United States Nuclear Regulatory Commission Official Hearing Exhibit

In the Matter of: Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3)

ASLBP #: 07-858-03-LR-BD01
Docket #: 05000247 | 05000286
Exhibit #: NYS000324-00-BD01
Identified: 10/15/2012
Admitted: 10/15/2012
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Rejected:
Stricken:
Other:

February 10, 2001

Mr. Roger A. Newton, Chairman
Westinghouse Owners Group
Wisconsin Electric Power Company
231 West Michigan
Milwaukee, Wisconsin 53201

SUBJECT: ACCEPTANCE FOR REFERENCING OF GENERIC LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLED, "LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR REACTOR INTERNALS", WCAP-14577, REVISION 1, OCTOBER 2000

Dear Mr. Newton:

The staff of the U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation has reviewed the topical report entitled, "License Renewal Evaluation: Aging Management for Reactor Internals," WCAP-14577, Revision 1, which the Westinghouse Owners Group (WOG) submitted in October 2000, as part of the Generic License Renewal Program (GLRP). The resultant final safety evaluation report (FSER) is transmitted to you as an enclosure to this letter.

As indicated in the FSER, the staff found the topical report acceptable for GLRP members' plants to reference in a license renewal application to the extent specified and under the limitations delineated in the staff FSER and the associated topical report. The limitations include committing to the accepted aging management programs defined in the topical report, and completing the renewal applicant action items described in Section 4.1 of the FSER. An applicant referencing the topical report and meeting these limitations will provide sufficient information for the staff to make a finding that there is reasonable assurance that the applicant will adequately manage the effects of aging so that the intended functions of the reactor internals covered by the scope of the report will be maintained consistent with the current licensing basis during the period of extended operation.

The staff does not intend to repeat its review of the matters described in the report and found acceptable in the FSER when the report appears as reference in a license renewal application, except to ensure that the material presented applies to the specified plant.

In accordance with the procedures established in NUREG-0390, "Topical Report Review Status," the staff requests that the WOG publish the accepted version of WCAP-14577 within three months after receiving this letter. In addition, the published version will incorporate this letter and the enclosed FSER between the title page and the abstract.

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To identify the version of the published topical report that was accepted by the staff, the staff requests the WOG include "-A" following the topical report number (e.g., WCAP-14577-A).

Sincerely,

/RA/

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 686

Enclosure: Final Safety Evaluation Report

cc w/encl: See next page

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To identify the version of the published topical report that was accepted by the staff, the staff requests the WOG include "-A" following the topical report number (e.g., WCAP-14577-A).

Sincerely,

/RA/

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
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Project No. 686

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FINAL SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF "LICENSE RENEWAL EVALUATION: AGING MANAGEMENT
FOR REACTOR INTERNALS"

WESTINGHOUSE OWNERS GROUP LIFE CYCLE MANAGEMENT/LICENSE RENEWAL
PROGRAM REPORT NO. WCAP-14577, REV. 1

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FINAL SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF "LICENSE RENEWAL EVALUATION: AGING MANAGEMENT
FOR REACTOR INTERNALS"

WESTINGHOUSE OWNERS GROUP LIFE CYCLE MANAGEMENT/LICENSE RENEWAL
PROGRAM REPORT NO. WCAP-14577, REV. 1

1.0 INTRODUCTION

Pursuant to Section 50.51 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.51), licenses to operate nuclear power plants are issued by the U.S. Nuclear Regulatory Commission (NRC) for a fixed period of time not to exceed 40 years. These licenses may be renewed by the NRC for a fixed period not to exceed 20 years beyond expiration of the current operating license term. The Commission's regulations in 10 CFR Part 54 published on May 8, 1995 (60 FR 22461), set forth the requirements for the renewal of operating licenses for commercial nuclear power plants (Ref.1).

Applicants for license renewal are required by the license renewal rule to perform an integrated plant assessment (IPA). The first step of the IPA, 10 CFR 54.21(a)(1), requires the applicant to identify and list structures and components that are subject to an aging management review (AMR); 10 CFR 54.21(a)(2) requires the applicant to describe and justify the methods used in meeting the requirements of 10 CFR 54.21(a)(1); and 10 CFR 54.21(a)(3) requires that, for each structure and component identified in 10 CFR 54.21(a)(1), the applicant demonstrate that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Furthermore, the applicant must provide an evaluation of time-limited aging analyses (TLAAs), as required by 10 CFR 54.21(c), and a list of TLAAs, as defined in 10 CFR 54.3.

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1.1 Westinghouse Owners Group Topical Report

By letter dated September 2, 1997, the Westinghouse Owners Group (WOG) submitted topical report WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals" (Ref. 2), for staff review and approval. This topical report is intended to provide a technical evaluation of the effects of aging of the reactor vessel internals (RVI) and demonstrate generically how aging management options maintain the intended functions of the RVI and how these options would remain effective during the extended period of operation. Applicants for renewed licenses are responsible for developing the aging management options into plant-specific programs for aging management. The topical report gives Westinghouse nuclear power plant utility owners the technical details related to aging management for the RVI that are necessary to develop and submit an application for license renewal.

1.2 Conduct of Staff Review

The staff reviewed the WOG topical report to determine whether the requirements set forth in 10 CFR 54.21(a)(3) and (c)(1) were met. The staff issued a request for additional information (RAI) after completing the initial review (Ref. 3), and the WOG responded to the staff's RAI (Ref. 4). After reviewing the RAI responses, the staff issued a draft safety evaluation (DSE) on the topical report (Ref. 5). Following the issuance of the DSE, the WOG representatives responded to the open items in the DSE (Ref. 6), including the submission of a line-in/line-out version of a draft of Revision 1 of WCAP-14577. Subsequently, Revision 1 of the topical report was submitted to the NRC (Ref. 7), in accordance with Open Item 4 of the DSE.

2.0 SUMMARY OF TOPICAL REPORT

The WOG topical report, WCAP-14577, contains a technical evaluation of aging degradation mechanisms and aging effects for Westinghouse RVI components. The WOG sent this report to the staff to demonstrate that WOG-member plant owners can adequately manage effects of aging during the period of extended operation, using approved aging management options to develop plant-specific aging management programs. This evaluation applies to all nuclear power plants with the Westinghouse nuclear steam supply system.

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The topical report identifies fatigue as the only TLAA of concern, as defined in 10 CFR 54.3, for the RVI. The topical report states that resolution of this TLAA is a plant-specific issue to be addressed in the license renewal application.

2.1 Components and Intended Functions

2.1.1 Intended Functions

Section 2.2 of the topical report describes the intended functions of the RVI at two levels. At a global level, the RVI support the following intended functions, consistent with 10 CFR 54.4(a):

- Provide the capability to shut down the reactor and maintain it in a safe shutdown condition.
- Prevent failure of all non-safety-related systems, structures, and components whose failure could prevent any of these functions.
- Ensure the integrity of the reactor coolant pressure boundary (bottom-mounted instrumentation flux thimbles only).

Revisions to the topical report in response to RAI #10 identify the following six intended functions for the individual subcomponents of the RVI, according to the requirements of 10 CFR 54.4(a):

- Provide support and orientation of the reactor core (i.e., the fuel assemblies).
- Provide support, orientation, guidance, and protection of the control rod assemblies.
- Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
- Provide a passageway for support, guidance, and protection for incore instrumentation.
- Provide a secondary core support for limiting the core support structure downward displacement.

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- Provide gamma and neutron shielding for the reactor pressure vessel.

2.1.2 Components

As described in the report, the reactor vessel internals (RVI) consist of two subassemblies within, but not welded to, the reactor pressure vessel (RPV). These subassemblies are the upper internals assembly (UIA) and the lower internals assembly (LIA). The topical report classifies the RVI components as either core support structures (CS) or core internals structures (IS), as defined in Subsection NG of Section III of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code. Core support structures are structures or parts of structures designed to directly support or restrain the core (fuel and blanket assemblies) inside the reactor pressure vessel. Internal structures are all RPV structures inside the RPV other than core support structures, fuel and blanket assemblies, control assemblies, and instrumentation.

Section 2.1 of the topical report lists the main components of the UIA and the LIA, and suggests classifications. This section also identifies components that interface between core supports and internals or between core supports and the RPV. These interface components are the upper core plate alignment pins, the internals holddown spring, head-to-vessel alignment pins, radial keys and clevis inserts, and driveline components (rod cluster control assemblies and driverods).

The scope of this report includes forged, cast, and rolled (plate) components, along with welds and threaded fasteners to join components.

Physical and functional descriptions of the individual items within the two subassemblies are given in Sections 2.3.1 and 2.3.2 of the report. A description of the design criteria for the RVI at Westinghouse plants are given in Sections 2.4.1 and 2.5.1 of the topical report.

2.2 Effects of Aging

Section 2.7 of the topical report discusses the aging degradation mechanisms and effects of aging applicable to the RVI for the period of extended operation. The topical report identifies the following aging effects that could result in adverse impact to or loss of any of the RVI intended functions:

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- fatigue-related cracking and crack growth for fatigue-sensitive components
- cracking and material degradation due to corrosion or stress corrosion cracking (SCC)
- cracking due to irradiation embrittlement and irradiation-assisted stress corrosion cracking (IASCC)
- reduction in fracture toughness resulting from thermal aging of austenitic stainless steel castings
- material wastage due to erosion and erosion/corrosion
- material loss caused by wear of interfacing components leading to loss of function
- reduction or loss of bolt preload because of creep or stress relaxation that leads to increased wear and fatigue usage

Section 3.1 of the topical report identifies the aging effects, and affected components, that require aging management, as determined by the WOG's evaluation. The evaluations include a review of industry operating experience to identify past incidents of aging effects applicable to the RVI (described in Section 2.6 of the topical report). Section 3.1.11 of the topical report also evaluates void swelling as a possible aging effect, concluding that there are no indications of discernible effects attributable to swelling.

Section 3.2 summarizes the aging mechanisms that require aging management for the license renewal period. These aging mechanisms are:

- irradiation embrittlement
- irradiation-assisted stress corrosion cracking
- irradiation creep and void swelling

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- stress relaxation

- wear

2.3 Aging Management Programs

Section 4 of the topical report describes the aging management options and the bases for demonstrating that the applicable aging effects identified in Section 3 of the topical report can be managed during the extended period of operation for Westinghouse plants. Tables 4-2 through 4-8 in the topical report provide summaries and program attributes of the activities that will manage aging effects applicable to each RVI component previously identified as requiring aging management.

For irradiation embrittlement, irradiation-assisted stress corrosion cracking, stress relaxation, and wear, Section 4.1 of the topical report states that current inspection activities suffice to adequately manage aging during the license renewal term. The effects of fatigue, and aging management of baffle/former and core barrel/former bolts are managed by additional activities described in Section 4.2 of the topical report.

2.4 Time-Limited Aging Analyses

Section 2.5 of the topical report identifies fatigue as the only TLAA applicable to the reactor vessel internals. Aging management of fatigue is described in Section 4.2.1 and Table 4-6 of the topical report.

3.0 STAFF EVALUATION

The staff reviewed the topical report and the additional information submitted by the WOG to determine if they demonstrated that the effects of aging of the RVI components covered by the report will be adequately managed so that the components' intended functions will be

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maintained consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(a)(3). This is the last step in the IPA described in 10 CFR 54.21(a).

Besides the IPA, Part 54 requires an evaluation of TLAAs in accordance with 10 CFR 54.21(c). The staff reviewed the topical report and additional information submitted by the WOG to determine if the TLAAs covered by the report were evaluated for license renewal in accordance with 10 CFR 54.21(c)(1).

To ensure applicability of the results and conclusions of WCAP-14577 to the applicant's plant(s), the license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant must commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel components. Applicants for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory controls. Any deviations from the aging management programs described in this topical report as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, must be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1). **This is Renewal Applicant Action Item 1.**

In accordance with 10 CFR 54.21(d), a summary description of the programs and activities for managing the effects of aging and an evaluation of TLAAs is to be provided in the license renewal final safety analysis report (FSAR) supplement. **This is Renewal Applicant Action Item 2.**

3.1 Intended Functions and Components

The staff reviewed Sections 1 and 2 of the topical report to determine whether there is reasonable assurance that the RVI components and supporting structures subject to AMR have been identified in accordance with the requirements of 10 CFR 54.21(a)(1). This was accomplished as described below.

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The topical report identifies the intended functions of the RVI (described in Section 2.1.1 of this evaluation). Consistent with 10 CFR 54.4(a), the topical report identifies global-level intended functions of the RVI. In response to RAI #10, as described in Section 2.1.1 of this evaluation, the WOG also identified component-level RVI intended functions and components fulfilling those intended functions. The staff found the list of intended functions to be complete and in accordance with 10 CFR 54.4(a).

As part of the evaluation, the staff determined whether the applicant had properly identified the systems, structures, and components within the scope of license renewal, pursuant to 10 CFR 54.4. The staff compared the information in the topical report with that for similar pressurized-water reactor (PWR) systems to assure that the list was complete and accurate. The staff found no omissions in the list of systems, structures and components, and, therefore, concludes that, except for the holddown spring and the guide tube support pins, there is reasonable assurance that the report has adequately identified those portions of the RVI and its associated supporting structures and components within the scope of license renewal and therefore subject to AMR, in accordance with 10 CFR Part 54.

Section 2.2 of the topical report states, "The reactor internals components listed in Table 2-1 that perform an intended function in a passive manner and which are long-lived are subject to an aging management review (see Table 2-2)." Tables 2-1 and 2-2 indicate that the holddown spring does not perform an intended function nor is subject to an aging management review, respectively. These findings are not internally consistent with other information provided in the topical report related to the purpose of the holddown spring, the safety significance of this purpose, and operating plant experience with and potential consequences of degradation of the holddown spring. In addition, the holddown spring is also included in an aging management program (AMP-4.2) for management of stress relaxation effects leading to wear and/or loss of preload leading to cracking.

Based on the topical report information provided, the staff concludes that the holddown spring performs an intended function in accordance with 10 CFR 54.4 and is subject to an aging management review and aging management during the extended plant operation period. For the holddown spring, applicants for license renewal are expected to address intended function,

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aging management review, and appropriate aging management program(s). **This is Renewal Applicant Action Item 3**.

Regarding the guide tube support pins, the topical report documents a history of degradation of the original pins and a pin design with a revised heat treatment. Based on the potential for guide tube support pin failure preventing other components from performing their intended safety function, aging management review, and appropriate aging management program(s), for guide tube support pins should be addressed by license renewal applicants. **This is Renewal Applicant Action Item 4**.

The descriptions of the components in the topical report generally do not specify the materials used to fabricate the components, and assumed in the aging management review. Renewal applicants should explicitly identify the materials of fabrication for each of the components within the scope of the topical report, and the applicable aging effect should be reviewed for each component based on the materials of fabrication and the environment. **This is Renewal Applicant Action Item 5.**

3.2 Aging Mechanisms

The aging mechanisms identified in WCAP-14577 are neutron irradiation embrittlement, stress corrosion cracking, irradiation-assisted stress corrosion cracking (IASCC), erosion and corrosion processes (including erosion/corrosion), creep/irradiation creep, stress relaxation, wear, thermal aging, corrosion, fatigue, and void swelling. The WOG reviewed these aging mechanisms for applicability to the RVI assemblies within the scope of the report. The WOG reviewed the operational history of RVI components. The WOG findings about these aging-degradation mechanisms and aging effects were incorporated into the aging management programs.

3.2.1 Neutron Irradiation Embrittlement

Determination of RVI components subject to neutron irradiation embrittlement was handled in the topical report using a fluence threshold to screen out components with a neutron fluence below 1×10^{21} n/cm^2 (E > 0.1 MeV). The components found to be subject to neutron irradiation

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embrittlement are the lower core barrel, the baffle/former assembly, the baffle/former bolts, the lower core-plate and the fuel pins, the lower support forging and the clevis bolts. The NRC staff does not agree that 1×10^{21} n/cm^2 should be the threshold fluence for screening components. However, the proposed aging management program does address the components with the highest fluences and hence most susceptible to this mechanism, so the threshold fluence approach does not affect the results of this review.

3.2.2 Stress Corrosion Cracking (SCC)

The topical report identifies cracking as a potential aging effect because of either SCC or IASCC. SCC results from the synergistic effects of tensile stresses and a corrosive environment on a susceptible material. The synergism is specific to the material. The material may be inherently susceptible or become sensitized during fabrication. The tensile stresses can be due to operational loading or residual fabrication stresses. The environmental parameters considered to be critical in SCC are dissolved oxygen, halide, and sulfide. In IASCC, neutron irradiation can make the material more susceptible to SCC.

For SCC, the report uses reactor coolant chemistry control, in particular control of dissolved oxygen, chlorides, and other halogens, as the basis for generally ruling out SCC as potentially significant for all components of the RVI. The staff does not agree with this assessment, particularly given the potential for occluded environmental conditions in the crevice areas typically associated with bolting. However, the aging management activities for bolting and IASCC discussed in Section 3.3.1 of this DSER suffice to address aging effects due to stress corrosion cracking.

3.2.3 Irradiation-Assisted Stress Corrosion Cracking (IASCC)

For IASCC, the report uses a neutron fluence threshold of 1×10^{21} n/cm^2 ($E > 0.1$ MeV) to determine susceptibility to IASCC. The NRC staff does not agree that 1×10^{21} n/cm^2 should be the threshold fluence for a component to be susceptible to IASCC; however, the proposed aging management program obviates the need for threshold fluence consideration (see Section 3.3). The list of RVI components determined to be susceptible to IASCC is consistent with the list of components susceptible to neutron embrittlement, in particular the lower core barrel, the

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baffle/former assembly, the baffle/former bolts, baffle/baffle bolts, the lower core plate and the fuel pins, and the lower support forging and the clevis bolts. Of these components, the baffle former and baffle/baffle bolts are expected to be the first to exhibit IASCC, because of their high neutron fluence and high stress.

Baffle/Former Bolt Cracking

Section 2.6.2 of the topical report reviews the historical performance of threaded and pinned fasteners in the reactor vessel internals (RVI) to identify and assess past incidents of aging effects applicable to these fasteners. The assessment is broad, covering operational experience for all PWR designs in the U.S. and overseas experience. There is at least one internal inconsistency in the report, as one part of Section 2.6.2 asserts that there have been no historical incidents of degradation of the baffle-former bolts in domestic PWRs. However, a subsequent discussion in this same section of the topical report describes degradation found in several domestic WOG plants. A more robust discussion of this domestic experience is provided in the WOG response to RAI #4 (Ref. 4). Management of baffle-former bolt cracking is described in AMP-4.6 of the topical report.

3.2.4 Erosion and Corrosion Processes

The topical report cites several possible mechanisms for loss of material by erosion and corrosion. These mechanisms are (1) erosion and erosion/corrosion, (2) uniform attack/general corrosion, (3) pitting corrosion, (4) crevice corrosion, and (5) intergranular corrosion attack.

Erosion and erosion/corrosion are not considered to be applicable since all of the RVI components are fabricated from stainless steel or nickel-based alloys, and these materials have been found to be resistant to erosion and erosion/corrosion in a PWR environment.

Uniform attack and general corrosion, pitting corrosion, and crevice corrosion are not considered to be applicable since all of the RVI components are fabricated from stainless steel or nickel-based alloys, and these materials have been found to be resistant due to the formation of protective passivation layers.

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Intergranular corrosion attack is not considered applicable because of the low oxygen levels in the reactor coolant as a result of water chemistry controls, along with halogen and sulfate controls.

Therefore, loss of material due to various erosion and corrosion processes is not considered to be an applicable aging effect for any of the RVI components.

3.2.5 Creep/Irradiation Creep

Creep is a plastic deformation that occurs over time in a material subjected to a stress below the elastic limit at a significantly elevated temperature. For stainless steel alloys and nickel-based alloys, creep is not a concern at PWR conditions with temperatures below 1000°F.

The topical report indicates that irradiation creep can be caused by defects that result from exposure to a neutron flux. The staff concurs with this conclusion. Besides stress relaxation, irradiation creep in the baffle plates could result in increased loads on the baffle/former and core barrel/former bolts, possibly making the bolts more likely to crack. Management of irradiation creep is addressed in AMP-4.6 and AMP-4.7.

3.2.6 Stress Relaxation

The topical report defines the stress relaxation mechanism as the unloading of preloaded components under conditions of long-term exposure of RVI materials to high constant strain, elevated temperatures, and/or neutron irradiation. The thermal effect is predominant at temperatures well above RCS operating temperatures; however, fast-neutron irradiation can induce stress relaxation at normal operating temperatures.

The topical report indicates that stress relaxation has significance only for substantially preloaded RVI components, such as springs and bolts, because these components cannot perform their functions without maintaining an adequate preload.

If the springs interfacing RVI components and the bolts connecting RVI components are subjected to stress relaxation, then the loss of the preload on the components will diminish RVI

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structural rigidity. As a result of the diminished rigidity, the RVI may experience increased vibration and loading, resulting in loss of function. The combination of bolt stress relaxation, changes in transient and high-cycle vibration of the RVI, and effects of increased RVI fatigue susceptibility may be significant and require further aging management during the license renewal period.

Section 4.1.2 of the topical report indicates that a loss of preload could result in higher cyclic and transient loads and an increased fatigue susceptibility. The RVI components that are affected by these stress relaxation effects, listed in Table 4-3 of the topical report, are the upper and lower support column bolts, the holddown spring, and the clevis insert bolts. Management of stress relaxation is addressed by AMP-4.2.

3.2.7 Wear

Wear of reactor vessel internals (RVI) components occurs due to relative motion between the interfaces and mating surfaces of components as a result of flow-induced vibration during plant operation, differential thermal expansion and contraction movements during plant heatup and cooldown and changes in power operating cycles. The severity of the wear depends upon the frequency of motion, duration, and the loads on the surfaces.

Section 3.2.7 of the topical report states: "The effects of wear are potentially significant at the interfaces of components having relative motion. Further evaluation of these components is provided in Section 4.1.3, AMP-4.3 and AMP-4.4. For all other (RVI) components, wear is nonsignificant and aging management of this effect will not be required during the extended period of operation."

3.2.8 Thermal Aging

The topical report identifies thermal aging in RVI components as an applicable aging effect. Thermal aging embrittlement can occur in cast austenitic stainless steel (CASS) exposed to high temperatures typical of reactor operating conditions. Neutron irradiation embrittlement occurs in all steels exposed to the high neutron flux conditions typical of many RVI components.

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Both of these mechanisms increase the hardness and tensile strength and reduce the ductility, impact strength, and fracture toughness of the material.

For RVI components fabricated from CASS and hence subject to thermal aging and neutron embrittlement, concurrent exposure to high neutron fluence levels can have synergistic effects whereby the service-degraded fracture toughness is reduced from the levels predicted independently for either of the mechanisms. Therefore, for components determined to be subject to thermal aging embrittlement, the license renewal applicant should also consider the neutron fluence of the component to determine the full range of degradation mechanisms applicable for the component.

The WOG response to RAI #7 (Ref. 4) states that the neutron fluence of the lower core support casting will be less than 1 dpa (~ 7×10^{20} n/cm^2) at the end of the license renewal period. With this low of a fluence level, the response states that neutron irradiation embrittlement should not be a concern for castings found acceptable by the screening criteria in EPRI-TR-106092 (Ref. 8), as supplemented by the additional criteria of RAI #7. The WOG response to RAI #7 (Ref. 4) states that the lower core support casting "will be evaluated in accordance with the guidelines of [EPRI] TR-106092 as modified according to the additional criteria listed in RAI #7." It is not clear whether this evaluation is generic in nature or intended as a plant-specific determination. With revisions to Section 3.2.8.2 of the topical report stating that evaluations of cast internals components demonstrate that the effects of thermal aging are not significant and an evaluation or an aging management program for this effect will not be required during an extended period of operation, the staff finds no support for this conclusion provided in the report.

The staff's concern is that the screening criteria for thermal aging susceptibility do not account for the neutron fluence level of the component and that synergistic degradation from neutron fluence and thermal aging could reduce the fracture toughness such that an aging management program is required. License renewal applicants should describe their aging management plans for cast austenitic stainless steel components during the license renewal period. **This is Renewal Applicant Action Item 6.**

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3.2.9 Fatigue

The staff's evaluation of fatigue is described in Section 3.4 of this report.

3.2.10 Void Swelling

Section 3.1.11 of the topical report says that void swelling is not a significant aging mechanism, and dismisses the aging effect of change of dimensions, for lack of evidence of void swelling under PWR conditions. However, EPRI TR-107521 (Ref. 9) cites several sources with conflicting results. One source predicts swelling as great as 14% for PWR baffle-former assemblies over a 40-year plant lifetime, whereas results from another source indicate that swelling would be less than 3% for the most highly irradiated sections of the internals at 60 years. The issue of concern to the staff is the impact of change of dimension due to void swelling on the ability of the RVI to perform their intended function. The topical report also cites ongoing work to develop an industry position on void swelling by considering the accumulated data, engineering evaluations of the ramifications of swelling, and field observations with this work scheduled to be completed in 2001. Until industry has developed sufficient data to demonstrate void swelling is not a significant aging mechanism, the staff believes that void swelling should be considered significant, and license renewal applicants should describe their aging management plans for void swelling for the license renewal period. **This is Renewal Applicant Action Item 7.**

3.2.11 Summary

The staff agrees with the WOG identification of applicable RV component aging effects that are subject to aging management as a condition of license renewal. However, the staff finds that license renewal applicants should consider the aging effects of (1) loss of fracture toughness due to synergistic effects of thermal aging and irradiation embrittlement of cast austenitic stainless steel components, (2) change of dimensions due to void swelling and provide their aging management plans for the license renewal period., (3) loss of preload of the holddown spring due to stress relaxation, and (4) cracking of guide tube support pins.

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3.3 Aging Management Programs

As described in Section 2.3, the aging management programs discussed by the WOG include current activities and new activities. Table 4-1 of the report lists the six basic attributes of the existing and additional AMPs. These attributes are the scope of the program, the surveillance techniques used to detect aging effects, the frequency of the surveillance, the acceptance criteria to determine when corrective actions are necessary, the corrective actions, and confirmation techniques. Section 4 of the topical report gives the program attributes of the AMPs, leaving the plant-specific details of the AMPs to the applicant to describe in the license renewal application. The staff has determined that there are 10 elements that should be addressed by the applicant in aging management programs. These are (1) scope of the program, (2) preventive actions, (3) parameters monitored or inspected (4) detection of aging effects (5) monitoring and trending (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience. The license renewal applicant must describe how each plant-specific AMP addresses these elements. **This is Renewal Applicant Action Item 8.**

3.3.1 AMP-4.1: Irradiation Embrittlement and Irradiation-Assisted Stress Corrosion Cracking

The topical report links management of neutron irradiation embrittlement to management of cracking caused by IASCC. This linkage is appropriate because the effects of neutron irradiation embrittlement, especially loss of fracture toughness, make existing cracks in the affected materials and components less resistant to growth. However, the report incorrectly cites “cracking” as the aging effect for neutron irradiation embrittlement; irradiation embrittlement does not cause cracking, but can cause smaller cracks to become critical under design basis loads.

This AMP covers the components with the highest neutron fluence levels, including the core barrel in the active fuel region, the baffle and former plates, the lower core plate, the fuel pins in the lower core plate, and the lower support forging (for plants with a 14 foot core). The baffle/former and core barrel/former bolting is exposed to high neutron fluence levels, with a

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potential for loss of fracture toughness, and has an increased propensity for cracking due to occluded environment conditions; this bolting is covered by other aging management activities (AMP-4.6 and AMP-4.7).

The description of AMP-4.1 covers the existing inservice inspection (ISI) program, which requires visual VT-3 examination to Examination Category B-N-3 of Section XI of the ASME Code, along with proposed requirements on inservice inspection of core support and internal structures (Subsection IWG). The topical report and the WOG response to RAI #6 are not clear regarding the surveillance techniques to be used during the license renewal period. For example, Table 4-2 of the topical report indicates supplemental examination (e.g., ultrasonic examination), but the text states that the ASME Section XI requirements are sufficient to manage the effects of irradiation embrittlement and IASCC "as supplemented when relevant conditions are detected." Since detection of relevant conditions is from visual VT-3 examination, it is not clear as to how/when supplemental examination would be employed. This is important because the visual VT-3 examination required by Examination Category B-N-3 may not be adequate to detect cracking of the susceptible RVI components. The examination technique ultimately used must demonstrate the capability to detect the types of cracking expected to occur in the RVI.

Reference to proposed requirements in a proposed Subsection IWG of Section XI of the ASME Code does not provide a document reviewable by the staff, since the details of the proposed requirements are not described in the topical report, and the requirements could undergo substantial modification before acceptance in the Code.

Since the staff concludes that augmented inspection is warranted for cracking (and loss of fracture toughness), each renewal applicant must address the plant-specific plans for management of cracking (and loss of fracture toughness) of RVI components, including any plans for augmented inspection activities. **This is Renewal Applicant Action Item 9.**

3.3.2 AMP-4.2: Stress Relaxation

As described in the topical report for AMP-4.2, including loose parts and neutron noise monitoring, the aging management activities and program attributes for the effects of stress

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relaxation on the RVI components rely on visual examination (VT-3) and other augmented methods of examination. The examination, frequency, acceptance criteria, and corrective actions are in accordance with the appropriate ASME Code requirements listed in Table 4-3 and described in Section 4.1.2 of the topical report.

The topical report states that for some cases, such as baffle/former and barrel/former assembly bolts, the applicable requirements of Section XI of the ASME Code may not be sufficient to detect significant stress relaxation and loss of preload. In these cases, aging management of stress relaxation is addressed by AMP-4.6 and AMP-4.7.

The staff reviewed the topical report to determine if it demonstrates that the stress relaxation effects of aging on the RVI components will be adequately managed. On the basis of its review the staff concludes that the aging effects of stress relaxation on RVI components, except for baffle/former and barrel/former assembly bolts, will be adequately managed during the extended period of operation by AMP-4.2. The staff's findings on aging management of stress relaxation for baffle/former and barrel/former assembly bolts are discussed in the staff evaluation of AMP-4.6 and AMP-4.7.

3.3.3 AMP-4.3 and AMP-4.4: Wear

The aging management activities and programs for wear described in the topical report address two different groups of RVI components. The bottom-mounted instrumented (BMI) flux thimbles are addressed in Table 4-4 of AMP-4.3. The aging management activities and attributes for wear of the upper-core-plate alignment pins and radial keys and the clevis inserts are addressed in Table 4-5 of AMP-4.4.

AMP-4.3 for the BMI flux thimble tubes lists ultrasonic and eddy current examination surveillance techniques performed at intervals and by acceptance criteria consistent with the licensee's commitments in responding to NRC Bulletin No. 88-09, "Thimble Tube Thinning in Westinghouse Reactors."

The AMP-4.4 wear program for the other components relies on visual and other augmented methods of examination. The examination methods, frequency, acceptance criteria, and

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corrective actions described in the topical report are consistent with the appropriate ASME Code requirements.

The staff reviewed the topical report to determine if it demonstrated that the wear effects of aging on the RVI will be adequately managed. On the basis of its review, the staff concludes that the aging effects of wear on the RVI will be adequately managed during the extended period of operation by AMP-4.3 and AMP-4.4, because as indicated above that, 1) AMP-4.3 requires that the plants that have experienced BMI flux tube thimble thinning will continue with the inspections methods and frequency, corrective actions and acceptance criteria commitments made to the NRC in response to NRC I&E Bulletin 88-09 and 2) AMP-4.4, wear effects on other RVI components, requires surveillance techniques and frequency, corrective actions and acceptance criteria that are acceptable to NRC and consistent with the ASME B&PV Code, Section XI, Subsection IWB requirements.

3.3.4 AMP-4.5: Fatigue

This aging management program is evaluated in Section 3.4 of this staff evaluation.

3.3.5 AMP-4.6 and AMP-4.7: Baffle/Former and Barrel/Former Bolts

These two AMPs manage similar aging effects for bolts in the same assembly and with similar intended functions. The aging mechanisms that these AMPs address are neutron irradiation embrittlement, IASCC, stress relaxation, irradiation creep, void swelling, and fatigue-related cracking.

Section 4.2.2 of the topical report describes the various surveillance techniques that will identify the effects of aging on the baffle/former and core barrel/former bolts. These include (1) visual VT-3 examination in accordance with Category B-N-3 of Subsection IWB of Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code, (2) visual examination in accordance with draft Subsection IWG of loose parts monitoring system and (3) the chemistry reactor coolant detection system (for baffle/former bolts), and augmented inspections. The topical report acknowledges that the VT-3 examinations will not detect cracking in these bolts since the industry experience is that the cracking occurs under the head of the bolt, an area that is not

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accessible for visual examination. In addition, loose parts monitoring and coolant reactivity monitoring are effective only after the aging effects have begun to manifest themselves in potentially serious ways (e.g., generation of loose parts and possible damage to fuel).

Therefore, augmented inspections, such as ultrasonic inspections, are proposed to provide effective management of the effects of aging on the baffle/former and core barrel/former bolts. The topical report states that details of these augmented inspections will be provided in the plant-specific license renewal applications. **This is Renewal Applicant Action Item 10.**

3.4 Time-Limited Aging Analyses

Time-limited aging analyses are defined in 10 CFR 54.3 as those licensee calculations and analyses that:

(1) involve systems, structures, and components within the scope of license renewal, as stated in 10 CFR 50.54(a);

(2) consider the effects of aging;

(3) involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) have been determined to be relevant by the licensee in making a safety determination;

(5) involve conclusions or provide the bases for conclusions about the capability of the system, structure, or component to perform its intended functions, as stated in 10 CFR 50.54(b); and

(6) are contained or incorporated by reference in the current licensing basis.

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Paragraph 54.21(c)(1) requires the applicant to demonstrate that:

(i) the analyses remain valid for the period of extended operation;

(ii) the analyses have been projected to the end of the period of extended operation; or

(iii) the effects of aging on the intended functions(s) will be adequately managed for the period of extended operation.

Based on the description of the engineering and design of the reactor vessel internals (RVI), Section 2.5 of the topical report concludes that fatigue is the only TLAA meeting all six of the TLAA criteria of 10 CFR 54.3.

3.4.1 Fatigue

Section 2.5.1 of the topical report describes fatigue as the structural deterioration resulting from repeated stress/strain cycles due to fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, damage can accumulate, initiating a crack in highly affected locations. Subsequent mechanical or thermal cyclic loading can cause the crack to grow.

Section 2.5.1 of the topical report states that the design bases for many RVI components contained fatigue evaluations, for RVI components designed to the ASME B&PV Code, Section III, Subsection NG, 1974 Edition, and earlier RVI components designed to the ASME Code as a guide for design. Only plants designed after the incorporation of the Subsection NG, 1974 Edition (i.e, Callaway, Wolf Creek, and South Texas Units 1 and 2) have complete fatigue analyses of RVI component low-cycle and high-cycle fatigue usage documented in a Code-required plant-specific "ASME Stress Report." All other domestic WOG plants were designed before the incorporation of the 1974 Edition of Subsection NG, and therefore do not have a plant-specific "ASME Stress Report."

3.4.2 Fatigue Evaluation

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RAI #5 (Ref. 3) requested a list of the TLAAs used to identify the fatigue-sensitive RVI components in Table 3-3 of the topical report, along with a summary description of each analysis and clarification of whether the identified fatigue-sensitive components apply to all Westinghouse-designed RVI. Based on its response to RAI #5 (Ref. 4), WOG indicates in Section 3.1.10 of the revised report that Section 2.5 identifies fatigue as the only TLAA related to the reactor internals. Section 3.1.10 also provides the overall approach that licensees will take in addressing the fatigue TLAA for the reactor vessel internals. If the TLAA cannot be dispositioned analytically, options are presented in Section 4.0 to manage the identified aging effects.

As described in the response to RAI #5 (Ref. 4), further modifications to the topical report provided in Section 3.1.10 include an extensive discussion of the conservatism in current analysis methods to better reflect the significant changes in the current thinking relative to evaluating the TLAA.

3.4.3 Staff Evaluation of AMP-4.5

The staff reviewed the topical report, the RAI responses and topical report modifications submitted by WOG to determine if they demonstrate that fatigue effects of aging of the RVI components will be adequately managed and if they require a fatigue-related TLAA to be performed and evaluated for license renewal applications in accordance with 10 CFR 54.21. On the basis of its review of the information provided with regard to the suggested overall approach as described in Section 4.2.1 of the topical report that licensees will adopt in addressing the fatigue TLAA for the reactor vessel internals, the staff concludes that (1) the aging effects of fatigue will be adequately managed, and (2) although the fatigue calculations needed for the TLAA have not been performed and/or have not been updated by WOG to reflect operations during the license renewal period, the screening process and methodology presented are acceptable for licensees' use in preparing plant-specific fatigue TLAA evaluations to support license renewal applications. The plant-specific requirements of 10 CFR 54.3 and 10 CFR 54.21(c)(1) for fatigue TLAAs must be addressed by the license renewal applicant. **This is Renewal Applicant Action Item 11.**

4.0 CONCLUSIONS

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The staff has reviewed the subject WOG topical report (Ref. 2) and additional information submitted by the WOG. On the basis of its review, upon satisfactory completion of the renewal applicant action items identified below in Section 4.1, the staff concludes that the WOG topical report provides an acceptable demonstration that the applicable effects of aging on reactor vessel internals components within the scope of this topical report will be adequately managed for the WOG plants, such that there is reasonable assurance that the RVI components will perform their intended functions in accordance with the CLB during the period of extended operation. The staff also concludes that, upon completion of the renewal applicant action items listed below in Section 4.1, the WOG topical report will provide an acceptable evaluation methodology of time-limited aging analyses for the reactor vessel internals within the scope of this report for the WOG plants during the period of extended operation.

Any WOG plant may reference this topical report in a license renewal application to satisfy the requirements of (1) 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the reactor vessel internals components within the scope of this topical report will be adequately managed and (2) 10 CFR 54.21(c)(1) for demonstrating the appropriate findings in the evaluation of time-limited aging analyses for the reactor vessel internals during the period of extended operation. Upon completion of the renewal applicant action items listed below in Section 4.1, a license renewal applicant that references this topical report in a license renewal application will be expected to provide a summary of aging management programs and TLAA evaluations in an FSAR Supplement, sufficiently detailed to enable the staff to make the necessary findings required by Sections 54.29(a)(1) and (a)(2) for components within the scope of this topical report.

4.1 Renewal Applicant Action Items

The following are license renewal applicant action items to be addressed in the plant-specific license renewal application when incorporating WOG topical report WCAP-14577 (Rev. 1) in a renewal application:

(1) To ensure applicability of the results and conclusions of WCAP-14577 to the applicant's plant(s), the license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant must commit to

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programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel components. Applicants for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory controls. Any deviations from the aging management programs described in this topical report as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, must be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

(2) A summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs must be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).

(3) For the holddown spring, applicants for license renewal are expected to address intended function, aging management review, and appropriate aging management program(s).

(4) The license renewal applicant must address aging management review, and appropriate aging management program(s), for guide tube support pins.

(5) The license renewal applicant must explicitly identify the materials of fabrication of each of the components within the scope of the topical report. The applicable aging effect should be reviewed for each component based on the materials of fabrication and the environment.

(6) The license renewal applicant must describe its aging management plans for loss of fracture toughness in cast austenitic stainless steel RVI components, considering the synergistic effects of thermal aging and neutron irradiation embrittlement in reducing the fracture toughness of these components.

(7) The license renewal applicant must describe its aging management plans for void swelling during the license renewal period.

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(8) Applicants for license renewal must describe how each plant-specific AMP addresses the following elements: (1) scope of the program, (2) preventative actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.

(9) The license renewal applicant must address plant-specific plans for management of cracking (and loss of fracture toughness) of RVI components, including any plans for augmented inspection activities.

(10) The license renewal applicant must address plant-specific plans for management of age-related degradation of baffle/former and barrel/former bolting, including any plans for augmented inspection activities.

(11) The license renewal applicant must address the TLAA of fatigue on a plant-specific basis.

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5.0 REFERENCES

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," 60 Fed Reg 22461 (1995).

2. WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," Westinghouse Owners Group, June 1997.

3. Letter from Raj K. Anand (NRC) to Roger A. Newton (WOG) dated June 14, 1999, "Request for Additional Information Regarding the Westinghouse Owners Group Generic License Renewal Program Topical Report Entitled 'License Renewal Evaluation: Aging Management for Reactor Internals,' WCAP-14577, June 1997."

4. Letter from Roger A. Newton (WOG) to Raj K. Anand (NRC) dated November 24, 1999, "Westinghouse Owners Group Response to NRC Request for Additional Information on WOG Generic Technical Reports: WCAP-14577,'License Renewal Evaluation: Aging Management for Reactor Internals,' (MUHP6110)."

5. Letter from Christopher I. Grimes (NRC) to Roger A. Newton (WOG) dated September 8, 2000, "Draft Safety Evaluation of Westinghouse Owners Group Topical Report 'License Renewal Evaluation: Aging Management for Reactor Internals,' WCAP-14577, Revision 0, June 1997."

6. Letter from Roger A. Newton (WOG) to Christopher I. Grimes (NRC) dated October 9, 2000, "Westinghouse Owners Group Transmittal of WCAP-14577, Revision 1,'License Renewal Evaluation: Aging Management of Internals,' October 2000, Showing Text Modification Identified in the WOG Responses to NRC RAIs."

7. WCAP-14577, Revision 1, "License Renewal Evaluation: Aging Management for Reactor Internals," Westinghouse Owners Group, October 2000.

8. EPRI Technical Report TR-106092, "Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components in LWR Reactor Coolant Systems," Electric Power Research Institute, September 1997.

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9. EPRI Technical Report TR-107521, "Generic License Renewal Technical Issues Summary," Electric Power Research Institute, April 1998.

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