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April 2, 2012

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Serial No. 12-151
LIC/JG/R0
Docket No.: 50-305
License No.: DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REACTOR VESSEL INTERNALS INSPECTION PLAN REVIEW REQUEST (ME7727)

By application dated December 12, 2011 (reference 1), Dominion Energy Kewaunee, Inc. (DEK), requested NRC approval of an inspection plan for reactor vessel internal (RVI) components at Kewaunee Power Station (KPS). The proposed inspection plan fulfils two commitments required by Renewed Operating License DPR-43, Section 2.C(15)(b), to submit an inspection plan for reactor internals to the NRC staff for review and approval to augment the current inspections.

Subsequently, the Nuclear Regulatory Commission (NRC) staff transmitted a request for additional information (RAI) regarding the proposed inspection plan (reference 2). Specifically, the staff requested a copy of Aging Management Program Technical Report AMP-KLR-1309A, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Reactor Vessel Internals Inspections, Kewaunee Power Station." The requested report is enclosed with this letter.

If you have questions or require additional information, please feel free to contact Mr. Jack Gadzala at 920-388-8604.

Very truly yours,

J. Alan Price
Vice President – Nuclear Engineering

Enclosure: Aging Management Program Technical Report AMP-KLR-1309A, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Reactor Vessel Internals Inspections, Kewaunee Power Station," Revision 3, dated September 30, 2011.

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References:

1. Letter from J. Alan Price (DEK) to Document Control Desk (NRC), "Reactor Vessel Internals Inspection Plan Review Request," dated December 12, 2011.
2. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK), Craig Sly (DEK), and Ganesh Cheruvenki (NRC), "ME7727 - Kewaunee - Reactor Vessel Internals Inspection Plan review - an Aging Management Program report is needed to prepare RAI items," dated March 2, 2012.

Commitments made by this letter: None

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ENCLOSURE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REACTOR VESSEL INTERNALS INSPECTION PLAN REVIEW REQUEST

Aging Management Program Technical Report AMP-KLR-1309A, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Reactor Vessel Internals Inspections, Kewaunee Power Station," Revision 3, dated September 30, 2011

(62 pages)

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

Dominion

Kewaunee Power Station
LICENSE RENEWAL
2013-2033

TECHNICAL REPORT: KLR-1309A

LICENSE RENEWAL PROJECT
AGING MANAGEMENT PROGRAM

ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD REACTOR VESSEL INTERNALS INSPECTIONS KEWAUNEE POWER STATION

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Effective Date: 09/30/2011
Revision Number: 3

REVISION RECORD

Page(s) Revised	Revision Number	Revision Date	Change Description
All	0	3/24/2010	Initial Issue
Various	1	10/29/2010	DMRs 2758, 2816
Various	2	06/30/2011	Incorporation of MRP-227 Requirements per Charter ER-31
14	3	09/30/2011	Alignment with NRC Letter for Commitments 1 & 2

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1.0 PURPOSE

The purpose of the aging management program (AMP) report is to update the basis for the aging management program information summarized in the Kewaunee Power Station (KPS) renewed operating license granted under 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Reference 8.1). The AMP information is applicable to systems, structures, and components (SSCs) that are in-scope, long-lived, passive, and perform a license renewal intended function.

This document implements commitments to the NRC to enhance the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program for managing aging effects on reactor internals and on limiting susceptible cast austenitic stainless steel reactor vessel internals components (Reference 8.7, Section 5.1 and 5.2).

In order to ensure the aging of reactor vessel internal components are adequately managed, the Nuclear Energy Institute initiated a commercial nuclear utility industry Materials Initiative. The result of this effort was NEI 03-08, which was issued through the Nuclear Strategic Issue Advisory Committee (NSIAC) for management of primary pressure boundary materials. NEI 03-08 is classified as a NSIAC initiative, which is a formal agreement by 80% vote of Chief Nuclear Officers and is binding for all US nuclear utilities. To implement the NEI 03-08 requirements, the U.S. industry, through the efforts of the EPRI Materials Reliability Program (MRP) and PWR Owners Group, developed and issued MRP-227 and MRP-228 to identify and document the components and subcomponents that require aging management to support continued reliable operation during the period of extended plant operation.

In addition, a secondary purpose of this AMP is to develop and document a PWR reactor internals aging management program (AMP) as outlined in Appendix A of MRP-227 Rev. 0 and as required by Condition Report 322266 (Reference 8.2).

This document, in conjunction with the integrated plant assessment reports, demonstrates that the effects of aging on identified SSCs will be adequately managed so that the intended function(s) will be maintained consistent with the plant-specific current licensing basis for the period of extended operation.

2.0 PROGRAM DESCRIPTION

The Reactor Vessel Internal Inspections program is a new plant-specific program that consists of the applicable ten elements as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components (Reference 8.3). These applicable ten elements are

the same as the ten elements included in MRP-227 Appendix A (Reference 8.4) and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal applications for Nuclear Power Plants" (Reference 8.5).

The program manages the aging effects of changes in dimensions due to void swelling: cracking due to stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and fatigue; loss of fracture toughness due to neutron irradiation embrittlement and thermal aging embrittlement; loss of material due to wear; and loss of preload due to stress relaxation.

Loss of material by pitting corrosion and crevice corrosion is controlled by plant chemistry by limiting the oxygen concentration in the coolant system. Additional management of such mechanisms is not required by the reactor internals program.

The Reactor Vessel Internals Inspections program performs one-time, periodic, and conditional examinations using visual, surface, and ultrasonic examination techniques in accordance with the ASME Code, Section XI and EPRI Report 1016596 (MRP-227) and NEI 03-08 (References 8.4, 8.6). The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program performs visual examinations of the reactor vessel removable core support structures under Table IWB-2500-1, Examination Category B-N-3. MRP-227 has developed additional inspection requirements for reactor vessel internals that are susceptible to the aging effects managed by the program, including the examination techniques to be used, the examination schedule, the examination acceptance criteria, and the inspection expansion criteria.

The MRP-227 requirements are administered by Dominion using Fleet Reactor Internals Inspection Program Description, ER-AA-RII-10 (Reference 8.10) and Fleet Reactor Internals Inspection Administrative Procedure ER-AA-RII-101 (Reference 8.11). The Reactor Vessel Internals Inspections program is implemented in accordance with the KPS ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program that include the inspection procedures, the acceptance criteria for each examination technique, and the process for evaluating unsatisfactory inspection results. For the ASME Code Section XI inspections, indications and relevant conditions detected during examination will be evaluated in accordance with ASME Section XI, Article IWB-3500. For the additional MRP-227 required inspections, the evaluations will be performed in accordance with the examination acceptance criteria and the expansion criteria provided in MRP-227.

The program inspections will provide support for the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program for managing the reactor vessel internal components in accordance with NEI 03-08 management of material issues.

Flaw indications detected during the required examinations are dispositioned in accordance with the Corrective Action Program.

3.0 SUMMARY PROGRAM INFORMATION

Attachment 1 provides the Kewaunee and Nuclear Business Unit procedures that implement this aging management program.

Attachment 2 provides a listing of the aging management programs that are credited by this program as performing activities required by the program and a listing of the aging management programs that credited this program to perform activities required by that program.

Attachment 3 identifies the materials and aging effects managed by this aging management program.

Attachment 4 identifies the Integrated Plant Assessment Reports that credits this aging management program and identifies the systems or commodities managed by the program.

Attachment 5 is a detailed assessment of the ten program elements at the implementation level.

4.0 EVALUATION OF NUREG-1800 AND MRP-227 PROGRAM ELEMENTS

This section provides a summary comparison of the Kewaunee aging management program to the ten program elements described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These applicable ten elements are the same as the ten elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." Age-related degradation in the reactor internals is managed through an integrated program. Specific features of the integrated program are listed in the following ten program elements. The comparison addresses the aging management program expectations for each element and provides a description of station implementation of these expectations.

Attachment 5 is a detailed assessment of the program elements at the implementation level. Attachment 5 provides the following information:

Column	Content
Attribute Number	Indicates the "attribute" or detailed element recommendation being addressed. A program element that contains multiple recommendations is divided into more than one attribute.
Attribute Description	Contains the detailed recommendation directly from NUREG-1800 and MRP-227, Appendix A.
Attribute Response	Evaluates the Kewaunee program against the NUREG-1800 and MRP-227 recommendations.
Document Number	Provides the procedure(s), preventive maintenance activities, or other documents, which implement the program.
Document Section	Indicates if all or only a portion of the document is required to implement the program.
Purpose	Indicates the specific activities performed by the document, which are required to address the NUREG-1800 and MRP-227 recommendations.
Document Change	Indicates the specific document changes to properly implement the program. Note: Some document changes that clarify or provide additional detail for existing activities have not been identified as a program enhancement since the change is not required to meet the NUREG-1800 and MRP-227 recommendations.
Follow-Up Action Item (FAI)	Indicates the internal tracking number used to track proposed document changes and other activities
CAP/CR	Indicates the number of the Corrective Action Program Action Request/Condition Report that will implement the proposed document changes and other activities.

4.1 PROGRAM ELEMENT 1 – SCOPE OF PROGRAM

The scope of the Reactor Vessel Internals Inspections program is defined in Sections 5.1 and 5.2 of the ASME Section XI Inservice Inspection, Subsections

IWB, IWC, and IWD program (Reference 8.7). These two sections identify two program enhancements to incorporate applicable industry reactor vessel internals inspection initiatives. The two enhancements state:

"5.1 Enhancement 1: Aging Management of Reactor Vessel Internals

The ASME Section XI Inservice Inspection Subsections IWB, IWC and IWD program will be enhanced to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval to augment the current inspections.

5.2 Enhancement 2: Aging Management of Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel

The ASME Section XI Inservice Inspection Subsections IWB, IWC and IWD program will be enhanced to include identification of the limiting susceptible cast austenitic stainless steel reactor vessel internals components from the standpoint of thermal aging susceptibility, neutron fluence, and cracking. For each identified component, a plan will be developed, which accomplishes aging management through either a supplemental examination or a component-specific evaluation. The plan will be submitted for NRC review and approval not less than 24 months before entering the period of extended operation."

The scope of the ASME Section XI ISI – Reactor Vessel Internals Inspections program is to implement the inspections required by the two above program enhancements.

The reactor vessel internals included in the scope of the program consist of two basic assemblies, the upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and the lower internals assembly that can be removed, if desired, following a complete core off-load.

Reactor Vessel Internals Inspections program implements the NEI 03-08 (Reference 8.6) "Mandatory" requirement to use MRP-227 and MRP-228 to develop and implement a program for reactor vessel internals no later than three years after the initial industry issuance of MRP-227, which was issued in December 2008.

The reactor vessel internals components included in the program are based on EPRI Report 1016596 (MRP-227) and Westinghouse letter LTR-ARIDA-08-63 Rev. 3, (References 8.4, 8.8).

The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are

required to be subject to an aging management review, as defined by the criteria set in 10 CFR 54.21(a)(1).

MRP-227 provides the results of the industry initiative via the Materials Reliability Program (MRP) to develop the inspection and evaluation guidelines for managing the long-term aging of pressurized water reactor (PWR) reactor vessel internals. The report provides a generic list of the reactor vessel internals components for Westinghouse PWRs. Westinghouse letter LTR-ARIDA-08-63 Rev. 3 evaluated the MRP-227 generic list of the reactor vessel internals components for applicability to KPS.

Fleet procedures provide the administrative and technical direction for the Reactor Vessel Internals Inspections program (References 8.9 – 8.11). The reactor vessel internals components, including the cast austenitic stainless steel components, in the scope of the program are identified in Table 2 and Table 4.

MRP-227 has been submitted to the NRC for review. Following NRC review and approval, MRP-227 will be revised to incorporate any necessary changes to the guidelines and reissued as MRP-227-A. The Reactor Vessel Internals Inspections program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A[1].

Dominion 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 scope of program will be consistent with the corresponding program elements as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.2 PROGRAM ELEMENT 2 – PREVENTIVE ACTIONS

The Reactor Vessel Internals Inspections program is a condition monitoring program and does not include any preventive or mitigative actions.

MRP-227 does not specify any preventive actions other than their applicability limitations to base-loaded plants. However, the guidelines in MRP-227 do rely on PWR water chemistry control to manage SCC and reduce the impact of IASCC. Therefore, an important adjunct to the aging management methodologies described by guidelines in MRP-227 is PWR water chemistry control. The water chemistry program for PWRs relies on monitoring and control of reactor water

[1] Any revisions to MRP-227 resulting from the issuance of MRP-227-A will not be incorporated for any inspections that are completed before MRP-227-A is issued.

chemistry as presented in Chapter XI.M2, "Water Chemistry," of NUREG-1801, Volume 2.

It is noted that KPS Primary Water Chemistry program manages the aging effects of cracking, loss of material, and reduction of heat transfer for nickel alloys, stainless steel and steel components due to stress corrosion cracking (SCC), including primary water stress corrosion cracking (PWSCC), and irradiation-assisted stress corrosion cracking (IASCC) pitting and crevice corrosion, and fouling (Reference 8.12).

The preventive actions will be consistent with the corresponding element as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.3 PROGRAM ELEMENT 3 – PARAMETERS MONITORED/INSPECTED

The Reactor Vessel Internals Inspections program manages the aging effects of Changes in dimensions due to void swelling: cracking due to stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and fatigue; loss of fracture toughness due to neutron irradiation embrittlement and thermal aging embrittlement; loss of material due to wear; and loss of preload due to stress relaxation.

The aging effects related to reactor vessel internals components in the program are identified in Table 2 and Table 4.

The parameters monitored/inspected will be consistent with the corresponding element as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.4 PROGRAM ELEMENT 4 – DETECTION OF AGING EFFECTS

The Reactor Vessel Internals Inspections program performs one-time, periodic, and conditional examinations using visual, surface, and ultrasonic examination techniques in accordance with the ASME Code, Section XI and MRP-227.

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program conducts a visual VT-3 examination of the reactor vessel removable core support structures under Table IWB-2500-1, Examination Category B-N-3, once per Inservice Inspection interval (Reference 8.13). The visual VT-3 examination determines the general mechanical and structural condition of the components by inspecting for structural distortion or displacement of parts,

loose/missing, cracked or fractured parts, loose/missing bolting or fasteners, debris, corrosion or erosion, and wear of mating surfaces. The inspections are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR Part 50, Appendix B.

Additional reactor vessel internals inspection requirements have been promulgated by MRP-227. MRP-227 developed three important precursor elements as a basis for the reactor vessel internals inspection requirements: screening criteria, categorization of PWR vessel internals, and functionality assessment of components and assemblies of components. The evaluation of the three precursor elements resulted in the assignment of the reactor vessel internals components into four inspection categories:

- Primary – those reactor vessel internals that are highly susceptible to the effects of at least one of the eight aging mechanisms.
- Expansion – those reactor vessel 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.
- Existing Programs – those reactor vessel internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which existing generic and plant-specific aging management programs are capable of managing those effects.
- No Additional Measures – those reactor vessel internals for which the effects of all eight aging mechanisms are below the screening criteria.

MRP-227 provided appropriate recommendations for aging management for each category (Reference 8.17). The categorization and resulting inspection requirements described above do not supersede the ASME Section XI, Inservice Inspection requirements for reactor vessel internals components.

The MRP-227 guidelines were based on a broad set of assumptions about plant operation, which encompass the range of current plant conditions for the U.S. domestic fleet of PWRs. The functionality analyses and supporting aging management strategies in MRP-231 and MRP-232 provided the basis for the MRP-227 guidelines (References 8.14, 8.15, 8.17). The following table compares KPS to the general assumptions used in the analysis.

Table 1 - Comparison to MRP-227 Assumptions

MRP Assumption	KPS Comparison	Assumption Met
30 years of operation with high leakage core loading patterns then remaining 30 years of operation with a low-leakage fuel management strategy.	KPS used new fuel at the peripheral locations during Fuel Cycles 1 and 2. Once burned fuel was used at peripheral locations during Fuel Cycles 3 through 15. In Fuel Cycle 16 KPS switched to use of a low leakage core design. KPS continues to use a Low Leakage core design for all subsequent fuel cycles.	Yes
Base load operation	KPS has implemented Base load operation over the life of the plant.	Yes
No design changes affecting Reactor Vessel Internals components beyond those identified in general industry guidance or recommended by the original vendors	Design changes to the KPS Reactor Vessel Internals components have been limited to Westinghouse replacement of split pins, installation of flexure less inserts, and replacement of the reactor vessel head.	Yes

The Reactor Vessel Internals Inspections program will perform the inspections of the reactor vessel internals in the Primary and Expansion categories consistent with the requirements of MRP-227. The aging management methodologies described in MRP-227 are based on either existing inservice examinations required by the ASME Code, Section XI or on well-documented and well-demonstrated examination methods with which the industry has considerable experience. The Reactor Vessel Internals Inspections program will perform visual examinations, surface examinations, volumetric examinations, and physical measurements. Visual examinations (VT-3) will be used to detect the general degradation conditions. Visual and enhanced visual examinations (VT-1 and EVT-1, respectively) will be conducted to detect discontinuities and imperfections on the surface of components. The surface examinations further characterize discontinuities on the surface of components, and the volumetric inspections

indicate the presence of discontinuities or flaws throughout the volume of material. Some aging effects may involve changes in clearances, settings, and physical displacements that can be monitored by visual means, supplemented by physical measurements.

The specific inspection methodologies and requirements for the NDE examinations of the components in the Primary and Expansion categories will be consistent with the guidance with EPRI Report 1016609 (MRP-228) (Reference 8.16). New procedures will be developed to perform the NDE examinations of the components in the Primary and Expansion categories as identified in Table 2 and Table 4.

The schedule for inspection of the reactor vessel internals components in the Primary and Expansion categories will be consistent with the requirements of MRP-227, as shown in Table 3.

Table 2 provides a breakdown of how each component maps to the four categories: Primary, Expansion, Existing, No Additional Measures. Those reactor vessel internals components included in the Existing Programs or No Additional Measures categories will require no additional measures for future inspections other than the ASME Code inspections per Section XI, Examination Category B-N-3 for removable internal structures, which were previously discussed.

The CASS items are identified in Table 2 as Expansion, Existing, or No Additional Measures. Per guidance from MRP-227, the CASS item in the Expansion category will be inspected if and when the Primary criteria for the RCCA guide tubes lower flange weld is exceeded. The CASS item in the Existing category will be inspected under the Section XI program on a frequency of approximately once every 10 years. Per MRP-227, the CASS items in the No Additional Measures category do not require scheduled inspections.

For the CASS items in the Expansion, Existing, or No Additional Measures category they will be inspected per the guidance in MRP-227 or evaluated per the following criteria.

For reactor vessel internal CASS components, in the Expansion and Existing category, that meet one of the following criteria: 1) have a neutron fluence of greater than [10^{17} n/cm^2][2] (E>1 MeV); OR 2) are determined to be susceptible to thermal embrittlement; the Reactor Vessel Internals Inspections program will:

- Perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required.

[2] Per MRP-175, Materials Reliability Program: PWR Internals Material Degradation Mechanism Screening and Threshold Values - EPRI Report 1012081, 2005, a threshold value of 10^{20} n/cm^2 (E>1 MeV) will be used for screening of neutron fluence for susceptibility to thermal aging embrittlement pending agreement with NRC.

- If the above criterion is not met, a supplemental inspection will be performed for the portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (i.e., applied stress, operating temperature, and environmental conditions). The inspection technique will be capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties.

For CASS components that meet neither of the two criteria specified above, the existing ASME Section XI inspection requirements are adequate in accordance with NUREG-1801, XI.M13.

Table 2 and Table 4 identify the reactor vessel internals components in the program, the aging effects applicable to each component, the examination techniques used to detect the aging effect(s), and the examination schedule for the Primary and Expansion category components.

The Reactor Vessel Internals Inspections program inspections are performed in accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program, which includes the procedures for performing the inspections, the acceptance criteria for each examination technique, and the review and disposition of inspection results.

The personnel performing the inspections will be trained in accordance with the guidance in MRP-228.

The detection of aging effects will be consistent with the corresponding element as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.5 PROGRAM ELEMENT 5 – MONITORING AND TRENDING

The Reactor Vessel Internals Inspections program performs one-time, periodic, and conditional examinations as scheduled and performed in accordance with the ASME Code, Section XI as well as MRP-227 to provide timely detection of aging effects.

In addition to the MRP-227 Primary components, Expansion components have been defined by MRP-227 should the scope of examination and re-examination need to be expanded beyond the Primary group due to detection of significant aging effects found in Primary components with the same material and potential aging mechanism.

Table 2 and Table 4 identify the reactor vessel internals components in the program, the aging effects applicable to each component, the examination

techniques used to detect the aging effect(s), and the examination schedule for the Primary and Expansion category components.

Those reactor vessel internals components included in the Existing Programs or No Additional Measures categories will be inspected on a schedule consistent with the requirements of the ASME Code for Section XI, Examination Category B-N-3, for removable internal structures, as outlined in the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program. This schedule is consistent with the recommendations of MRP-227.

The EPRI MRP Reactor Internals Inspection and Evaluation Guidelines Core Group will evaluate the results of the first and subsequent rounds of augmented examinations as documented in Section 7 of MRP-227. This evaluation may result in recommendations for changes in the examination periodicity, which would be evaluated for applicability to KPS (Reference 8.17).

Flaw indications detected during the required examinations are dispositioned in accordance with the Corrective Action Program.

The monitoring and trending will be consistent with the corresponding element as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.6 PROGRAM ELEMENT 6 – ACCEPTANCE CRITERIA

As discussed in Section 4.4, the Reactor Vessel Internals Inspections program is implemented in accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program that include the inspection procedures, the acceptance criteria for each examination technique, and the process for evaluating unsatisfactory inspection results. For ASME Code Section XI inspections, indications and relevant conditions detected during examination are required to be evaluated in accordance with ASME Section XI, Article IWB-3500.

In addition to the ASME Code Section XI requirements, MRP-227, MRP-228, WCAP-17096, and WCAP-17020-P provide the examination acceptance criteria for the Primary and Expansion components and the expansion criteria for expanding the examinations beyond the Primary components to include the Expansion components (References 8.18, 8.19).

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

- For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that detected and sized for length by VT-1/EVT-1 examinations;

- For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and
- For physical measurements, the required examination acceptance criterion is related to the hold down spring. At KPS the hold down spring is fabricated from alloy 403SS and therefore does require inspection.

For the examination of the baffle-former bolts and the baffle-former edge bolts, a plant specific evaluation to determine the minimum acceptable bolting pattern in accordance with the NRC reviewed and approved methodology in WCAP-15029-P-A has been completed and is documented in WCAP-15425, (References 8.20, 8.21).

Table 2 and Table 4 identify the specific reactor vessel internals components in the program, the aging effects applicable to each component, the examination techniques used to detect the aging effect(s), and the additional examination acceptance criteria and inspection expansion criteria for the Primary and Expansion category components.

The acceptance criteria will be consistent with the corresponding element as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.7 PROGRAM ELEMENT 7 – CORRECTIVE ACTIONS

As discussed in Section 4.4, the Reactor Vessel Internals Inspections program is implemented for applicable components in accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program that include the inspection procedures, the acceptance criteria for each examination technique, and the process for evaluating unsatisfactory inspection results.

For the additional MRP-227-required inspection of the Primary and Expansion category components, MRP-227 provides information on methodology that can be used for the evaluation of detected conditions that exceed the examination acceptance criteria. The flaw evaluation methodology accounts for the accumulated neutron exposure and the resulting loss of fracture toughness due to radiation embrittlement in assessing the suitability of the component for continued service. Justification for flaw evaluation fracture toughness limits is provided in Section 6 of MRP-227.

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 functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. For example, previous NRC-endorsed alternative corrective actions bases include the corrective actions bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7, and 4-8 of Westinghouse Report No. WCAP-14577 Rev 1-A (Reference 8.22).

The Quality Assurance Program and sub-tier procedures fulfill and implement the requirements of Criterion XVI of 10 CFR Part 50, Appendix B. The process includes provisions for timely evaluation of adverse conditions and implementation of corrective actions, including root cause determinations and prevention of recurrence, where appropriate. Provisions are in place for tracking, monitoring, reviewing, and approving corrective actions to ensure effective corrective actions are taken, along with monitoring the process for adverse trends (References 8.23, 8.24).

The corrective actions will be consistent with the corresponding element as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.8 PROGRAM ELEMENT 8 – CONFIRMATION PROCESS

As discussed in Section 4.4, the Reactor Vessel Internals Inspections program is implemented in accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program that include the inspection procedures, the acceptance criteria for each examination technique, and the process for evaluating unsatisfactory inspection results.

The corrective action program, as presented above, includes provisions to ensure that all significant conditions adverse to quality receive a cause determination and that corrective actions are implemented to preclude reoccurrence (References 8.23, 8.24).

The confirmation process will be consistent with the corresponding element as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.9 PROGRAM ELEMENT 9 – ADMINISTRATIVE CONTROLS

The Quality Assurance Program and sub-tier procedures fulfill and implement the requirements of 10 CFR Part 50, Appendix B. Administrative controls include

management of document issue, revision, review and approval (References 8.23 and 8.25 – 8.28).

The MRP-227 requirements are administered by Dominion using Fleet Reactor Internals Inspection Program Description, ER-AA-RII-10 and Fleet Reactor Internals Inspection Administrative Procedure ER-AA-RII-101.

The administrative controls will be consistent with the corresponding element as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

4.10 PROGRAM ELEMENT 10 – OPERATING EXPERIENCE

The Reactor Vessel Internals Inspections program is a new program. Therefore, no internal programmatic experience is available. As operating experience is obtained, lessons learned will be used to adjust this program as needed.

Relatively few incidents of PWR vessel internals aging degradation have been reported in operating U.S. commercial PWR plants. However, a considerable amount of PWR vessel internals aging degradation has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. Reactor internals failures (both domestic and international), research data, and vendor evaluations have been considered in the development of MRP-227, which forms the basis for the Reactor Vessel Internals Inspections program.

The PWROG has performed a survey under PA-MSC-0568 to gather reactor internals inspection results to date. Reactor vessel internals inspection results performed under MRP-227 in the future will be reported to the EPRI MRP Issue Program within 120 days following completion of the inspections. The EPRI MRP program plans to publish future inspection results under MRP-219 (Reference 8.29).

The Operating Experience Program ensures that additional operating experience is factored into the aging management programs to ensure program effectiveness (Reference 8.30).

5.0 PROGRAM ENHANCEMENTS

The Reactor Vessel Internals Inspection program is a new plant-specific program that will be consistent with the applicable ten elements as described in Fleet GARD ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components." These program elements are the same as the program elements included in MRP-227 Appendix A and Appendix A of

NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

6.0 PROGRAM EXCEPTIONS

The Reactor Vessel Internals Inspection program is a new plant-specific aging management program.

7.0 CONCLUSION

The Reactor Vessel Internals Inspections program ensures that the effects of aging associated with the in-scope components will be adequately managed so that there is reasonable assurance that their intended functions will be maintained consistent with the current licensing basis throughout the period of extended operation.

8.0 REFERENCES

8.1 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Code of Federal Regulations, U.S. Nuclear Regulatory Commission, Washington, D.C.

8.2 Central Reporting System, Condition Report 322266, "Mandatory and Needed Requirements for Reactor Vessel Internals," February 3, 2009.

8.3 Nuclear Fleet Guidance and Reference Document ER-AA-AMP-1003, "Aging Management Programs for License Renewal Structures, Systems, and Components."

8.4 EPRI Report 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," December 2008, Electric Power Research Institute, Palo Alto, California

8.5 NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," U. S. Nuclear Regulatory Commission, September 2005.

8.6 NEI 03-08, "Guideline for the Management of Materials Issues," April 2007.

8.7 Technical Report KLR-1309, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

8.8 Memo from Westinghouse Advanced Reactor Internals Design & Analysis to Donna Rogosky, "Summary Report for the Fabrication and Design Information for KPS Reactor Vessel Internals," dated December 5, 2008, [LTR-ARIDA-08-63 Rev. 3].

8.9 Dominion Equipment Reliability (ER) Peer Group Project Charter ER-31, "Fleet Reactor Internals Inspection Program."

8.10 Nuclear Fleet Program Description ER-AA-RII-10, "Fleet Reactor Internals Inspection."

8.11 Nuclear Fleet Administrative Procedure ER-AA-RII-101, "Fleet Reactor Internals Inspection Program."

8.12 Technical Report KLR-1310, "Primary Water Chemistry."

8.13 Nuclear Fleet Nondestructive Examination Procedure ER-AA-NDE-VT-608, "VT-3 Visual Examination Procedure for Examination Category B-N-1, Interior of Reactor Vessel."

8.14 EPRI Report 1016592, "Materials Reliability Program: Aging Management Strategies for B&W PWR Internals (MRP-231)," Electric Power Research Institute, Palo Alto, CA.

8.15 EPRI Report 1016593, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232)," Electric Power Research Institute, Palo Alto, CA.

8.16 EPRI Report 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)," July 2009, Electric Power Research Institute, Palo Alto, CA.

8.17 Letter from Terry McAlister (EPRI) to Tanya M. Mensah (NRC), "EPRI MRP Responses to: Request for Additional Information Re: Electric Power Research Institute Topical Report 1016596, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-REV. 0)' (TAC NO. ME0680), August 24, 2009," dated February 1, 2010 [MRP 2010-004].

8.18 WCAP-17096, Revision 0, "Reactor Internals Acceptance Criteria Methodology and Data Requirements".

8.19 WCAP-17020-P, Revision 0, "Point Beach Unit 1 Upper Internals Guide Tube – Guide Card Wear Evaluation".

8.20 WCAP-15029-P-A, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Loaded Conditions".

8.21 WCAP-15425, "Determination of Acceptable Baffle-Barrel-Bolting for Kewaunee and Prairie Island Plants."

8.22 WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals."

8.23 Topical Report DOM-QA-1 "Dominion Nuclear Facility Quality Assurance Program Description."

8.24 Nuclear Fleet Administrative Procedure PI-AA-200, "Corrective Action "

8.25 KPS Nuclear Administrative Directive NAD-03.01, "Directive, Implementing Document, and Procedure Control."

8.26 KPS General Nuclear Procedure GNP-03.01.01, "Directive, Implementing Document, and Procedure Administrative Controls."

8.27 DNAP-0501, "Dominion Nuclear Procedure Administrative Control Program."

8.28 Nuclear Fleet Program Description AD-AA-10, "Administrative Controls Program."

8.29 MRP-219, "Inspection Data Survey Report."

8.30 Nuclear Fleet Guidance and Reference Document PI-AA-100-1007, "Operating Experience Program."

9.0 ATTACHMENTS

Attachment 1 – Implementing Procedures

Attachment 2 – Associated Aging Management Programs

Attachment 3 – Materials and Aging Effects Managed

Attachment 4 – Integrated Plant Assessment Reports / Systems or Commodities

Attachment 5 – Detailed NUREG-1800 Program Evaluation

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Baffle Former Assembly Including : • Baffle Assembly baffle/former plates • Baffle-edge bolts including bolting lock bars • Baffle-former bolts including bolting lock bars	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	Primary • Baffle Former Assembly	Baffle Former Assembly • VT-3[6] • Core side surface as indicated	Baffle Former Assembly • Baseline examination between 20 and 40 EFPY. • Subsequent examinations on a ten-year interval.	New procedure[8] required to perform the VT-3 inspections.	
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling		Primary • Baffle-edge bolts and locking devices	Baffle-edge bolts and locking devices • VT-3[6] • Bolts and locking devices on high fluence seams. 100% of components accessible from core side.	Baffle-edge bolts and locking devices • Baseline examination between 20 and 40 EFPY. • Subsequent examinations on a ten-year interval.		
	Loss of material	pitting and crevice corrosion						
	Loss of preload	stress relaxation		NAM[3] • Baffle bolting lock bars Primary • Baffle-former bolts Expansion • Baffle-former bolts	Baffle-former bolts • UT • 100% of accessible bolts or as supported by plant specific justification. Heads accessible from the core side.	Baffle-former bolts • Baseline examination between 25 and 35 EFPY. • Subsequent examination after 10 to 15 additional EFPY Re-examination for high leakage core	New procedure[8] required to perform the UT inspections.	

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
						designs requires continuing examinations on a ten-year interval.		

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Bottom-Mounted Instrumentation columns Including: • BMI column bodies • BMI column bolts • BMI column collars • BMI column cruciforms • BMI column extension bars • BMI column extension tubes • BMI column lock caps • BMI column nuts	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM Except as listed below.				CASS - BMI column cruciforms
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Loss of fracture toughness	thermal aging and neutron irradiation embrittlement		Expansion • BMI column bodies	BMI column bodies • VT-3[6] • 100% of BMI column bodies for which difficulty is detected during flux thimble insertion/ withdrawal.	BMI column bodies • As indicated by difficulty of insertion/withdrawal of flux thimbles. Flux thimble insertion/ withdrawal to be monitored at each inspection interval.	New procedure[8] required to perform the VT-3 inspections.	
	Loss of material	pitting and crevice corrosion						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Core Barrel Including: • Core Barrel flange • Core Barrel outlet nozzles • Lower Core Barrel • Upper Core Barrel • Ring Sector	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM Except as listed below.				
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking		Expansion- • Core barrel flange weld	Core barrel flange weld • EVT-1 • 100% of one side of the accessible surfaces of the selected weld and adjacent base metal.	Core barrel flange weld • Initial examination and reexamination frequency dependent on the examination results for upper core barrel flange.	New procedure[8] required to perform the EVT-1 inspections.	
				Expansion- • Core barrel outlet nozzles	Core Barrel outlet nozzles • EVT-1 • 100% of one side of the accessible surfaces of the selected weld and adjacent base metal.	Core Barrel outlet nozzles • Initial examination and reexamination frequency dependent on the examination results for upper core barrel flange	New procedure[8] required to perform the EVT-1 inspections.	
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling		Expansion- • Lower core barrel flange weld	Lower core barrel flange weld • EVT-1 • 100% of one side of the accessible surfaces of the	Lower core barrel flange weld • Initial examination and reexamination frequency dependent on the	New procedure[8] required to perform the EVT-1 inspections.	

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
					selected weld and adjacent base metal.	examination results for upper core barrel		
				Primary • Upper core barrel flange weld	Upper core barrel flange weld • EVT-1 • 100% of one side of the accessible surfaces of the selected weld and adjacent base metal.	Upper core barrel flange weld • No later than 2 refueling outages from the beginning of the license renewal period. • Subsequent examination on a ten-year interval.	New procedure[8] required to perform the EVT-1 inspections.	

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Interfacing Components Including: • Clevis inserts • Clevis insert bolts • Clevis insert lock keys • Head and vessel alignment pins • Head and vessel alignment pin bolts • Head and vessel alignment pin lock cups • Hold-down spring	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM • Except as noted below.				
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking		Existing[4] • Clevis insert bolts				
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						
	Loss of preload	stress relaxation		Primary • Hold-down spring (304SS only)	NA – KPS does not have 304SS hold-down springs[5].			

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Irradiation specimen guide Including: Irradiation specimen guide bolts Irradiation specimen lock caps Irradiation specimen plugs			ER-AA-NDE-VT-608	NAM				
Lower core plate Including: • Fuel alignment pins • Fuel alignment pin bolts • Fuel alignment pin lock caps	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM Existing				
	Cracking	stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Lower core plate support columns Including: • Column bodies • Column bolts • Lug • Nuts (Bearing Nut) • Sleeves	Changes in dimensions	void swelling		NAM • Except as noted below.				
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking		Expansion • Column bodies	Column bodies • EVT-1 • 100% of accessible surfaces.	Column bodies • Initial examination and reexamination frequency dependent on the examination results for upper core barrel flange weld.	New procedure[8] required to perform the EVT-1 inspections	
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling		Expansion • Column bolts	Column bolts • UT • 100% of accessible bolts or as supported by plant-specific justification.	Column bolts • Initial and subsequent examinations dependent on results of baffle-former bolt examinations.	New procedure[8] required to perform the UT inspections	
	Loss of preload	stress relaxation						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Lower Internals Including: • Diffuser plate • Flux thimble tubes • Flux thimble tube plugs • Head cooling spray nozzles • Lower support forging (13 inch tk. plate) • Radial support keys • Radial support key bolts • Radial support key lock keys	Changes in dimensions	void swelling stress corrosion cracking, irradiation-assisted stress corrosion cracking	ER-AA-NDE-VT-608	NAM Existing				
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
RCCA guide tubes Includes: • Anti-rotation studs and nuts • Bolts • C tubes • Enclosure pins • Upper guide tube enclosures • Flanges-intermediate • Flanges-lower • Guide plates/cards • Housing plates • Inserts • Lock bars • Sheaths • Support pins • Support pin cover plate • Support pin cover plate locking caps	Changes in dimensions	void swelling		NAM • Except as noted below.				
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking		Primary • Flanges-lower welds	Flanges-lower welds • EVT-1 • 100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal.	Flanges-lower welds • No later than 2 refueling outages from the beginning of the license renewal period. • Subsequent examination on a ten-year interval.	New procedure[8] required to perform the EVT-1 inspections	
				Primary • Guide plates/cards	Guide plates/cards • VT-3[6] • 20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined.	Guide plates/cards • 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 exams on a ten-year basis.	New procedure[8] required to perform the VT-3 inspections	

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
and tie straps • Support pin nuts • Water flow slot ligaments								
• Flexures • Flexureless Inserts	Loss of fracture toughness	thermal aging and neutron irradiation embrittlement		Primary • Flexures	NA KPS does not have RCCA guide tubes flexures[5].			

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Secondary Core Support (SCS) Assembly Including: • SCS base plate • SCS bolts • SCS energy absorber • SCS guide post • SCS housing • Tie Plate • SCS lock keys	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM				
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Thermal shield Including: • Thermal shield bolts • Thermal shield flexures • Thermal shield dowels • Thermal shield bumper bar • Locking Bar • Adjustment Plug • Neutron panel • Neutron panel bolts • Neutron panel lock caps	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM • Except as noted below.				
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Cracking	fatigue		Primary • Thermal shield flexures	Thermal shield flexures • VT-3[6] • 100% of thermal shield flexures.	Thermal shield flexures • No later than 2 refueling outages from the beginning of the license renewal period. • Subsequent examinations on a ten-year interval.	New procedure[8] required to perform the VT-3 inspections	
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						
	Loss of material	wear						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Upper core plate Including : • Upper core plate alignment keyways and pins • Upper core plate fuel guide pins • Upper core plate mixing devices	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM Existing				CASS • Upper core plate mixing devices
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Upper core plate instrumentation columns Including: • Bolting • Brackets, clamps, terminal blocks, and conduit straps • Conduit seal assembly–body, tubesheets • Conduit seal assembly–tubes • Conduits • Flange base • Locking caps • Support tubes	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM				CASS • conduit support • clamp
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Upper core plate support columns Including • Adapters • Bolts • Column bases • Column bodies; Upper Support Column - Type II • Extension tubes • Flanges • Lock keys • Thermocouple stop - located at flow mixing device • Nuts	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM				CASS • support column bases • thermocouple stop
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking						
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						
	Loss of preload	stress relaxation						

Table 2 - Reactor Vessel Internals Evaluation

RVI Component	LR Aging Effect	LR Aging Mechanism	Existing Inspections	MRP-227 Information				CASS Components[2]
				Category[1]	Required Inspections[7]	Inspection Schedule	Procedure Requirements	
Upper core support plate assembly Including • Bolts • Deep beam ribs • Deep beam stiffeners • Upper core support plate - welded to deep beam assy. • Lock keys • Upper core support ring or skirt	Changes in dimensions	void swelling	ER-AA-NDE-VT-608	NAM • Except as noted below.				
	Cracking	stress corrosion cracking, irradiation-assisted stress corrosion cracking		Existing • Upper support ring or skirt				
	Loss of fracture toughness	neutron irradiation embrittlement, void swelling						

Table Notes:

1. MRP-227 inspection categories are discussed in Section 4.4.
2. The CASS (Cast Austenitic Stainless Steel) components identified in Reference 8.8 for evaluation in accordance with KLR-1309, Section 5.2. A component specific evaluation must be performed determine the susceptibility of each CASS component to thermal embrittlement or an inspection is required to detect cracking as discussed in Section 4.4.
3. NAM is the MRP-227 No Additional Measures category.
4. Existing is the MRP-227 Existing Programs category.
5. Reference 8.8, Table 3: Kewaunee Components.

6. The MRP-227 inspection recommendations for VT-3 to monitor for the effects of cracking are appropriate because they are limited to cases where the intent of the examination is to monitor the general condition of the component. The VT-3 examination is specified where the objective of the examination is detection of broken or missing bolt locking devices and welds; protruding bolts; or broken or missing pieces (e.g., supports, spider arms, dowels). VT-3 is not specified where the examination objective the detection of the onset of cracking with accompanying tight crack opening displacements (CODs). VT-3 as currently defined in Section XI of the ASME Code is capable of this level of detection. The MRP-227 recommendations are consistent with the approach used in the ASME Section XI examinations, which require VT-3 inspections for accessible core support structures (Reference 8.17).
7. The coverage requirements for the required inspections based on MRP-227 are adequate to ensure timely detection of aging effects in the reactor vessel components. Coverage for a visual examination of a single component is defined as the percentage of the target surface area observed in the examination. Consideration of accessibility in terms of target surface is included in the coverage requirements for a number of the components. Additionally, accessibility was a factor in determining coverage for inspection of a variety of bolts, locking devices and lock welds. In these cases, coverage referred to the fraction of components inspected rather than the coverage associated with any particular examination (Reference 8.17).
8. Additional NDE procedures (either fleet, plant-specific, or vendor) will be needed to implement the revised ISI Plan for Reactor Vessel Internals Inspections.

Table 3 - Aging Management Program Enhancement and Inspection Implementation Schedule[1]

Refueling Outage	Target Month/Year	Estimated EFPY	AMP - Scope	Inspection Method Criteria	Comment
KR-31	Spring 2011	30.16	Not Applicable	Not Applicable	
KR-32	Spring 2012	31.15	RCCA guide tubes Guide Plate/Cards	MRP-227 inspections in accordance with MRP-228 specifications and ASME Section XI	These items are planned for KR-32. If not performed in KR-32 they will be inspected in KR-33.
KR-33	Fall 2013	32.64	Baffle Former Bolts	MRP-227 inspections in accordance with MRP-228 specifications and ASME Section XI	Original license expires December 21, 2013. Expansion scope will be implemented on an as needed basis.
KR-34	Spring 2015	34.03	Baffle Former Assembly, Baffle-edge bolts and locking devices, Baffle-former bolts, Upper core barrel flange weld, RCCA guide tubes flanges lower welds, RCCA guide tubes guide plates/ cards, Thermal shield flexures, CASS items	MRP-227 inspections in accordance with MRP-228 specifications and ASME Section XI	Core barrel removed this outage. Expansion scope will be implemented on an as needed basis.
KR-35	Fall 2016	35.52	Not Applicable	Not Applicable	
KR-36	Spring 2018	36.91	Not Applicable	Not Applicable	
KR-37	Fall 2019	38.40	Not Applicable	Not Applicable	
KR-38	Spring 2021	39.79	Not Applicable	Not Applicable	
KR-39	Fall 2022	41.28	Not Applicable	Not Applicable	
KR-40	Spring 2024	42.67	Baffle Former Assembly, Baffle-edge bolts and locking devices, Baffle-former bolts, Upper core barrel flange weld, RCCA guide tubes flanges lower welds, RCCA guide tubes guide plates/ cards, Thermal shield flexures, CASS items	MRP-227 inspections in accordance with MRP-228 specifications and ASME Section XI	Core barrel may be removed this outage. Expansion scope will be implemented on an as needed basis.
KR-41	Fall 2025	44.16	Not Applicable	Not Applicable	
KR-42	Spring 2027	45.55	Not Applicable	Not Applicable	
KR-43	Fall 2028	47.04	Not Applicable	Not Applicable	
KR-44	Spring 2030	48.43	Not Applicable	Not Applicable	
KR-45	Fall 2031	49.92	Not Applicable	Not Applicable	
KR-46	Spring 2033	51.31	Not Applicable	Not Applicable	

Footnote 1 – Table 3 is for illustration, refer to the KPS ISI Program and IDDEAL for the implementation inspection schedule.

Table 4 – Detailed MRP-227 Reactor Vessel Internals – Component Inspection Requirements[1]

RVI Component	LR Effect	LR Aging Mechanism	Examination Coverage	*MRP-227 Information*				
				Category	Required Inspections	Inspection Schedule	Re-inspection Schedule	Expansion
Control Rod Guide Tube Assembly *Guide plates (cards)*	Loss of material	Wear	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined.	Primary	VT-3 - measurements	No later than 2 RFO from beginning of 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.	Subsequent exams required on a ten year interval per MRP-227 Rev. 0	None
Control Rod Guide Tube Assembly *Lower flange welds*	Cracking	Stress corrosion cracking, Fatigue	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal.	Primary	EVT-1	No later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Subsequent exams required on a ten year interval per MRP-227 Rev. 0	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast)
Core Barrel Assembly *Upper core barrel flange weld*	Cracking	Stress corrosion cracking	100% of one side of the accessible surfaces of the selected weld and adjacent base metal.	Primary	EVT-1	No later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Subsequent examinations on a ten-year interval	Remaining core barrel welds, Lower support column bodies (non cast)

Table 4 – Detailed MRP-227 Reactor Vessel Internals – Component Inspection Requirements[1]

RVI Component	LR Effect	LR Aging Mechanism	Examination Coverage	*MRP-227 Information*				
				Category	Required Inspections	Inspection Schedule	Re-inspection Schedule	Expansion
Baffle-Former Assembly *Baffle-edge bolts*	Cracking that results in •Lost or broken locking devices •Failed or missing bolts •Protrusion of bolt heads)	Irradiation-assisted stress corrosion cracking, Fatigue	Bolts and locking devices on high fluence seams. 100% of components accessible from core side.	Primary	VT-3	Baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Subsequent exams required on a ten year interval per MRP-227 Rev. 0	None
Baffle-Former Assembly *Baffle-former bolts*	Cracking	Irradiation-assisted stress corrosion cracking, Fatigue	100% of accessible bolts or as supported by plant specific justification Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs.	Primary	UT	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high leakage core designs requires continuing examinations on a ten-year interval.	Subsequent exams required after 10 to 15 additional EFPY	Lower support column bolts, Barrel-former bolts

Table 4 – Detailed MRP-227 Reactor Vessel Internals – Component Inspection Requirements[1]

RVI Component	LR Effect	LR Aging Mechanism	Examination Coverage	*MRP-227 Information*				
				Category	Required Inspections	Inspection Schedule	Re-inspection Schedule	Expansion
Baffle-Former Assembly *Assembly*	Changes in dimensions and cracking 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 joint	Void swelling or Irradiation-assisted stress corrosion cracking	Core side surface as indicated.	Primary	VT-3	Baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Subsequent exams required on a ten year interval per MRP – 227 Rev. 0	None
Thermal Shield Assembly *Thermal shield flexures*	Cracking or loss of material that results in thermal shield flexures excessive wear, fracture, or complete separation	Fatigue or wear	100% of thermal shield flexures	Primary	VT-3	No later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	Subsequent exams required on a ten year interval per MRP – 227 Rev. 0	None

Table 4 – Detailed MRP-227 Reactor Vessel Internals – Component Inspection Requirements[1]

RVI Component	LR Effect	LR Aging Mechanism	Examination Coverage	*MRP-227 Information*				
				Category	Required Inspections	Inspection Schedule	Re-inspection Schedule	Expansion
Core Barrel Assembly *Barrel-former bolts*	Cracking	Irradiation-assisted stress corrosion cracking, Fatigue	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads.	Expansion	UT	Initial and subsequent examinations dependent on results of baffle-former bolt examinations.	Subsequent examinations dependent on results of baffle-former bolt examinations.	None
Lower Support Assembly *Lower support column bolts*	Cracking	Irradiation-assisted stress corrosion cracking, Fatigue	100% of accessible bolts or as supported by plant-specific justification.	Expansion	UT	Initial and subsequent examinations dependent on results of baffle-former bolt examinations.	Subsequent examinations dependent on results of baffle-former bolt examinations.	None
Core Barrel Assembly *Core barrel flange, Core barrel outlet nozzles, Lower core barrel flange weld*	Cracking	Stress corrosion cracking, fatigue	100% of one side of the accessible surfaces of the selected weld and adjacent base metal.	Expansion	EVT-1	Initial examination and reexamination frequency dependent on the examination results for upper core barrel flange.	Reexamination frequency dependent on the examination results for upper core barrel flange.	None
Lower Support Assembly *Lower support column bodies* ***(non cast)***	Cracking	Irradiation-assisted stress corrosion cracking	100% of accessible surfaces.	Expansion	EVT-1	Initial examination and reexamination frequency dependent on the examination results for upper core barrel flange weld.	Reexamination frequency dependent on the examination results for upper core barrel flange weld.	None

Table 4 – Detailed MRP-227 Reactor Vessel Internals – Component Inspection Requirements[1]

RVI Component	LR Effect	LR Aging Mechanism	Examination Coverage	*MRP-227 Information*				
				Category	Required Inspections	Inspection Schedule	Re-inspection Schedule	Expansion
Lower Support Assembly *Lower support column bodies* ***(cast)***	Cracking including the detection of fractured support columns	Irradiation-assisted stress corrosion cracking	100% of accessible support columns.	Expansion	EVT-1	Initial examination and reexamination frequency dependent on the examination results for control rod guide tube (CRGT) lower flanges.	Reexamination frequency dependent on the examination results for control rod guide tube (CRGT) lower flanges.	None
Bottom Mounted Instrumentation System *Bottom-mounted instrumentation (BMI) column bodies*	Cracking including the detection of completely fractured column bodies	Fatigue	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/ withdrawal.	Expansion	VT-3	Examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	Flux thimble insertion/withdrawal to be monitored at each inspection interval.	None
Core Barrel Assembly Core Barrel Flange	Loss of material	Wear	All accessible surfaces at specified frequency.	Existing Programs	VT-3	Each 10 Year Interval Per Section XI Category B-N-3.	Per Section XI	None
Upper Internals Assembly Upper support ring	Cracking	SCC, Fatigue	All accessible surfaces at specified frequency.	Existing Programs	VT-3	Each 10 Year Interval Per Section XI Category B-N-3.	Per Section XI	None
Lower core plate	Loss of material	Wear	All accessible surfaces at specified frequency.	Existing Programs	VT-3	Each 10 Year Interval Per Section XI Category B-N-3.	Per Section XI	None

Table 4 – Detailed MRP-227 Reactor Vessel Internals – Component Inspection Requirements[1]

RVI Component	LR Effect	LR Aging Mechanism	Examination Coverage	*MRP-227 Information*				
				Category	Required Inspections	Inspection Schedule	Re-inspection Schedule	Expansion
Clevis insert bolts	Loss of material and cracking	Wear and stress relaxation	All accessible surfaces at specified frequency.	Existing Programs	VT-3	Each 10 Year Interval Per Section XI Category B-N-3.	Per Section XI	None
Upper core plate alignment pins	Loss of material	Wear	Eddy current surface examination as defined in plant response to IEB 88-09.	Existing Programs	VT-3	Each 10 Year Interval Per Section XI Category B-N-3.	Per Section XI	None
Flux thimble tubes	Loss of material	Wear	All accessible surfaces at specified frequency.	Existing Programs	Eddy Current (ET)	ER-AA-NDE-ET-501 WDI-ISI-088	Every five years	None

Footnotes:

[1] This table is based on the guidelines of MRP-227 for Westinghouse-designed plants. The following are *not* listed in this table:
"No Additional Measures" components from MRP-227 for Westinghouse Plants (no additional aging management is necessary for this group).
A comprehensive list of Section XI B-N-3 components and inspections (see KPS "ASME Section XI Inservice Inspection, Subsection IWB, IWC, IWD" program).

ATTACHMENT 1
IMPLEMENTING PROCEDURES

(1) Nuclear Fleet Nondestructive Examination Procedure ER-AA-NDE-VT-608[1], "VT-3 Visual Examination Procedure for Examination Category B-N-1, Interior of Reactor Vessel."

(2) Topical Report DOM-QA-1 "Dominion Nuclear Facility Quality Assurance Program Description."

(3) Nuclear Fleet Administrative Procedure PI-AA-200, "Corrective Action."

(4) KPS Nuclear Administrative Directive NAD-03.01, "Directive, Implementing Document, and Procedure Control."

(5) KPS General Nuclear Procedure GNP-03.01.01, "Directive, Implementing Document, and Procedure Administrative Controls."

(6) DNAP-0501, "Dominion Nuclear Procedure Administrative Control Program."

(7) Nuclear Fleet Program Description AD-AA-10, "Administrative Controls Program."

(8) Nuclear Fleet Guidance and Reference Document PI-AA-100-1007, "Operating Experience Program."

(9) Nuclear Fleet Program Description ER-AA-RII-10, "Fleet Reactor Internals Inspection."

(10) Nuclear Fleet Administrative Procedure ER-AA-RII-101, "Fleet Reactor Internals Inspection Program."

[1] Additional NDE procedures (either fleet, plant-specific, or vendor) will be needed to implement the revised ISI Plan for Reactor Vessel Internals Inspections.

ATTACHMENT 2
ASSOCIATED AGING MANAGEMENT PROGRAMS

Aging Management Programs Supporting KLR-1309A.

KLR-1310, "Primary Water Chemistry"

Aging Management Program Supported by KLR-1309A.

KLR-1309, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"

Aging Management Programs Verifying the Effectiveness of KLR-1309A.

None

Aging Management Programs – Effectiveness Validated by KLR-1309A.

None

ATTACHMENT 3
MATERIALS AND AGING EFFECTS MANAGED

Materials	Aging Effects
Nickel Alloys	Cracking
Nickel Alloys, Stainless Steel, and CASS (PWR Reactor Internals Under MRP-227)	Cracking (SCC, PWSCC, IASCC, Fatigue) Loss of material (Wear) Loss of Fracture Toughness (Neutron Irradiation or Thermal Aging Embrittlement) Loss of Preload (Thermal and Irradiation-Enhanced Stress relaxation or Creep) Dimensional Changes (Void Swelling)

ATTACHMENT 4
INTEGRATED PLANT ASSESSMENT REPORTS SYSTEMS OR COMMODITIES

KLR-1104, "Reactor Vessel, Internals, and Reactor Coolant System"

Section 4.2, "Reactor Vessel"

Section 4.3, "Reactor Vessel Internals"

Section 4.4, "Steam Generator"

Attachment 5
Detailed NUREG-1800 Program Evaluation

Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
Program Element: 1 - Scope of Program								
1	The specific program necessary for license renewal should be identified. The scope of the program should include the specific structures and components of which the program manages the aging.	KLR-1309, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," addresses the inservice inspections for the components included in Section XI, Subsections IWB-1100, IWC-1100, and IWD-1100 for ASME Class 1, 2, and 3 components, respectively. The aging of the Class 1 Components, Table IWB-2500-1, Examination category B-N-3 items is managed in accordance with the AMSE Code, Section XI requirements. KLR-1309 identifies two program enhancements in Sections 5.1 and 5.2 to incorporate applicable industry reactor vessel internals inspection initiatives. The two enhancements state: "5.1 Enhancement 1: Aging Management of Reactor Vessel Internals "The ASME Section XI Inservice Inspection Subsections IWB, IWC and IWD program will be enhanced to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval to augment the current inspections. "5.2 Enhancement 2: Aging Management of Thermal Aging And Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel "The ASME Section XI Inservice Inspection Subsections IWB, IWC and IWD program will be enhanced to include identification of the limiting susceptible cast austenitic stainless steel reactor vessel internals components from the standpoint of thermal aging susceptibility, neutron fluence, and cracking. For each identified component, a plan will be developed, which accomplishes aging management through either a supplemental examination or a component-specific evaluation. The plan will be submitted for NRC review and approval not less than 24 months before entering the period of extended operation."The scope of the KLR-1309A, "ASME S XI ISI - Reactor Vessel Internals Inspections" program is to implement the inspections required by the two program enhancements." The reactor vessel internals (RVI) included in the scope of the program consist of two basic assemblies, the upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and the lower internals assembly that can be removed, if desired, following a complete core off-load. EPRI Report 1016596 (MRP-227) provides the	None	None	None	The identified FAI and CAP/CR numbers were initiated in KLR-1309 and are included here to provide a tie to the basis for this program. No documents to implement the scope of this program were identified.	62	LA001042 LA001043

Attachment 5
Detailed NUREG-1800 Program Evaluation

Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
		results of the industry initiative via the Materials Reliability Program (MRP) to develop the inspection and evaluation guidelines for managing the long-term aging of pressurized water reactor (PWR) RVI. The report provides a generic list of the RVI structures for Westinghouse PWRs. Westinghouse letter LTR-ARIDA-08-63 Rev. 3 (Ref **) evaluated the generic MRP-227 scope with respect to KPS. The RVI components included in the program are identified in Table 2 and Table 4. These tables are based on the MRP-227 identified RVI components as modified by the Westinghouse evaluation documented in LTR-ARIDA-08-63 Rev. 3. Note: MRP-227 has been submitted to the NRC for review. Following NRC review and approval, MRP-227 will be revised to incorporate any necessary changes to the guidelines and reissued as MRP-227-A. The Reactor Vessel Internals Inspections program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A. However, any revisions to MRP-227 resulting from the issuance of MRP-227-A will not be incorporated for any inspections that are completed before MRP-227-A is issued. Not included in the program are consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation and welded attachments to the reactor vessel.						
Program Element: 2 – Preventive Actions								
1	The activities for prevention and mitigation programs should be described. These actions should mitigate or prevent aging degradation.	The Reactor Vessel Internals Inspections program s a condition monitoring program and does not include any preventive or mitigative actions.	None	None	None	None	None	None
2	For condition or performance monitoring programs, they do not rely on preventive actions and thus, this information need not be provided. More than one type of aging management program may be implemented to ensure that aging effects are managed.	As noted in Element 2, Attribute 1, the Reactor Vessel Internals Inspections program does not perform any preventive actions. The Reactor Vessel Internals program provides verification that The Primary Water Chemistry program has been effective in preventing primary water stress corrosion cracking.	None	None	None	None	None	None
Program Element: 3 – Parameters Monitored/Inspected								
1	The parameters to be monitored or inspected should be identified and linked to the degradation of the particular structure and component intended function(s).	The program manages the follow aging effects for the RVI by monitoring for the parameters identified: Stress Corrosion Cracking – SCC refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. The observed aging effect of SCC on component integrity is cracking. Irradiation-Assisted Stress Corrosion Cracking -- IASCC is a unique form of SCC that occurs only in highly irradiated components. The observed aging effect of IASCC on component integrity is cracking. Wear -- Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent	None	None	None	None	None	None

Attachment 5
Detailed NUREG-1800 Program Evaluation

Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
		materials and their surface condition. The observed aging effect of wear on component integrity is loss of material. Fatigue -- Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. The observed aging effect of fatigue on component integrity is cracking. Thermal Aging Embrittlement -- Thermal embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardened (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. The observed aging effect of thermal embrittlement on component integrity is cracking. Irradiation Embrittlement -- When exposed to high energy neutrons, the mechanical properties of stainless steel and nickel-base 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 observed aging effect of irradiation embrittlement on component integrity is cracking. Void Swelling and Irradiation Growth -- Void swelling is defined as a gradual increase in the volume of a component caused by the formation of microscopic cavities in the material. 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. The aging effect is a loss of mechanical closure integrity (or, preload) that can lead to unanticipated loading which, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking. The aging effects related to RVI components are identified in Element 4, Attribute 1.						
2	For a condition monitoring program, the parameter monitored or inspected should detect the presence and extent of aging effects. Some examples are measurements of wall thickness and detection and sizing of cracks.	The aging effects managed by the RVI Inspections program and associated parameters monitored are discussed in Element 3, Attribute 1.	None	None	None	None	None	None
3	For a performance monitoring program, a link should be established between the degradation of the particular structure or component intended function(s) and the parameter(s) being monitored. An example of linking the degradation of a passive component intended function with the performance being monitored is linking the fouling of heat exchanger tubes with the heat transfer intended function. This could be monitored by periodic heat balances. Since this example deals only with one intended function of the tubes, heat transfer, additional programs may be necessary to manage	The Reactor Vessel Internals Inspections program is not a performance monitoring program.	None	None	None	None	None	None

Attachment 5
Detailed NUREG-1800 Program Evaluation

Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
	other intended function(s) of the tubes, such as pressure boundary. A performance monitoring program may not ensure the structure and component intended function(s) without linking the degradation of passive intended functions with the performance being monitored. For example, a periodic diesel generator test alone would not provide assurance that the diesel will start and run properly under all applicable design conditions. While the test verifies that the diesel will perform if all the support systems function, it provides little information related to the material condition of the support components and their ability to withstand DBE loads. Thus, a DBE, such as a seismic event, could cause the diesel supports, such as the diesel embedment plate anchors or the fuel oil tank, to fail if the effects of aging on these components are not managed during the period of extended operation.							
4	For prevention and mitigation programs, the parameters monitored should be the specific parameters being controlled to achieve prevention or mitigation of aging effects. An example is the coolant oxygen level that is being controlled in a water chemistry program to mitigate pipe cracking.	The Reactor Vessel Internals Inspection program is not a prevention and mitigation program.	None	None	None	None	None	None
Program Element: 4 – Detection of Aging Effects								
1	Detection of aging effects should occur before there is a loss of the structure and component intended function(s). The parameters to be monitored or inspected should be appropriate to ensure that the structure and component intended function(s) will be adequately maintained for license renewal under all CLB design conditions. This includes aspects such as method or technique (e.g., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects. Provide information that links the parameters to be monitored or inspected to the aging effects being managed.	The aging management methodologies described in MRP-227 are based on either existing inservice examinations required by the ASME Code, Section XI or on well-documented and well-demonstrated examination methods with which the industry has considerable experience. The RVI Inspections program will use the follow methods to detect the parameters monitored identified in Element 3, Attribute 1: Visual (VT-3) Examination -- VT-3 examination has been determined to be an appropriate NDE method for the detection of general degradation conditions in many of the susceptible components. A VT-3 can detect structural distortion or displacement of parts to the extent that component function may be impaired; loose, missing, cracked, or fractured parts, bolting, or fasteners; corrosion or erosion that reduces the nominal section thickness by more than 5%; wear of mating surfaces that may lead to loss of function; and structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5%. Visual (VT-1) Examination -- VT-1 examination is defined in the ASME Code Section XI as an examination "conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion." Enhanced Visual (EVT-1) Examination -- EVT-1 examination has the same requirements VT-1 examination, with additional requirements intended to improve the detection and characterization of	None	None	None	None	None	None

Attachment 5
Detailed NUREG-1800 Program Evaluation

Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
		discontinuities taking into account the remote visual aspect of reactor internals examinations. Surface Examination -- Surface ET (eddy current) examination is specified as an alternative or as a supplement to visual examinations and provides for detection and length sizing of surface-breaking or near-surface cracks. Volumetric Examination – VT examinations are specified for bolts to detect planar defects. The RVI Inspections program inspections are performed in accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program, which includes the procedures for performing the inspections, the acceptance criteria for each examination technique, and the review and disposition of inspection results.						
2	Nuclear power plants are licensed based on redundancy, diversity, and defense-in-depth principles. A degraded or failed component reduces the reliability of the system, challenges safety systems, and contributes to plant risk. Thus, the effects of aging on a structure or component should be managed to ensure its availability to perform its intended function(s) as designed when called upon. In this way, all system level intended function(s), including redundancy, diversity, and defense-in-depth consistent with the plant's CLB, would be maintained for license renewal. A program based solely on detecting structure and component failure should not be considered as an effective aging management program for license renewal.	MRP-227 categorizes the RVI components into the following four categories: • Primary -- those RVI that are highly susceptible to the effects of at least one of the eight aging mechanisms. • Expansion – those RVI 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. • Existing Programs (Existing) -- those RVI that are susceptible to the effects of at least one of the eight aging mechanisms and for which existing generic and plant-specific aging management programs are capable of managing those effects. • No Additional Measures (NAM) -- those RVI for which the effects of all eight aging mechanisms are below the screening criteria. The examination techniques and schedule for inspection of the RVI components in the Primary and Expansion categories will be consistent with the requirements of MRP-227 and are provided in Tables 2 and 4. EPRI Report 1016609 (MRP-228) provides additional specific guidance to the inspection methodologies and requirements for the NDE examinations of the components in the Primary and Expansion categories. The implementing procedures must be consistent with the MRP-228 guidance. The personnel performing the inspections will be trained in accordance with the guidance in MRP-228. The RVI components in the Existing and NAM categories will be in accordance with the	ER-AA-NDE-VT-608	All	The purpose of this procedure is to provide a process for the application of VT-3 visual examination techniques for the examination of Examination Category B-N-1, Interior of Reactor Vessel in accordance with the requirements of ASME Section XI.	None	None	None
						The personnel performing inspections of reactor vessel internals for MRP-227 will be trained in accordance with the guidance in MRP-228.	138	None

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Detailed NUREG-1800 Program Evaluation

Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
		requirements of the ASME Code inspections per Section XI, Examination Category B-N-3, for removable internal structures, as outlined in the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program. For Examination category B-N-3, ER-AA-NDE-VT-608 specifies a VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel. The examinations are performed once during the inspection interval when the core barrel is removed. The visual VT-3 examination determines the general mechanical and structural condition of the components by inspecting for structural distortion or displacement of parts, loose/missing, cracked or fractured parts, loose/missing bolting or fasteners, debris, corrosion or erosion, and wear of mating surfaces. The inspections are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR Part 50, Appendix B. In accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program enhancements, a plant-specific inspection plan for the RVI components will be developed and will address the requirements established by MR-227 for Westinghouse designed PWRs, as approved or modified by the NRC. The plan will also define any proposed alternatives determined to be necessary. The plant-specific inspection plan will be developed to address the requirements established by MR-227 for Westinghouse designed PWRs, as approved or modified by the NRC, including any proposed alternatives, and will be submitted for NRC review and approval at least 24 months prior to entry into the period of extended operation. Note: MRP-227 has been submitted to the NRC for review. Following NRC review and approval, MRP-227 will be revised to incorporate any necessary changes to the guidelines and reissued as MRP-227-A. The Reactor Vessel Internals Inspections program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A. However, any revisions to MRP-227 resulting from the issuance of MRP-227-A will not be incorporated for any inspections that are completed before MRP-227-A is issued.						
			New Procedures	None	None	New procedures will be developed to perform the NDE examinations of the components in the Primary and Expansion categories as identified in Tables 2 and 4.	62	LA001042 LA001043
3	This program element describes "when," "where," and "how" program data are collected (i.e., all aspects of activities to collect data as part of the program).	The "when," "where," and "how" program information is discussed in Element 4, Attributes 1 and 2.	None	None	None	None	None	None
4	The method or technique and frequency may be linked to plant-specific or industry-wide operating	The examination methods and frequencies are consistent with industry guidance.	None	None	None	None	None	None

Attachment 5
Detailed NUREG-1800 Program Evaluation

Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
	experience. Provide justification, including codes and standards referenced, that the technique and frequency are adequate to detect the aging effects before a loss of SC intended function. A program based solely on detecting SC failures is not considered an effective aging management program.	As discussed in Element 4, Attribute 1, the examination techniques to be used in the examination of the RVI components are the accepted techniques used in the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program. As discussed in Element 4, Attribute 2, the inspection schedule is consistent with either MRP-227 or the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program, as applicable.						
5	When sampling is used to inspect a group of SCs, provide the basis for the inspection population and sample size. The inspection population should be based on such aspects of the SCs as a similarity of materials of construction, fabrication, procurement, design, installation, operating environment, or aging effects. The sample size should be based on such aspects of the SCs as the specific aging effect, location, existing technical information, system and structure design, materials of construction, service environment, or previous failure history. The samples should be biased toward locations most susceptible to the specific aging effect of concern in the period of extended operation. Provisions should also be included on expanding the sample size when degradation is detected in the initial sample.	The Reactor Vessel Internals Inspections program does not use sampling to characterize a group of components.	None	None	None	None	None	None
Program Element: 5 - Monitoring and Trending								
1	Monitoring and trending activities should be described, and they should provide predictability of the extent of degradation and thus effect timely corrective or mitigative actions. Plant-specific and/or industry-wide operating experience may be considered in evaluating the appropriateness of the technique and frequency.	As discussed in Element 4, Attribute 2, the inspection schedule is consistent with either MRP-227 or the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program, as applicable.	None	None	None	None	None	None
2	This program element describes "how" the data collected are evaluated and may also include trending for a forward look. This includes an evaluation of the results against the acceptance criteria and a prediction regarding the rate of degradation in order to confirm that timing of the next scheduled inspection will occur before a loss of SC intended function. Although aging indicators may be quantitative or qualitative, aging indicators should be quantified, to the extent possible, to allow trending. The parameter or indicator trended should be described. The methodology for analyzing the inspection or test results against the acceptance criteria should be described. Trending is a comparison of the current monitoring results with previous monitoring results in order to make predictions for the future.	The acceptance criteria and inspection expansion criteria are provided in Tables 2 and 4 for the RVI components in the MRP-227 Primary and Expansion categories. In addition to the Primary components, Expansion components have been defined should the scope of examination and re-examination need to be expanded beyond the Primary group due to detection of significant aging effects. Tables 2 and 4 identify the reactor vessel internals components in the program, the aging effects applicable to each component, the examination techniques used to detect the aging effect(s), and the examination schedule for the Primary and Expansion category components. The RVI components in the Existing and NAM categories will be evaluated in accordance with the requirements of the ASME Code for Section XI, Examination Category B-N-3, for removable internal structures, as outlined in the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program.	None	None	None	None	None	None

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Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
Program Element: 6 – Acceptance Criteria								
1	The acceptance criteria of the program and its basis should be described. The acceptance criteria, against which the need for corrective actions will be evaluated, should ensure that the structure and component intended function(s) are maintained under all CLB design conditions during the period of extended operation. The program should include a methodology for analyzing the results against applicable acceptance criteria. For example, carbon steel pipe wall thinning may occur under certain conditions due to erosion-corrosion. An aging management program for erosion-corrosion may consist of periodically measuring the pipe wall thickness and comparing that to a specific minimum wall acceptance criterion. Corrective action is taken, such as piping replacement, before reaching this acceptance criterion. This piping may be designed for thermal, pressure, deadweight, seismic, and other loads, and this acceptance criterion must be appropriate to ensure that the thinned piping would be able to carry these CLB design loads. This acceptance criterion should provide for timely corrective action before loss of intended function under these CLB design loads.	As discussed in Element 4, Attribute 1, the RVI Inspections program is implemented in accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program. The program includes the inspection procedures, the acceptance criteria for each examination technique, and the process for evaluating unsatisfactory inspection results. For ASME Code Section XI inspections, indications and relevant conditions detected during examination are required to be evaluated in accordance with ASME Section XI, Article IWB-3500. The acceptance criteria and inspection expansion criteria are provided in Tables 2 and 4 for the RVI components in the MRP-227 Primary and Expansion categories. The RVI components in the Existing and NAM categories will be evaluated in accordance with the requirements of the ASME Code for Section XI, Examination Category B-N-3, for removable internal structures, as outlined in the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program.	None	None	None	None	None	None
2	Acceptance criteria could be specific numerical values, or could consist of a discussion of the process for calculating specific numerical values of conditional acceptance criteria to ensure that the structure and component intended function(s) will be maintained under all CLB design conditions. Information from available references may be cited.	Refer to Element 6, Attribute 1.	None	None	None	None	None	None
3	It is not necessary to justify any acceptance criteria taken directly from the design basis information that is included in the FSAR because that is a part of the CLB. Also, it is not necessary to discuss CLB design loads if the acceptance criteria do not permit degradation because a structure and component without degradation should continue to function as originally designed. Acceptance criteria, which do permit degradation, are based on maintaining the intended function under all CLB design loads.	Refer to Element 6, Attribute 1.	None	None	None	None	None	None
4	Qualitative inspections should be performed to same predetermined criteria as quantitative inspections by personnel in accordance with ASME Code and through approved site specific programs.	Refer to Element 4, Attribute 1.	None	None	None	None	None	None
Program Element: 7 - Corrective Actions								
1	Actions to be taken when the acceptance criteria are not met should be described. Corrective actions, including root cause determination and prevention of recurrence, should be timely.	Topical Report DOM-QA-1, "Dominion Nuclear Facility Quality Assurance Program Description" and sub-tier procedures fulfill and implement the requirements of 10 CFR Part 50, Appendix B. The corrective action process provides reasonable assurance that events or conditions potentially adverse to quality are promptly identified, evaluated, and corrected.	DOM-QA-1	All	This topical report provides the quality assurance program description (QAPD) for Dominion's nuclear power stations and independent spent fuel storage installations.	None	None	None

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Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
			PI-AA-200	All	This procedure establishes the expectations for the identification and reporting of conditions that are adverse to quality, potentially adverse to quality, affect personnel safety, affect nuclear safety, affect plant reliability, adverse trends, or other conditions that do not meet expectations.	None	None	None
2	If corrective actions permit analysis without repair or replacement, the analysis should ensure that the structure and component intended function(s) will be maintained consistent with the CLB.	Refer to Element 7, Attribute 1.	None	None	None	None	None	None
Program Element: 8 - Confirmation Process								
1	The confirmation process should be described. It should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.	The Corrective Action program, as presented above, includes provisions to ensure that all significant conditions adverse to quality receive a cause determination and that corrective actions are implemented to preclude reoccurrence.	DOM-QA-1	All	This topical report provides the quality assurance program description (QAPD) for Dominion's nuclear power stations and independent spent fuel storage installations.	None	None	None
			PI-AA-200	All	This procedure establishes the expectations for the identification and reporting of conditions that are adverse to quality, potentially adverse to quality, affect personnel safety, affect nuclear safety, affect plant reliability, adverse trends, or other conditions that do not meet expectations.	None	None	None
2	The effectiveness of prevention and mitigation programs should be verified periodically. For example, in managing internal corrosion of piping, a mitigation program (water chemistry) may be used to minimize susceptibility to corrosion. However, it may also be necessary to have a condition monitoring program (ultrasonic inspection) to verify that corrosion is indeed insignificant.	Refer to Element 8, Attribute 1 for discussion of the Corrective Action Program.	None	None	None	None	None	None
3	When corrective actions are necessary, there should be follow-up activities to confirm that the corrective actions were completed, the root cause determination was performed, and recurrence is prevented.	Refer to Element 8, Attribute 1 for discussion of the Corrective Action Program.	None	None	None	None	None	None
Program Element: 9 - Administrative Controls								
1	The administrative controls of the program should be described. They should provide a formal review and approval process.	Topical Report DOM-QA-1, "Dominion Nuclear Facility Quality Assurance Program Description" and sub-tier procedures fulfill and implement the requirements of 10 CFR Part 50, Appendix B. Administrative and implementing procedures are reviewed, approved, and maintained as controlled documents in accordance with the procedure control process.	DOM-QA-1	All	This topical report provides the quality assurance program description (QAPD) for Dominion's nuclear power stations and independent spent fuel storage installations.	None	None	None
			GNP-03.01.01	All	This procedure provides the requirements for the preparation, review, approval, and distribution of directives, implementing documents, and procedures.	None	None	None
			NAD-03.01	All	This directive provides instruction for establishing responsibilities and requirements for the management of Nuclear Administrative Directives, Dominion Fleet Procedures, NMC Fleet Procedures, and site procedures or other implementing	None	None	None

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Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
					documents.			
			DNAP-0501	All	This procedure establishes the hierarchical procedural relationships; establishes the program requirements to administratively control Dominion nuclear common administrative and technical procedures; provides a description of the process necessary to develop and maintain the quality of Dominion nuclear common administrative and technical procedures; provides guidance and requirements for implementation and adherence for Dominion nuclear common administrative and technical procedures; and describes the methods by which Dominion nuclear common administrative procedure forms are controlled.	None	None	None
			AD-AA-10	All	This procedure defines the major elements of the administrative controls program; establishes the standard management controls as required to define the policies, programs and processes; supports management controls via the implementation tools, including the document hierarchy; and institutionalizes standardization throughout the business processes.	None	None	None
			ER-AA-RII-10	All	The purpose of this Program Description is to establish and implement the Reactor Internals Inspection Program (RIIP) at all Dominion nuclear sites. The RIIP is based upon requirements from the Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0) and site specific license renewal commitments.	None	None	None
			ER-AA-RII-101	All	The purpose of this procedure is to establish and implement the Reactor Internals Inspection Program (RIIP) at all Dominion nuclear sites. The RIIP is based upon requirements from the Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227 - Rev. 0) and site specific license renewal commitments.	None	None	None
2	Any aging management programs to be relied on for license renewal should have regulatory and administrative controls. That is the basis for 10 CFR 54.21(d) to require that the FSAR supplement includes a summary description of the programs and activities for managing the effects of aging for license renewal. Thus, any informal programs relied on to manage aging for license renewal must be administratively controlled and included in the FSAR supplement.	Refer to Element 9, Attribute 1 for discussion of the administrative controls	None	None	None	None	None	None
Program Element: 10 - Operating Experience								

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Attribute Number	Attribute Description	Attribute Response	Document Number	Document Section	Purpose	Document Change	FAI	CAP/CR
1	Operating experience with existing programs should be discussed. The operating experience of aging management programs, including past corrective actions resulting in program enhancements or additional programs, should be considered. A past failure would not necessarily invalidate an aging management program because the feedback from operating experience should have resulted in appropriate program enhancements or new programs. This information can show where an existing program has succeeded and where it has failed (if at all) in intercepting aging degradation in a timely manner. This information should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.	Operating Experience is discussed in Section 4.10.	PI-AA-100-1007	All	This document provides expectations for use of Operating Experience (OE) in daily work activities, direction on evaluating OE information and guidance on sharing a Nuclear Network Operating Experience (NN OE) entry with the industry.	None	None	None
2	An applicant may have to commit to providing operating experience in the future for new programs to confirm their effectiveness.	Operating Experience is discussed in Section 4.10.	None	None	None	None	None	None