Enclosure to WCNOC Letter RA 17-0088

Wolf Creek RVI Inspection Plan

(51 pages)

WOLF CREEK
NUCLEAR OPERATING CORPORATION

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Wolf Creek RVI Inspection Plan

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1.0 BACKGROUND

Commitment 19-B of the NRC's Safety Evaluation (NUREG 1915) [1] of the Wolf Creek License Renewal Application [2] states that Wolf Creek will perform the following tasks to address aging management of the reactor vessel internals (RVI): (1) Participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of 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, WCNOC will submit an inspection plan for reactor internals to the NRC for review and approval.

To address task (1), Wolf Creek actively participated in the EPRI joint industry program (IP) efforts to address age related degradation of the RVI components. These efforts culminated in the publication of MRP-227 Rev. 0, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines", [3] in December 2008. Upon receiving the NRC's safety evaluation, MRP-227-A [4] was issued in December 2011.

Wolf Creek satisfies task (2) by its continued participation in both the EPRI and the PWROG joint IP efforts to better define and manage the aging effects of the reactor vessel internals during periods of extended operation. All new recommendations issued by these IPs for improved aging management of the RVI are evaluated and implemented by Wolf Creek on a timely basis.

To address task (3), this document serves as the Wolf Creek RVI Inspection Plan which summarizes the overall Wolf Creek RVI Aging Management Program (AMP).

The Wolf Creek RVI AMP is implemented by a plant specific program document (WCRE-27) [5] that was first issued in October 2011, thereby satisfying the NEI 03-08 Mandatory Requirement of MRP-227, Rev. 0: *Each commercial U.S. PWR unit shall develop and document a program for management of aging of reactor internal components within thirty-six months following issuance of MRP-227-Rev. 0 (that is, no later than December 31, 2011).* Wolf Creek program document WCRE-27 is revised as needed to address ongoing industry efforts, including the issuance of MRP-227-A.

2.0 DESCRIPTION OF RVI AMP

The Wolf Creek RVI AMP is currently based upon EPRI MRP-227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." This document contains a description summary of the degradation mechanisms of concern, the categorization of components, and the inspection requirements. The methodology of the Wolf Creek RVI AMP is described below.

2.1 Degradation Mechanisms

A total of eight age related degradation mechanisms are considered applicable to the RVI components: 1) stress corrosion cracking (SCC); 2) irradiation assisted stress corrosion cracking (IASCC); 3) fatigue; 4) irradiation embrittlement (IE); 5) thermal embrittlement (TE); 6) wear; 7) void swelling; and 8) irradiation and thermal enhanced stress relaxation/creep. A brief description of these degradation mechanisms and the associated aging effects follows:

Stress Corrosion Cracking (SCC)

SCC is a localized, non-ductile failure caused by a combination of stress, susceptible material, and an aggressive environment. The fracture path of SCC can be either transgranular or intergranular in nature. The aggressive contaminants most commonly associated with SCC of austenitic stainless steels are dissolved chlorides and oxygen. Nickel base alloys such as Alloy 600 and X-750 have exhibited susceptibility to intergranular SCC in primary water without the presence of aggressive contaminants, commonly referred to as primary water stress corrosion cracking (PWSCC). SCC of SS in primary water is also considered feasible at high stress levels. The aging effect of SCC is cracking.

Irradiation Assisted SCC (IASCC)

IASCC is a form of intergranular SCC that results from the combined influence of neutron irradiation and an aggressive environment. A limited number of IASCC failures of RVI components, specifically fasteners, constructed of austenitic stainless steels and nickel base alloys have been observed. The aging effect of IASCC is cracking.

Fatigue

Fatigue is defined as the structural deterioration that can occur as a result of the periodic application of stress by mechanical, thermal, or combined effects. High cycle fatigue results from relatively low cyclic stress (<yield strength) applied for many ($>10^5$) cycles. Low cycle fatigue results from relatively high cyclic stress (≥yield strength) applied for low number of cycles. The aging effect of fatigue is cracking.

Irradiation Embrittlement (IE)

IE refers to a gradual and progressive change in mechanical properties of a material resulting from exposure to high levels of neutron irradiation. These changes include an increase in yield and tensile strengths, and a corresponding decrease in ductility and toughness. The aging effect of IE is loss of fracture toughness.

Thermal Embrittlement (TE)

Thermal embrittlement refers to the same gradual and progressive change in mechanical properties of a material as IE except it results from exposure to elevated temperatures rather than neutron irradiation. For the RVI components, TE is only a concern for SS castings and welds with duplex microstructures containing both ferrite and austenite. The aging effect of TE is loss of fracture toughness.

Wear

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

Void Swelling (VS)

Void swelling is the gradual increase in volume of a component caused by the formation of microscopic cavities. These cavities result from the nucleation and growth of vacancies created by exposure to high levels of neutron irradiation. During the initial licensing periods of domestic PWRs, field experience has not revealed any evidence of VS in RVI components; however, it is

postulated as a possibility during periods of extended operation based upon accelerated laboratory testing. The aging effect of VS is dimensional change.

Irradiation and Thermally Enhanced Stress Relaxation/Creep (SR/C)

Stress relaxation involves the short term unloading of preloaded components upon exposure to elevated temperatures or high levels of neutron irradiation. Creep is a longer term process in which plastic deformation occurs within a loaded component. The temperatures of RVI are typically not high enough to support creep; however, it can develop upon exposure to high levels of neutron irradiation over an extended period. The aging effect of stress relaxation and creep is loss of preload.

2.2 Component Categorization

Wolf Creek is a Westinghouse pressurized water reactor (PWR), four-loop reactor coolant system (RCS) configuration. A detailed description of the typical Westinghouse internals design characteristics is given in Section 3.1.3 of MRP-277-A [4]. Based upon the MRP-227-A methodology, the Wolf Creek RVI components were screened for susceptibility to the eight degradation mechanisms considering their chemical compositions, neutron fluence exposures, operating temperatures, and stress levels. Functionality assessments were then performed on the screened-in components to determine the effects of the applicable degradation mechanism(s) on functionality. Each of the RVI components was then categorized as an Existing Program, Primary, Expansion, or No Additional Measurements Component based upon the functionality analysis, component accessibility, operating history, existing evaluations, and prior examination results. A description of the component categories follows:

Primary Components

Primary Components are highly susceptible to at least one of the eight degradation mechanisms, for which augmented inspections are required on a periodic basis to manage the associated aging effect(s). Primary Components are considered lead indicators for the onset of the applicable degradation mechanism(s). Details of the required inspections for Primary Components are provided in Table 2-1, Westinghouse Plants Primary Components.

Expansion Components

Expansion Components are highly or moderately susceptible to at least one of the eight degradation mechanisms, but exhibit a high degree of tolerance to the associated aging effect(s). Augmented inspections are required once a specified level of degradation is detected in a linked Primary Component. Details of the required inspections for Expansion Components are provided in Table 2-2, Westinghouse Plants Expansion Components.

Existing Program Components

Existing Program Components are susceptible to at least one of the eight degradation mechanisms, for which existing plant programs are capable of managing the associated aging effect(s). Details of the required inspections for Existing Program Components are provided in Table 2-3, Westinghouse Plants Existing Programs Components. The Existing Programs are further discussed in Section 4.0, Licensee Action Items.

No Additional Measures Components

No Additional Measures Components are either not susceptible to any of the eight degradation mechanisms, or if susceptible the impact of failure on the functionality of the RVI components is insignificant. No further action is required for managing the aging of these RVI components.

2.3 Inspection of RVI Components

Inspections detailed in Table 2-1 and Table 2-3 are required to manage aging effects in Primary and Existing Program Components. Additionally, inspections detailed in Table 2-2 are required should evidence of aging degradation be detected in linked Primary Components.

Inspection Methodologies

Proven inspection methodologies are utilized to detect evidence of the relevant aging mechanism(s) for the Primary, Expansion, and Existing Program Components. These include the following:

- Direct physical measurements to monitor for loss of material or preload

- VT-3 exams to monitor for general degradation associated with loss of material or preload
- EVT-1 exams to monitor for surface breaking linear discontinuities indicative of cracking
- UT exams to monitor directly for cracking
- ECT to further characterize conditions detected by visual (VT-3 and EVT-1) exams

Requirements for the inspection methodologies and qualification of NDE systems used to perform those inspections are provided in EPRI MRP-228, "Inspection Standard for PWR Internals" [6].

Inspection Frequencies

Specified inspection frequencies are considered adequate to manage aging effects; however more frequent inspections may be warranted based upon an internal and external OE.

Inspection Coverage

The required inspection coverage for Primary and Expansion Components is specified in Tables 4-1 and 4-2, respectively. If the specified coverage cannot be obtained, the condition shall be addressed in the Wolf Creek Corrective Action Program (CAP).

Acceptance Criteria

The acceptance criteria for Primary and Expansion Components are provided in Table 2-4, Westinghouse Plants Examination Acceptance and Expansion Criteria. All detected relevant conditions must be addressed in the CAP prior to plant start-up. Possible disposition options include: 1) supplemental exams to further characterize a detected condition; 2) engineering evaluation for continued service until the next inspection; 3) repair; or 4) replacement. Engineering evaluations for continued service shall be conducted in accordance with NRC approved methodologies, described in WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" [7]. The potential loss of fracture toughness must be considered in any flaw evaluations.

Expansion Components

The criteria for expanding the scope of examination from the Primary to the linked Expansion Components are also provided in Table 2-4. Generally, the inspection of the Expansion Components is required in the RFO following that in which degradation of the linked Primary Component was detected.

It should be noted that the component categorizations and associated inspection requirements described above do not replace or relieve current ASME Section XI inspection requirements for the RVI components.

Table 2-1: Westinghouse Plants Primary Components

COMPONENT	APPLICABILITY	EFFECT (MECHANISM)	EXPANSION LINK (NOTE 1)	EXAMINATION METHOD	EXAMINATION COVERAGE
Control Rod Guide Tube Assembly Guide plates (cards)	Wolf Creek	Loss of material (wear)	None	Visual (VT-3) Per the schedule requirements of WCAP-17451-P Section 5 [16] including subsequent examinations (note 7)	Minimum examination of 20% of the number of CGRT assemblies, and as per the requirements of WCAP-17451-P Revision 1 Section 5 (note 7)
Control Rod Guide Tube Assembly Lower flange Welds	Wolf Creek	Cracking (SCC, fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast), Upper core plate, Lower support casting/forging	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the adjacent base metal on the individual periphery CRGT assemblies. (Note 2)
Core Barrel Assembly Upper Core Barrel Flange Weld	Wolf Creek	Cracking (SCC)	Lower support column bodies (non cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4)
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	Wolf Creek	Cracking (SCC, IASCC, Fatigue)	Upper and lower cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4)
Core Barrel Assembly Lower core barrel flange weld (Note 5)	Wolf Creek	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4)

COMPONENT	APPLICABILITY	EFFECT (MECHANISM)	EXPANSION LINK (NOTE 1)	EXAMINATION METHOD	EXAMINATION COVERAGE
Baffle-Former Assembly Baffle-edge bolts	Not applicable to Wolf Creek which has no baffle-edge bolts	Cracking (IASCC, fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. (Note 3)
Baffle-Former Assembly Baffle-former bolts	Wolf Creek	Cracking (IASCC, fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs.
Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	Wolf Creek	Distortion (void swelling), or cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated

COMPONENT	APPLICABILITY	EFFECT (MECHANISM)	EXPANSION LINK (NOTE 1)	EXAMINATION METHOD	EXAMINATION COVERAGE
Alignment and Interfacing Components Internals hold down spring	Not Applicable to Wolf Creek's 403 SS Spring	Distortion (loss of load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms.	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to eliminate uncertainty.
Thermal Shield Assembly Thermal Shield Flexures	Not Applicable to Wolf Creek's neutron panels which are bolted and pinned to core barrel and do not incorporate flexures	Cracking (fatigue) Loss of material (wear) that results in thermal shield flexures excessive wear, fracture, or complete separation.	None	Visual (VT-3) examination non later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures.

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 2-4.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 2-4, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 2-4, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.
7. WCAP-17451-P Revision 1 requires a remote visual examination consistent with visual (VT-3) for minimum compliance and examination coverage of a minimum of 20% of the number of CRGT guide card assemblies. The baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results.

Table 2-2: Westinghouse Plants Expansion Components

Component	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Internals Assembly Upper core plate	Wolf Creek	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Reexamination every 10 years following initial inspection.	100% of accessible surfaces (Note 2)
Lower Internals Assembly Lower support forging or castings	Wolf Creek	Cracking Aging Management (TE in casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Reexamination every 10 years following initial inspection.	100% of accessible surfaces (Note 2)
Core Barrel Assembly Barrel-former bolts	Wolf Creek	Cracking (IASCC, fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads. (Note 2)
Lower Support Assembly Lower support column bolts	Wolf Creek	Cracking (IASCC, fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant specific justification. (Note 2)
Core Barrel Assembly Core barrel outlet nozzle welds	Wolf Creek	Cracking (SCC, fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. (Note 2)
Core Barrel Assembly Upper and lower core barrel cylinder axial welds	Wolf Creek	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. (Note 2)
Lower Support Assembly Lower support column bodies (non cast)	Wolf Creek	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2)

Component	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Support Assembly Lower support column bodies (cast)	Wolf Creek	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination. Reexamination every 10 years following initial inspection.	100% of accessible support columns (Note 2)
Bottom Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	Wolf Creek	Cracking (fatigue) including the detection of completely fractured column bodies. Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Reexamination every 10 years following initial inspection. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal.

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

Table 2-3: Westinghouse Plants Existing Program Components

COMPONENT	APPLICABILITY	EFFECT (MECHANISM)	REFERENCE	EXAMINATION METHOD	EXAMINATION COVERAGE
Core Barrel Assembly Core barrel flange	Wolf Creek	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear	All accessible surfaces at specified frequency.
Upper Internals Assembly Upper support ring or skirt	Wolf Creek	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate	Wolf Creek	Cracking (IASCC, fatigue),	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate	Wolf Creek	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.
Bottom Mounted Instrumentation System Flux thimble tubes	Wolf Creek	Loss of material (wear)	NUREG-1801, Rev. 1	Surface (ET) examination	ET surface examination of full length tubes at frequency specified in BMI-FTT-IP. Tube selection and frequency based upon engineering evaluation of previous examination results.
Alignment and Interfacing Components Clevis insert bolt	Wolf Creek	Loss of material (wear) (Note 1)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.
Alignment and Interfacing Components Upper core plate alignment pins	Wolf Creek	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.

Note:

1. Bolt was screened in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue

Table 2-4: Westinghouse Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	Wolf Creek	Visual (VT-3) examination (Note 3) The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A
Control Rod Guide Tube Assembly Lower flange welds	Wolf Creek	Enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast), and upper core plate and lower support forging or casting	a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage. b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies, upper core plate and lower support forgings/castings within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, upper core plate and lower support forgings/castings, the specific relevant condition is a detectable crack-like surface indication.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	Wolf Creek	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Core barrel outlet nozzle welds b. Lower support column bodies (non cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core barrel outlet nozzle welds by the completion of the next refueling outage. b. If extensive cracking in the remaining core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles following the initial observation.	a and b. The specific relevant condition for the expansion core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel flange weld (Note 2)	Wolf Creek	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	None	None
Core Barrel Assembly Upper core barrel cylinder girth welds	Wolf Creek	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel cylinder girth welds	Wolf Creek	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower core barrel cylinder axial weld examination is a detectable crack-like surface indication.
Baffle-Former Assembly Baffle-edge bolts	Not applicable to Wolf Creek	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A
Baffle-Former Assembly Baffle-former bolts	Wolf Creek	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles. b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Assembly	Wolf Creek	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A
Alignment and Interfacing Components Internals hold down spring	Not Applicable to Wolf Creek's 403 SS Spring	Direct physical measurement of spring height. The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.	None	N/A	N/A
Thermal Shield Assembly Thermal shield flexures	Not Applicable to Wolf Creek's neutron panels which are bolted and pinned to core barrel and do not incorporate flexures	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation.	None	N/A	N/A

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. The lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
3. WCAP-17451-P Revision 1 specifies a remote visual examination consistent with visual (VT-3) but allows for various supplemental measurement techniques which if employed increase wear estimate accuracy and allow use of acceptance criteria (wear projections) to determine the appropriate re-examination interval.

3.0 AGING MANAGEMENT PROGRAM ATTRIBUTES

The attributes of the Wolf Creek RVI AMP and compliance with NUREG-1801, "Generic Aging Lessons Learned Report (GALL)" [8], Section XI.M16, PWR Vessel Internals, are described in this section. The GALL identifies 10 attributes for successful component aging management. The framework for assessing the effectiveness of the projected program is established by the use of the 10 elements of the GALL.

Table 3-1: RPV Program Attributes

GALL Element	Plan Attribute	Approach and supplemental information
1	Scope of Program	The Wolf Creek RVI AMP includes all RVI components which were built to the Westinghouse NSSS design. Using the guidance provided in MRP-227-A, the Wolf Creek RVI AMP was developed to manage the aging of these components during the initial and extended periods of operation. Components considered for inspection under MRP-227-A include core support structures, RVI components that serve an intended license renewal safety function pursuant to criteria in 10CFR54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any other functions identified in 10 CFR 54.4(a)(i), (ii), or (iii). The program does not included consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. The program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed by the Wolf Creek Reactor Vessel Surveillance Program.
2	Preventive Measures	The Wolf Creek Water Chemistry Program is credited for limiting the levels of corrosive chemical species (e.g. halogens, sulfur compounds, oxygen) in the RCS to extremely low levels as a preventative measure for corrosion related degradation mechanisms including pitting, crevice corrosion, SCC, PWSCC and IASCC.
3	Parameters Monitored	The Wolf Creek RVI AMP manages the following age-related degradation effects and mechanisms: 1) cracking induced by SCC, PWSCC, IASCC, or fatigue; 2) loss of material induced by wear; 3) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; 4) changes in dimension due to void swelling and irradiation growth, distortion or defection; and 5) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the Wolf Creek RVI AMP monitors for evidence of surface breaking linear discontinuities using visual (EVT-1) exams, or directly using volumetric (UT) or surface (ECT) exams. For the management of loss of material, the RVI AMP monitors for surface conditions that may be indicative of wear using visual (VT-3) exams. For the management of changes in dimension and loss of preload, the RVI AMP monitors for gross surface conditions using visual (VT-3) exams or direct physical measurements. The RVI AMP does not directly monitor for loss of fracture toughness but

GALL Element	Plan Attribute	Approach and supplemental information
		relies on visual or volumetric examination techniques to monitor for cracking in components. Specifically, the Wolf Creek RVI AMP implements the parameters monitored/inspected criteria for Westinghouse Designed Primary Components in Table 4-3 of MRP-227-A [4] and for guide card wear per WCAP-17451-P [16]. Additionally, the program implements the parameters monitored/inspected criteria for Westinghouse designed Expansion Components in Table 4-6 of MRP-227-A [4]. The parameters monitored/inspected for Existing Program Components follow the bases for the Wolf Creek ASME Section XI Program [9].
4	Detection of Aging Effects	Discussion and justification of the inspection methods selected for detection of the aging effects managed by the Wolf Creek RVI AMP are provided in MRP-227-A [4] and MRP-228 [6]. In all cases, well established methods described above were selected. Additionally, the RVI AMP adopts the recommended guidance in MRP-227-A for defining Expansion criteria that need to be applied to inspections of Primary and Existing Program Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are in conformance with the inspection criteria, sampling basis criteria and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Technical Position RLSB-1. Specifically, the Wolf Creek RVI AMP implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for Westinghouse designed Primary Components in Table 4-3 of MRP-227-A, WCAP-17451-P and for Westinghouse designed Expansion Components in Table 4-6 of MRP-227-A [4].
5	Monitoring and Trending	The methods for monitoring, recording, evaluating, and trending the data that result from the Wolf Creek RVI AMP inspections are given in Section 6 of MRP-227-A [4]. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227-A guidance, together with the requirements specified in MRP-228 [6] for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.
6	Acceptance Criteria	Section 5 of MRP-227-A [4] provides the examination acceptance criteria for the Primary and Expansion Components in the Wolf Creek RVI AMP. For Existing Program components referenced to ASME Section XI, the IWB-3500 acceptance criteria apply.
7	Corrective Actions	Components with identified relevant conditions shall be entered into the Wolf Creek Corrective Action Program (CAP). The disposition may include a supplementary examination to further characterize the relevant condition, an engineering evaluation to show that the component is capable of continued

GALL Element	Plan Attribute	Approach and supplemental information
		operation with a known relevant condition until the next planned inspection, or repair/replacement to remediate the relevant condition. Additional inspections of expansion category components may also be required. The disposition will insure that the design basis function of the RVI will continue to be fulfilled for all licensing basis loads and events.
8	Confirmation Process	The Wolf Creek Quality Assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the recommendations of NEI 03-08 [10] and the requirements of 10 CFR Part 50, Appendix B. The implementation of the guidance in MRP-227-A, in conjunction with NEI 03-08, and other guidance documents, reports or methodologies referenced in this AMP, provide an acceptable level of quality and basis for confirming the quality of inspections, flaw evaluations and corrective actions.
9	Administrative Controls	The administrative controls for the Wolf Creek RVI AMP, including its implementing procedure and review and approval processes, are implemented in accordance with the recommended industry guidelines and criteria in NEI 03-08, and are under the site 10 CFR 50 Appendix B, Quality Assurance Program.
10	Operating Experience	The review and assessment of relevant operating experience for impact on the Wolf Creek RVI AMP are governed by NEI 03-08 and Appendix A of MRP-227-A [4]. The reporting of inspection results and operating experience is treated as a "Needed" category item under NEI 03-08.

4.0 LICENSEE ACTION ITEMS

This section provides the Wolf Creek response to the eight Licensee Action Items (LAI) noted in the NRC Safety Evaluation Report (SER) issued by the NRC for MRP-227-A. Additionally, the three bounding assumptions included in Section 2.4 of MRP-227-A [4] are also addressed in the response to LAI #1.

LAI #1 Applicability of FMECA and Functionality Analysis Assumptions

As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. ***This is Applicant/Licensee Action Item 1.***

Wolf Creek Response:

The process used to provide reasonable assurance that Wolf Creek is reasonably represented by the generic industry program assumptions (with regard to neutron fluence, temperature, stress values, and materials used in the development of MRP-227-A) is:

1. Identification of typical Westinghouse (W)-designed pressurized water reactor (PWR) RVI components (Table 4-4 of MRP-191[11]).

2. Identification of Wolf Creek PWR components.

3. Comparison of the typical W-designed PWR RVI components to the Wolf Creek RVI components:

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

b. Confirmation that the materials from Table 4-4 of MRP-191 [11] are consistent with Wolf Creek RVI component materials.

c. Confirmation that the design and fabrication of Wolf Creek RVI components are the same as, or equivalent to, the typical W-designed PWR RVI components.

4. Confirmation that the Wolf Creek operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns and base load operation.

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

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

7. Confirmation that any changes to the Wolf Creek RVI components do not impact the application of the MRP-227-A generic aging management strategy.

The Wolf Creek RVI components are reasonably represented by the design and operating history assumptions regarding neutron fluence, temperature, materials, and stress values in the MRP-191 generic FMECA and in the MRP-232 [13] functionality analysis based on the following:

1. Wolf Creek operating history is consistent with the assumptions in MRP-227-A with regard to neutron fluence and fuel management.

a. FMECA and functionality analysis for MRP-227-A made the following assumption of 30 years of operation with high-leakage core loading patterns followed by 30 years of low-leakage core fuel management strategy. Wolf Creek went to a low leakage core starting in cycle 4 and had a full low leakage core at the start of cycle 7. The assumption of 30 years operation with a high leakage core is bounding for Wolf Creek. Therefore, Wolf Creek meets the fluence and fuel management assumptions in MRP-191 and requirements for MRP-227-A application.

b. Wolf Creek typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule. Therefore, Wolf Creek satisfies the assumptions in MRP documents regarding operational parameters affecting fluence.

c. Wolf Creek has also performed an evaluation of its fuel design and fuel management in accordance with the guidance provided in EPRI MRP 2013-025 (ML13322A454) [14]. The results of that evaluation indicate that Wolf Creek has not utilized atypical fuel designs or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative, including power changes/uprates that have occurred over the operating lifetime of the unit. This conclusion is based on comparisons of the Wolf Creek core geometry and operating characteristics with the MRP-227-A applicability guidelines for Westinghouse-designed reactors specified in MRP 2013-025. The details of this evaluation are included in Appendix A (PWROG-16010-NP).

2. The Wolf Creek reactor coolant system operates between T_{cold} and T_{hot}. T_{cold} is no lower than 555.8°F and T_{hot} is no higher than 621.1°F. The design temperature for the vessel is 650°F. Therefore, Wolf Creek operating history is within original design basis parameters and is consistent with the assumptions used to develop the MRP-227-A aging management strategy with regard to temperature operational parameters.

3. As discussed below, the Wolf Creek RVI components and materials are comparable to the typical W-designed PWR RVI components (MRP-191, Table 4-4 [11]).

a. The components required to be in in the Wolf Creek program are consistent with those contained in MRP-191.

b. Wolf Creek RVI component materials are consistent with, or equivalent to, those materials identified in Table 4-4 of MRP-191 [11] for W-designed plants. Where differences exist, there is no impact on the Wolf Creek RVI program (see response to LAI #2 below).

c. Design and fabrication of Wolf Creek RVI components are the same as, or equivalent to, the typical W-designed PWR RVI components.

d. Wolf Creek is actively participating in a joint industry program under the PWROG aimed at addressing the 20% cold work issue for non-weld or bolting austenitic stainless steel components on a generic rather than plant-specific basis. A discussion of this ongoing program (PA) follows:

PA-MSC-1288, PWR Materials Assessment, was discussed with the NRC at the June 2-4, 2015 Annual Materials Programs Technical Information Exchange Public Meeting (Ref. ML15155B431). This PA utilizes a statistical approach for determining and assessing material or fabrication factors for PWR internals components. To date, plant-specific component manufacturing records have been gathered for over 50% of the domestic PWRs. A review of these records in accordance with the guidance provided in MRP 2013-025 (ML1322A454) has revealed the following:

- 20% cold work limitation was already recognized at the time of plant construction, i.e. from 1970's
- Plant fabricators quality programs were in place to adhere to limitations in cold work in austenitic stainless steels in these times
- Plant specific assessments conducted to date confirm that no non-fastener materials contain cold work greater than 20%
- Correlation of data based on searches to date demonstrates consistency across the PWR fleet - B&W, CE and, W show no cold worked non-fastener materials used in reactor vessel internals

A final report (PWROG-15105-NP [12]) for this PA was issued in April 2016 and submitted to the NRC for information in June 2016 by letter OG-16-209. Wolf Creek will continue to participate in and follow the progress of PA-MSC-1288, including interactions with the NRC. If necessary, plant specific information will be provided to the NRC to supplement this joint industry program, if requested.

4. Modifications to the Wolf Creek reactor internals made over the lifetime of the plant are those specifically directed by Westinghouse, the Original Equipment Manufacturer (OEM). The OEM has developed or evaluated design changes and satisfied assumptions for LAI 1. The design has been maintained over the lifetime of the plant as specified by the OEM, operational parameters are compliant with MRP-227-A requirements with regard to fluence and temperature, and the components are consistent with those considered in MRP-191. The materials for the components are also consistent with those considered in MRP-191. Therefore, the Wolf Creek RVI stress values are represented by the assumptions in MRP-191, MRP-227-A, and MRP-232, confirming the applicability of the generic FMECA.

Conclusion:

The FMECA and Functionality Analysis Assumptions of MRP-227-A are applicable to the Wolf Creek RVI components.

LAI #2 PWR Vessel Internal Components within the Scope of License Renewal

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

This Wolf Creek RVI components were compared to those components contained in Table 4-4 of MRP-191 [11], as Wolf Creek is a W-plant design. Based on a comparison of the plant-specific components to Table 4-4 of MRP-191, the components required to be in the Wolf Creek RVI

AMP are mostly consistent with those contained in MRP-191, with the exceptions described below.

The following Wolf Creek RVI component materials differ from those specified in MRP-191.

Reactor Internals Assembly	RVI Component	MRP-191, Table 4-4 Material	Wolf Creek RVI Component Material
(1) Upper Internals Assembly, Upper instrumentation conduit and supports.	bolting	316 SS	304 SS
	brackets, clamps, terminal blocks and conduit straps	304 SS	304 SS and 302 SS
	locking caps	304 SS	304L SS
(2) Upper Internals Assembly, Upper support column assemblies	column bases	CF8	304 SS
(3) Lower Internals Assembly, Lower support column	bolts	304 SS	316 SS
(4) Lower Internals Assembly, Neutron panels	lock caps	304 SS	304L SS
(5) Lower Internals Assembly, Radial support key	bolts	304 SS	316 SS

The upper instrumentation conduit and support parts (1), lower support column bolts (3), neutron panels lock caps (4), and the radial support bolts (5) are fabricated from different grades of austenitic stainless steel (SS) than what are listed in the generic evaluation of MRP-191. Since the Wolf Creek plant-specific 302 SS, 304 SS, 304L SS, and 316 SS materials are in the austenitic SS category, there are no differences in the MRP screening criteria for these grades of materials; thus, there are no changes to screening and no additional degradation mechanisms are applicable.

The Wolf Creek upper support column bases (2) are fabricated from 304 SS, rather than the CF8 that is listed in MRP-191. In MRP-191, CF8 materials were screened-in for the aging mechanism of thermal embrittlement (TE), along with other applicable degradation mechanisms. Since the

Wolf Creek upper support column bases are 304 SS, the material is in the austenitic SS category and is not subject to TE. This material difference reduces the number of applicable degradation mechanisms for the upper support column bases at Wolf Creek Unit 1.

The differing Wolf Creek RVI component materials were determined to be suitable alternative materials during the original design process and, therefore, are equally capable of meeting component functional requirements. The material differences have no effect on the recommended MRP categorization and aging strategy; therefore, no modifications to the program details in MRP-227-A need to be proposed.

Conclusion:

The Wolf Creek RVI components are mostly consistent with those contained in Table 4-4 of MRP-191 [11]. Differing materials have no impact on the implementation requirements of MRP-227-A.

LAI #3 Evaluation of the Adequacy of Plant-Specific Existing Programs

As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). ***This is Applicant/Licensee Action Item 3.***

Wolf Creek Response:

Existing plant programs credited for adequately managing specific aging effects of selected components in the Wolf Creek RVI AMP include:

- Water Chemistry Program
- ASME Section XI In-Service Inspection Program
- Control Rod Guide Tube (CRGT) Support Pin (Split) Replacement
- Flux Thimble Tube Inspection Program

Water Chemistry Program

The Wolf Creek Water Chemistry Program is credited for controlling the levels of corrosive contaminants in the Primary Water System, thereby preventing or mitigating cracking of RVI components by SCC and IASCC. The Water Chemistry Program is not listed in Table 2-3 since it does not include any inspections of RVI components.

ASME Section XI In-Service Inspection Program

ASME Section XI, IWB-2500, provides inspection requirements for B-N-2 (Welded Core Support Structures and Interior Attachments to Reactor Vessels) and B-N-3 (Removable Core Support Structures). Visual inspections (VT-3) of the applicable components' accessible surfaces are required one time per interval. Relevant conditions requiring correction are described in ASME Section XI, IWB-3520.

Clevis insert bolts are addressed as an Existing Program Component in the Wolf Creek RVI AMP. The clevis inserts and fastening devices (bolts, pins and welded lock bars) are included in the Wolf Creek ASME Section XI Program as B-N-2 components. During ASME Section XI 10-year ISI exams, remote visual (VT-3) inspections are conducted using a submersible mini-submarine. The last ASME Section XI 10-year ISI inspections of the clevis inserts, bolts, pins, and welded lock bars were performed in 2005 (RF14). There were no indications of age related degradation observed. The Wolf Creek clevis insert bolts are constructed of X-750.

Wolf Creek has evaluated industry OE concerning failure of X-750 clevis insert bolts at another Westinghouse NSSS designed plant in 2010 [4, Appendix A] under its corrective action program (CAP). In response to this event, Wolf Creek performed an opportunistic remote visual (VT-3) inspection of the clevis inserts, bolts, pins, and welded lock bars in 2011 (RF18). Similar to the 2005 (RF14) inspection, there was no evidence of age related degradation observed. The next

scheduled ASME Section XI exam of the clevis inserts, bolts, pins, and welded lock bars is in 2016 (RF21).

The Wolf Creek clevis insert design differs from the Westinghouse NSSS designed plant that experienced the clevis insert bolt failure in 2010 but is similar to two other Westinghouse NSSS designed plants that have submitted requested evaluations to the NRC (Ref. ML14093A780 and ML15197A029). These evaluations demonstrated that structural adequacy of the clevis inserts would be maintained should failure of one or more X-750 clevis insert bolts occur. These evaluations also demonstrated that there would be no significant degradation of other mechanical components due to loose parts in the event of a clevis insert bolt or associated lock bar failure. Based upon these evaluations and the absence of any observed age related degradation to date, Wolf Creek concludes that the current inspection method and frequency of its ASME Section XI In-Service Inspection Program are adequate to maintain continued functionality of the clevis inserts during the initial and extended periods of operation.

CRGT Support Pin Replacement Program

The original Wolf Creek CRGT support pins were fabricated from X-750 alloy with a heat treatment (Rev. A) that was later determined to have rendered them susceptible to stress corrosion cracking based upon external OE. Replacement support pins, also manufactured of X-750 but with a modified heat treatment (Rev. B), were installed prior to commercial operation, per recommendations from Westinghouse. In 2002, Wolf Creek suffered a failure of an X-750 (Rev. B) support pin that was attributed to SCC. Based on recommendations from Westinghouse replacement support pins constructed of cold worked (CW) 316 SS were installed in 2003.

Prior to installation, the replacement CW 316 SS support pins were evaluated for resistance to the eight degradation mechanisms of concern for the RVI : 1) SCC; 2) IASCC; 3) fatigue; 4) IE; 5) TE; 6) wear; 7) void swelling; and 8) irradiation and thermal enhanced stress relaxation/creep. None of the eight degradation mechanisms were found to be a concern for the CW 316 SS material over a specified 60-year design life (90% capacity factor).

Wolf Creek performs a full core offload every refueling outage and a foreign object search and retrieval (FOSAR) inspection of the reactor vessel is conducted prior to reloading the core. A visual inspection is also conducted inside the steam generators' primary channel heads during outages when eddy current testing of the steam generator tubes is performed. These inspections should detect the presence of support pin fragments in the unlikely event that a future failure does occur.

The CRGTs and locking devices support pins are included in the Wolf Creek ISI Program as B-N-3 components, requiring a VT-3 inspection during 10 year ISI exams. These inspections provide a partial view of the support pin heads associated with the peripheral guide tube assemblies from the top of the upper core plate, and the support pin leaves associated with all guide tube assemblies from the bottom of the upper core plate.

The CRGT Support Pin Replacement Program is not listed in Table 2-3 since it does not include any augmented inspections of RVI components. However, routine inspections of the reactor vessel and steam generator bowls during normal refueling outages and 10 year ISI exams are considered adequate to detect age related degradation of the CRGT support pins. Furthermore, should additional recommendations for CRGT support pin replacement be issued by Westinghouse, Wolf Creek will take appropriate actions.

Flux Thimble Tubes.

The Wolf Creek Flux Thimble Tube Inspection Program performs wall thickness eddy current testing of all flux thimble tubes that form part of the reactor coolant system pressure boundary. The pressure boundary includes the length of the tube inside the reactor out to the seal fittings outside the reactor vessel. Eddy current testing is performed on the portion of the tubes inside the reactor vessel. The program implements the recommendations of NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors."

During each refueling outage, flux thimble tube wear is evaluated and inspections are performed based upon evaluation results. Wall thickness measurements are trended and wear rates are calculated. If the predicted wear (as a measure of percent through wall) for a given flux thimble tube is projected to exceed the established acceptance criteria prior to the next scheduled refueling outage, corrective actions are taken to reposition, cap or replace the tube. Program documentation maintains details regarding the core location, wear location and number of times a tube has been previously repositioned or replaced.

Conclusion:

Wolf Creek Existing Programs adequately manage the effects of aging for applicable components during the period of extended operation. No changes to the current programs are required.

LAI #4 B&W Core Support Structure Upper Flange Stress Relief

As discussed in Section 3.2.5.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval. ***This is Applicant/Licensee Action Item 4.***

Wolf Creek Response:

LAI #4 pertains to B&W Core Support Structure Upper Flange Stress Relief issue and is not applicable to Wolf Creek which is a Westinghouse NSSS design.

LAI #5 Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components

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

Wolf Creek Response:

The Wolf Creek hold down spring is constructed of 403 SS which is not susceptible to loss of preload due to stress relaxation. Therefore, LAI #5 is not applicable to Wolf Creek.

LAI #6 Evaluation of Inaccessible B&W Components

As addressed in Section 3.3.6 in this SE, MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques. Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval. ***This is Applicant/Licensee Action Item 6.***

Wolf Creek Response:

LAI #6 pertains to B&W Inaccessible Components and is not applicable to Wolf Creek which is a Westinghouse NSSS design.

LAI #7 Plant-Specific Evaluation of CASS Materials

As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. ***This is Applicant/Licensee Action Item 7.***

Wolf Creek Response:

Wolf Creek has identified only the lower internals assembly - BMI column cruciform as being constructed of CASS. MRP-191 screened all CASS BMI cruciforms in for TE without concern for their ferrite content. Wolf Creek conducted a search of the manufacturing records and located the certified material test reports for all 26 BMI column cruciforms. The cruciforms were statically cast and manufactured in accordance with SA-351, Grade CF8. Based upon the

guidance provided in NUREG/CR4513, the ferrite contents of the cruciforms were calculated using Hull's equivalent factors. The calculated ferrite contents for all 26 BMI column cruciforms were below the 20% threshold for TE Susceptibility of static castings as described in NRC letter of May 19, 2000, "License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*".

The BMI cruciforms also screened in for IE but were placed in the No Additional Measures Group by the FMECA described in MRP-191 and MRP-227-A. This placement indicates that the effects of IE, both singularly and synergistically with TE, will be adequately managed by the overall inspection strategy of MRP-227-A.

Wolf Creek has identified only the hold down spring as being constructed of martensitic 403 SS. The 403 SS hold down spring screened in for TE but was also placed in the No Additional Measures Group by the FMECA described in MRP-191 and MRP-227-A. This placement indicates that the effects of TE will be adequately managed by the overall inspection strategy of MRP-227-A.

Wolf Creek has no PH SS RVI components.

Conclusion:

Age related degradation of the Wolf Creek CASS and martensitic SS RVI components due to TE and IE will be adequately managed by the overall inspection strategy of MRP-227-A.

LAI #8 Submittal of Information for Staff Review and Approval

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

Wolf Creek Response:

During the license renewal process, Wolf Creek committed to submit an inspection plan for reactor internals to the NRC for review and approval not less than 24 months prior to entering the period of extended operation (March 12, 2025). This document serves as the Wolf Creek RVI Inspection Plan. It provides a summary of the Wolf Creek RVI AMP in Sections 1.0 to 3.0, and the responses to the eight Licensee Action Items in Section 4.0.

5.0 REFERENCES

1. NUREG-1915, Safety Evaluation Report Related to the License Renewal of Wolf Creek Generating Station.
2. Wolf Creek Generating Station License Renewal Application (ML062770308).
3. MRP-227 Rev. 0, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines."
4. MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines."
5. WCRE-27, Rev. 2, "Program Plan for Reactor Vessel Internals Inspection Program", Wolf Creek Generating Station.
6. MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals", (ML092750569).
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APPENDIX A

PWROG-16010-NP

Wolf Creek Summary Report for the Fuel Design/Fuel Management Assessment to Demonstrate MRP-227-A Applicability

PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG
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PWROG-16010-NP
Revision 0

WESTINGHOUSE NON-PROPRIETARY CLASS 3

Wolf Creek Summary Report for the Fuel Design / Fuel Management Assessment to Demonstrate MRP-227-A Applicability

Materials Committee

PA-MSC-0983, Revision 1, Task 7

March 2016

Westinghouse

PWROG-16010-NP
Revision 0

Wolf Creek Summary Report for the Fuel Design / Fuel Management Assessment to Demonstrate MRP-227-A Applicability

PA-MSC-0983, Revision 1, Task 7

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1 MRP 2013-025 GUIDELINES TO DEMONSTRATE MRP-227-A APPLICABILITY FOR WOLF CREEK REACTOR INTERNALS AGING MANAGEMENT FUEL DESIGN / FUEL MANAGEMENT ASSESSMENT

Applicant/Licensee Action Item 1 from the U.S. Nuclear Regulatory Commission (NRC) staff's final safety evaluation of MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," (Reference 1) states that:

> "*The Materials Reliability Program indicated that each applicant/licensee was responsible for assessing its plant's operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227, and each applicant/licensee shall describe the process used for determining plant-specific differences in the design of their reactor vessel integrity components or plant operating conditions, which result in different component inspection categories.*"

The NRC staff indicated (Reference 2) that the information provided by the industry to the NRC staff demonstrated that the MRP-227-A Inspection and Evaluation (I&E) Guidelines are applicable for the range of conditions expected at the currently operating Westinghouse and CE-designed plants in the United States. As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC's Request for Additional Information (RAI) to demonstrate compliance with MRP-227-A for originally licensed and uprated conditions was determined to be satisfied with plant-specific responses to the following two questions (References 2 and 3):

Question 1 *"Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)"*

Question 2 *"Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?"*

This report is concerned with Question 2 in the not yet received RAI regarding the reactor vessel internals aging management program for Wolf Creek.

1.1 WOLF CREEK EVALUATION FOR QUESTION 2

Westinghouse has evaluated the Wolf Creek reactor internals components with regard to fuel designs and fuel management according to industry guideline MRP 2013-025 (Reference 4) in Reference 5.

Wolf Creek has not utilized atypical fuel designs or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative, including power changes/uprates that have occurred over the operating lifetime of the unit. This conclusion is based on comparisons of the Wolf Creek core geometry and operating characteristics with the MRP-227-A applicability guidelines for Westinghouse-designed reactors specified in MRP 2013-025 (Reference 4).

Specifically, the following comparisons with the MRP-227-A applicability guidelines in MRP 2013-025 (Reference 4) were established for the key reactor internals components at Wolf Creek.

1.1.1 Components Located Beyond the Outer Radius of the Reactor Core

Guideline 1 - The reactor has been operated with out-in fuel management for 30 years or less and all future operation will use low-leakage fuel management.

Comparison - Wolf Creek initiated low-leakage fuel management strategy in the twelfth fuel cycle following 15.07 years of operation and has been implementing low-leakage core designs since that time. There are no current plans to return to out-in fuel management.

Guideline 2 - For operation going forward, the average power density of the reactor core (as defined in MRP 2013-025 (Reference 4)) shall not exceed 124 W/cm^3.

Comparison - For the last five operating fuel cycles (Cycles 16 through 20), Wolf Creek has been operating at a rated power level of 3565 MWt. For the 193 fuel assembly Wolf Creek core geometry, the 3565 MWt power level corresponds to a core power density of 109.21 W/cm^3. This level of power generation is also representative of anticipated future operation.

Guideline 3 - For operation going forward, the nuclear heat generation rate figure of merit (HGR-FOM) (as defined in MRP 2013-025 (Reference 4)) shall not exceed 68 W/cm^3.

Comparison - For the last five operating fuel cycles at Wolf Creek, the HGR-FOM at key baffle locations has ranged between [ ]^{a,c}. This range of HGR-FOM is representative of anticipated future operation.

1.1.2 Components Located Above the Reactor Core

Guideline 1 - Considering the entire operating lifetime of the reactor, the average power density of the core (as defined in MRP 2013-025 (Reference 4)) shall not exceed 124 W/cm³ for a period of more than two years.

Comparison - Over the operating lifetime of the Wolf Creek reactor, the rated core power level, including power uprates, has varied between 3411 MWt and 3565 MWt. This variation of rated power level corresponds to a power density range of 104.49 W/cm³ to 109.21 W/cm³.

Guideline 2 - Considering the entire operating lifetime of the reactor, the distance between the top of the active fuel stack and the bottom of the upper core plate (UCP) shall not be less than or equal to 12.2 inches for a period of more than two years.

Comparison - For the Wolf Creek reactor internals and fuel assembly geometry, the nominal distance between the top of the active fuel stack and the bottom of the UCP averaged over the first 20 fuel cycles of operation was [ ]^{a,c}. During that period of time, the nominal distance between the UCP and the top of the active fuel was not less than 12.2 inches for any fuel cycle.

1.1.3 Components Located Below the Reactor Core

Based on the discussion provided in MRP 2013-025 (Reference 4), plant-specific applicability of MRP-227-A for components located below the reactor core with no further evaluation required is demonstrated by meeting the MRP-227-A, Section 2.4 criteria.

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]a,c