Alignment and Interfacing Components**<br>Upper core plate alignment pins | All plants | Loss of material (Wear) | ASME Code Section XI | Visual (VT-3) examination. | All accessible surfaces at specified frequency. |

Notes:
1. XL = "Extra Long," referring to Westinghouse plants with 14-foot cores.
2. Bolt was screened-in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue.

**Table C-4. MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals**

| Item | Applicability | Examination Acceptance Criteria(1) | Expansion Link(s) | Expansion Criteria | Additional Examination Acceptance Criteria |
|---|---|---|---|---|---|
| **Control Rod Guide Tube Assembly**<br>Guide plates (cards) | All plants | Visual (VT-3) Examination<br>The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion. | None | N/A | N/A |

**Table C-4. MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Examination Acceptance Criteria(1) | Expansion Link (s) | Expansion Criteria | Additional Examination Acceptance Criteria |
|---|---|---|---|---|---|
| **Control Rod Guide Tube Assembly**<br>Lower flange welds | All plants | Enhanced visual (EVT-1) examination<br>The specific relevant condition is a detectable crack-like surface indication. | a. Bottom-mounted instrumentation (BMI) column bodies<br>b. Lower support column bodies (cast), upper core plate and lower support forging or casting | a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage.<br>b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies, upper core plate and lower support forging/castings within three fuel cycles following the initial observation. | a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies.<br>b. For cast lower support column bodies, upper core plate and lower support forging/castings, the specific relevant condition is a detectable crack-like surface indication. |

**Table C-4. MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Examination Acceptance Criteria(1) | Expansion Link (s) | Expansion Criteria | Additional Examination Acceptance Criteria |
|---|---|---|---|---|---|
| **Core Barrel Assembly**<br>Upper core barrel flange weld | All plants | Periodic enhanced visual (EVT-1) examination.<br>The specific relevant condition is a detectable crack-like surface indication. | a. Core barrel outlet nozzle welds<br>b. Lower support column bodies (non-cast) | a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core outlet nozzle welds by the completion of the next refueling outage.<br>b. If extensive cracking in the remaining core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles follow the initial observation. | a and b. The specific relevant condition for the expansion core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication. |

**Table C-4. MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Examination Acceptance Criteria[1] | Expansion Link (s) | Expansion Criteria | Additional Examination Acceptance Criteria |
|---|---|---|---|---|---|
| **Core Barrel Assembly**<br>Lower core barrel flange weld[2] | All plants | Periodic enhanced visual (EVT-1) examination.<br>The specific relevant condition is a detectable crack-like surface indication. | None | None | None |
| **Core Barrel Assembly**<br>Upper core barrel cylinder girth welds | All plants | Periodic enhanced visual (EVT-1) examination.<br>The specific relevant condition is a detectable crack-like surface indication. | Upper core barrel cylinder axial welds | The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage. | The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication. |

**Table C-4. MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Examination Acceptance Criteria(1) | Expansion Link (s) | Expansion Criteria | Additional Examination Acceptance Criteria |
|---|---|---|---|---|---|
| **Core Barrel Assembly**<br>Lower core barrel cylinder girth welds | All plants | Periodic enhanced visual (EVT-1) examination.<br>The specific relevant condition is a detectable crack-like surface indication. | Lower core barrel cylinder axial welds | The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage. | The specific relevant condition for the expansion lower core barrel cylinder axial weld examination is a detectable crack-like surface indication. |
| **Baffle-Former Assembly**<br>Baffle-edge bolt | All plants with baffle-edge bolts<br>Note:<br>FNP Unit 2 has baffle-edge bolts. | Visual (VT-3) examination.<br>The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads. | None | N/A | N/A |

**Table C-4. MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Examination Acceptance Criteria(1) | Expansion Link (s) | Expansion Criteria | Additional Examination Acceptance Criteria |
|---|---|---|---|---|---|
| **Baffle-Former Assembly**<br>Baffle-former bolts | All plants | Volumetric (UT) examination.<br>The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification. | a. Lower support column bolts<br>b. Barrel-former bolts | a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles.<br>b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts. | a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification. |

**Table C-4. MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Examination Acceptance Criteria(1) | Expansion Link (s) | Expansion Criteria | Additional Examination Acceptance Criteria |
|---|---|---|---|---|---|
| **Baffle-Former Assembly** Assembly | All plants | Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints. | None | N/A | N/A |

**Table C-4. MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals (cont.)**

| Item | Applicability | Examination Acceptance Criteria(1) | Expansion Link (s) | Expansion Criteria | Additional Examination Acceptance Criteria |
|---|---|---|---|---|---|
| **Alignment and Interfacing Components**<br>Internals hold down spring | All plants with 304 stainless steel hold down springs<br>Note:<br>FNP Unit 2 hold down spring is 304 SS | Direct physical measurement or spring height.<br>The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold down forces within the plant-specific design tolerance. | None | N/A | N/A |
| **Thermal Shield Assembly**<br>Thermal shield flexures | All plants with thermal shields<br>Note:<br>FNP Unit 2 does not have a thermal shield. | Visual (VT-3) examination.<br>The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation. | None | N/A | N/A |

Notes:
1. The examination acceptance criterion for visual examination is the absence of the specified relevance condition(s).
2. The lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs.