ATTACHMENT 1

Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals (Application to Implement MRP-227-A)

WCAP-18265-NP, Rev. 0

(Non-Proprietary)

WCAP-18265-NP
Revision 0

December 2017

Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals

(Application to Implement MRP-227-A)

Westinghouse

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LIST OF ACRONYMS

A/LAI	Applicant/Licensee Action Item
AMP	Aging Management Program
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
EFPY	effective full-power year
EOC	end of cycle
EPRI	Electric Power Research Institute
ET	eddy current testing
EVT	enhanced visual testing (a visual NDE method that includes EVT-1)
FME	foreign material exclusion
FMECA	failure modes, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	Inspection and Evaluation
IASCC	irradiation-assisted stress corrosion cracking
IE	irradiation embrittlement
IMI	in-core monitoring instrumentation
INPO	Institute of Nuclear Power Operations
ISI	inservice inspection
ISR	irradiation-enhanced stress relaxation
LAR	License Amendment Request
LR	license renewal
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-hardening
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group

LIST OF ACRONYMS (cont.)

PWSCC	primary water stress corrosion cracking
QA	quality assurance
RCS	reactor coolant system
RIS	Regulatory Issue Summary
RVI	reactor vessel internals
SCC	stress corrosion cracking
SE	Safety Evaluation
SER	Safety Evaluation Report
SRP	Standard Review Plan
SS	stainless steel
TE	thermal embrittlement
TLAA	time-limited aging analysis
UFSAR	Updated Final Safety Analysis Report
UHI	upper head injection
UT	ultrasonic testing (a volumetric NDE method)
VT	visual testing (a visual NDE method that includes VT-1 and VT-3)
WOG	Westinghouse Owners Group
XL	Extra-long Westinghouse Fuel

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1 PURPOSE

The purpose of this report is to document the Duke Energy, McGuire Nuclear Station Unit 1 and McGuire Nuclear Station Unit 2 (hereafter referred to as McGuire) Reactor Vessel Internals (RVI) Aging Management Program (AMP) and Inspection Plan for submittal to the United States (U.S.) Nuclear Regulatory Commission (NRC). The purpose of the McGuire RVI Program is to manage the effects of aging on RVI through the license renewal period (which begins June 13, 2021 for McGuire Unit 1 and March 4, 2023 for McGuire Unit 2). This document demonstrates that the McGuire RVI Program manages the effects of aging for RVI components and establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function throughout the McGuire license renewal period of extended operation. This document is supported by existing Duke Energy documents and procedures. As needed by industry experience or directive in the future, the McGuire RVI Program will be updated or supported by additional documents to provide clear and concise direction for the effective management of aging degradation in RVI components. These actions provide assurance that operations at McGuire will continue to be conducted in accordance with the current licensing basis (CLB) for the RVI by fulfilling license renewal [1], American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI Inservice Inspection (ISI) programs [2] (and the applicable Section XI Edition and Addenda [3]), and industry requirements [53]. The McGuire RVI Program fully captures the intent of the industry guidance for RVI 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 McGuire RVI AMP and Inspection Plan are to:

- Demonstrate that the effects of aging on the RVI will be adequately managed for the period of extended operation in accordance with 10 CFR 54 [5].
- Summarize the role of relevant existing McGuire programs and activities in the RVI Program.
- Define the McGuire RVI Program, based on industry-defined (EPRI/MRP and PWROG) RVI requirements and guidance, addressing the ten AMP elements in NUREG-1801, Revision 2 [7] as updated by LR-ISG-2011-04 [46].
- Address the applicable Applicant/Licensee Action Items identified in the Safety Evaluation (SE) on MRP-227, Revision 0 (contained in MRP-227-A) [27].
- Provide an inspection plan for the McGuire reactor internals.

During the review of the License Renewal Application [1], Duke Energy made commitments related to the McGuire RVI Program. In a letter dated June 16, 2010 [38], Duke Energy notified the NRC of its intent to revise its commitments for RVI inspections from those that currently exist in McGuire's Updated Final Safety Analysis Report (UFSAR) to the inspection guidelines provided by MRP-227 as approved by the NRC (i.e., MRP-227-A). The existing inspection commitments are contained in Section 18.2.23 of the McGuire UFSAR. The UFSAR section contains an allowance that permits Duke Energy to modify or eliminate these inspections if plant-specific justification is provided to demonstrate the basis for the

modification or elimination. Additionally, as part of its license renewal, Duke Energy stated they would participate in industry activities associated with RVI-related issues and that the McGuire RVI Program is subject to future enhancements as the industry understanding of degradation continues to change. 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 serve as the basis for identifying augmented inspections that are required to complete the McGuire RVI Program. Duke Energy is revising its commitments for RVI Inspections from those that currently exist in the McGuire UFSAR to the inspection guidelines provided in MRP-227, as approved by the NRC. This MNS RVI AMP and Inspection Plan, as documented herein, is based on MRP-227-A. Once the McGuire RVI AMP and Inspection Plan is approved by the NRC, the McGuire UFSAR will be updated as required.

In a letter dated March 19, 2014 [31], Duke Energy notified the NRC of its intent to submit a plant-specific RV Internals inspection plan for McGuire to implement MRP-227-A no later than two years before the initial inspection. In letter dated February 23, 2017 [39], Duke Energy notified the NRC that the expected inspection plan submittal date for McGuire is Fall 2017. The expected initial inspection dates are Spring 2019 and Spring 2023 for McGuire Units 1 and 2, respectively, but may be subject to change.

2 BACKGROUND

The management of aging degradation effects in RVI is required for nuclear plants considering or entering license renewal, as specified in NUREG-1800, " Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (SRP-LR)" [8]. To accomplish this task the nuclear industry has been engaged in several efforts to provide general guidelines to manage the aging of RVI for the industry as a whole. On a plant-specific basis, McGuire has defined its RVI and has demonstrated that the aging effects associated with the McGuire RVI will be adequately managed throughout the period of extended operation.

2.1 INDUSTRY EFFORT

The U.S. nuclear power industry has been actively engaged in supporting the goal of managing aging degradation effects in RVI. Various programs have been underway within the industry over the past decades to develop guidelines for managing the effects of aging within PWR internals. In 1997, the Westinghouse Owners Group (formerly WOG, now PWROG) issued WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals" [9], which was reissued as Revision 1-A in 2001 after receiving NRC staff review and approval. Other efforts were engaged by the EPRI MRP to address the PWR internals aging management issue for the following three currently operating domestic 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, as well as accumulated industry research data, the following elements were further developed [4] [9] [10]:

- Screening criteria, considering material properties, 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 document)

- Categorization of PWR internals components, based on the screening criteria and the likelihood and severity of safety and economic consequences, into groups: components with insignificant effects from aging degradation, components with the potential to have moderately significant effects from aging degradation, and components with the potential to be significantly affected by aging degradation

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

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

The industry effort, as coordinated by the EPRI MRP, has finalized initial Inspection and Evaluation (I&E) Guidelines for RVI. The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" [27] (hereafter referred to as "the I&E Guidelines" or simply "MRP-227-A") provides industry background for the guidelines, lists of RVI components requiring inspection, and the timing for initial inspections of those components. For each component, the guidelines require a specific type of nondestructive examination (NDE) and give 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). The document was submitted to the NRC for a formal evaluation and review. The SER was issued on June 22, 2011 [11].

- MRP-228 [12], "Inspection Standard for Pressurized Water Reactor Internals – 2015 Update," provides guidance on the qualification/demonstration of the required NDE techniques and other criteria pertaining to the actual performance of the inspections. Reference 12 is the most recent revision of MRP-228 at the time of this report; the revision of MRP-228 in effect at the time of the inspection will be utilized for future inspections.

Additionally, the PWROG has developed WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" for the MRP-227 inspections. In 2016, the PWROG reissued WCAP-17096-NP as WCAP-17096-NP-A Revision 2 after receiving NRC staff review and approval [13].

2.2 MCGUIRE REACTOR VESSEL INTERNALS

The RVI for McGuire are integral with the reactor coolant system (RCS) of a Westinghouse four-loop nuclear steam supply system (NSSS). Illustrations of typical RVI are provided in Figure A-1 through Figure A-11. As described in the LRA [14], the RVI consist of the upper core support structure, the lower core support structure, and the in-core instrumentation support structure; each of these major component assemblies has a distinct purpose. The flux thimble tubes, although not part of the RVI, are being addressed because of their inclusion in MRP-227-A. The flux thimble tubes extend from the penetrations in the reactor vessel lower head to the seal table.

The upper core support structure consists of the upper support plate assembly, the upper core plate, support columns, and the control rod guide tube assemblies. The support columns establish the spacing between the upper support plate assembly and the upper core plate. The upper core support structure is positioned in its proper orientation with respect to the lower support structure by slots in the upper core plate that engage flat-sided upper core plate alignment pins welded to the core barrel [6]. The control rod guide tube assemblies shield and guide the control rod drive shafts and control rods. The guide tubes are fastened to the upper support plate assembly using hold down bolts, and are guided by pins into the upper core plate. A large, circumferential hold-down spring restrains axial movements of the RVI. The entire upper core support structure is removed as a unit during refueling operations to permit access to the fuel assemblies.

Vertical loads from dead weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns to the upper support plate assembly, and

then into the reactor vessel head. Transverse loads from coolant crossflow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support plate and upper core plate. The upper support plate is particularly stiff to minimize deflection [6].

The lower core support structure remains in place in the reactor vessel during most refueling operations. Typically, it is only removed to perform scheduled reactor vessel inspections or inspection of the lower core support structure itself. The lower core support structure is supported at its upper flange by a ledge in the reactor vessel and its lower end is restrained from transverse motion by a radial support system attached to the reactor vessel wall. The lower core support structure consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support (which is welded to the core barrel).

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried by the lower core plate into the lower core plate support flange on the core barrel shell and through the lower support columns to the core support, and then through the core barrel shell to the core barrel flange supported by the vessel flange. Transverse loads from earthquake acceleration, coolant crossflow, and vibration are carried by the core barrel shell and distributed between the lower radial support and the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins, which are welded into the core barrel [6].

The lower core support structure directs coolant flow upwards through the reactor vessel. The internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and guide the in-core instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper core support structure, and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant from the lower plenum flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

The purpose of the lower core support structure is to form a peripheral enclosure of the core including core baffles and a bottom flow distribution plate for efficient flow distribution, provide neutron shielding by means of the neutron panels, and provide structural support to withstand transverse loadings from coolant crossflow and other design conditions. The lower core support structure also provides structural support for vertical loads from the fuel, hydraulic forces, control rod dynamics, and other design loadings.

The purpose of the in-core instrumentation support structure is to provide structural support for the bottom-mounted in-core instrumentation (flux thimbles and thermocouples).

2.3 MCGUIRE LICENSE RENEWAL

Duke Energy submitted an LRA for a renewed operating license for McGuire [14]. In the SER (NUREG-1772) [1] for the LRA, the NRC concluded that the applicant has demonstrated that the aging effects associated with the RVI will be adequately managed, so there is reasonable assurance that these components will perform their intended function(s) consistent with the CLB throughout the period of extended operation, as required by 10 CFR 54.21(a)(3).

A listing of the McGuire RVI components and subcomponents that are subject to AMP requirements is in Table 3.1-1 of the LRA. In the SER, the NRC concluded that the Duke LRA adequately identified the RVI system structures and components that are subject to an AMR, as required by 10 CFR 54.21(a)(1). A listing of the McGuire Unit 1and Unit 2 reactor vessel internals components and subcomponents already reviewed by the NRC is included in Table B-1. Included in the SER, Appendix D is Commitment 14 associated with the Reactor Vessel Internals Inspection.

The existing inspection commitments are contained in Section 18.2.23 of the McGuire UFSAR. The UFSAR section contains an allowance that permits Duke Energy to modify or eliminate these inspections if plant-specific justification is provided to demonstrate the basis for the modification or elimination. Additionally, as part of its license renewal, Duke Energy stated they would participate in industry activities associated with RVI-related issues and that the McGuire RVI Program is subject to future enhancements as the industry's understanding of degradation continues to improve. 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 serve as the basis for identifying any augmented inspections that are required to complete the McGuire RVI Program. In a letter dated June 16, 2010 [38], Duke Energy notified the NRC of its intent to revise its commitments for RVI inspections from those that currently exist in McGuire's UFSAR to the inspection guidelines provided by MRP-227 as approved by the NRC (i.e., MRP-227-A). This MNS RVI AMP and Inspection Plan, as documented herein, is based on MRP-227-A. Once the McGuire RVI AMP and Inspection Plan is approved by the NRC, the McGuire UFSAR will be updated as required.

The original license renewal commitment did not include the submittal of an inspection plan. The 2010 letter [38] did not provide a schedule date for implementation of the MRP-227 guidelines. However, in letter dated March 19, 2014 [31], Duke Energy notified the NRC of its intent to submit a plant-specific RV Internals inspection plan for McGuire to implement MRP-227-A no later than two years before the initial inspection. In a letter dated February 23, 2017 [39], Duke Energy notified the NRC that the expected inspection plan submittal date for McGuire is Fall 2017. The expected initial inspection dates are Spring 2019 and Spring 2023 for McGuire Units 1 and 2, respectively, but may be subject to change.

3 PROGRAM OWNER

The successful implementation and comprehensive long-term management of the McGuire RVI program will require the integration of Duke Energy organizations, corporately and at McGuire, and interaction with multiple industry organizations including, but not limited to, the ASME, MRP, NRC, and PWROG. Duke Energy 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 administration of the RVI program is McGuire senior management.

Roles and responsibilities for establishing, maintaining, and implementing the McGuire RVI Program are established in the applicable Duke Energy administrative and program procedures [36] and [37].

4 DESCRIPTION OF MCGUIRE 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 RVI plant-specific programs based on inspection and evaluation. The intent of the McGuire RVI program is to ensure the long-term integrity and safe operation of the RVI components. McGuire developed this AMP and Inspection Plan in conformance with NUREG-1801, Revision 2 as updated by LR-ISG-2011-04 [7, 46] and MRP-227-A [27].

This RVI program utilizes a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs, such as water chemistry [16] and inspections prescribed by the ASME Section XI ISI Program [2] and [6] and past and future mitigation projects, such as control rod guide tube support pin replacement. These existing programs are augmented with the inspections and evaluations recommended by MRP-227-A.

Aging degradation mechanisms that impact internals have been identified and documented in the LRA submitted by Duke Energy [14]. 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 RVI components to provide early detection of the degradation mechanisms of concern. Therefore, this AMP and Inspection Plan is consistent with the existing McGuire RVI Aging Management Review (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 RVI:

- <u>Stress Corrosion Cracking</u>

  Stress corrosion cracking (SCC) refers to local, nonductile 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.

- <u>Irradiation-Assisted Stress Corrosion Cracking</u>

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

- <u>Wear</u>

  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.

- <u>Fatigue</u>

  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.

- <u>Thermal Aging Embrittlement</u>

  Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardening (PH) stainless steel to high in-service 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 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.

- <u>Irradiation Embrittlement</u>

  Irradiation embrittlement 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 when the material is subjected to 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 program to manage the aging of the McGuire RVI, which will include this AMP and Inspection Plan once it is approved by the NRC, incorporates programs and activities that are credited for managing the aging effects produced by the aging degradation mechanisms listed above.

4.1 SUPPORTING MCGUIRE PROGRAMS AND ACTIVITIES

McGuire has a number of programs and activities that support the aging management of the RVI; these include:

- Chemistry Control Program
- ASME Section XI ISI Program
- Clevis Insert and Bolt Inspections

- Bottom-Mounted Instrumentation Thimble Tube Inspection Program
- Control Rod Guide Tube Support Pin Replacement Project
- Power Uprating Project
- Upflow Conversion

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

4.1.1 Chemistry Control Program

The McGuire Chemistry Control Program is an existing program that provides activities for monitoring and controlling the chemical environments of the McGuire primary cycle systems such that aging effects of system components are minimized. This program manages the aging effects of cracking and loss of material. The program mitigates damage caused by corrosion and SCC and other aging mechanisms. This program includes provisions specified for the verification of proper chemistry control and aging management, such that the intended functions of plant components will be maintained during the period of extended operation for McGuire.

The McGuire Chemistry Control Program includes periodic sampling of primary water for the known detrimental contaminants specified in the EPRI PWR water chemistry guidelines [17] to maintain their concentrations below levels known to result in loss of material or cracking. Sampling frequencies and action limits for each control parameter are defined in McGuire-specific procedures [16].

McGuire follow the guidance set forth in the EPRI PWR Primary Water Chemistry Guidelines [17]. The limits imposed by the McGuire Program meet the intent of the industry standard for addressing primary water chemistry [16].

4.1.2 ASME Section XI Program

The McGuire ASME Section XI Program [2, 47, 48] is an existing program that includes examinations of the Reactor Vessel core support structure components in accordance with ASME Section XI, Subsection IWB-2500. Core support structures are examined using visual VT-3 examination methods each interval (Examination Category B-N-3). Table 4-9 in MRP-227-A lists the existing programs that are credited for aging management in the Westinghouse-design plants. Many of these component items are considered core support structures that are typically examined during the 10-year in-service inspection [49]. For these component items, with the exception of the clevis inserts and bolts, the Code requirements for visual VT-3 are considered sufficient to monitor for the applicable aging effects. The Code required exam for the clevis inserts and bolts is supplemented by Technical Bulletin 14-5 [40].

4.1.3 Clevis Inserts and Bolts - Technical Bulletin 14-5

Technical Bulletin 14-5 [40] provides a summary of the operating experience as well as the root cause findings and the applicability of these findings on Westinghouse and Combustion Engineering pressurized water reactor designs. Technical Bulletin 14-5 also reviews the safety implications of the operating experience and root cause analysis results, as well as provides inspection recommendations for licensees to consider including as part of their aging management program to address this operating experience.

Clevis insert and clevis insert bolt inspection recommendations for Westinghouse-designed plants are described within Technical Bulletin 14-5 as follows:

> In MRP-227-A, the clevis insert bolts for the Westinghouse-designed plants are classified as an Existing Programs component managed for loss of material (wear). To ensure that the functional performance of the clevis inserts is being managed, it is recommended to ensure that the MRP-227-A inspections (ASME Section XI, VT-3) include the following scope [40]:
>
> - Radial key/clevis insert interfacing surfaces; look for aggressive or abnormal wear as compared to previous inspection if available. The radial key does not make contact with the full length of the clevis insert; if wear is significant it would be visible as a step located toward the bottom end of the clevis insert.
> - Interface between the clevis insert and vessel lug; look for signs of looseness or dislocation. Faces of the insert and vessel lug are generally flush; dislocations may be visible by the insert protruding toward the vessel centerline as compared to the vessel lug.
>
> As Note 2 of Table 4-9 of MRP-227-A indicates, the clevis insert bolts were screened in because of stress relaxation and associated cracking. However, the failures of the bolts alone do not result in a loss of the intended safety function of the lower radial supports. Although managing the functional performance of the clevis inserts is the primary goal of the recommended inspections, [40] recommends that the scope of the inspection also include the following as a means of reducing maintenance risk associated with clevis insert bolt degradation:
>
> - Look for wear between the bolt head and lock bar and/or bolt head dislocation.
> - Look for broken tack welds and dislocation of the dowel pin.

McGuire committed to perform a VT-1 of the clevis insert fasteners during the review of the License Renewal Application. McGuire, as stated previously, will follow the recommendations of MRP-227-A, as supplemented by TB-14-5 for the aging management of this component.

4.1.4 Bottom-Mounted Instrumentation Thimble Tube Inspection

Flux thimble tubes are long, slender, stainless steel tubes that are seal welded at one end with flux thimble tube plugs, which pass through the vessel penetration, through the lower internals assembly, and finally extend to the top of the fuel assembly. The bottom-mounted instrumentation (BMI) column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor. The flux thimble provides a path for the neutron flux detector into the core and is subject to reactor coolant pressure on the outside and containment pressure on the inside [6].

The McGuire Bottom-Mounted Instrumentation Thimble Tube Inspection Program is a program that manages loss of material due to wear of the flux thimble tube materials [30]. It implements the recommendations of NRC Bulletin 88-09 [18] that a tube wear inspection procedure be established and maintained for Westinghouse-supplied reactors that use bottom-mounted flux thimble tube instrumentation. The program utilizes an inspection methodology, such as eddy current testing (ET), to

inspect the flux thimble tubes on a periodic frequency and to monitor wall thinning and predict when tubes would require repair or replacement. The program implements a wall thickness trending report.

The Bottom-Mounted Instrumentation Thimble Tube Inspection Program establishes appropriate acceptance criteria (percent through-wall wear), based on industry guidance and including margin to include allowances for factors such as instrument uncertainty, uncertainties in wear scar geometry, and other potential inaccuracies, as applicable, to the inspection methodology. Table 4-9 in MRP-227-A lists the Existing examinations that are credited for aging management in the Westinghouse-design plants. Included in Table 4-9 are the Bottom Mounted Instrumentation System Flux Thimble Tubes. For this item, the Eddy current examination as defined in the plant's response to NCR Bulletin 88-09 is considered sufficient to monitor for the applicable aging effect.

4.1.5 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 upper core plate.

Because of support pin cracking experienced in the industry the original McGuire support pins, at both Unit 1 and Unit 2, were changed to a replacement material prior to startup [56]. The new support pins were still fabricated from INCONEL®[1] Alloy X-750, but with a modified heat treatment. Support pins made of this material with the associated heat treatments continued to be susceptible to PWSCC and proved likely to fail during the lifetime of a nuclear power plant. Westinghouse developed an improved support pin design made of 316 SS material with a fabrication technique that significantly reduced the susceptibility to PWSCC while maintaining the fatigue and wear requirements necessary to support continued uninterrupted service [19, 26]. In response to industry concern, the Alloy X-750 support pins were replaced with 316 SS support pins at McGuire Unit 1 in fall 1999 and at McGuire Unit 2 in fall 2000; the replacement support pins utilized improved materials that support the proactive management of aging in RVI components.

Support pins fabricated from 316 SS are not susceptible to PWSCC which was the primary failure mechanism for Alloy X-750 support pins [43]. MRP-191 also indicates that the three screened-in degradation mechanisms of wear, fatigue, and stress relaxation for 316 SS support pins would have minimal likelihood to cause failure based on the 316 SS support pin assignment to FMECA Category A [43]. MRP-227-A states that subsequent performance monitoring of the support pins should follow the supplier recommendations. They thus are not included in Table 4-9 of MRP-227-A. There are no supplier recommendations for performance monitoring of 316 SS support pins. Additionally, the Type 316 SS support pins were evaluated in MRP-191. While they screened in for wear, fatigue, and stress relaxation, the FMECA and ranking resulted in them being assigned to Category A. Therefore no additional inspections are required by the supplier or per MRP-227-A, and the support pins should remain functional for the period of extended operation.

[1] INCONEL® is a trademark or registered trademark of Special Metals; a Precision Castparts Corp. company. Other names may be trademarks of their respective owners.

4.1.6 Power Uprating Project

McGuire Unit 1 and Unit 2 were each licensed to 3411 MWt [34]. McGuire sought to increase core power at each unit to 3469 MWt through the use of more accurate feedwater flow measurement instrumentation. Performance of this power uprate was approved by the NRC per McGuire Nuclear Station Units 1 and 2 License Amendments 269 and 249 and resulted in greater power generation of electricity [35]. The power uprate design evaluation [34] ensured operation of McGuire remained consistent with safety-related analyses and remained below design basis limits.

The evaluation of the power uprate performed at McGuire and detailed description of the changes are available in plant records [34]. The NRC staff reviewed the licensee's evaluation of the impact of the LAR on the structural integrity assessments for the RV internals. The NRC staff determined within[35] that the licensee's RV internals evaluation considering the effect of the LAR was acceptable because "(1) the licensee confirmed that the McGuire 1 and 2 RV internals aging management was developed before December 31,2011, and the inspection program (plan) will be implemented by December 31,2013, as required by the MRP-227-A and (2) the LAR would result in very small changes to aging parameters such as temperature and neutron flux".

4.1.7 Upflow Conversion

The original plant design at McGuire Unit 1 and Unit 2 was "downflow" such that the flow in the region between the core barrel and the baffle plates was downward while the direction of flow in the core region was upward. As a result of the opposing flows, a differential pressure existed across the baffle plates with the higher pressure being on the core barrel side of the baffle plates. In a downflow configuration, when gaps develop between adjacent baffle plates the differential pressure can force a jet of water into the core. Impingement of these water jets on the fuel rods can cause critical vibrations which can lead to failures. This phenomenon is referred to as "baffle jetting."

In order to prevent fuel damage due to baffle jetting, an upflow conversion modification was performed on McGuire Unit 1 during the Fall 1991 outage [50] and on McGuire Unit 2 during the Fall 1990 outage [51], to reverse the direction of flow through the core barrel/ baffle region [50, 51]. The flow reversal resulted in a significant reduction of differential pressure across the baffle plate, and the potential for baffle jetting was minimized.

The upflow conversion was accomplished by plugging existing holes in the core barrel and opening new holes in the top former plate. In addition, several lower former plate holes were plugged to further reduce the differential pressure across the baffle plates. After the modification, coolant flow enters the core barrel/baffle region through the remaining flow holes in the lower former plate, flows upward through the intermediate former plates, exits through the new holes in the top former plate, and joins the main coolant flow at the top of the core above the baffle plates [50, 51].

The upflow conversion included fatigue usage calculations for the flow holes in the modified upper former plate, the core barrel flow hole plugs, and the lower former plate flow hole plugs. In addition, irradiation-induced stress relaxation between the core barrel and plug and lower former plate and plug was evaluated [50, 51].

The up flow conversion was not addressed in the original LRA. An item in Duke Energy's Corrective Action Program [52] has been initiated to address the 10 CFR 54.37(b) requirement to include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with § 54.21.

These components will be addressed through the A/LAI process as discussed in Section 6.

4.2 INDUSTRY PROGRAMS

4.2.1 WCAP-14577, Aging Management for Reactor Internals

The WOG topical report WCAP-14577 [9] 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 AMPs.

Duke Energy did not use WCAP-14577 as a reference to complete the AMR for the McGuire internals. However, the NRC referred to WCAP-14577 in their review of the McGuire LRA and requested additional information where the McGuire LRA was inconsistent with WCAP-14577.

4.2.2 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 RVI. The following subsections briefly describe the industry process.

4.2.2.1 MRP-227-A Reactor Vessel Internals Component Categorizations

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 U.S. were evaluated in the MRP Program and 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, described as follows:

- <u>Primary</u>

    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 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 will depend 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 the failure modes, 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 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.

4.2.2.2 MRP-227-A, Applicability to McGuire Unit 1 and Unit 2

The applicability of MRP-227-A to McGuire 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.*

  McGuire Applicability: McGuire Unit 1 and Unit 2 fuel management program changed from a high- to a low-leakage core-loading pattern for each unit prior to 30 years of operation.

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

McGuire Applicability: McGuire Unit 1 and Unit 2 operate as base load units.

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

  McGuire 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 identified. Duke Energy has not made any modifications to either of the McGuire unit reactor internals components since May 2007.

Based on the applicability, as stated above, the MRP-227-A work is representative for McGuire.

4.2.3 WCAP-17451-P, Reactor Internals Guide Tube Wear

The PWROG developed a tool to facilitate prediction of upper internals control rod guide tube assembly guide card and lower guide tube continuous guidance wear. An initial inspection schedule based on the various guide tube designs for the utilities participating in this program and acceptance criteria was then established. WCAP-17451-P Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections" [29], documents the guide plate (card) initial inspection schedule and acceptance criteria for Westinghouse NSSS designed plants. Per EPRI interim guidance letter, MRP 2014-006 [32], MRP endorsed the guide plate (card) inspection requirements [27].

McGuire Unit 1 is a four-loop plant with a 17x17 standard guide tube design and will follow the guidance in [29]. McGuire Unit 2 is a four-loop plant with a 17x17A guide tube design that uses ion nitride RCCAs. Recent OE at U.S. Westinghouse NSSS plants that have 17x17 A or 17x17 AS style GTs and have switched to ion nitride RCCAs indicate that the rate of guide card wear has outpaced the predications in WCAP-17451-P, as stated in NSAL-17-1 [60]. This issue was determined to have a potential nuclear safety consequence and therefore was reported to the NRC as a defect, pursuant to 10 CFR Part 21. The recommended timing for performing baseline guide card wear inspections, as well as the wear projection methodology used for determining the remaining life of the guide tubes documented in [29], is non-conservative for these types of plants. McGuire Unit 2 will therefore follow the initial inspection guidance in [60].

Additionally, there is a provision within [29] that allows for a schedule extension or scope reduction for performing guide card wear measurement based on guide card wear screening through video inspection. Duke Energy requested Westinghouse review guide tube foreign material exclusion (FME) video tapes for McGuire Unit 1 within [41] and for McGuire Unit 2 within [42]. The guide card wear screening in accordance with [29] concluded that McGuire Unit 1 should perform initial guide card inspections by EFPY 31.66 [41] and that McGuire Unit 2 should perform initial guide card inspections by Fall 2018 [42].

4.2.4 Baffle-Former Bolt Degradation

Recently, a larger-than-expected number of degraded baffle-former bolts were discovered in 4-loop downflow plants through MRP-227 inspection requirements and reactionary inspections at like plants. The pattern of these degraded bolts was also concentrated, or clustered, more than anticipated based on OE gained from previous analyses and inspections. As a result, Westinghouse performed a 10 CFR 21 evaluation, the results of which are discussed in NSAL-16-1 [57]. NSAL-16-1 also provided guidance for

affected utilities by grouping plants based on their susceptibility to baffle former bolt degradation, and offered recommendations on re-inspection techniques and intervals. Interim guidance has been developed and published by the Electric Power Research Institute's (EPRI) Materials Reliability Program (MRP) for the Tier 1 plants in [59] and for all of the NSAL-16-1 plants in [58], respectively. Both McGuire Unit 1 and Unit 2 are Tier 3 plants and have converted to upflow prior to 20 years as discussed in Section 4.1.7. Upflow modifications have been shown to reduce the incidence of baffle jetting damage to fuel and to reduce the bolt loads due to pressure differentials across the baffle under both normal operating and expected faulted conditions. Based on [57], McGuire is recommended to continue to follow the current MRP-227 guidelines and implement any revisions to the MRP-227 recommendations. If visually damaged baffle-former bolts or lock bars are detected, it is recommended that the fuel assemblies that were adjacent to the baffle in the previous cycle, and are scheduled for use in the next cycle, be inspected for fretting wear on the face that was adjacent to the baffle.

4.2.5 WCAP-17096, Reactor Internals Acceptance Criteria Methodology and Data Requirements

The industry, through various cooperative efforts, is working to construct a consensus set of tools in line with accepted and proven methodologies to support GALL Revision 2 Element 6, related to acceptance criteria. One of these tools is the PWROG document WCAP-17096-NP-A, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" [13], which details acceptance criteria methodology for the MRP-227-A Primary and Expansion components.

Revision 2 of WCAP-17096-NP was transmitted to the NRC on May 19, 2010 for review. The NRC issued a draft SE on November 12, 2015 and received comments from EPRI on January 28, 2016. The NRC accepted WCAP-17096-NP Revision 2 on May 3, 2016 allowing the issuance of WCAP-17096-NP-A [13]. WCAP-17096-NP-A provides the methodology to evaluate inspection findings at McGuire and the status of WCAP-17096-NP-A is monitored through direct Duke Energy cognizance of industry (including PWROG) activities related to PWR internals inspection and aging management.

4.2.6 On-Going Industry Programs and NEI 03-08 Guidelines

As part of its license renewal, Duke Energy stated they would participate in industry activities associated with RVI-related issues and that the McGuire RVI Program is subject to future enhancements as the industry's understanding of degradation continues to improve. 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 serve as the basis for identifying any augmented inspections that are required.

Several of the industry work products developed in support of the aging management of PWR internals are issued by Issue Programs (including MRP and PWROG) under the implementation protocol of NEI 03-08 [15]. Included work products are MRP-227, MRP-228, WCAP-17451-P, and any interim guidance associated with these documents. Appendix B to NEI 03-08, Implementation Protocol, defines the processes and expectations for implementing industry guidance issued under the Materials Initiative, and requires that Issue Programs identify the specific implementation category for 'requirements' identified by guideline-type work products. While McGuire is basing the AMP and Inspection Plan for RVI on the scope defined in MRP-227-A, McGuire will also implement work products issued under the

implementation protocol of NEI 03-08 in accordance with Duke Energy administrative procedures [53], including later revisions and interim guidance to the work products listed above. A failure to meet a Needed or a Mandatory requirement is a deviation from the guidelines and a written justification for the deviation must be prepared and approved as described in Appendix B to NEI 03-08 and Duke Energy administrative procedures, including notification to the NRC for information only.

The U.S. industry, through both the EPRI/MRP and the PWROG, continues to sponsor activities related to RVI aging management. Duke Energy 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 [53].

4.3 SUMMARY

This section contains pertinent McGuire and industry programs and activities used for the development and implementation of MRP-227-A, the McGuire RVI AMP and Inspection Plan, and the McGuire RVI Program.

The augmented inspections described in this document, as summarized in Appendix C, combined with the ASME Code Section XI ISI Program inspections, existing McGuire Programs, and use of Operating Experience, provide reasonable assurance that the reactor internals at McGuire will continue to perform their intended functions throughout the period of extended operation.

5 MCGUIRE NUCLEAR STATION UNIT 1 AND UNIT 2 REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES

Based on Duke Energy's revised commitment, the McGuire RVI AMP and Inspection Plan 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 (loss of material)
- Fatigue (cracking)
- Thermal aging embrittlement (reduction in fracture toughness)
- Irradiation embrittlement (reduction in fracture toughness)
- Void swelling and irradiation growth (distortion)
- Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep (loss of preload or loss of mechanical closure integrity)

The attributes of the McGuire RVI Program and compliance with NUREG- 1801 (GALL Report), Section XI.M16A, "PWR Vessel Internals" [7], as updated via LR-ISG-2011-04 [46] 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.

Duke Energy utilized a process similar to the GALL process contained in NUREG-1801 to perform the AMR of the RVI.

This AMP and Inspection Plan is consistent with that process, considers the augmented inspections identified in MRP-227-A, and fully meets the requirements of the current commitment. Specific details of the McGuire RVI Program 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 based on the plant's applicable nuclear steam supply system design. The scope of the program applies the methodology and guidance in MRP-227-A, which provides an 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 Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse. The scope of components considered for inspection in MRP-227-A includes core support structures, 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). In addition, ASME Code, Section XI includes inspection requirements for PWR removable core support structures in Table IWB-2500-1, Examination Category B-N-3, which are in addition to any inspections that are implemented in accordance with MRP-227-A.

The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. 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 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." [46]

McGuire Nuclear Station Unit 1 and Unit 2 Program Scope

The McGuire RVI components consist of the upper core support structure, the lower core support structure, and the in-core instrumentation support structure, where each of these major components has a distinct purpose. The flux thimble tubes, although not part of the RVI, are being addressed because of their inclusion in MRP-227-A. The flux thimble tubes extend from the penetration on the reactor vessel lower head up to the seal table. Additional RVI details are provided in the McGuire LRA [14].

The McGuire RVI subcomponents that are subject to AMP requirements were provided in Table 3.1-1 in the McGuire LRA. The table listed each subcomponent's intended function(s) and material. The aging effects that required management were identified in the table. A column in the table lists the aging management program that was 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 LRA, as documented in the SER on license renewal [1]. Table 3.1-1 from the LRA is included in Appendix B as Table B-1.

Duke Energy's commitment to implement MRP-227-A necessitates that the aging management strategy in the original LRA be updated. McGuire utilizes NUREG-1801, Revision 2 as updated by LR-ISG-2011-04 to ensure the aging effects are managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

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. This supersedes the aging management review performed in the LRA. The information provided in MRP-227-A is rooted in the GALL methodology.

As discussed in Section 4.1.2, core support structures are examined in accordance with Examination Category B-N-3 of ASME Section XI. The inspections credited in the McGuire LRA are based on utilizing these ASME Section XI exams and the augmented inspections derived from MRP-227-A. The MRP-227-A inspections only augment and do not reduce, alter, or otherwise affect the ASME Section XI requirements.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

5.2 GALL REVISION 2 ELEMENT 2: PREVENTIVE ACTIONS

GALL Report AMP Element Description

"MRP-227-A 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, as described in GALL AMP XI.M2, 'Water Chemistry.'" [46]

McGuire Unit 1 and Unit 2 Preventive Action

The McGuire RVI Program does not prevent degradation due to aging effects; rather, it provides measures for monitoring to detect degradation prior to loss of intended function. Preventative measures to mitigate aging effects such as loss of material and cracking in the primary water system are established and implemented in accordance with the McGuire Chemistry Control Program [16]. A description and applicability to the McGuire RVI Program is provided in the following subsection.

Chemistry Control Program

To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. The McGuire PWR Chemistry Control Program [16] is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines [17].

This program is consistent with the corresponding program described in the GALL Report [7] and [46].

The limits of known detrimental contaminants imposed by the chemistry monitoring program are consistent with the EPRI PWR Primary Water Chemistry Guidelines.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

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 RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclic 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

dimensions due to void swelling or distortion; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

For the management of cracking, the program monitors for evidence of surface-breaking linear discontinuities if a visual inspection technique is used as the non-destructive examination (NDE) method, or for relevant flaw presentation signals if a volumetric ultrasonic testing (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. Instead, the impact of loss of fracture toughness on component integrity is indirectly managed by: (1) using visual or volumetric examination techniques to monitor for cracking in the components, and (2) applying applicable reduced fracture toughness properties in the flaw evaluations, in cases where cracking is detected in the components and is extensive enough to necessitate a supplemental flaw growth or flaw tolerance evaluation. The program uses physical measurements to monitor for any dimensional changes due to void swelling or distortion.

Specifically, the program implements the parameters monitored/inspected criteria consistent with the applicable tables in Section 4, 'Aging Management Requirements,' in MRP-227-A." [46]

McGuire Unit 1 and Unit 2 Parameters Monitored or Inspected

The McGuire RVI Program monitors the following aging effects by inspection, in accordance with the guidance of MRP-227-A. Relevant Indications are as defined by Table 5-3 of MRP-227-A (included as Table C-4 of this document), MRP-228, or the associated Existing Program.

1) Cracking

Cracking is due to SCC, PWSCC, IASCC, or fatigue/cyclical loading. Cracking is monitored with visual or volumetric examination for evidence of relevant indications. Surface examinations may also be used to supplement visual examinations for detection and sizing of relevant indications.

2) Loss of Material

Loss of material is due to wear. Loss of material is monitored with a visual examination for relevant indications, physical measurement, or eddy current testing.

3) Loss of Fracture Toughness

Loss of fracture toughness is due to TE or IE. The impact of loss of fracture toughness on component integrity is indirectly managed by monitoring for cracking by using visual or volumetric examination techniques, and by applying applicable reduced fracture toughness properties in flaw evaluations if any detected cracking is determined to be extensive enough to necessitate a supplemental flaw growth or flaw tolerance evaluation.

4) Changes in Dimension

Changes in dimension are due to void swelling or distortion. Changes in dimension are monitored by visual or volumetric examination.

5) Loss of Preload

Loss of preload is due to thermal and ISR or irradiation-enhanced creep. Loss of preload is monitored with a visual or volumetric examination for relevant indications that may be indicative of loosening in applicable bolted, fastened, keyed or pinned connections.

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

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

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

5.4 GALL REVISION 2 ELEMENT 4: DETECTION OF AGING EFFECTS

GALL Report AMP Element Description

"The inspection methods are defined and established in Section 4 of MRP-227-A. Standards for implementing the inspection methods are defined and established in MRP-228. In all cases, well-established inspection methods are selected. These methods include volumetric UT examination methods for detecting flaws in bolting 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 surfacebreaking 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). VT-3 visual methods may be applied for the detection of cracking in non-redundant RVI components only when the flaw tolerance of the component, as evaluated for reduced fracture toughness properties, is known and the component has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. VT-3 visual methods are acceptable for the detection of cracking in redundant RVI components (e.g., redundant bolts or pins used to secure a fastened RVI assembly).

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.

The program adopts the guidance in MRP-227-A for defining the "Expansion Criteria" that need to be applied to the inspection findings of "Primary" components and for expanding the examinations to include additional "Expansion" components. RVI component inspections are performed consistent with the inspection frequency and sampling bases for "Primary" components, "Existing Programs" components, and "Expansion" components in MRP-227-A.

In some cases (as defined in MRP-227-A), 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 dimensions due to void swelling or distortion.

Inspection coverages for "Primary" and "Expansion" RVI components are implemented consistent with Sections 3.3.1 and 3.3.2 of the NRC SE, Revision 1, on MRP-227." [46]

McGuire Unit 1 and Unit 2 Detection of Aging Effects

Detection of indications that are required by the ASME Code Section XI ISI Program is well established and field-proven through the application of the Section XI ISI Program [2]. The McGuire RVI Program 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 document for reference. These tables include the inspection frequency and sampling bases. For the Expansion Components of MRP-227-A, the McGuire RVI Program implements the expansion requirements of Table 5-3 of MRP-227-A (included in Appendix C of this document 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. Inspection standards developed by the industry for the application of these techniques for augmented reactor internals inspections are documented in MRP-228.

<u>VT-3 Examination for General Condition Monitoring</u>

One examination method selected for use, which has an extensive history of use for PWR internals, is visual (VT-3) examination. Such visual examinations are relied upon for detection of general degradation of PWR internals subject to Table IWB-2500-1 Category B-N-3 requirements. Visual (VT-3) examinations are conducted to determine the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and by identifying conditions that could affect operational or functional adequacy of components. This type of examination has been determined to be acceptable for the continued monitoring of many of the internals within the scope of the McGuire RVI Program. When specified, a visual (VT-3) examination is conducted in accordance with the requirements

of the MRP-228 Inspection Standard. All examination personnel, equipment, examinations, classification and measurement of indications, and documentation associated with visual examinations will meet the requirements of MRP-228. Visual (VT-3) examinations of internals are conducted using remote examination techniques, because of personnel radiation exposure issues.

VT-1 Visual Examinations and EVT-1 Enhanced Visual Examinations

Two examination methods selected for use are visual (VT-1) and enhanced visual (EVT-1) examinations. The visual (VT-1) examinations and the enhanced visual (EVT-1) examinations were selected where a greater degree of detection capability, as well as sizing capability, is required – over and above the capability inherent in visual (VT-3) examinations to manage the aging effects. Unlike the detection of general degradation conditions by visual (VT-3) examination, visual (VT-1) and enhanced visual (EVT-1) examinations are conducted to detect and size discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion. Specifically, VT-1 is used for the detection and sizing of surface discontinuities such as gaps, while EVT-1 is used for the detection of surface breaking flaws.

When specified in these guidelines, a visual (VT-1) examination is conducted in accordance with the requirements of the MRP-228 Inspection Standard. Enhanced visual (EVT-1) examination is also conducted in accordance with the requirements described for visual (VT-1) examination with additional requirements (such as camera scanning speed) as specified in the MRP-228 Inspection Standard. All examination personnel, equipment, examinations, classification and measurement of indications, and documentation associated with visual examinations will meet the requirements of MRP-228. Visual (VT-1) and enhanced visual (EVT-1) examinations of internals are conducted using remote examination techniques, because of personnel radiation exposure issues.

Ultrasonic Testing

Another method selected for use is volumetric examination. An ultrasonic examination (UT) was selected where visual or surface examination is unable to detect the effect of the age-related degradation for some PWR internals. For example, irradiation-assisted stress corrosion cracking (IASCC) in baffle/former bolts may occur underneath the bolt head, and will be undetectable by visual or surface examination unless the bolt is removed and subject to examination over its entire length. When specified in these guidelines, an ultrasonic examination (UT) is conducted in accordance with the requirements of the MRP-228 Inspection Standard. All examination personnel, examinations, classification of indications, and documentation associated with UT will meet the requirements of MRP-228. Additionally, technical justifications as described in MRP-228 are required for qualification of ultrasonic examinations. While UT has only been selected for use in these guidelines for detection of aging effects in bolting, UT is also permissible as an alternative or supplement to the specified visual examinations for other configurations such as plates and welds.

Physical Measurement Examination

The effects of loss of material caused by wear, the loss of pre-load or clamping force caused by such mechanisms as thermal and irradiation-enhanced stress relaxation, and excessive distortion or deflection caused by void swelling can be managed in some cases by physical measurements. Table C-1 requires

guide card wear measurements in accordance with WCAP-17451-P. These guide card wear measurements will be performed in accordance with WCAP-17451-P and the MRP-228 Inspection Standard. Additionally, technical justifications, as described in MRP-228, are required when determining measurement uncertainty for flaw, degradation, or wear measurements (e.g., guide card wear).

Surface Examination

Surface examination, specifically eddy current examination (ET), can supplement either visual (VT-3) or (VT-1/EVT-1) examinations. This supplemental examination may thus be used to reject or accept relevant indications. When selected for use, an ET examination is conducted in accordance with the requirements of MRP-228 [12]. The ET examination, as defined in McGuire's response to IEB 88-09 [18], is also considered sufficient to monitor for the applicable aging effect on Bottom-Mounted Instrumentation Thimble Tubes.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

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-A and its subsections. Flaw evaluation methods, including recommendations for flaw depth sizing and for crack growth determinations as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications, are defined in MRP-227-A. The examination and re-examinations that are implemented in accordance with MRP-227-A, together with the criteria specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide for timely detection, reporting, and implementation of corrective actions for the aging effects and mechanisms managed by the program.*
>
> *The program applies applicable fracture toughness properties, including reductions for thermal aging or neutron embrittlement, in the flaw evaluations of the components in cases where cracking is detected in a RVI component and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.*
>
> *For singly-represented components, the program includes criteria to evaluate the aging effects in the inaccessible portions of the components and the resulting impact on the intended function(s) of the components. For redundant components (such as redundant bolts, screws, pins, keys, or fasteners, some of which are accessible to inspection and some of which are not accessible to inspection), the program includes criteria to evaluate the aging effects in the population of components that are inaccessible to the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components." [46]*

McGuire Unit 1 and Unit 2 Monitoring and Trending

The methods for monitoring, recording, evaluating, and trending the results from the RVI inspections under the McGuire RVI Program are in accordance with MRP-227-A [27], MRP-228 [12], WCAP-17096-NP-A [13], and ASME Section XI [2, 54].

Monitoring is accomplished through implementation of the MRP-227-A Primary, Expansion, and Existing inspections (Tables 4-3, 4-6, and 4-9 of MRP-227-A) according to the criteria specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel. Implementation of these guidelines provides timely detection, reporting, and corrective actions to manage the effects of the age-related degradation mechanisms within the scope of the program. In Appendix C, Tables C-1, C-2, and C-3 identify the augmented Primary and Expansion inspections 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 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.

Per the guidance of MRP-227-A and WCAP-17096-NP-A, the McGuire RVI Program applies applicable fracture toughness properties, including reductions for thermal aging or neutron embrittlement, when a flaw evaluation is conducted. Through implementation of the inspection and evaluation requirements of MRP-227-A, the McGuire RVI Program also addresses potential aging effects in the inaccessible portions of components or redundant component populations and the resulting impact on the intended function(s) of the components.

Recording requirements for the MRP-227-A examinations are provided within MRP-228. Methodologies for evaluation and disposition of off-normal conditions observed by the inspections conducted under the McGuire RVI Program are based on Section 6 of MRP-227-A and WCAP-17096-NP-A, which include flaw depth sizing, crack growth determination, and recommendations for performing applicable limit load, linear elastic, and elastic-plastic fracture analyses of relevant indications.

Trending is supported by the implementation requirements documented in Section 7 of MRP-227, which includes an NEI-03-08 "Needed" requirement for data reporting. 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 updates of the industry guidelines.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

5.6 GALL REVISION 2 ELEMENT 6: ACCEPTANCE CRITERIA

GALL Report AMP Element Description

> *"Section 5 of MRP-227-A, which includes Table 5-1 for B&W-designed RVIs, Table 5-2 for CE-designed RVIs, and Table 5-3 for Westinghouse-designed RVIs, provides the specific examination and flaw evaluation acceptance criteria for the "Primary" and "Expansion" RVI component examination methods. For RVI components addressed by examinations performed in accordance with the ASME Code, Section XI, the acceptance criteria in IWB-3500 are applicable. For RVI components covered by other "Existing Programs," the acceptance criteria are described within the applicable reference document. As applicable, the program establishes acceptance criteria for any physical measurement monitoring methods that are credited for aging management of particular RVI components." [46]*

McGuire Unit 1 and Unit 2 Acceptance Criteria

The McGuire RVI Program acceptance criteria for the Westinghouse-designed Primary and Expansion component examinations are consistent with MRP-227-A, Section 5. For the Westinghouse-designed Existing Programs components, the acceptance criteria are described within the applicable program documents. The McGuire RVI Program establishes acceptance criteria for the physical measurement monitoring of the control rod guide cards using the criteria fromWCAP-17451.

Examination acceptance and expansion criteria for the MRP-227-A inspections are provided in Appendix C, Table C-4. The Existing Programs Components in Table C-3 will continue to be examined in accordance with the credited Existing Program requirements. Augmented inspections, as defined by the MRP-227-A requirements included in Appendix C, Table C-1 and Table C-2, that result in relevant indications will be entered into the Corrective Action Program [24] and addressed by appropriate actions that may include enhanced inspection, repair, replacement, mitigation actions, or analytical evaluations.

Methodologies for evaluation and disposition of relevant indications observed by the inspections are based on Section 6 of MRP-227-A and WCAP-17096-NP-A.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

5.7 GALL REVISION 2 ELEMENT 7: CORRECTIVE ACTIONS

GALL Report AMP Element Description

> *"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. The implementation of the guidance in MRP-227-A, 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 actions bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Alternative corrective actions not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation." [46]

McGuire Unit 1 and Unit 2 Corrective Action

Relevant indications discovered during examinations of the McGuire RVI will be entered into the Corrective Action Program [24] and addressed by appropriate actions that may include enhanced inspection, repair, replacement, mitigation actions, or analytical evaluations. The Work Management System [45, 27, 2, 54] is also part of the Corrective Action Program. The Corrective Action Program charges personnel with the responsibility to identify Undesired Conditions (including conditions adverse to quality), requires that conditions adverse to quality be corrected, and, in the case of significant conditions adverse to quality, ensure that the cause of the condition is determined and actions are taken to preclude repetition. The inspection and evaluation guidance of MRP-227-A [27] provides the basis for what relevant conditions adverse to quality must be addressed by the Corrective Action Program. The ASME Section XI Program Functional Area Manual [2, 54] establishes the McGuire repair and replacement requirements of ASME Code Section XI and will also be credited for this element.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

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 recommendations of NEI 03-08 and the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. The implementation of the guidance in MRP-227-A, in conjunction with NEI 03-08 and other guidance documents, reports, or methodologies referenced in this AMP, provides an acceptable level of quality and an acceptable basis for confirming the quality of inspections, flaw evaluations, and corrective actions." [46]

McGuire Unit 1 and Unit 2 Confirmation Process

McGuire has an established 10 CFR 50, Appendix B Program [20] that addresses the elements of corrective actions, confirmation process, and administrative controls. Quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. McGuire implements work products issued under the implementation protocol of NEI 03-08 in accordance with Duke Energy administrative procedures [53]. A failure to meet a Needed or a Mandatory requirement is a deviation from the guidelines and a written

justification for the deviation must be prepared and approved as described in Appendix B to NEI 03-08 and Duke Energy administrative procedures, including notification to the NRC for information only.

The implementation of the guidance in MRP-227-A, in conjunction with the requirements of NEI 03-08 and other guidance documents, reports or methodologies referenced in this document, provides an acceptable level of quality and an acceptable basis for confirming the quality of inspection, flaw evaluation and other elements of aging management of the McGuire RVI.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

5.9 GALL REVISION 2 ELEMENT 9: ADMINISTRATIVE CONTROLS

GALL Report AMP Element Description

> *"The administrative controls for these types of programs, including their implementing procedures and review and approval processes, are implemented in accordance with the recommended industry guidelines and criteria in NEI 03-08, and are under existing site 10 CFR 50 Appendix B, Quality Assurance Programs, or their equivalent, as applicable. The evaluation in Section 3.5 of the NRC's SE, Revision 1, on MRP-227 provides the basis for endorsing NEI 03-08. This includes endorsement of the criteria in NEI 03-08 for notifying the NRC of any deviation from the I&E methodology in MRP-227-A and justifying the deviation no later than 45 days after its approval by a licensee executive." [46]*

McGuire Unit 1 and Unit 2 Administrative Controls

McGuire has an established 10 CFR 50, Appendix B Program [20] that addresses the elements of corrective actions, confirmation process, and administrative controls. Quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

McGuire implements work products issued under the implementation protocol of NEI 03-08 in accordance with Duke Energy administrative procedures [53]. A failure to meet a Needed or a Mandatory requirement is a deviation from the guidelines and a written justification for the deviation must be prepared and approved as described in Appendix B to NEI 03-08 and Duke Energy administrative procedures, including notification to the NRC for information only.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

5.10 GALL REVISION 2 ELEMENT 10: OPERATING EXPERIENCE

GALL Report AMP Element Description

"The review and assessment of relevant operating experience for its impacts on the program, including implementing procedures, are governed by NEI 03-08 and Appendix A of MRP-227-A. Consistent with MRP-227-A, the reporting of inspection results and operating experience is treated as a "Needed" category item under the implementation of NEI 03-08.

The program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry operating experience, as discussed in Appendix B of the GALL Report, which is documented in LR-ISG-2011-05." [46]

McGuire Unit 1 and Unit 2 Operating Experience

Extensive industry and McGuire operating experience has been reviewed during the development of the reactor vessel internals AMP and Inspection Plan. The experience reviewed includes NRC Information Notices 84-18, "Stress Corrosion Cracking in PWR Systems" [22] and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants" [23]. Most of the industry operating experience 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. Recent operating experience associated with clevis inserts and control rod guide tube guide cards has also been reviewed.

Early plant operating experience related to hot functional testing and reactor internals is documented in plant historical records. Inspections performed as part of the ASME Section XI Program [2 and 6] have been conducted as designated by existing commitments and are expected to discover general internals structure degradation. To date, very little degradation has been observed industry-wide.

A review of industry and plant-specific experience with reactor vessel internals reveals that the U.S. industry, including Duke Energy and McGuire, has responded proactively to industry issues relative to reactor internals degradation. An example that demonstrates this proactive response by Duke Energy is the replacement of control rod guide tube support pins at both McGuire Units between 1999 and 2000. The replacement pins included a material upgrade to Type 316 stainless steel in support of managing aging in the component. Duke Energy will follow industry guidance on the McGuire baffle-former bolts, as discussed in Section 4.2.4, and on the McGuire clevis insert bolts, as discussed in Section 4.1.3. Industry related issues with accelerated guide card wear will be managed by Duke Energy as discussed in Section 4.2.3.

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

Industry operating experience is routinely reviewed by Duke Energy using Institute of Nuclear Power Operations (INPO) Operating Experience (OE), the Nuclear Network, and other information sources as directed under the McGuire operating experience procedure [25], for the determination of additional actions and lessons learned.

Conclusion

This element is consistent with the corresponding aging management attribute in NUREG-1801, Section XI.M16A [7], as updated via LR-ISG-2011-04 [46].

6 MRP-227-A SAFETY EVALUATION CONDITIONS AND ACTION ITEMS

Revision 0 of MRP-227 was finalized in December 2008 and transmitted to the NRC in January 2009. It was requested that the NRC issue a safety evaluation (SE) on MRP-227-Rev. 0. In June 2011, the NRC issued Revision 0 of an SE for MRP-227, Revision 0. The NRC issued Revision 1 of the SE in December 2011, which incorporated technical changes required to ensure MRP-227 included all NRC required changes and defined the basis for acceptance of MRP-227. Revision 1 of the SE, includes eight applicant/licensee-specific action items and seven topical report conditions. The accepted version of the I&E guidelines, containing the SE, was issued as MRP-227-A in December, 2011.

Section 3.5.1 of the SE to MRP-227 states that to ensure the MRP-227 program and the plant-specific action items will be carried out by licensees, licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components.

Table 6-1 lists the seven Topical Report Conditions, and Section 6.2 addresses the eight licensee action items that resulted from the NRC review contained in MRP-227-A.

6.1 TOPICAL REPORT CONDITIONS

Table 6-1 Topical Report Conditions

Topical Condition	Applicable/ Not Applicable	Compliance
1. High consequence components in the "No Additional Measures" Inspection Category	Applicable	Fulfilled for McGuire Unit 1 and Unit 2 by publication of MRP-227-A.
2. Inspection of components subject to irradiation-assisted stress corrosion cracking	Applicable	Fulfilled for McGuire Unit 1 and Unit 2 by publication of MRP-227-A.
3. Inspection of high consequence components subject to multiple degradation mechanisms	Not Applicable	Not applicable to McGuire Unit 1 and Unit 2
4. Imposition of minimum examination coverage criteria for "Expansion" inspection category components	Applicable	Fulfilled for McGuire Unit 1 and Unit 2 by publication of MRP-227-A.

Table 6-1 Topical Report Conditions		
Topical Condition	**Applicable/ Not Applicable**	**Compliance**
5. Examination frequencies for baffle-former bolts and core shroud bolts	Applicable	Fulfilled for McGuire Unit 1 and Unit 2 by publication of MRP-227-A.
6. Periodicity of the re-examination of "Expansion" inspection category components	Applicable	Fulfilled for McGuire Unit 1 and Unit 2 by publication of MRP-227-A.
7. Updating of MRP-227, Revision 0, Appendix A	Applicable	Fulfilled for McGuire Unit 1 and Unit 2 by publication of MRP-227-A.

6.2 APPLICANT/LICENSEE ACTION ITEMS

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" [27].

McGuire Unit 1 and Unit 2 Compliance

The process used to verify that McGuire Unit 1 and Unit 2 are reasonably represented by the generic industry program assumptions with regard to neutron fluence, temperature, materials, and stress values used in the development of MRP-227-A is as follows:

1. Identification of typical Westinghouse PWR internal components.

   The typical Westinghouse PWR internal components are listed within MRP-191, Table 4-4.

2. Identification of McGuire PWR internal components.

   The McGuire components were identified from the McGuire AMR [14]. Additionally, access plug assembly springs, anti-vibration sleeves, core barrel plugs, lower former plate plugs, and modified upper former plate were also identified for consideration.

3. Comparison of the typical Westinghouse PWR internal components to the McGuire PWR internal components.

   a. Confirmation that no additional items were identified by this comparison (primarily supports Applicant/Licensee Action Item 2).

      The McGuire components were compared to MRP-191 within [55]. The McGuire AMR identifies a component, "Irradiation specimen holder (spring)" [14] that has no corresponding MRP-191 component. The irradiation specimen holder (spring) is a subcomponent of the access plug assembly, which is a subcomponent of the irradiation specimen holder assembly. This component will be referred to as the access plug assembly spring throughout this document. The access plug assembly spring was the only additional component identified for McGuire when comparing the McGuire AMR to MRP-191 [55]. Additionally, several other components outside of the AMR have been identified that do not have corresponding MRP-191 components: the anti-vibration sleeves, core barrel plugs, lower former plate plugs, and modified upper former plate. The anti-vibration sleeves are protective sleeves for the flux thimbles. They were installed within McGuire Unit 1 and Unit 2 to reduce the clearance within the lower internals BMI columns and limit the amount of vibratory motion of the flux thimbles. The access plug assembly spring and anti-vibration sleeves will be assessed within MRP-191 Rev. 2. Therefore, Duke Energy plans to provide a separate submittal to the NRC to apply the FMECA and ranking results of MRP-191 Rev. 2 for the access plug assembly spring and anti-vibration sleeves to close these open items.

      As discussed in Section 4.1.7, McGuire Unit 1 and Unit 2 each had an upflow conversion performed by a vendor other than the McGuire Original Equipment Manufacturer (OEM). The core barrel plugs, lower former plate plugs, and modified upper former plate are a result of this upflow conversion and do not have a corresponding MRP-191 component. These new and modified components will be evaluated for compliance with MRP-227-A by the vendor that performed the upflow conversion and provided to the NRC in a separate submittal.

   b. Confirmation that the materials identified for McGuire are consistent with those materials identified in MRP-191, Table 4-4.

      Most of the materials for McGuire are identical or equivalent to those materials identified in MRP-191 Rev. 0, Table 4-4 for Westinghouse-designed plants. Components considered equivalent are those that were fabricated from a different material of the same material class considered in MRP-191, such as a different type of austenitic stainless steel. Examples include components fabricated from Type 316 SS instead of Type 304 SS, Type 304L SS instead of Type 304 SS, or Type 304 SS instead of Type 316 SS. Types 304, 316, and 304L SS fall under the austenitic stainless steel category and there are no differences in the screening criteria for the materials. With no changes to the susceptibility or degradation mechanisms of concern, the FMECA and functionality analysis are still acceptable.

The potential for alternate materials, specifically CF8, to be used for the guide plates/cards, housing plates, upper guide tube enclosures, and brackets, clamps, terminal blocks, and conduit straps – conduit positioners was identified within [55] in comparison to MRP-191 Rev. 0 [10] for both McGuire Units. When reviewing the potential CASS components for McGuire Unit 1, it was determined that the upper guide tube enclosures, guide plates/cards, and housing plates were constructed from Type 304 SS [65]. When reviewing the potential CASS components for McGuire Unit 2, it was determined that the housing plates were constructed from Type 304 SS [65]. The components within McGuire Unit 1 that were determined to be potentially CASS but were not considered to be CASS in MRP-191, Rev. 0 are the brackets, clamps, terminal blocks, and conduit straps – conduit positioners [65]. The components within McGuire Unit 2 that were determined to be potentially CASS but were not considered to be CASS in MRP-191, Rev. 0 are the upper guide tube enclosures, the guide plates/cards, and the brackets, clamps, terminal blocks, and conduit straps – conduit positioners [65].

MRP-191 Rev. 1 [43] considers the use of CF8 materials for the upper guide tube enclosures, guide plates/cards, and brackets, clamps, terminal blocks, and conduit straps – conduit positioners. Although CF8 for the upper guide tube enclosures and guide plates/cards introduces TE and IE, the components were still categorized into the same FMECA categories in MRP-191 Rev. 1 as their material counterparts in MRP-191 Rev. 0. Although CF8 for the brackets, clamps, terminal blocks, and conduit straps – conduit positioners introduces TE, the component was still placed in the same FMECA category in MRP-191 Rev. 1 as its material counterpart in MRP-191 Rev. 0. The FMECA and ranking resulted in the CF8 guide plates/cards remaining in Category C, the CF8 housing plates remaining in Category A, the CF8 upper guide tube enclosures remaining in Category A, and the CF8 brackets, clamps, terminal blocks, and conduit straps – conduit positioners remaining in Category A [43]. Therefore, no change to the McGuire MRP-227-A inspection requirements is required as a result of the inclusion of CF8 for these components within MRP-191 Rev. 1.

c. Confirmation that the McGuire internals are the same as, or equivalent to, the typical Westinghouse PWR internals regarding design and fabrication and that any modifications do not impact the applicability of MRP-227-A.

Modifications to the McGuire reactor internals include a control rod guide tube support pin replacement, a power uprating project, and an upflow conversion. Details of these replacements are retained in plant records and described in Section 4 of this document. With the exception of the upflow conversion, the design has been maintained over the lifetime of the plant as specified by the Original Equipment Manufacturer (OEM) [55]. McGuire Unit 1 and Unit 2 each had an upflow conversion performed by a vendor other than the McGuire Original Equipment Manufacturer (OEM). Duke Energy has identified two upflow modification components outside of the AMR that do not have corresponding MRP-191 components, the core barrel plugs and the lower former plate plugs. Additionally, the upper former plate was modified as part of the upflow conversion and was not considered in MRP-191. These new components from the upflow conversion will

be evaluated for compliance with MRP-227-A by the vendor that performed the upflow conversion and will be provided to the NRC in a separate submittal.

4. Confirmation that the McGuire operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns.

   The McGuire operating history is consistent with the assumptions in MRP-227-A with regard to neutron fluence.

   a. The FMECA and functionality analyses for MRP-227-A 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. McGuire Unit 1 had approximately 16.71 years of operation with fresh fuel assemblies at peripheral locations (high-leakage core loading pattern) [63]. McGuire Unit 2 had approximately 10.95 years of operation with fresh fuel assemblies at peripheral locations (high-leakage core loading pattern) [63]. The low-leakage loading pattern has been applied to all subsequent core designs through current operation. No change to the low-leakage core design philosophy is anticipated for the extended plant operating license [55]. By operating with a high-leakage core design for less than 30 years of operation, McGuire has taken a conservative approach. Therefore, McGuire meets the fluence and fuel management assumptions in MRP-191 and requirements for MRP-227-A application.

   b. McGuire Unit 1 and Unit 2 have operated under base load conditions for the majority of the life of the plant [55]. Therefore, McGuire satisfies the assumptions in the MRP documents regarding operational parameters affecting fluence.

5. Confirmation that the McGuire RVI materials operated at temperatures within the original design basis parameters.

   The McGuire reactor coolant systems operate between Tcold and Thot [55], 556°F for Tcold and 614°F for Thot for both McGuire Unit 1 and Unit 2. The design temperature for the reactor vessel is 650°F [55]. The operating history for McGuire is within original design basis parameters and therefore consistent with the assumptions used to develop the MRP-227-A aging management strategy with regard to temperature operational parameters.

6. Confirmation that the McGuire stress values are consistent with the assumptions in MRP-227-A.

   With the exception of the upflow conversion, the design has been maintained over the lifetime of the plant as specified by the Original Equipment Manufacturer (OEM) [55]. Upflow conversion has a number of beneficial effects on the stresses in the reactor vessel internals, such as reducing the differential pressure across the baffle plates. Therefore, the upflow conversion would not be expected to increase the stress values on the internals components, and the McGuire stress values are represented by the assumptions in MRP-191, MRP-232, and MRP-227-A.

7. Resolution of the MRP 2013-025 Applicability Guidelines

The NRC held several public meetings to discuss their expectations and concerns regarding industry responses to MRP-227-A, A/LAIs 1 and 2. These concerns were addressed to owners of currently operating pressurized water reactor plants designed by Westinghouse and Combustion Engineering (CE). At these meetings, the NRC, Westinghouse, EPRI, and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate the applicability of MRP-227-A. As a result, [64] was developed to provide utilities with the basis for a plant to respond to the NRC's Request for Additional Information (RAI) to demonstrate compliance with the basic technical applicability assumptions in MRP-227-A for originally licensed and uprated conditions. Question 1 and 2 from [64] are discussed below.

a. Does the plant have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)

   MRP-232 [62] calls for the demonstration of plant specific materials, fabrication, and design to meet the assumptions inherent in MRP-191 Rev. 0. The cold work assessment for McGuire performed a detailed material fabrication and design assessment of each unit and concluded that no non-fastener materials of 20% cold work or greater were used in construction [61]. Therefore, the inspection sampling approach of MRP-232 is applicable for McGuire. The cold work assessment for McGuire also concluded that the plant-specific material fabrication and design was consistent with the MRP-191 basis, and the MRP-227-A sampling inspection aging management requirements as related to cold work are directly applicable to McGuire [61].

b. Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

   McGuire has not utilized atypical fuel design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for these units, including power changes/uprates that have occurred over their operating lifetimes [63]. This conclusion is based on comparisons of the McGuire Unit 1 and Unit 2 core geometries and operating characteristics with the MRP-227-A applicability guidelines for Westinghouse-designed reactors specified in [64].

Conclusion

The McGuire evaluation for A/LAI 1 of the NRC SE on MRP-227, revision 0, confirms that MRP-227-A is applicable to McGuire Unit 1 and Unit 2, with two exceptions:

- The McGuire AMR identifies a component, access plug assembly spring [14], that is not contained in the FMECA and functionality analysis [10]. The anti-vibration sleeves were also identified as falling outside the scope of MRP-191. Duke Energy plans to provide a separate

submittal to the NRC to apply the FMECA and ranking results of MRP-191 Rev. 2 for the access plug assembly spring and anti-vibration sleeves.

- McGuire Unit 1 and Unit 2 each had an upflow conversion performed by a vendor other than the McGuire OEM. The core barrel plugs and lower former plate plugs are a result of this upflow conversion and do not have a corresponding MRP-191 component. Additionally, the upper former plate was modified as part of the upflow conversion and was not considered in MRP-191. These new and modified components from the modification will be evaluated for compliance with MRP-227-A by the vendor that performed the upflow conversion and provided to the NRC in a separate submittal.

McGuire meets the requirements for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components, with two exceptions. The open items have been captured in Table 8-1 of this document. Therefore this applicant/licensee action item is considered fulfilled for McGuire with the submittals that will be sent separately to the NRC to confirm application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals 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" [27].

McGuire Unit 1 and Unit 2 Compliance

This Applicant/Licensee Action Item (A/LAI) requires comparison of the RVI components that are within the scope of license renewal for McGuire to those components contained in MRP-191 Rev. 0, Table 4-4 [10]. A detailed tabulation of the McGuire RVI components was completed within [55] and compared favorably to the typical Westinghouse PWR internals components in MRP-191 Rev. 0. The McGuire AMR identifies a component, access plug assembly spring [14] that has no corresponding MRP-191 component. This was the only additional component identified for both McGuire Unit 1 and Unit 2 when comparing the AMR to MRP-191. Additionally, several other components outside of the AMR have been identified that do not have corresponding MRP-191 components: the anti-vibration sleeves, core barrel plugs, lower former plate plugs, and modified upper former plate. The access plug assembly spring and anti-vibration sleeves will be assessed within MRP-191 Rev. 2. Therefore, Duke Energy plans to provide a separate submittal to the NRC to apply the FMECA and ranking results of MRP-191 Rev. 2 for the access plug assembly spring and anti-vibration sleeves to close these open items. As discussed in Section 4.1.7, McGuire Unit 1 and Unit 2 each had an upflow conversion performed by a vendor other than the

McGuire Original Equipment Manufacturer (OEM). The core barrel plugs and lower former plate plugs are a result of this upflow conversion and do not have a corresponding MRP-191 component. Additionally, the upper former plate was modified as part of the upflow conversion. These new and modified components will be evaluated for compliance with MRP-227-A by the vendor that performed the upflow conversion and provided to the NRC in a separate submittal.

Several components have different materials than specified in MRP-191 Rev. 0, Table 4-4. The potential for alternate materials, specifically CF8, to be used for the guide plates/cards, housing plates, upper guide tube enclosures, and brackets, clamps, terminal blocks, and conduit straps – conduit positioners was identified within [55] in comparison to MRP-191 Rev. 0 [10] for both McGuire Units. When reviewing the potential CASS components for McGuire Unit 1, it was determined that the upper guide tube enclosures, guide plates/cards, and housing plates were constructed from Type 304 SS [65]. When reviewing the potential CASS components for McGuire Unit 2, it was determined that the housing plates were constructed from Type 304 SS [65]. The components within McGuire Unit 1 that were determined to be potentially CASS but were not considered to be CASS in MRP-191, Rev. 0 are the brackets, clamps, terminal blocks, and conduit straps – conduit positioners [65]. The components within McGuire Unit 2 that were determined to be potentially CASS but were not considered to be CASS in MRP-191, Rev. 0 are the upper guide tube enclosures, the guide plates/cards, and the brackets, clamps, terminal blocks, and conduit straps – conduit positioners [65].

MRP-191 Rev. 1 [43] considers the use of CF8 materials for the upper guide tube enclosures, guide plates/cards, and the brackets, clamps, terminal blocks, and conduit straps – conduit positioners. Although CF8 for the upper guide tube enclosures and guide plates/cards introduces TE and IE, the components were still placed in the same FMECA categories in MRP-191 Rev. 1 as their material counterparts in MRP-191 Rev. 0. Although CF8 for the brackets, clamps, terminal blocks, and conduit straps – conduit positioners introduces TE, the component was still placed in the same FMECA category in MRP-191 Rev. 1 as its material counterpart in MRP-191 Rev. 0. The FMECA and ranking resulted in the CF8 guide plates/cards remaining in Category C, the CF8 housing plates remaining in Category A, the CF8 upper guide tube enclosures remaining in Category A, and the brackets, clamps, terminal blocks, and conduit straps – conduit positioners remaining in Category A [43]. Therefore, no change to the McGuire MRP-227-A inspection requirements is required as a result of the inclusion of CF8 for these components within MRP-191 Rev. 1.

This supports the requirement to provide assurance that the effects of aging on the McGuire RVI components within the scope of license renewal, but not included in the generic Westinghouse-designed RVI components from Table 4-4 of MRP-191, will be managed for the period of extended operation.

The generic scoping and screening of the RVI, as summarized in MRP-191 and MRP-232, is applicable to McGuire with no modifications, with the exception of the access plug assembly spring, anti-vibration sleeves, core barrel plugs, lower former plate plugs, and modified upper former plate.

Conclusion

With the exception of the access plug assembly spring, anti-vibration sleeves, core barrel plugs, lower former plate plugs, and modified upper former plate, all of the reactor vessel internal components within the scope of license renewal at McGuire were addressed in MRP-191, Rev. 0 or Rev. 1. Material

differences found between McGuire and MRP-191, Rev. 0, have been addressed within MRP-191 Rev. 1. Thus, the age-related degradation management strategy of MRP-227-A for the reactor vessel internals components is applicable to McGuire Unit 1 and Unit 2 for all of the components, with two exceptions:

- The McGuire AMR identifies a component, access plug assembly spring [14], that is not contained in the FMECA and functionality analysis [10]. The anti-vibration sleeves were also identified as falling outside the scope of MRP-191. Duke Energy plans to provide a separate submittal to the NRC to apply the FMECA and ranking results of MRP-191 Rev. 2 for the access plug assembly spring and anti-vibration sleeves.
- McGuire Unit 1 and Unit 2 each had an upflow conversion performed by a vendor other than the McGuire OEM. The core barrel plugs and lower former plate plugs are a result of this upflow conversion and do not have a corresponding MRP-191 component. Additionally, the upper former plate was modified as part of the upflow conversion and was not considered in MRP-191. These new and modified components from the modification will be evaluated for compliance with MRP-227-A by the vendor that performed the upflow conversion and provided to the NRC in a separate submittal.

McGuire meets the requirements for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components, with two exceptions. The open items have been captured in Table 8-1 of this document. Therefore this applicant/licensee action item is considered fulfilled for McGuire with the submittals that will be sent separately to the NRC to confirm application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals 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"* [27].

McGuire Unit 1 and Unit 2 Compliance

McGuire is compliant with the applicable requirements in Table 4-9 of MRP-227-A as shown in Appendix C, Table C-3. This is detailed in the plant-specific McGuire program documents for ASME Section XI [2] and the McGuire flux thimble tube program [30]. McGuire has a number of programs and activities that support the aging management of the RVI as discussed in Section 4.1.

MRP-227-A states that performance monitoring of the support pins should follow supplier recommendations, and thus they are not included in Table 4-9 of MRP-227-A. As discussed in Section

4.1.5, McGuire Unit 1 and Unit 2 replaced the INCONEL® Alloy X-750 support pins with 316 SS support pins [26]. There are no supplier recommendations for performance monitoring of the 316 SS support pins. The replacement support pins utilized improved materials (strain hardened austenitic stainless steel) that support the proactive management of aging in reactor internals components. Support pins fabricated from 316 SS are not susceptible to PWSCC which was the primary failure mechanism for Alloy X-750 support pins [43]. The Type 316 SS support pins were evaluated in MRP-191. While they screened in for wear, fatigue, and stress relaxation, the FMECA and ranking resulted in them being assigned to Category A [43]. Therefore no additional inspections of the McGuire 316 SS support pins are required by the supplier or per MRP-227-A.

Conclusion

The McGuire Existing Programs credited in Table 4-9 of MRP-227-A, as well as the support pin replacements [26], are adequate to ensure aging degradation is adequately managed. McGuire complies with Applicant/Licensee Action Item 3 of the NRC SE on MRP-227, Revision 0, and therefore meet the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components. Therefore this applicant/licensee action item is considered fulfilled for McGuire.

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"* [27].

McGuire Unit 1 and Unit 2 Compliance

This Applicant/Licensee Action Item is not applicable to McGuire 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 McGuire.

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" [27].

McGuire Unit 2 Compliance

As shown in Table C-1, McGuire Unit 1 and Unit 2 contain a Type 403 hold-down spring. MRP-227-A requirements apply only to plants with Type 304 SS hold-down springs. Therefore, this Applicant/Licensee Action Item is not applicable to McGuire.

Conclusion

Applicant/Licensee Action Item 5 of the NRC SE on MRP-227, Revision 0 is not applicable to McGuire.

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" [27].

McGuire Unit 1 and Unit 2 Compliance

This Applicant/Licensee Action Item is not applicable to McGuire 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 McGuire.

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" [27].

McGuire Unit 1 and Unit 2 Compliance

Section 3.3.7 from the Staff's final SE on MRP-227 Revision 0 states the following is an acceptable approach:

"For CASS, if the application of applicable screening criteria for the component's material demonstrates that the components are not susceptible to either thermal embrittlement or irradiation embrittlement, or the synergistic effects of thermal embrittlement and irradiation embrittlement combined, then no other evaluation would be necessary. For assessment of CASS materials, the licensee or applicant 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" (NRC ADAMS Accession No. ML003717179) as the basis for determining whether the CASS materials are susceptible to the thermal aging mechanism."

<u>Thermal Embrittlement</u>

The McGuire Unit 1 and Unit 2 RVI CASS components and the assessment of their susceptibility to TE are summarized in Table 6-2 and Table 6-3, respectively.

The upper guide tube enclosures, guide plates/cards, housing plates, brackets, clamps, terminal blocks, and conduit straps – conduit positioners, upper head injection (UHI) flow column bases, upper support column bases, bottom-mounted instrumentation (BMI) column cruciforms, intermediate flanges, and lower flanges were determined to either be constructed of CASS or potentially constructed of CASS for McGuire [55]. When reviewing the potential CASS components for McGuire Unit 1, it was determined that the upper guide tube enclosures, guide plates/cards, lower flanges, intermediate flanges, and housing plates were constructed from Type 304 SS [65]. When reviewing the potential CASS components for McGuire Unit 2, it was determined that the lower flanges and housing plates were constructed from Type 304 SS [65]. The lower support column bodies are not CASS material for either McGuire Unit 1 or Unit 2.

Based on the criteria in [28], the McGuire CASS upper support column bases, CASS UHI flow column bases, and CASS BMI column cruciforms are not susceptible to TE [65].

The McGuire brackets, clamps, terminal blocks, and conduit straps – conduit positioners are also CASS. Conclusive confirmation of material composition under TE susceptibility thresholds was not demonstrated for McGuire Unit 1. It is conservatively assumed that the conduit positioners in McGuire Unit 1 are potentially susceptible to TE. 513 of the 520 conduit positioners in McGuire Unit 2 can be shown to be below the TE susceptibility threshold. Conclusive confirmation of material composition under TE susceptibility thresholds was not demonstrated for the remaining seven conduit positioners, and so they must be assumed potentially susceptible to TE. The development of MRP-191 Revision 1 evaluated CASS material for the conduit positioners through the FMECA process. Although CF8 for the conduit positioners introduces TE, the component was placed in the same FMECA category in MRP-191 Rev. 1 as its material counterpart in MRP-191 Rev. 0. The FMECA and ranking resulted in the CF8 conduit positioners remaining in Category A [43]. Therefore, no change to the McGuire MRP-227-A inspection requirements is required as a result of the inclusion of CF8 for these components within MRP-191 Rev. 1.

For McGuire Unit 2, conclusive confirmation of material composition under TE susceptibility thresholds was not demonstrated for the upper guide tube enclosures, guide plates/cards, or for the intermediate flanges; thus, it is conservatively assumed that they are potentially susceptible to TE. Of these, only the intermediate flanges were evaluated for TE in the development of MRP-227-A [27]. MRP-191 Revision 1 evaluated CASS material for the upper guide tube enclosures and guide plates/cards through the FMECA process. Although CF8 for these components introduces TE and IE, the components were still categorized into the same FMECA categories in MRP-191 Rev. 1 as their counterparts in MRP-191 Rev. 0. The FMECA and ranking resulted in the CF8 guide plates/cards remaining in Category C and the CF8 upper guide tube enclosures remaining in Category A [43]. Therefore, no change to the McGuire MRP-227-A inspection requirements is required as a result of the inclusion of CF8 for these components within MRP-191 Rev. 1.

Irradiation Embrittlement

Irradiation may also cause a material to become embrittled. The CASS upper support column bases, CASS UHI flow column bases, and the CASS BMI column cruciforms screened-in at the MRP-191 Rev. 0 irradiation screening level [10]; thus, for these components, susceptibility to

irradiation embrittlement (IE) was considered in the development of MRP-227-A [27]. In MRP-191 Rev. 1, the CASS intermediate flanges, upper guide tube enclosures, guide plates/cards, and the brackets, clamps, terminal blocks, and conduit straps - conduit positioners were screened for IE. MRP-191 Rev. 1 considered the upper guide tube enclosures and guide plates/cards to be susceptible to IE. Although these components were conservatively assumed to be susceptible to IE, the FMECA and ranking resulted in the CF8 upper guide tube enclosures remaining in Category A and the CF8 guide plates/cards remaining in Category C. The CF8 intermediate flanges and brackets, clamps, terminal blocks, and conduit straps – conduit positioners did not screen in for IE within MRP-191 Rev. 1 and both components remained in Category A. Therefore, no change to the McGuire MRP-227-A inspection requirements is required as a result of the inclusion of CF8 for these components within MRP-191 Rev. 1.

The only martensitic stainless steel component identified in the McGuire RVI is the hold-down spring. The Type 403 SS hold-down spring is evaluated in MRP-191, Rev. 0. The Type 403 SS hold-down spring is below the screening threshold for IE. Though the Type 403 SS hold-down spring screens in for TE, the FMECA and ranking resulted in them being assigned to Category A. Therefore no additional inspections are required per MRP-227-A.

No martensitic precipitation-hardened stainless steel components were identified in the McGuire RVI.

Conclusion

The McGuire CASS RVI components will maintain their functionality during the period of extended operation. Continued application of the strategy of MRP-227-A will meet the requirement for managing age-related degradation of the McGuire CASS and martensitic stainless steel RVI components. Therefore this applicant/licensee action item is considered fulfilled for McGuire.

Table 6-2 Summary of McGuire Unit 1 CASS Components and Their Susceptibility to TE

CASS Component	Molybdenum Content	Casting	Ferrite Content (%)	Susceptibility to TE (Based on the NRC Criteria [28])
Upper Instrumentation Conduits and Supports – Brackets, clamps, terminal blocks, and conduit straps (conduit positioners)	Low, 0.5 max	Static	> 20(1)	Potentially Susceptible(1)
Upper Plenum – UHI Flow Column Bases	Low, 0.5 max	Static	≤ 20	Not Susceptible
Upper Support Column Assemblies – Column Bases	Low, 0.5 max	Static	≤ 20	Not Susceptible
BMI Column Assemblies – BMI Column Cruciforms	Low, 0.5 max	Static	≤ 20	Not Susceptible

Note:
1. Where a component-specific certified material test report is not available, the ferrite content is calculated based on permitted variations in ASTM A351, Grade CF8 chemistry requirements. Allowable variants of the material specification chemistry requirements may result in ferrite content estimations higher than 20%; thus, the ferrite content is identified as potentially exceeding 20 percent.

Table 6-3 Summary of McGuire Unit 2 CASS Components and Their Susceptibility to TE

CASS Component	Molybdenum Content	Casting	Ferrite Content (%)	Susceptibility to TE (Based on the NRC Criteria [28])
Control Rod Assemblies and Flow Downcomers – Upper Guide Tube Enclosures(1), (3)	Low, 0.5 max	Static	> 20(2)	Potentially Susceptible(2)
Control Rod Assemblies and Flow Downcomers – Guide Plates/Cards(1), (3)	Low, 0.5 max	Static	> 20(2)	Potentially Susceptible(2)
Control Rod Assemblies and Flow Downcomers – Flanges – intermediate(1), (3)	Low, 0.5 max	Static	> 20(2)	Potentially Susceptible(2)

Table 6-3 Summary of McGuire Unit 2 CASS Components and Their Susceptibility to TE

CASS Component	Molybdenum Content	Casting	Ferrite Content (%)	Susceptibility to TE (Based on the NRC Criteria [28])
Upper Instrumentation Conduits and Supports – Brackets, clamps, terminal blocks, and conduit straps (conduit positioners)	Low, 0.5 max	Static	≤ 20	513 of 520 Not Susceptible
			> 20(2)	7 Potentially Susceptible(2)
Upper Plenum – UHI Flow Column Bases	Low, 0.5 max	Static	≤ 20	Not Susceptible
Upper Support Column Assemblies – Column Bases	Low, 0.5 max	Static	≤ 20	Not Susceptible
BMI Column Assemblies – BMI Column Cruciforms	Low, 0.5 max	Static	≤ 20	Not Susceptible

Notes:
1. This component has CASS listed as an alternate material.
2. Where a component-specific certified material test report is not available, the ferrite content is calculated based on permitted variations in ASTM A351, Grade CF8 chemistry requirements. Allowable variants of the material specification chemistry requirements may result in ferrite content estimations higher than 20%; thus, the ferrite content is identified as potentially exceeding 20 percent.
3. Potentially TE susceptible components are on the unrodded 15x15 guide tube assemblies.

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" [27].

McGuire Unit 1 and Unit 2 Compliance

In a letter dated March 19, 2014, Duke Energy notified the NRC of its intent to submit a plant-specific RV Internals inspection plan for McGuire to implement MRP-227-A [31]. In a letter dated February 23, 2017, Duke Energy notified the NRC that the expected inspection plan submittal date for McGuire is Fall 2017 [39].

Section 3.5.1 of the final SE on MRP-227, Revision 0 states:

> *"In addition to the implementation of MRP-227 in accordance with NEI 03-08, applicants/licensees whose licensing basis contains a commitment to submit a PWR RVI AMP and/or inspection program shall also make a submittal for NRC review and approval to credit*

their implementation of MRP-227, as amended by this SE. An applicant's/licensee's application to implement MRP-227, as amended by this SE shall include the following items (1) and (2)...

1. An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.

2. To ensure the MRP-227 program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components."

LR-ISG-2011-04 was later issued and revised the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A. Thus, the AMP should address LR-ISG-2011-04. The 10 program elements are addressed in Section 5 of this document.

This document also provides the McGuire RVI inspection plan, including examination method, frequency, acceptance criteria, coverage, etc. The plant-specific action items are addressed in Section 6 of this document.

Conclusion

This document includes the information identified in Section 3.5.1 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. Therefore, this applicant/licensee action item is considered fulfilled for McGuire.

7 INSPECTION PLAN AND IMPLEMENTATION SCHEDULE

The components identified in Table 7-1 and Table 7-2 cover the Primary Component inspection scope for McGuire Unit 1 and Unit 2, respectively, and are based on Table 4-3 of MRP-227-A as modified by any applicable interim guidance. Additionally, Table C-1 of this document provides the degradation mechanism(s) being managed, the examination coverage, and any associated Expansion Component links for each of the Primary Components.

Table 7-1 and Table 7-2 provide the due dates for the listed Primary Components; these represent the latest opportunity for McGuire to perform the initial examinations and McGuire may elect to perform examinations earlier than the due dates listed. Many of the initial inspection due dates in MRP-227-A are based on the start of the period of extended operation. The period of extended operation begins June 13, 2021 for McGuire Unit 1 and March 4, 2023 for McGuire Unit 2. Additionally, the McGuire refueling outages corresponding to inspection due dates that are based on EFPY in Table 7-1 and Table 7-2 are conservative estimates based on 18-month fuel cycles and are subject to change. Subsequent exams will be performed in accordance with MRP-227-A as modified by any applicable interim guidance.

The schedule for implementation of aging management requirements for Expansion Components will depend on the findings from the examinations of the Primary components at McGuire. Table C-2 of this document provides the Expansion Components from MRP-227-A, the applicability to McGuire, the degradation mechanism(s) being managed, the examination method/frequency, the examination coverage, and the Primary Component Links.

Table C-4 of this document provides the examination acceptance criteria and expansion criteria for each of the Primary Components. The examinations specified in Table 7-1, Table 7-2, Table C-1, and Table C-2 of this document will be conducted in accordance with the revision of MRP-228 in effect at the time of the examination.

Table C-3 of this document provides the degradation mechanism(s) being managed, the Existing Program being credited (e.g., ASME Section XI, IEB 88-09), the examination method, and the examination coverage for each of the Existing Programs Components from MRP-227-A. The Existing Programs Components will continue to be examined in accordance with the credited Existing Program requirements.

Should a change occur in plant operational practices or should 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 McGuire Unit 1 Primary Component Inspection Plan

RVI Component	Examination Method	Examination Frequency	McGuire Unit 1 Due Date (Refueling Outage)	Projected EFPY
Control Rod Guide Tube Assembly Guide plate (cards)	Visual (VT-3) examination	Per the schedule requirements of WCAP-17451-P Section 5 including subsequent examinations (Note 1)	1R26 Spring 2019	30.3
Control Rod Guide Tube Assembly Lower flange welds	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.	1R29 Fall 2023	34.8
Core Barrel Assembly Upper core barrel flange weld	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.	1R29 Fall 2023	34.8
Core Barrel Assembly Upper and lower core barrel cylinder girth 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.	1R29 Fall 2023	34.8
Core Barrel Assembly Lower core barrel flange weld	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.	1R29 Fall 2023	34.8
Baffle-Former Assembly Baffle-edge bolts	Visual (VT-3) examination	Baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	1R32 Spring 2028	39.3

Table 7-1 McGuire Unit 1 Primary Component Inspection Plan

RVI Component	Examination Method	Examination Frequency	McGuire Unit 1 Due Date (Refueling Outage)	Projected EFPY
Baffle-Former Assembly Baffle-former bolts	Volumetric (UT) examination.	Baseline examination between 25 and 35 EFPY and subsequent examinations on a ten-year interval.	1R29 Fall 2023	34.8
Baffle-Former Assembly Assembly	Visual (VT-3) examination	Baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	1R32 Spring 2028	39.3

Note:

1. MRP-227-A provided an examination frequency for Guide plates (cards) of "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." MRP issued interim guidance to MRP-227-A in Letter MRP 2014-006, changing the examination frequency to "Per the schedule requirements of WCAP-17451-P Section 5 including subsequent examinations." Per WCAP-17451-P [29], McGuire Unit 1 was due for initial guide card wear inspections between 24 to 28 EFPY. However, there is a provision within [29] that allows for a schedule extension or scope reduction for performing guide card wear measurement based on guide card wear screening through video inspection. A guide card wear screening was performed for McGuire Unit 1, resulting in a recommendation to perform initial guide card wear measurements by 31.66 EFPY [41].

Table 7-2 McGuire Unit 2 Primary Component Inspection Plan

RVI Component	Examination Method	Examination Frequency	McGuire Unit 2 Due Date (Refueling Outage)	Projected EFPY
Control Rod Guide Tube Assembly Guide plate (cards)	Visual (VT-3) examination	Per the schedule requirements of WCAP-17451-P Section 5 including subsequent examinations (Note 1)	2R25 Fall 2018	29.7
Control Rod Guide Tube Assembly Lower flange welds	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.	2R29 Fall 2024	35.7
Core Barrel Assembly Upper core barrel flange weld	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.	2R29 Fall 2024	35.7
Core Barrel Assembly Upper and lower core barrel cylinder girth 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.	2R29 Fall 2024	35.7
Core Barrel Assembly Lower core barrel flange weld	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.	2R29 Fall 2024	35.7
Baffle-Former Assembly Baffle-edge bolts	Visual (VT-3) examination	Baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	2R31 Fall 2027	38.7
Baffle-Former Assembly Baffle-former bolts	Volumetric (UT) examination.	Baseline examination between 25 and 35 EFPY and subsequent examinations on a ten-year interval.	2R28 Spring 2023	34.2

Table 7-2 McGuire Unit 2 Primary Component Inspection Plan

RVI Component	Examination Method	Examination Frequency	McGuire Unit 2 Due Date (Refueling Outage)	Projected EFPY
Baffle-Former Assembly Assembly	Visual (VT-3) examination	Baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	2R31 Fall 2027	38.7

Note:

1. MRP-227-A provided an examination frequency for Guide plates (cards) of "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." MRP issued interim guidance to MRP-227-A in Letter MRP 2014-006, changing the examination frequency to "Per the schedule requirements of WCAP-17451-P Section 5 including subsequent examinations." Recent OE found the schedule requirements in WCAP-17451-P to be non-conservative for the guide tube design of McGuire Unit 2. According to NSAL-17-1 and McGuire Unit 2 guide card wear screening through video inspection [42], McGuire Unit 2 should perform initial guide card inspection no later than Fall 2018.

8 SUMMARY AND CONCLUSIONS

This report documents and provides a description of the McGuire RVI AMP and Inspection Plan and how it relates to the McGuire RVI Program for management of aging effects through the period of extended operation. This McGuire RVI AMP and Inspection Plan is based on MRP-227-A. Section 5.0 of this document demonstrates that the McGuire RVI Program will meet the intent of the ten AMP elements described in NUREG-1801, Rev. 2, Chapter XI, AMP XI.M16A, as modified by LR-ISG-2011-04. Section 6.0 of this document fulfills application/licensee plant-specific action items from MRP-227-A.

The McGuire RVI Program will include this AMP and Inspection Plan and will demonstrate that the program adequately manages the effects of aging for RVI components and establishes the basis for providing reasonable assurance that the RVI components will remain functional through the period of extended operation.

Duke Energy is revising its commitments for RVI Inspections from those that currently exist in the McGuire UFSAR to the inspection guidelines provided in MRP-227, as approved by the NRC. This MNS RVI AMP and Inspection Plan, as documented herein, is based on MRP-227-A. Once the McGuire RVI AMP and Inspection Plan is approved by the NRC, the McGuire UFSAR will be updated as required.

Table 8-1 McGuire Aging Management Program and Inspection Plan Open Items

	Open Item Component	Component Reference Sections
1	Access Plug Assembly Spring	Section 6.2.1, Section 6.2.2, and Appendix B
2	Anti-Vibration Sleeves	Section 6.2.1 and Section 6.2.2
3	Upflow Conversion Modification (This includes the core barrel plugs, lower former plate plugs, and the modified upper former plate)	Section 4.1.7, Section 6.2.1, and Section 6.2.2

9 REFERENCES

1. U.S. Nuclear Regulatory Commission NUREG-1772, "Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station Units 1 and 2, and Catawba Nuclear Station Units 1 and 2," Docket Nos. 50-369, 50-370, 50-413, 50-414. January 2003.

2. Duke Energy Functional Area Manual, Rev. 28, "ASME Section XI Program Functional Area Manual," November 3, 2016.

3. ASME Boiler and Pressure Vessel Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Applicable Edition and Addenda, ASME International.

4. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP 227-Rev. 0).* EPRI, Palo Alto, CA: 2008. 1016596.

5. U.S. Nuclear Regulatory Commission, Title 10 of the Code of Federal Regulations (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," August 2007.

6. McGuire Nuclear Station Units 1 and Unit 2 Updated Final Safety Analysis Report, Revision 19, October 9, 2015.

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

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

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

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

11. U.S. Nuclear Regulatory Commission Safety Evaluation, Accession Number ML111600498, "Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Rev. 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC NO. ME0680)," June 22, 2011.

12. *Materials Reliability Program: Inspection Standard for Pressurized Water Reactor Internals – 2015 Update (MRP-228, Rev. 2).* EPRI, Palo Alto, CA: 2015. 3002005386.

13. Westinghouse Report WCAP-17096-NP-A, Rev. 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 2016.

14. Duke Energy License Renewal Application, "Application to Renew the Operating Licenses of McGuire Nuclear Station Units 1 & 2 and Catawba Nuclear Station Units 1 & 2," June 2001.

15. Nuclear Energy Institute Document, NEI 03-08, Rev. 3, "Guideline for the Management of Materials Issues," Nuclear Energy Institute, Washington, DC, February 2017.

16. Duke Energy Document, CSD-CP-MNS-0001, Rev. 3, "McGuire Primary Chemistry Strategic Plan," June 26, 2017.

17. *Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1, Revision 7*. EPRI, Palo Alto, CA; 2014, 3002000505.

18. U.S. Nuclear Regulatory Commission Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 26, 1988.

19. Westinghouse Report WCAP-15028-NP, Rev. 1, "Guide Tube Cold-Worked 316 Replacement Support Pin Development Program," June 2011.

20. Duke Energy Topical Report, DUKE-QAPD-001, "Quality Assurance Program Description," Amendment 42.

21. Westinghouse Letter, DCP-01-007, "Duke Power Company McGuire Unit 2 Reactor Internals Material Listing," February 7, 2001.

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

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

24. Duke Energy Nuclear Development / Operating Fleet Administrative Procedure, AD-PI-ALL-0100, Rev. 7, "Corrective Action Program," September 27, 2016.

25. Duke Energy Administrative Procedure, AD-PI-ALL-0400, Rev. 3, "Operating Experience Program," January 25, 2017.

26. Westinghouse Report WCAP-15252, Rev. 1, "Duke Energy Corporation, McGuire and Catawba Units 1 & 2 Replacement CW 316 Support Pin Design Qualification Report," March 2000.

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

28. U.S. Nuclear Regulatory Commission Letter, "License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components*," May 19, 2000.

29. Westinghouse Report WCAP-17451-P, Rev. 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections," October 2013.

30. McGuire Units 1 and 2 Engineering Calculation, MCC-1553.03-00-0014, Rev. 7, "Incore Instrumentation Thimble Tube Wear Review," March 28, 2017.

31. Duke Energy Letter, RA-14-0005, "Letter of Intent to Submit Plant Specific Reactor Vessel Internals Inspection Plans," March 19, 2014.

32. EPRI Letter MRP 2014-006, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), EPRI, Palo Alto, CA: 2011. 1022863, Transmittal of Interim Guidance," February 18, 2014.

33. Westinghouse Letter, DCP-01-006, "Duke Power Company McGuire Unit 1 Reactor Internals Material Listing," February 7, 2001.

34. Duke Energy License Amendment Request, "License Amendment Request for Measurement Uncertainty Recapture Power Uprate," March 5, 2012.

35. U.S. Nuclear Regulatory Commission ML13073A041, "McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME8213 and ME8214)," May 16, 2013.

36. Duke Energy Administrative Procedure, AD-EG-ALL-1911, Rev. 1, "Reactor Internals Program Implementation," March 24, 2016.

37. Duke Energy Administrative Program and Process Description, PD-EG-ALL-1911, Rev. 1, "Reactor Internals Program," March 24, 2016.

38. Duke Energy Letter, "Letter of Intent to adopt Materials Reliability Program 227, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," June 16, 2010.

39. Duke Energy Letter, MNS-17-008, "Letter of Intent to Submit Plant-Specific Reactor Vessel Internals Inspection Plans," February 23, 2017.

40. Westinghouse Technical Bulletin, TB-14-5, "Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation," August 25, 2014.

41. Westinghouse Letter, LTR-RIDA-15-147, Rev. 0, "McGuire Unit 1 Guide Tube Foreign Material Exclusion Video Review," August 28, 2015.

42. Westinghouse Letter, LTR-RIDA-16-38, Rev. 0, "Duke Energy Foreign Material Exclusion (FME) Video Inspection for Guide Card Wear," April 8, 2016.

43. *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191, Revision 1)*. EPRI, Palo Alto, CA: 2016. 3002007960.

44. Westinghouse Letter, LTR-RIAM-12-105, "MRP-191 Failure Modes, Effects, and Criticality Analysis (FMECA) Impact Assessment as a Result of Alternate Materials for Reactor Vessel Internals Upper Instrumentation Conduit and Supports Brackets, Clamps, Terminal Blocks, and Conduit Straps," September 13, 2012.

45. AD-WC-ALL-0210, Work Request Initiation, Screening, Prioritization and Classification

46. LR-ISG-2011-04, Updated Aging Management Criteria for Reactor Vessel Internal Components For Pressurized Water Reactors, June 3, 2013.

47. AD-EG-ALL-1701, ASME Section XI Plan Development, Rev.1.

48. AD-EG-ALL-1702, ASME Section XI Inservice Inspection Program Administration, Rev. 3.

49. AD-EG-ALL-1704, Augmented Inservice Inspection Program Administration, Rev. 0.

50. EC 68062, MG12021 /00/ /1 Perform Reactor Internals Upflow Modification to Reduce Baffle Jetting (Unit 1).

51. EC 57683, MG22021 /00/ /2 Perform Reactor Internals Upflow Modification to Reduce Baffle Jetting (Unit 2).

52. AR 02139025, Missed Components in License Renewal Application, July 25, 2017.

53. AD-EG-ALL-1912, Materials Degradation Management Program Implementation, Rev. 0.

54. AD-EG-ALL-1703, ASME Section XI Repair/Replacement Program Administration, Rev. 1.

55. Westinghouse Calculation Note, CN-RIDA-13-59, Rev. 3, "McGuire Units 1 and 2 Reactor Internals MRP-227-A Licensee Action Items 1 and 2," November 30, 2017.

56. Westinghouse Letter, NS-TMA-2214, "Control Rod Guide Tube Support Pin Cracking," March 14, 1980.

57. Westinghouse Nuclear Safety Advisory Letter, NSAL-16-1, "Baffle-Former Bolts," August 1, 2016.

58. Materials Reliability Program, MRP 2017-009, "Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01 Rev. 1," March 15, 2017.

59. Materials Reliability Program, MRP 2016-021, "Transmittal of NEI-03-08 "Needed" ***Interim Guidance Regarding Baffle Former Bolt inspections for Tier 1 plants as Defined in Westinghouse NSAL 16-01***," July 25, 2016.

60. Westinghouse Nuclear Safety Advisory Letter, NSAL-17-1, "Guide Tube Guide Card Wear Attributed to Ion Nitride Rod Cluster Control Assembly," January 16, 2017.

61. Pressurized Water Reactor Owners Group Report, PWROG-14054-P, Rev. 0, "McGuire Units 1 and 2 Summary Report for the Cold Work Assessment," December 2, 2014.

62. *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232).* EPRI, Palo Alto, CA: 2008. 1016593.

63. Westinghouse Letter, LTR-REA-17-108, Rev. 0, "McGuire Units 1 & 2 Summary Report for the Fuel Design / Fuel Management Assessments to Demonstrate MRP-227-A Applicability," September 28, 2017.

64. Materials Reliability Program, MRP 2013-025, "MRP-227-A Applicability Template Guideline," October 14, 2013.

65. Westinghouse Calculation Note, CN-RIDA-13-113, Rev. 0, "McGuire Units 1 and 2 Reactor Internals MRP-227-A Aging Management Program Plan Update Licensee Action Item 7, Evaluation of CASS," February 3, 2014.

APPENDIX A
ILLUSTRATIONS

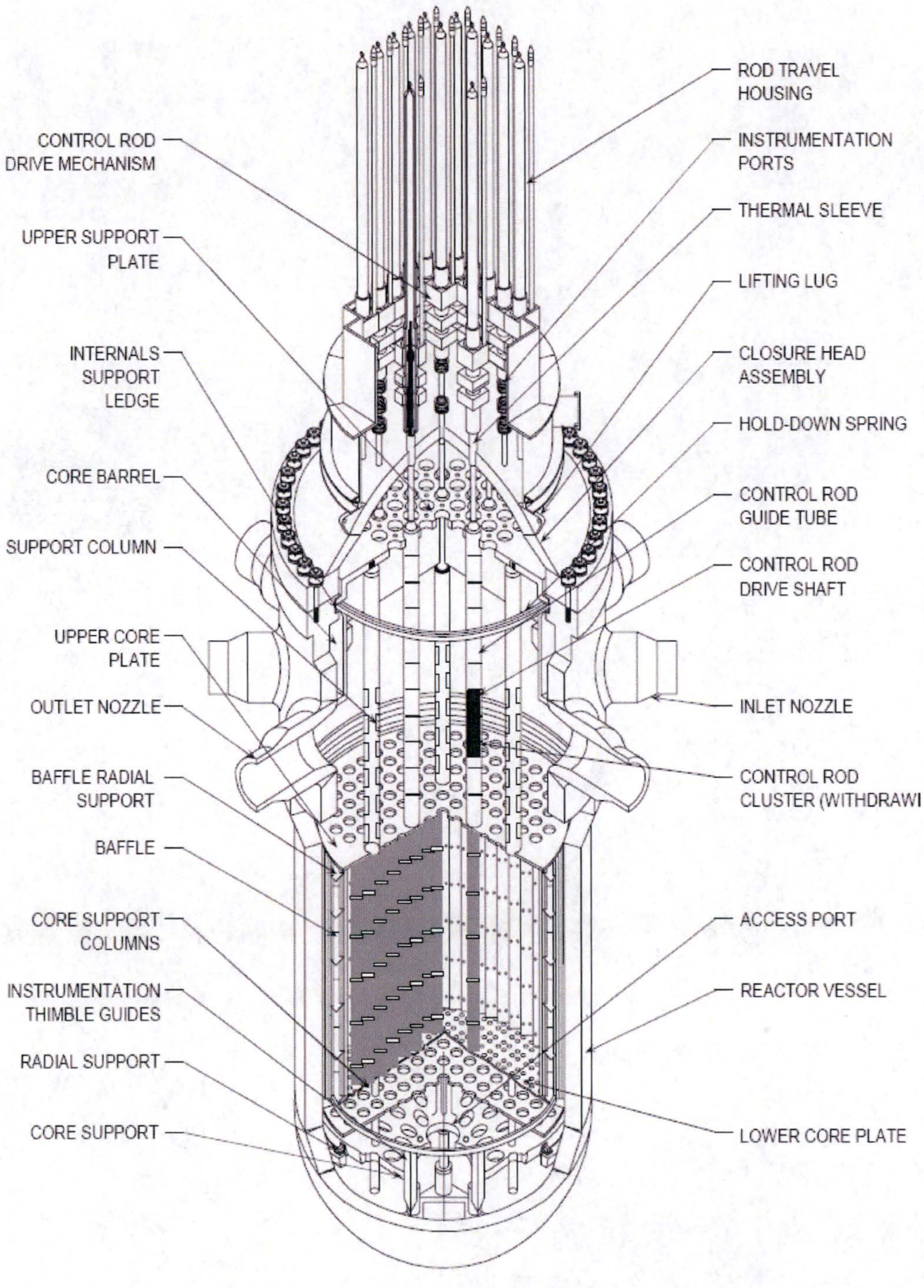

Figure A-1 Illustration of Typical Westinghouse Internals

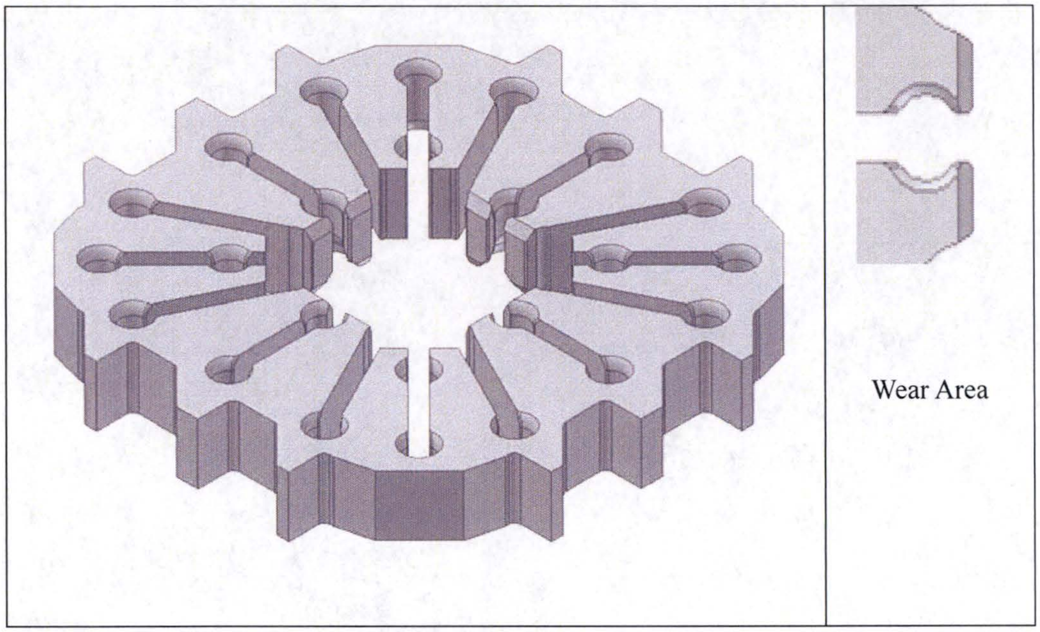

Figure A-2 Typical Westinghouse Control Rod Guide Card

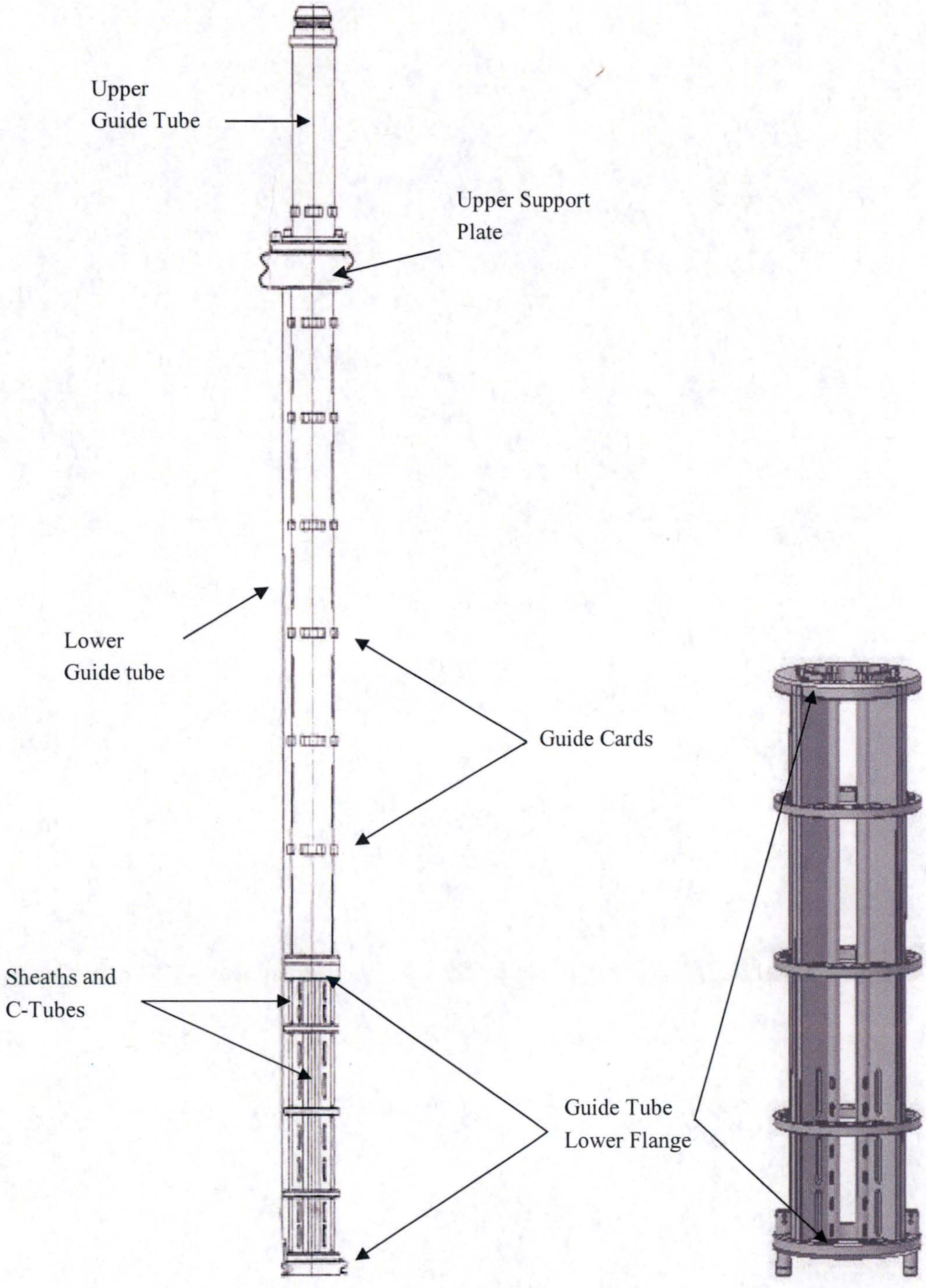

Figure A-3 Typical Lower Section of Control Rod Guide Tube Assembly

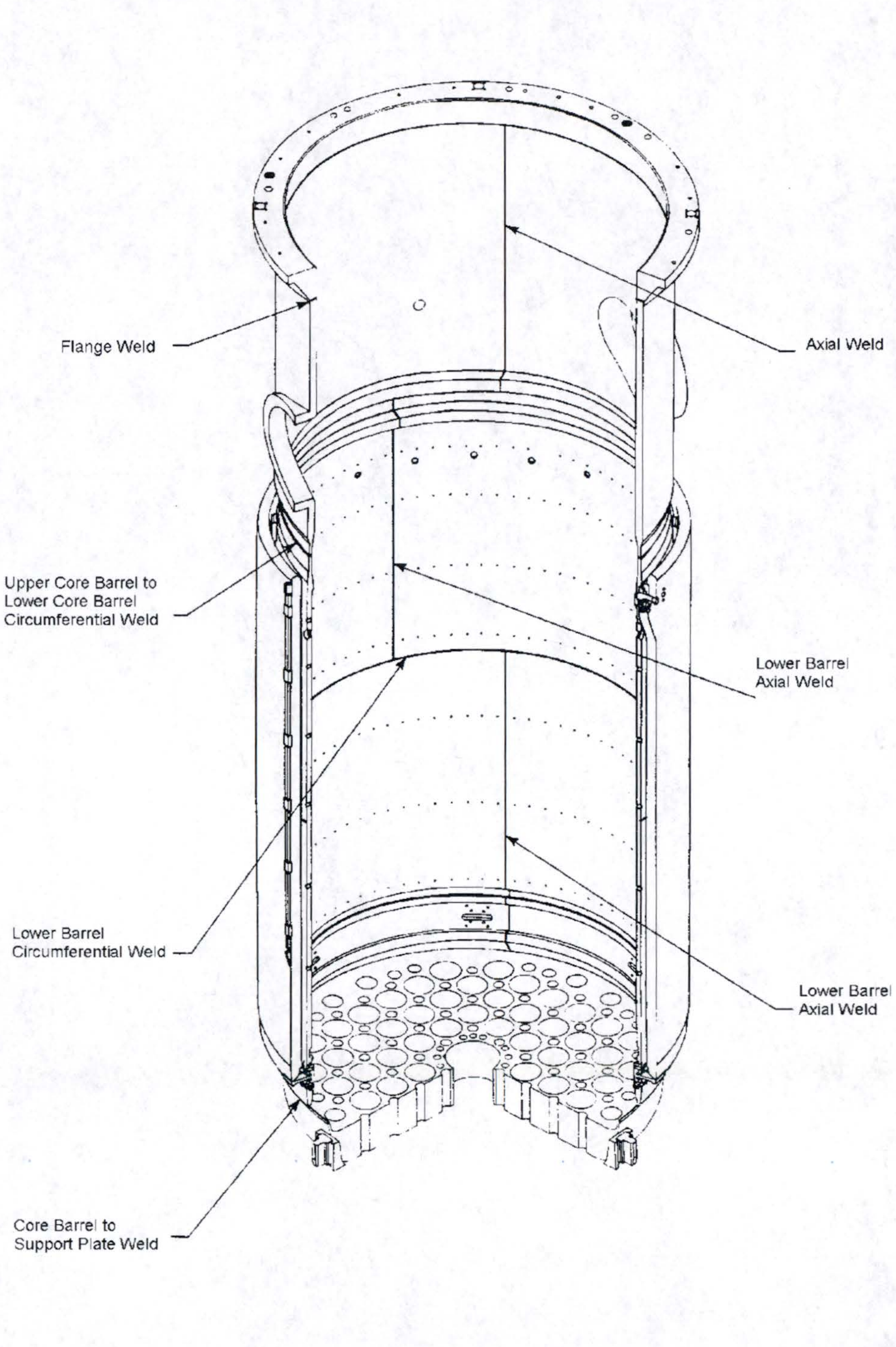

Figure A-4 Major Core Barrel Welds

(Note: The thermal shield and thermal shield flexures are not applicable to McGuire Unit 1 and Unit 2.)

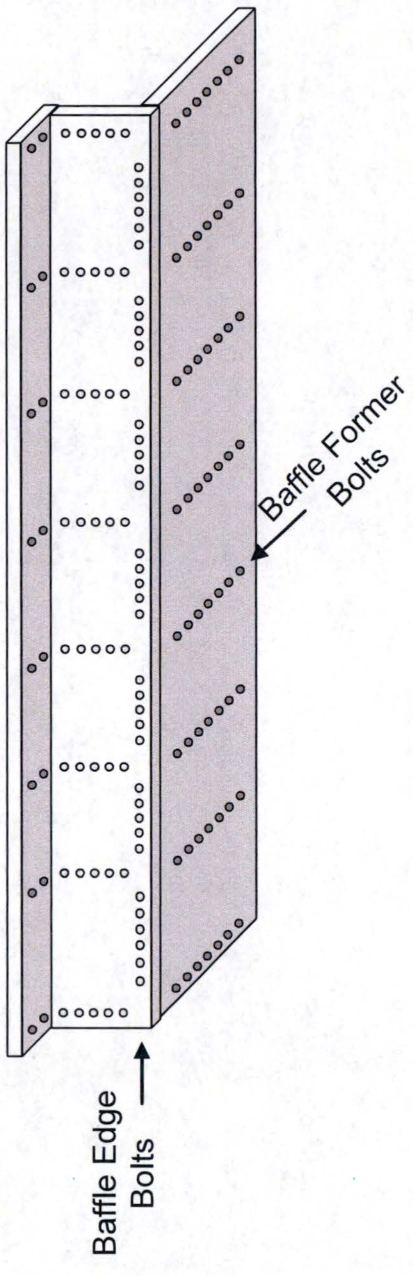

Figure A-5 Bolting Systems Used in Westinghouse Core Baffles

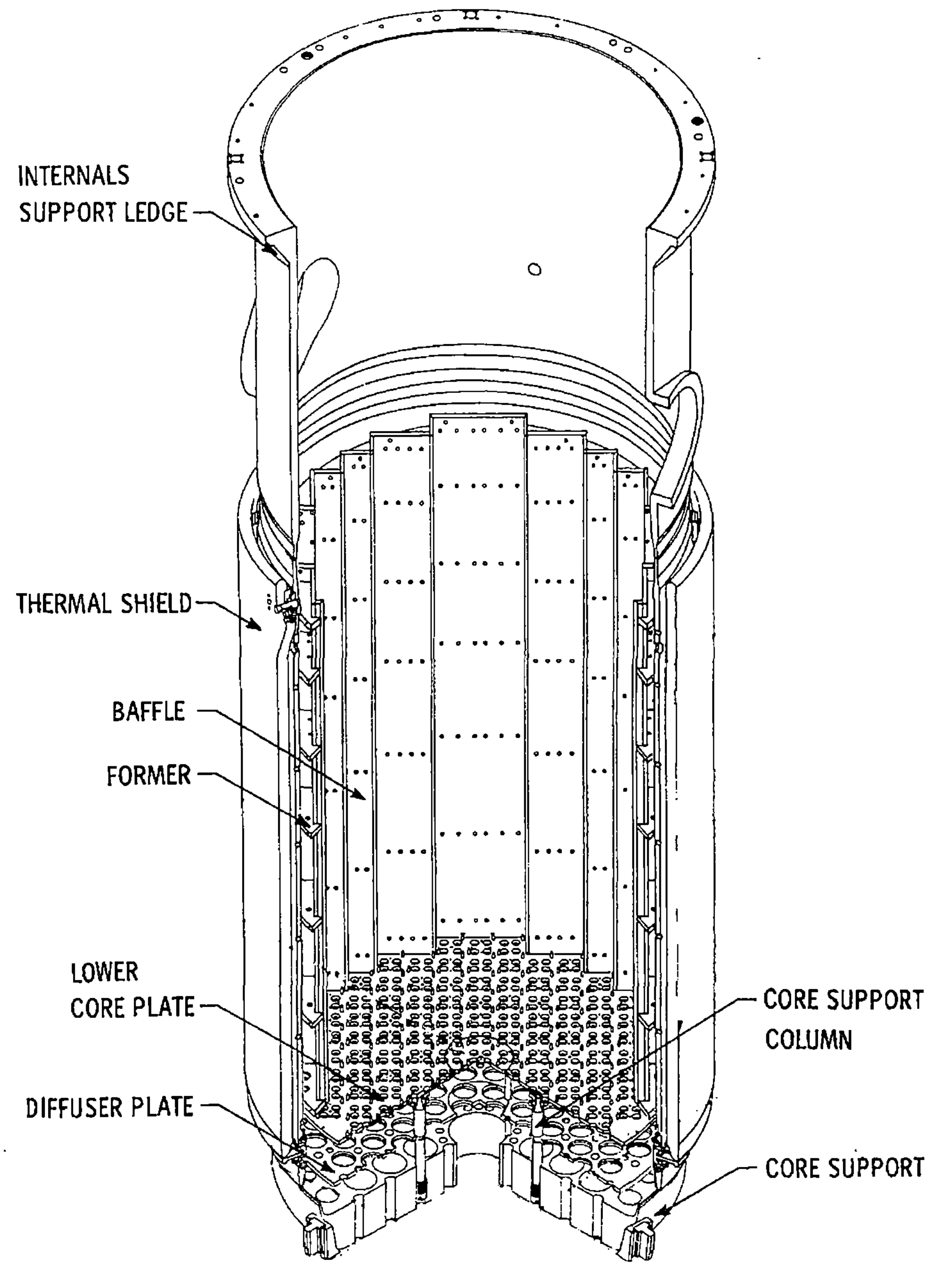

Figure A-6 Core Baffle/Barrel Structure
(Note: The thermal shield and thermal shield flexures are not applicable to McGuire Unit 1 and Unit 2.)

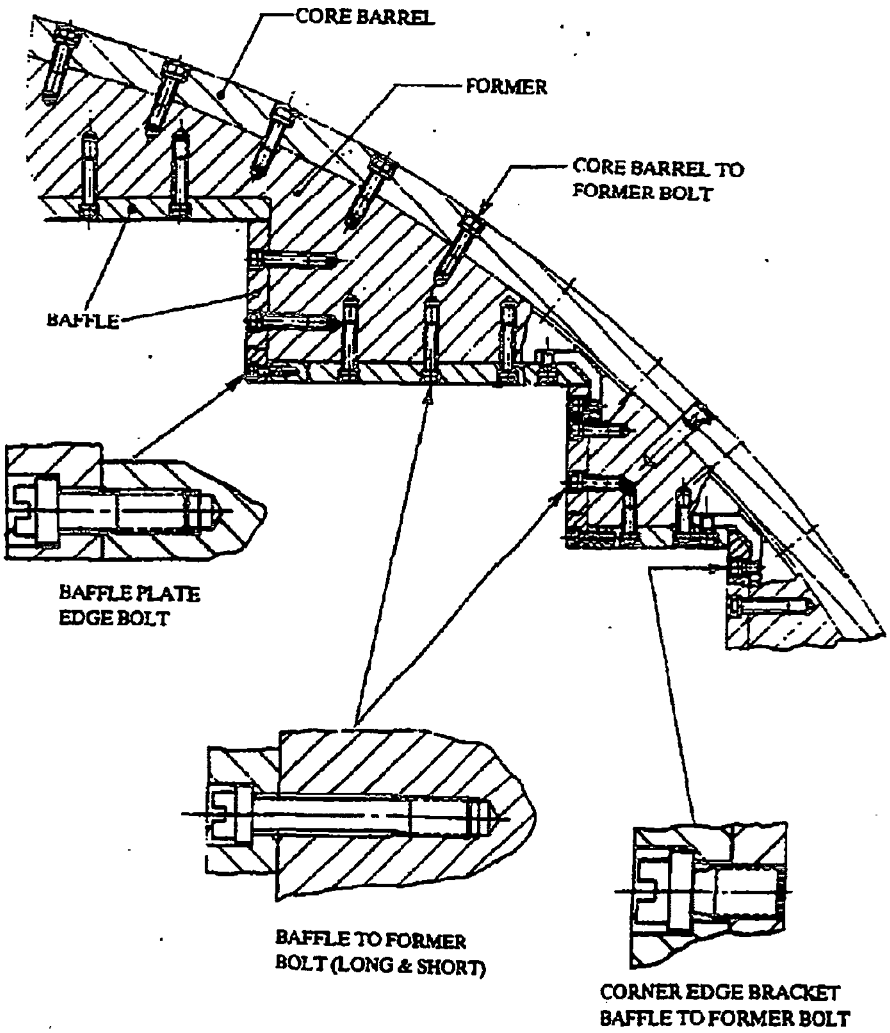

Figure A-7 Bolting in a Typical Westinghouse Baffle-Former Structure

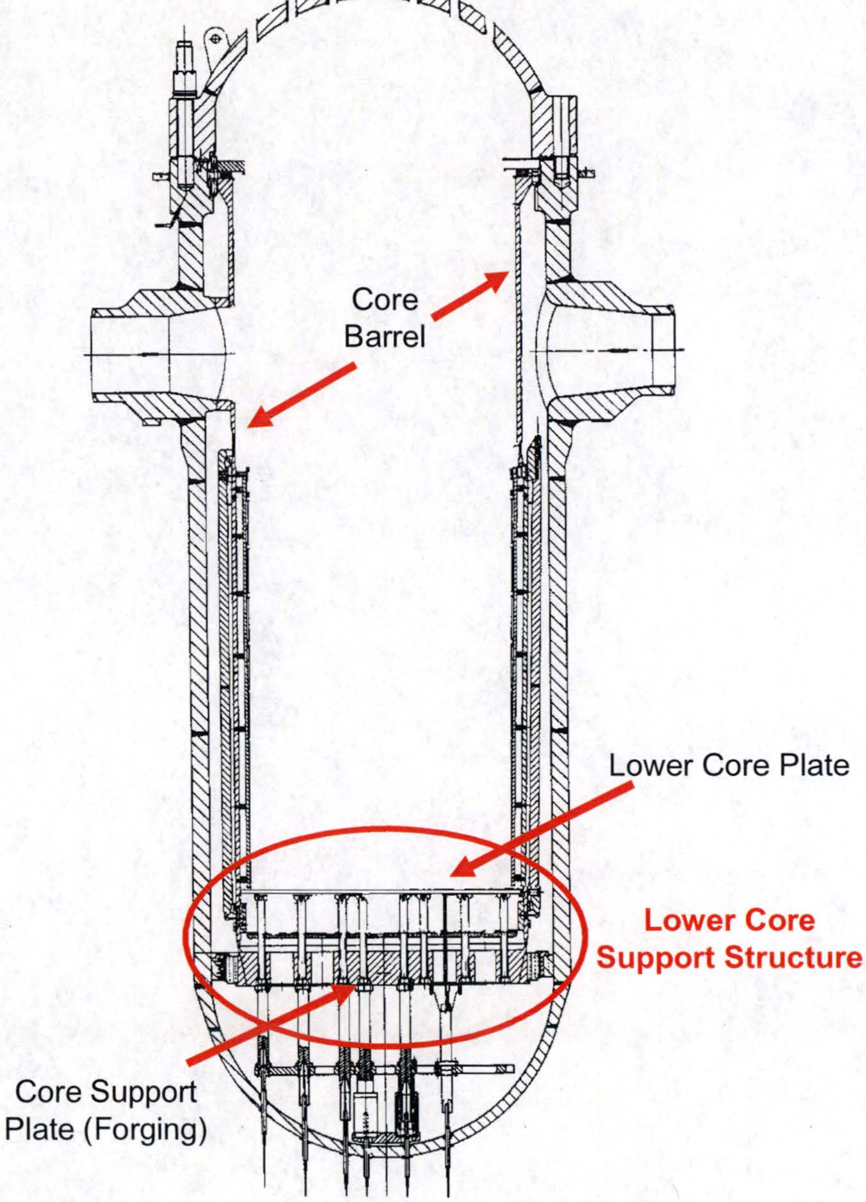

Figure A-8 Lower Core Support Structure

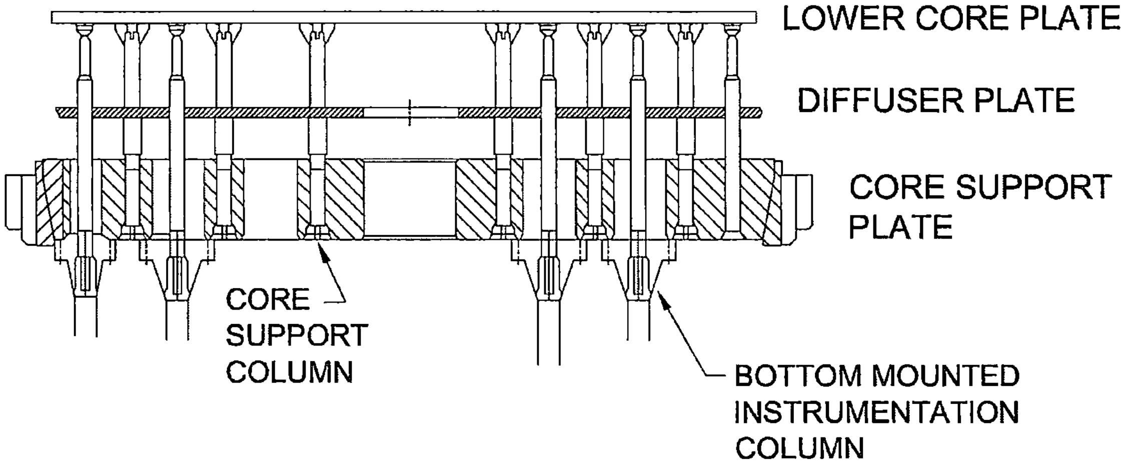

Figure A-9 Lower Core Support Structure – Core Support Plate Cross-Section

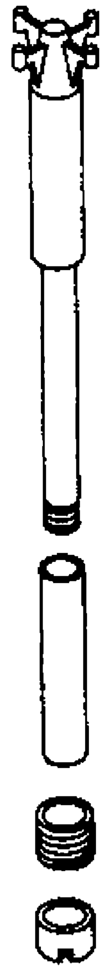

Figure A-10 Typical Core Support Column

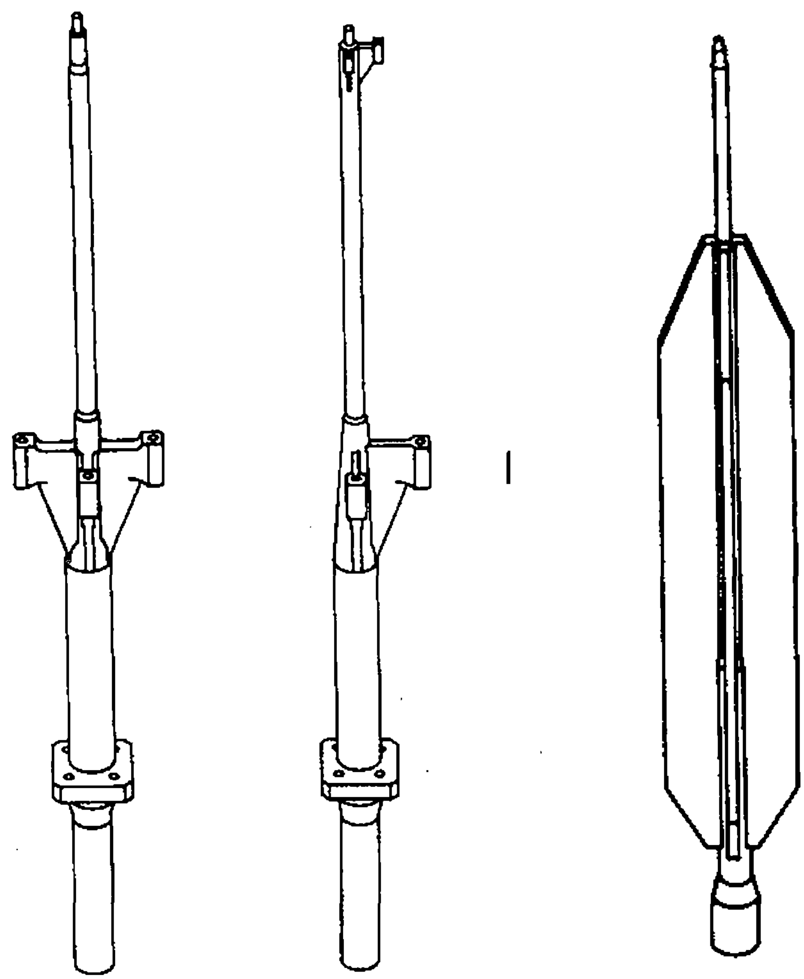

Figure A-11 Examples of Bottom-Mounted Instrumentation (BMI) Column Designs

APPENDIX B
MCGUIRE UNIT 1 AND UNIT 2 LICENSE RENEWAL AGING MANAGEMENT REVIEW SUMMARY TABLES

The content and numerical identifiers in Table B-1 are extracted from Table 3.1-1, "Aging Management Review Results – Reactor Coolant System," of the McGuire LRA. Only those items applicable to reactor vessel internals (according to the LRA) were imported into Table B-1 from the LRA.

Table B-1 Aging Management Review Results – Reactor Coolant System (Reactor Vessel Internals Components)

Component Type	Component Function (Note 1)	Material	Environment	Aging Effect	Aging Management Programs and Activities
Reactor Vessel Internals					
Upper Core Support Structure					
Upper Support Assembly (Forging, Plates, Weld)	1,2,3,4	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Upper Support Column	1,2,4	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Upper Support Column (Base, Conduit Support, Thermocouple stop (U1))	1,2,4	CASS	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table B-1 Aging Management Review Results – Reactor Coolant System (Reactor Vessel Internals Components)					
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Upper Support Column Bolts	1,2,4	Stainless Steel	Borated Water	Cracking Loss of Material Loss of Preload (bolting)	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Upper Core Plate	1,2,3,4	Stainless Steel	Borated Water	Cracking Loss of Material Dimensional Changes Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Upper Core Plate Alignment Pins	1	Stainless Steel	Borated Water	Cracking Loss of Material Dimensional Changes	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table B-1 Aging Management Review Results – Reactor Coolant System (Reactor Vessel Internals Components)

Component Type	Component Function (Note 1)	Material	Environment	Aging Effect	Aging Management Programs and Activities
Fuel Alignment Pins	1	Stainless Steel	Borated Water	Cracking Loss of Material Dimensional Changes Loss of Preload	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Hold-down Spring	1	Stainless Steel	Borated Water	Cracking Loss of Material Loss of Preload	Chemistry Control Program Inservice Inspection Plan
Thermocouple Column and Crossrun Assemblies	4	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
17×17 and 15×15 Guide Tube Assembly	2 (17×17 only), 3	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan

Table B-1 Aging Management Review Results – Reactor Coolant System (Reactor Vessel Internals Components)

Component Type	Component Function (Note 1)	Material	Environment	Aging Effect	Aging Management Programs and Activities
15×15 and 17×17 Guide Tube Assembly	3	CASS	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
UHI Flow Columns	3	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
UHI Flow Columns (Base)	3	CASS	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table B-1 Aging Management Review Results – Reactor Coolant System (Reactor Vessel Internals Components)

Component Type	Component Function (Note 1)	Material	Environment	Aging Effect	Aging Management Programs and Activities
Lower Core Support Structure					
Core Barrel Flange Core Barrel Outlet Nozzles Neutron Panels Irradiation Specimen Holder Fasteners	1,3,4,5,6	Stainless Steel	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness Dimensional Changes Loss of Preload (bolting)	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Irradiation Specimen Holder (spring)	5, 8	Nickel Based Alloy	Borated Water	Loss of Material Cracking	Alloy 600 Aging Management Review Inservice Inspection Plan Chemistry Control Program

Table B-1 Aging Management Review Results – Reactor Coolant System (Reactor Vessel Internals Components)					
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Baffle and Former Plates	1,3,6	Stainless Steel	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness Dimensional Changes	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Baffle Bolts (baffle to baffle, baffle to former)	1,3	Stainless Steel	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness Dimensional Changes Loss of Preload (bolting)	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table B-1 Aging Management Review Results – Reactor Coolant System (Reactor Vessel Internals Components)

Component Type	Component Function (Note 1)	Material	Environment	Aging Effect	Aging Management Programs and Activities
Lower Core Plate Fuel Alignment Pins Lower Support Column Bolts	1,3,4,5	Stainless Steel	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness Dimensional Changes Loss of Preload (bolting)	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Lower Support Plate (forging) Lower Core Support Columns	1,3,4,5,6	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Radial Keys and Fasteners	1	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Clevis Inserts and Fasteners	1	Nickel-Based Alloy	Borated Water	Cracking Loss of Material	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan

Table B-1 Aging Management Review Results – Reactor Coolant System (Reactor Vessel Internals Components)

Component Type	Component Function (Note 1)	Material	Environment	Aging Effect	Aging Management Programs and Activities
Bottom-Mounted Instrumentation					
Bottom-Mounted Instrumentation (Plates, forgings, welds, energy absorber, fasteners)	3,4,5	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Bottom-Mounted Instrumentation (upper end, cruciform)	4	CASS	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Thimble Assembly	7	Stainless Steel	Borated Water	Loss of Material and Cracking	Chemistry Control Program Bottom Mounted Thimble Tube Instrumentation Inspection Program

Notes:

Reactor Vessel Internals Functions:

1. Provide support and orientation of the reactor core (i.e., the fuel assemblies)
2. Provide support, orientation, guidance and protection of the control rod assemblies.
3. Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
4. Provide a passageway for the support, guidance, and protection for the in-core instrumentation.
5. Provide secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.
6. Provide neutron shielding to the reactor vessel and provide support for the vessel material test specimens.
7. Provide pressure boundary.
8. This item specifically refers to the Access Plug Assembly Spring. The Irradiation Specimen Holder (spring) is a subcomponent of the Access Plug Assembly which is a subcomponent of the Irradiation Specimen Holder Assembly.

APPENDIX C
MRP-227 AUGMENTED INSPECTIONS

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

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of Material (Wear)	None	Refer to WCAP-17451-P, Revision 1 [29] (Note 7)	Refer to WCAP-17451-P, Revision 1 [29] See Figure A-2 (Note 7)
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) 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 two refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies (Note 2). See Figure A-3
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Lower support column bodies (non-cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than two refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure A-4

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

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly 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 two refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure A-4
Core Barrel Assembly Lower core barrel flange weld (Note 5)	All plants	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, no later than two refueling outages from the beginning of the license renewal period and subsequent examinations on a 10-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4).
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, Fatigue) that results in: • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a 10-year interval.	Bolts and locking devices on high-fluence seams. 100% of components accessible from core side (Note 3). See Figures A-5, A-6, and A-7

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

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a 10-year interval.	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures A-5, A-6, and A-7

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

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Assembly (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: • Abnormal interaction with fuel assemblies • Gaps along high-fluence baffle joint • Vertical displacement of baffle plates near high-fluence joint • 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 10-year interval.	Core side surface, as indicated.

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

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Alignment and Interfacing Components Internals hold-down spring	All plants with 304 stainless steel hold-down springs NOTE: The McGuire Unit 1 and Unit 2 hold-down spring is 403 SS; therefore, this component is not applicable to McGuire Unit 1 and Unit 2.	Distortion (Loss of Load) 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.
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields Not applicable to McGuire Unit 1 and Unit 2	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 two refueling outages from the beginning of the license renewal period. Subsequent examinations on a 10-year interval.	100% of thermal shield flexures. See Figure A-4

Notes:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are listed 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 are managed through management of void swelling on the entire baffle-former assembly.
7. WCAP-17451-P, Revision 1 [29] requires a remote visual examination consistent with visual (VT-3) for minimum compliance and examination coverage of a minimum of 20% of the number of CRGT guide card assemblies. The baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation, and examination results. The content of this note and the noted cells is included according to the interim guidance provided in the EPRI transmittal MRP 2014-006 [32].

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Internals Assembly Upper Core Plate	All plants	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
Lower Internals Assembly Lower support forging or castings	All plants NOTE: McGuire Unit 1 and Unit 2 have a lower support forging	Cracking Aging Management (TE in Casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure A-8 and A-9.
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination. Reinspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads (Note 2). See Figure A-7
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Reinspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant-specific justification (Note 2). See Figures A-8, A-9, and A-10

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Core barrel outlet nozzle welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection. Figure A-4	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figures A-4 and A-8
Core Barrel Assembly Upper and lower core barrel cylinder axial welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure A-4
Lower Support Assembly Lower support column bodies (non-cast)	All plants	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figures A-8, A-9, and A-10
Lower Support Assembly Lower support column bodies (cast)	All plants NOTE: Not applicable to McGuire Unit 1 and Unit 2	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of accessible support columns (Note 2). See Figures A-8, A-9, and A-10

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Bottom-Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies 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. Reinspection every 10 years following initial inspection. 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. See Figures A-9 and A-11

Notes:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are listed 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 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 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 Lower core plate XL lower core plate (Note 1)	All plants	Cracking (IASCC, Fatigue) 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 Lower core plate XL lower core plate (Note 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 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.
Alignment and Interfacing Components Clevis insert bolts	All plants	Loss of material (Wear)	ASME Code Section XI as supplemented by TB-14-5 (Note 2)	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

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
Alignment and Interfacing Components 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. The clevis inserts are attached to integrally welded reactor vessel lugs and the inserts are bolted to the lugs. The ASME Code examination of accessible surfaces is considered to include all details of the clevis configuration, including the bolting and locking devices. The bolting is fabricated from nickel-based materials and is susceptible to stress corrosion cracking (SCC). Although failure of the bolting does not itself cause loss of support function, asset impairment or issues with core barrel removal are a subsequent possibility. Westinghouse technical bulletin TB 14-5 dated 8/25/2014 provides additional information regarding possible visual indications that clevis bolting failure may have occurred. This information should be reviewed to ensure a heightened awareness of the examiners is applied to this Code inspection.

Table C-4 MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals					
Item	**Applicability**	**Examination Acceptance Criteria (Note 1)**	**Expansion Link(s)**	**Expansion Criteria**	**Additional Examination Acceptance Criteria**
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) Examination (Note 3) 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

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Lower flange welds	All plants	Enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast), upper core plate, and lower support plate 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. 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. 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

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Core barrel outlet nozzle welds b. Lower support column bodies (non-cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than 2 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. b. If extensive cracking in the core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper 6 inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles following 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

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel flange weld (Note 2)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	None	None
Core Barrel Assembly Upper core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. 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 2 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

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. 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 2 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 Baffle-edge bolts	All plants with baffle-edge bolts	Visual (VT-3) examination. 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

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-former bolts	All plants	Volumetric (UT) examination. 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 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. 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

Item	Applicability	Examination Acceptance Criteria (Note 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

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Alignment and Interfacing Components Internals hold-down spring	All plants with 304 stainless steel hold-down springs NOTE: Not applicable to McGuire Unit 1 and Unit 2	Direct physical measurement or spring height. 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 Thermal shield flexures	All plants with thermal shields NOTE: Not applicable to McGuire Unit 1 and Unit 2	Visual (VT-3) examination. 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.
3. WCAP-17451-P Revision 1 specifies a remote visual examination consistent with visual (VT-3) but allows for various supplemental measurement techniques which if employed increase wear estimate accuracy and allow use of acceptance criteria (wear projections) to determine the appropriate re-examination interval.