ELECTRIC POWER RESEARCH INSTITUTE

Materials Reliability Program: Pressurized Water Reactor Internals Aging Management Program Development Template (MRP-342)

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ABSTRACT

The Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) has completed and published guidance for managing the effects of aging degradation in pressurized water reactor (PWR) internals. The initial version of this report, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 0)*, was submitted to the staff of the U. S. Nuclear Regulatory Commission (NRC) on January 12, 2009, for formal review and comment and received its final Safety Evaluation (SE) on June 22, 2011.

The initial version of this approved guidance was published by EPRI as *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)* in December 2011. In addition, the essential elements of MRP-227-A that define an aging management program (AMP) in support of a PWR License Renewal Application (LRA) have been included in Revision 2 of NUREG-1801, the Generic Aging Lessons Learned (GALL) report, published by the NRC in December 2010, as AMP XI.M16A (PWR Vessel Internals).

This report uses the approved version of the industry guidance (MRP-227-A) and NUREG-1801, Revision 2 AMP XI.M16A to develop a template for creating a PWR reactor vessel internals AMP. For those utilities with an existing AMP for PWR reactor vessel internals, this template can also provide the basis for either a revision of that AMP or for the development of an inspection plan to implement that AMP.

Keywords
Aging management program
Degradation
License renewal application
Materials Reliability Program
PWR

CONTENTS

1 INTRODUCTION

U. S. Pressurized Water Reactor (PWR) licensees that apply for license renewal to operate their units beyond the original 40-year license commit to the submittal for approval by the U. S. Nuclear Regulatory Commission (NRC) of a program for managing the effects of aging of reactor internals. To assist in this regard, the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) has developed and published *Materials Reliability Program: Pressurized Water Reactor Inspection and Evaluation Guidelines (MRP-227, Revision 0*, *and MRP-227-A)* [1a, 1b], along with a supporting document, *Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228, Revision 0)* [2].

These guidelines were submitted to the NRC for generic approval in January 2009, and received approval from the NRC Staff in the form of a Final Safety Evaluation (SE) in June and December 2011 [3a, 3b]. In addition to the approval of the industry guidance, the Staff imposed two sets of additional requirements – referred to as Licensee/Applicant Actions Items (to be performed by each licensee or applicant adopting some or all of the industry guidance) and Topical Report Conditions (to be included in the revised version of the industry guidance, MRP-227-A). Reference 3 also requested that the revised industry guidance be designated as MRP-227-A, and that this revised version be published within three months of the receipt of the final SE. In addition, shortly after issuing the final SE, the NRC Staff issued a Regulatory Issue Summary [4] describing the procedures by which the industry guidance should be implemented in the license renewal process. The required procedures depend upon the status of a particular license renewal application and, if that application has been approved, the status of the submittal of an inspection plan to the Staff at least 24 months prior to entering the period of extended operation. Further, the Staff has issued draft Interim Staff Guidance (ISG) in March 2012 [7] that, while still under review as the result of extensive industry comments, must be considered to some extent in any licensee submittals.

It should be noted that the industry guidance embodied in MRP-227, Revision 0, and in its successor documents, MRP-227-A et seq., were not intended to contain the complete set of information necessary to support the preparation of licensee renewal AMP submittal. Nevertheless, a large portion of the necessary information covering at least six of the ten AMP attributes (see NUREG-1800, Revision 2 [5]) is included, and the remainder of the information for three of the four remaining attributes is now available in Revision 2 of NUREG-1801 [6], in the form of a recommended AMP for PWR Reactor Internals designated AMP XI.M16A. Finally, Topical Report Condition 7 in the final SE required MRP-227-A to include an Appendix A that not only referenced AMP XI.M16A in Reference 6, but also included a comprehensive Operating Experience Summary that addressed the tenth AMP attribute. The information contained in that Appendix A is also summarized in Section 4 of this document for convenience.

In view of the relative complexity of dealing with the information contained in the revised MRP-227, including its new Appendix A and the other six Topical Report Conditions, the additional requirements imposed by the final SE in the form of Licensee/Applicant Action Items, and the RIS, the industry has chosen to prepare a PWR Reactor Internals Aging Management Program template. The purpose of this template is to provide a road map for assimilating the sources of information into directed license renewal documents, such as an AMP or an inspection plan. This template draws heavily on recent work performed by the EPRI MRP in support of the technical review of MRP-227 by the NRC staff, support that has involved assisting the staff in revising two documents that provide the spine of the license renewal and implementation process – NUREG-1800 [5], the Standard Review Plan used by the NRC staff for reviewing license renewal applications, and NUREG-1801 [6], the Generic Aging Lessons Learned (GALL) report, which contains (among other information) the templates for aging management reviews and the descriptions of acceptable aging management programs. Revision 1 of each of these documents did not completely address the aging management review and aging management program development processes, since the results of the industry efforts to address these topics have only recently achieved the necessary level of maturity. However, the intent was to incorporate the industry efforts into Revision 2 of each document, as needed.

In support of the NRC staff Revision 2 process, the EPRI MRP undertook three tasks:

1. Preparation of a draft AMP for PWR reactor internals that covers all ten AMP attributes (no such AMP currently exists in Revision 1 of Reference 6);

2. Preparation of revised GALL Tables IV.B2, IV.B3, and IV.B4, pointing to the information contained in MRP-227, Revision 0, and its successor versions that forms the basis for aging management of specific PWR reactor internals components; and

3. Preparation of a generic operating experience historical record that can be used in support of GALL AMP Attribute number ten, and which forms the basis for the new Appendix A in MRP-227-A.

It is important to note that these three tasks were undertaken by the MRP to justify the waiver of the fees charged by the NRC staff to review and hopefully approve MRP-227. However, these three tasks also assembled information that assisted greatly in defining this template for preparing a licensee AMP or inspection plan submittal to NRC staff.

Section 2 of this document provides the licensee submittal development context and process, including the connections to both regulatory and industry sources of information. Section 3 provides the template outline, but also includes actual submittal language with locations for insertion of licensee-specific information where needed, including insertable paragraphs taken from the current draft of the AMP in Chapter XI.M16A of Reference 6, the PWR reactor internals aging management program description that was included in Revision 2 of NUREG-1801. In view of the proposed changes to Chapter XI.M16A documented in the Staff draft ISG [7], alternate wording is provided to assist in submittal preparation. Then, Section 4 provides a summation of the current industry-prepared historical record of PWR reactor internals operating experience, information that can be supplemented and used directly in the licensee-prepared submittal under the attribute that deals with operating experience. Section 5 reviews the Applicant/Licensee Action Items identified by the staff in the SE for Reference 1, including suggested language for responding to the SE requirements. Finally, Section 6 summarizes the content of this template, with Section 7 containing the references.

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SUBMITTAL DEVELOPMENT PROCESS

2.1 Regulatory Basis and Requirements

Regulatory Issue Summary 2011-07 [4] provides the framework for submittal of license renewal information relative to aging management of PWR reactor vessel internals. Each licensee will need to review Reference 4 to determine the extent of their commitment to the requirements of MRP-227-A, the timing of their submittal of the associated inspection plan, and the process for any revisions to previously submitted AMPs or inspection plans. For example, licensees of Category A facilities (a facility for which an inspection plan may have already been submitted and inspections possibly conducted) may choose to withdraw their current inspection plan in writing, and resubmit the required information in accordance with MRP-227-A. If this path is selected, the resubmission must be in place within one year of the date of the RIS publication. If inspections have been carried out that do not comply with the MRP-227-A requirements, and if credit for those inspections is sought, the submittal must include justified deviations from the MRP-227-A requirements. The process for licensees of Category B facilities (a facility following existing license renewal commitments) is equally complex, while the path forward for Category C facilities (a facility with a license renewal application currently under review) or Category D facilities (a facility for which a license renewal application has not yet been submitted) is very straightforward.

The actual process to be followed is well defined in current regulatory documents. First, Regulatory Branch Technical Position RLSB-1 (see Appendix A.1 of NUREG-1800 [5]) describes the steps for the licensee AMP submittal and its relationship to industry guidance such as MRP-227-A. Assuming that the licensee is planning to, or has already completed the scoping and screening step and the aging management review step, the next step is to develop the aging management program elements necessary to manage aging effects during the extended operating period, based on the guidance provided by MRP-227-A and MRP-228, including eventual implementation in the form of the inspection plan. Most importantly, Branch Technical Position RLSB-1 outlines and describes the ten AMP attributes that comprise an acceptable program for managing the effects of aging.

2.2 Industry Basis and Requirements

The guidance in MRP-227-A and MRP-228 contains the bulk of the information needed to complete the licensee AMP submittal for seven of the ten attributes – Scope of Program, Preventive Actions, Parameters Monitored/Inspected, Detection of Aging Effects, Monitoring and Trending, Acceptance Criteria, and Operating Experience. Three of the attributes – Corrective Actions, Confirmation Process, and Administrative Controls – require licensee-specific information that is best adapted from the licensee's Corrective Action Program (CAP) and Quality Assurance (QA) Program. However, when developing the seven attributes for licensee AMP submittal through reference to MRP-227-A and MRP-228 information, several caveats apply:

Section 2.4 of MRP-227-A identifies the major assumptions underlying the guidelines, including base load operation and fuel management limitations; these assumptions were felt to be readily met by all U. S. operating PWR units; however, review of Section 2.4 and concurrence with the assumptions is a necessary step in the process. This caveat is covered by Applicant/Licensee Action Item 1 (see Section 5 of this document), and language for addressing this action item is included in that section.

MRP-227-A contains relatively conservative examination acceptance criteria that call for recording and potential engineering evaluation of all relevant conditions that are detected during the required examinations. However, no formal engineering evaluation acceptance criteria are provided in MRP-227-A. Some useful information is provided in Section 6 of MRP-227-A on the process that could be used for developing engineering evaluation acceptance criteria, but the acceptance criteria themselves are outside the scope of the guidance. The PWR Owners Group has an effort underway in this regard that has led to Reference 7, and further efforts continue to refine those criteria.

Chapter XI.M16A of Revision 2 of NUREG-1801 [6] contains a thorough description of an aging management program (AMP) for PWR reactor vessel internals, and that description is an excellent starting point for the preparation of an individual licensee AMP. However, it should be noted that the Staff has proposed a number of revisions to Chapter XI.M16A (see Reference 7) that should be considered in the preparation of any submittal to the Staff. Wording is reproduced from both the GALL Revision 2 version of Chapter XI.M16A and the proposed draft ISG changes, in order to assist in the preparation of any submittal, including the locations in Chapter XI.M16A where licensee-specific information needs to be added to the information derived from MRP-227-A and MRP-228.

MRP-227-A [1b] contains the revisions to MRP-227, Revision 0 [1a] needed to address all of the Topical Report Conditions described in the SE [3a, 3b], as determined by staff review of the draft MRP-227-A submitted for final review in October 2011. Therefore, care should be taken to assure that the information extracted from the guidance is taken from the latest published version of MRP-227-A, or its successor documents.

Finally, in addition to Applicant/Licensee Action Item 1, seven other Applicant/Licensee Action Items were identified by the staff in the final SE for Reference 1b [3b]. An attempt has been made in this document (see Section 5) to not only clarify these action items, but also to suggest language that can be used to demonstrate satisfaction of those requirements.

2.3 Typical Submittal Format

Section 1. Introduction. A brief discussion about the unit or units covered, references to the license renewal status, the purpose of the submittal, and appropriate references to the principal documents that serve as the basis for the submittal.

Section 2. Background. A more detailed discussion of the license renewal status for the unit or units, including a discussion of license renewal commitments related to the reactor vessel internals aging management program or the associated inspection plan. In some cases, it may be appropriate to discuss in greater depth the scope and extent of the industry efforts with respect to MRP-227-A and MRP-228, and their supporting documentation, and to provide background information on the NEI 03-08 process for managing materials issues. Some submittals use this section to provide a top-level scoping description of the aging management program or the inspection plan.

Section 3. Owner Responsibilities. A brief discussion of the Owner of the unit or units, plus the assignment of responsibilities related to the aging management program or inspection plan.

Section 4. Aging Management Program or Inspection Plan Basis. It may be convenient to group the discussion of various aging management program or inspection plan bases into this section, with the sub-sections associated with each basis able to be referenced, as needed in Section 5 of the submittal. Typical bases that can be described are MRP-227-A or MRP-228 themselves, various PWR Owners Group program elements, any applicable time-limited aging analyses (TLAAs), ASME Code Section XI requirements, existing plant programs, and other existing plant commitments.

Section 5. Aging Management Program Scope and Attribute Affirmation. This section describes the program and affirms its attributes, possibly include references back to sub-sections in Section 4 that provide necessary justifications for the affirmation of attributes. This template is intended to supply the bulk of the information needed to complete this section.

Section 6. Plant-Specific Topical Report Conditions and Action Items. This section has turned out to be a very important section in the submittal, and the section that generates the most interest during review by the Staff. This is the section that addresses the eight Licensee/Applicant Action Items defined by the SE on MRP-227-A, some of which can be very challenging. This template is intended to assist to some extent in addressing these items on a plant-specific basis, but obviously cannot supply the complete level of detail that may be required.

Section 7. Summary and Conclusions. A brief discussion that summarizes the content of the submittal.

Section 8. References. This template has attempted to gather a large cross section of potential references that might be included in this section.

Appendices. Appendices may be needed for tables or portions of tables extracted from MRP-227-A, or from one of its supporting documents, in order to supplement fully an action item or discussion point in the body of the submittal.

3
LICENSEE SUBMITTAL TEMPLATE

With the approval by the U. S. Nuclear Regulatory Commission of the guidance in MRP-227, Revision 0, and the issuance of MRP-227-A or any subsequent NRC-approved revision of that document, U. S. PWR licensees have an obligation under both the guidance itself or under the Regulatory Issue Summary (RIS) to possibly revise an existing AMP or to submit a new AMP based on that guidance, and to identify and justify any deviations from that guidance. In the following sub-sections, the actual licensee submittal template is provided. The template is based on the aging management program guidance provided in Chapter XI.M16A (PWR Vessel Internals), NUREG-1801, Revision 2 [6]; however, in view of the potential changes in Chapter XI.M16A represented in the draft Interim Staff Guidance issued in March and April 2012 [7], an attempt has been made to provide alternate wording for specific aging management program attributes.

3.1 Purpose and Background

The submittal should state that the purpose of the submittal is to document the plant or station reactor internals aging management program, and to specifically identify those activities that are intended to be credited for license renewal. It is useful in this part of the submittal to reference the plant or station aging management review (AMR) process for the reactor internals that led to the identification of the aging effects. It is also useful to reference the major supporting documents upon which the submittal relies, such as MRP-227, MRP-228, and any owners' group documents.

A background discussion that covers the plant or station involvement with the EPRI MRP effort to develop MRP-227 and MRP-228 would also be useful here, along with a discussion of the plant or station license renewal commitment to follow the program.

This part of the submittal may also contain specific background information about the plant or station Owner, a historical record of the plant or station license renewal activity, and a summary statement of the following type:

(insert plant or station identifier) has a license renewal commitment to work with the industry to develop a reactor vessel internals aging management program. The development and implementation of this aging management program meets this license renewal commitment.

3.2 Description of Program

The submittal should have a section that describes the aging management program for the plant or station PWR reactor vessel internals. The information needed to prepare this section was part of the task undertaken by the EPRI MRP in support of the NRC review of MRP-227. The following text may be used in its entirety as the opening discussion for the program description, and for the opening discussion of each of the ten aging management program attributes; however, in all cases, declarative statements about the intention to meet the MRP-227-A guidance, or a discussion of any potential deviations from that guidance, are needed as a conclusion for each sub-section. Note again that alternative wording – either from Reference 6 or from Reference 7, is provided.

3.2.1 General Program Description

The first sub-section of the PWR reactor vessel internals aging management program should provide an overall description of the program. It is suggested that the language from the draft ISG [7], shown below, which updates the language from Revision 2 of the GALL report, Chapter XI.M16A, be used to prepare that overall description, as needed.

This program relies on implementation of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Report No. 1022863 (MRP-227-A) and EPRI Report No. 1016609 (MRP-228) to manage the aging effects on the pressurized-water-reactor (PWR) reactor vessel internal (RVI) components. The MRP recommended activities in MRP-227-A and any additional plant-specific activities that need to be defined for this program are implemented in accordance with the guidelines of Nuclear Energy Institute (NEI) 03-08. The staff approved the MRP's augmented inspection and evaluation (I&E) criteria for PWR RVI components in a safety evaluation (i.e., the NRC SE (Revision 1) on MRP-227) dated December 16, 2011.

This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at the facility. These aging effects include: (a) various forms of cracking, including stress-corrosion cracking (SCC), primary water stress-corrosion cracking (PWSCC), irradiation-assisted stress-corrosion cracking (IASCC), and cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

The program applies the guidance in MRP-227-A for inspecting, evaluating, and, if applicable, dispositioning non-conforming RVI components at the facility. The program conforms to the definition of a sampling-based condition monitoring program, as defined by the NRC Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections, if the extent of the degradation effects exceeds the expected levels.

The MRP-227-A guidance for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process. Through this process, the reactor internals for all three PWR designs were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions of each group are provided in "Generic Aging Lessons Learned Report" (GALL Report), Revision 2, Chapter IX.B.

The result of this four-step sample selection process is a set of Primary internals component locations for each of the three plant designs that are expected to show the leading indications of the degradation effects, with another set of Expansion internals component locations that are specified to expand the sample should the indications be more severe than anticipated.

The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs, such as American Society of Mechanical Engineers (ASME) Code, Section XI[1], Examination Category B-N-3, examinations of core support structures. A fourth set of internals locations are deemed to require no additional measures. As a result, the program typically identifies 5 to 15 percent of the RVI locations as Primary Component locations for inspections, with another 7 to 10 percent of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15 percent of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. This process thus uses appropriate component functionality criteria, age-related degradation susceptibility criteria, and failure consequence criteria to identify the components that will be inspected under the program in a manner that conforms to the sampling criteria for sampling-based condition monitoring programs in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Consequently, the sample selection process is adequate to assure that the intended function(s) of the PWR reactor internal components are maintained during the period of extended operation.

The program's use of visual examination methods in MRP-227-A for detection of relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code, Section XI rules for visual examination. However, the program's adoption of the MRP-227-A guidance for visual examinations goes beyond the ASME Code, Section XI visual examination criteria because additional guidance is incorporated into MRP-227-A to clarify how the particular visual examination methods will be used to detect relevant conditions and describes in more detail how the visual techniques relate to the specific RVI components and how to detect their applicable age-related degradation effects.

The technical basis for detecting relevant conditions using volumetric ultrasonic testing (UT) inspection techniques can be found in MRP-228, where the review of existing bolting UT examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations. Specifically, the capability of program's UT volumetric methods to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as the baffle-former bolts in Babcock and Wilcox (B&W)-designed units and Westinghouse-designed units, has been well demonstrated by operating experience.

In addition, the program's adoption of the MRP-227-A guidance and process incorporates the UT criteria in MRP-228, which calls for the technical justifications that are needed for volumetric examination method demonstrations, required by the ASME Code, Section V.

The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227-A. The program thus provides reasonable assurance for the long-term integrity and safe operation of reactor internals in all commercial operating PWR nuclear power plants in the United States of America (U.S. PWR nuclear power plants).

Age-related degradation in the reactor internals is managed through an integrated program. Specific features of the integrated program are listed in the following ten program elements. Degradation due to changes in material properties (e.g., loss of fracture toughness) was considered in the determination of the inspection recommendations and is managed by the

[1] Refer to the GALL Report, Chapter I, for applicability of various editions of the ASME Code Section XI.

requirements to use appropriately degraded properties in the evaluation of identified defects. The integrated program is implemented by the applicant through an inspection plan that is submitted to the NRC for review and approval with the application for license renewal.

3.3 Evaluation and Technical Basis

The next sub-section of the aging management program description contains the evaluations of the ten aging management program attributes. Two different sets of wording are provided for each attribute – one from Chapter XI.M16A of Revision 2 of the GALL report and an alternate from the Staff draft ISG [7]. Either version may be used in part or in its entirety as a part of the licensee-specific submittal, with plant or station identifiers inserted where needed.

3.3.1a Scope of Program (GALL). The scope of the program includes all reactor vessel internals components at the [*as an administrative action item for the AMP, the applicant to fill in the name of the applicant's nuclear facility, including applicable units*], which [*is/are*] built to a [*applicant to fill in Westinghouse, CE, or B&W, as applicable*] NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC-endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those reactor vessel internals components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other reactor vessel internals components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).

The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's reactor vessel internals components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAIs as well. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA.

The guidance in MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227.

3.3.1b Scope of Program (ISG) The scope of the program includes all RVI components at the [*as an administrative action item for the aging management program (AMP), the applicant to fill in the name of the applicant's nuclear facility, including applicable units*], which [*is/are*] built to a [*applicant to fill in Westinghouse, CE, or B&W, as applicable*] NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC-endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227-A guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).

The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

The scope of the program includes the responses to applicable applicant/licensee action items (A/LAIs) on the MRP-227-A methodology (as identified in Revision 1 of the NRC SE on MRP-227), and to applicable subsections in Section 3.1.2.2 of NUREG-1800, Revision 2 (i.e., SRP-LR "further evaluation" criteria sections), and any additional programs, actions, or activities that are discussed in these A/LAI or "further evaluation" responses and are credited for aging management of the applicant's RVI components[2].

Additional criteria are provided in the latest NRC endorsed version of Westinghouse Commercial Atomic Power (WCAP) Report No. WCAP-17096-NP, " Reactor Internals Acceptance Criteria Methodology and Data Requirements." If WCAP-17096-NP is used as a basis for superseding applicable criteria in MRP-227-A, the application of the report is subject to the conditions and A/LAIs established in the staff's SE endorsing the use of the WCAP report.

[2] Consistent with the NRC recommendations in the applicable SRP-LR "further evaluation" sections, those A/LAI response bases or SRP-LR "further evaluation" response bases that result in the need for augmentation of the AMP beyond the I&E criteria recommended in the MRP-227-A report, are requested to be identified as plant-specific enhancements of the AMP, along with the applicable program element criteria that they impact. Justifications for the enhancements are to be provided in the applicable responses to the A/LAI items or to the SRP-LR "further evaluation" items.

3.3.1c Scope of Program Following either the GALL or ISG Scope of Program discussion, two short sub-sections follow – one that describes any plant-specific Scope of Program items, and another sub-section that affirms the applicability or points out exceptions to the basic Scope of Program.

3.3.2a Preventive Actions (GALL) The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."

3.3.2b Preventive Actions (ISG) The guidance in MRP-227-A relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress-corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."

3.3.2c Preventive Actions Following either the GALL or ISG Preventive Actions discussion, two short sub-sections follow – one that describes any plant-specific Preventive Actions items, and another sub-section that affirms the applicability or points out exceptions to the basic Preventive Actions element.

3.3.3a Parameters Monitored/Inspected (GALL) The program manages the following age-related degradation effects and mechanisms that are applicable in general to the reactor vessel internals components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep.

For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Primary Components in Table 4-1 of MRP-227"; "for CE designed Primary Components in Table 4-2 of MRP-227"; and "for Westinghouse designed Primary Components in Table 4-3 of MRP-227"*]. Additionally, the program implements the parameters monitored/inspected criteria for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Expansion Components in Table 4-4 of MRP-227"; "for CE designed Expansion Components in Table 4-5 of MRP-227"; and "for Westinghouse designed Expansion Components in Table 4-6 of MRP-227"*].

The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing Programs, such as the requirements for ASME Code Class reactor vessel internals components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse designed flux thimble tubes in GALL AMP XI.M37, "Flux Thimble Tube Inspection." No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measures," in accordance with the analyses reported in MRP-227.

3.3.3b Parameters Monitored/Inspected (ISG) The program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep.

For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destructive examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDFE method. For management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by: indirectly managed by: (1) using visual or volumetric examination techniques to monitor for cracking in the components, and (2) applying applicable reduced fracture toughness properties in the flaw evaluations, if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria consistent with the applicable condition monitoring tables in MRP-227-A, as applicable to the nuclear steam supply vendor for the plant's RVI components, or as enhanced based on the applicant's response bases to applicable A/LAIs in Revision 1 of the NRC SE on MRP-227, or to applicable NRC further evaluation recommendations in the SRP-LR (Refer to Footnote 2).

3.3.3c Parameters Monitored/Inspected Following either the GALL or ISG Parameters Monitored/Inspected discussion, two short sub-sections follow – one that describes any plant-specific Parameters Monitored/Inspected items, and another sub-section that affirms the applicability or points out exceptions to the basic Parameters Monitored/Inspected element.

3.3.4a Detection of Aging Effects (GALL) The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected.

These methods include volumetric UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimension, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface-breaking discontinuities. Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that need to be applied to inspections of Primary Components and Existing Requirement Components, and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the reactor vessel internals components are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.

Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "B&W designed Primary Components in Table 4-1 of MRP-227"; "CE designed Primary Components in Table 4-2 of MRP-227;" or "Westinghouse designed Primary Components in Table 4-3 of MRP-227"*] and for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Expansion Components in Table 4-4 of MRP-227;" "for*

CE designed expansion components in Table 4-5 of MRP-227;" and "for Westinghouse designed Expansion Components in Table 4-6 of MRP-227"].

The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): [*As a relevant license renewal applicant action item, the applicant is to list (using criteria in MRP-227) each additional reactor vessel internals component that needs to be inspected as an additional plant-specific Primary Component for the applicant's program and each additional reactor vessel internals component that needs to be inspected as an additional plant-specific Expansion Component for the applicant's program. For each plant specific component added as an additional Primary or Expansion Component, the list should include the applicable aging effects that will be monitored for, the inspection method or methods used for monitoring, and the sample size and frequencies for the examinations*].

In addition, in some cases (as defined in MRP-227), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program include [*Applicant to input physical measure methods identified by the MRP in response to NRC RAI No. 11 in the NRC's Request for Additional Information to Mr. Christen B. Larson, EPRI MRP on Topical Report MRP-227 dated November 12, 2009*].

3.3.4b Detection of Aging Effects (ISG) Since the program is consistent with MRP-227-A and MRP-228, the detection of aging effects is introduced in Section 4 of MRP-227-A and standards for the examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface-breaking discontinuities, and the program may apply physical measurement techniques for the detection of loss of preload in fastened or bolted assemblies or for measuring changes in dimension as a result of distortion, irradiation-assisted creep, or void swelling.

Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking if justified in accordance with the NRC's "further evaluation" criteria in SRP-LR Section 3.1.2.2.9.A.7, which provides the NRC's further evaluation "acceptance criteria" recommendations on this matter.

In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation- enhanced stress relaxation and creep.

In addition, the program adopts the recommended guidance in MRP-227-A for defining the Expansion criteria that need to be applied to inspections of Primary Components and Existing Requirement Components and for expanding the examinations to include additional Expansion Components. RVI component inspections are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227-A, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.

3.3.4c Detection of Aging Effects Following either the GALL or ISG Detection of Aging Effects discussion, two short sub-sections follow – one that describes any plant-specific Detection of Aging Effects items, and another sub-section that affirms the applicability or points out exceptions to the basic Detection of Aging Effects element.

3.3.5a Monitoring and Trending (GALL) The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and reexaminations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.

3.3.5b Monitoring and Trending (ISG) The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227-A and its subsections. Flaw evaluation methods, including recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications, are given in the latest NRC-endorsed version of the WCAP-17096-NP report. The examinations and re-examinations required by the MRP-227-A guidance, together with the flaw evaluation criteria in the WCAP-17096-NP report and the criteria specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions of the aging effects and mechanisms that are managed by the program.

3.3.5c Monitoring and Trending Following either the GALL or ISG Monitoring and Trending discussion, two short sub-sections follow – one that describes any plant-specific Monitoring and Trending items, and another sub-section that affirms the applicability or points out exceptions to the basic Monitoring and Trending element.

3.3.6a Acceptance Criteria (GALL) Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.

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

- For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;
- For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and
- For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants are given in Table 5-1 of MRP-227. The acceptance criterion for physical measurements performed on the height limits of the Westinghouse-designed hold-down springs are [*The incorporation of this sentence is a license renewal applicant action item for Westinghouse PWR applicants only – insert the applicable sentence incorporating the specified physical measurement criteria only if the applicant's facility is based on a Westinghouse NSSS design: the Westinghouse applicant is to incorporate the applicable language and then specify the fit up limits on the hold down springs, as established on a plant-specific basis for the design of the holddown springs at the applicant's Westinghouse-designed facility*].

3.3.6b Acceptance Criteria (ISG) Section 5 of MRP-227-A, as supplemented by information in NRC approved versions of WCAP-17096-NP, provides specific Primary and Expansion Component examination methods, and include those for visual, surface, and volumetric techniques. For components addressed by examinations referenced to ASME Code Section XI, the IWB-3500 acceptance criteria apply. For components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document (e.g., those for Westinghouse-design flux thimble tubes under a Westinghouse applicant's NRC Bulletin 88-09 implementation program.

The program adopts the acceptance criteria for the physical measurement monitoring methods recommended in MRP-227-A, as qualified in Section 3.3.5 and A/LAI No. 5 in Revision 1 of the NRC SE on MRP-227[3].

[3] The EPRI MRP's recommendations are applicable to physical measurements of Westinghouse-design hold-down springs and to those CE-design nuclear facilities that are designed with welded core shrouds made from two vertical shroud segments. Based on the recommendations, Westinghouse applicants should define, as enhancements to the program, the acceptance criteria that will be applied to physical measurement methods used to manage loss of compressibility (loss of preload) in their hold-down springs, as based on the applicant's response bases to the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.B.2. Similarly, CE applicants, whose plants are designed with welded core shrouds containing the specified gap area, should define, as enhancements of the program, the acceptance criteria that will be applied to physical measurement methods used to manage distortion (changes in dimension) in the gap areas of the shrouds, as based on the applicant's response bases to the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.C.2. The MRP did not recommend physical measurement monitoring bases for Babcock and Wilcox (B&W) designed plants or for CE plants with bolted core shroud designs.

3.3.6c Acceptance Criteria Following either the GALL or ISG Acceptance Criteria discussion, two short sub-sections follow – one that describes any plant-specific Acceptance Criteria items, and another sub-section that affirms the applicability or points out exceptions to the basic Acceptance Criteria element.

3.3.7a Corrective Actions (GALL) Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227.

Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for Westinghouse-designed reactor vessel internals components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577, Rev. 1-A, or for B&W-designed reactor vessel internals components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577, Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies on behalf of the B&W Owners Group, dated December 9, 1999.

Alternative corrective action bases not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation.

3.3.7b Corrective Actions (ISG) Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events.

The implementation of the guidance in MRP-227-A, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC, such as the corrective action bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, or for B&W-designed RVI components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies on behalf of the B&W Owners Group, dated December 9, 1999. Alternative corrective action bases not approved or endorsed by the NRC, will be submitted for NRC approval prior to their implementation.

3.3.7c Corrective Actions Following either the GALL or ISG Corrective Actions discussion, two short sub-sections follow – one that describes any plant-specific Corrective Actions items, and another sub-section that affirms the applicability or points out exceptions to the basic Corrective Actions element.

3.3.8a Confirmation Process (GALL) Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls.

3.3.8b Confirmation Process (ISG) Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the recommendations of NEI 03-08 and the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. The implementation of the guidance in MRP-227-A, in conjunction with NEI 03-08 and other guidance documents, reports, or methodologies referenced in this AMP, provides an acceptable level of quality and an acceptable basis for confirming the quality of inspections, flaw evaluations, and corrective actions performed under this program.

3.3.8c Confirmation Process Following either the GALL or ISG Confirmation Process discussion, two short sub-sections follow – one that describes any plant-specific Confirmation Process items, and another sub-section that affirms the applicability or points out exceptions to the basic Confirmation Process element.

3.3.9a Administrative Controls (GALL) The administrative controls for such programs, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation.

3.3.9b Administrative Controls (ISG) The administrative controls for these types of programs, including their implementing procedures and review and approval processes, are implemented in accordance with the recommended industry guidelines and criteria in NEI 03-08, and in accordance with the existing site 10 CFR 50, Appendix B, Quality Assurance Programs, or their equivalent, as applicable. The evaluation in Section 3.5 of Revision 1 of the SE on the MRP-227 methodology provides the NRC's basis for endorsing the NEI 03-08 implementation process for these programs. This includes NRC's endorsement of the NEI 03-08 criteria for notifying the NRC of any deviation from the I&E methodology in MRP-227-A and justifying the deviation no later than 45 days after approval by a licensee executive. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation.

3.3.9c Administrative Controls Following either the GALL or ISG Administrative Controls discussion, two short sub-sections follow – one that describes any plant-specific Administrative Controls items, and another sub-section that affirms the applicability or points out exceptions to the basic Administrative Controls element.

3.3.10a Operating Experience (GALL) Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program or to participate in industry initiatives that perform this function. The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience.

3.3.10b Operating Experience (ISG) The review of relevant operating experience (OE) and the assessment of OE for its impacts on the program elements of an applicant's PWR Vessel Internals Program, and on the program's implementing procedures and review and approval processes, are governed by the recommended industry guidelines and criteria in NEI 03-08, and additionally in accordance with the existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. The evaluation in Section 3.5 of Revision 1 of the SE on MRP-227 provides the staff's basis for endorsing the NEI 03-08 implementation process for these programs.

This includes NRC's endorsement of the NEI 03-08 criteria for notifying the NRC of any deviation from the I&E methodology in MRP-227-A and justifying the deviation no later than 45 days after approval by a licensee executive. As discussed in Appendix B of the GALL Report, the ongoing effectiveness of the program is ensured through the systematic review of both plant-specific and industry operating experience.

Consistent with MRP-227-A, the reporting of PWR RVI inspection and OE results is treated as a "Needed" category item under the applicant's NEI 03-08 implementation process. Based on these criteria, such programs are expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-tem implementation.

3.3.10c Operating Experience Following either the GALL or ISG Operating Experience discussion, two short sub-sections follow – one that describes any plant-specific Operating Experience items, and another sub-section that affirms the applicability or points out exceptions to the basic Operating Experience element.

3.3.11 References (as needed)

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR Part 50.55a, Codes and Standards, Office of the Federal Register, National Archives and Records Administration, 2011.

ASME Boiler & Pressure Vessel Code, Section V, Nondestructive Examination, 2004 Edition, American Society of Mechanical Engineers, New York, NY.

ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

B&W Report No. BAW-2248, Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, Framatome Technologies (now AREVA Technologies), Lynchburg VA, July 1997. (NRC Microfiche Accession Number A0076, Microfiche Pages 001 - 108).

EPRI 1014986, PWR Primary Water Chemistry Guidelines, Volume 1, Revision 6, Electric Power Research Institute, Palo Alto, CA, December 2007. (Non-publicly available ADAMS Accession Number ML081140278). The non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number ML081230449.

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

EPRI 1022863, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), Electric Power Research Institute, Palo Alto, CA: Final Report, December 2011 (Transmittal Letter to NRC in ADAMS Accession Number ML12017A193, Report is given in four parts in ADAMS Accession Numbers ML12017A194, ML12017A196, ML12017A197, and ML12017A191).

EPRI 1016609, Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228), Electric Power Research Institute, Palo Alto, CA, July 2009. (Non-publicly available ADAMS Accession Number ML092120574). The non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number ML092750569.

NRC RAI No. 11 in the NRC's Request for Additional Information to the Mr. Christen B. Larson, EPRI MRP on Topical Report MRP-227 dated November 12, 2009.

NRC Safety Evaluation from C. I. Grimes [NRC] to R. A, Newton [Chairman, Westinghouse Owners Group], Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled "License Renewal Evaluation: Aging Management for Reactor Internals," WCAP-14577, Revision 1, February 10, 2001. (ADAMS Accession Number ML010430375).

NRC Safety Evaluation from C. I. Grimes [NRC] to W. R. Gray [Framatome Technologies], Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," February 10, 2001. (ADAMS Accession Number ML993490288).

NUREG-1800, Revision 2, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, Appendix A.1, "Aging Management Review - Generic (Branch Technical Position RLSB-1)," U.S. Nuclear Regulatory Commission, Washington, DC, 2010.

Westinghouse Non-Proprietary Class 3 Report No. WCAP-14577-Rev. 1-A, License Renewal Evaluation: Aging Management for Reactor Internals, Westinghouse Electric Company, Pittsburgh, PA [March 2001]. Report was submitted to the NRC Document Control Desk in a letter dated April 9, 2001. (ADAMS Accession Number ML011080790).

Westinghouse Non-Proprietary Class 3 Report No. WCAP-17096-NP (latest version), Reactor Internals Acceptance Criteria Methodology and Data Requirements, Westinghouse Electric Company, Pittsburgh, PA. (Most recent version was reported in Revision 2 of the report [December 2009], which was submitted in ADAMS Accession Number ML101460156).

NRC Interim Staff Guidance LR-ISG No. 2011-04, "Aging Management Criteria for PWR Reactor Vessel Internal Components," March 13, 2012 (ADAMS Accession Number ML12004A149).

Nuclear Energy Institute (NEI) Report No. 03-08 (latest version), Guideline for the Management of Materials Issues. (Most recent edition is given in Revision 2 of the report, April 5, 2010; Refer to ADAMS Accession Number ML101050334).

NRC Safety Evaluation from Robert A. Nelson (NRC) to Neil Wilmshurst (EPRI), Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," December 16, 2011 (ADAMS Accession Number ML11308A770).

3.4 Reporting

Following the section on evaluation of aging management program attributes, the licensee-specific submittal should have a section on reporting, with an emphasis on reporting the results of PWR reactor internals examinations to the EPRI MRP. Section 7 of MRP-227 describes the **Good Practice** reporting requirements.

Each commercial U.S. PWR unit should provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals are examined.

These results will be compiled into an overall industry report which will track industry progress, aid in evaluation of significant issues, identification of fleet trends and determination of any needed revisions to MRP-227. It is planned to update this industry report biennially for the benefit of the fleet, the regulator, owners groups, and other industry stakeholders. The biennial report will also serve to assist in review of operating experience, and required monitoring and trending for aging management programs established by the industry.

3.5 Plant or Station Specific Sections

As pointed out in Sub-Section 3.3, in addition to the GALL or ISG template material, the licensee-specific submittal should add one or more plant or station specific sections covering topics such as program enhancements over and above MRP-227 requirements. In addition, the submittal may choose to extract information directly from MRP-227 (e.g., design-specific tables in Sections 4 and 5 of MRP-227) and place them in subsequent sections or in appendices. A reference section that includes all sources of information used in preparing the submittal should be attached.

4
OPERATING EXPERIENCE

The tenth aging management program (AMP) attribute covers operating experience, both historically for the industry, as well as for the individual licensee's unit(s). This section is intended to provide a substantive historical record of operating experience for U. S. and, to some extent, foreign PWR internals. The licensee will need to supplement this historical record with any relevant plant-specific operating experience. The historical record may need updating, as well. While relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants, a summary of the current operating experience is useful for licensees developing aging management programs. This summary is organized by age-related degradation mechanism (effect). This compilation does not replace efforts by licensees to review and document plant-specific operating experience for impact on its program, or participate in industry initiatives that perform this function.

4.1 Cracking

4.1.1 IGSCC

Multiple reactor vessel internals bolt failures of the lower thermal shield bolts were discovered during the 1981 and 1982 in-service inspections performed at three B&W-design PWRs. The thermal shield bolt locking clips at these three plants were visually observed to be missing or loose. Subsequent examinations during 1982, 1983, and 1984 revealed bolt failures at four additional units. These failures included upper core barrel, lower core barrel, upper thermal shield, and surveillance specimen holder tube bolts. All of the affected fasteners were fabricated from Alloy A-286 ASTM A 453, Grade 660, Condition A or B material. The results of an extensive evaluation program revealed the failure mechanism was predominantly due to an environmentally-assisted IGSCC mechanism. However, for some bolts, there was evidence that fatigue was also a contributor, likely in the form of corrosion fatigue.

In general, the primary mechanism causing cracking and failure of the Alloy A286 reactor vessel internals bolts was IGSCC. All the failures occurred in the bolt head-to-shank fillet. NRC Information Notice (IN) 90-68 provides information about IGSCC cracking in Alloy A-286 bolts used to hold the turning vanes to reactor coolant pumps at a foreign plant. The IN 90-68 document includes a general discussion of the problems experienced with cracking of Alloy A-286 bolting materials, including the problems identified with respect to B&W reactor vessel internals bolting.

In 2005, cracking of replacement core barrel-to-former plate bolts fabricated from cold-worked Type 316Ti stainless steel was observed in a German PWR by visual inspection. These bolts had replaced the original Alloy X-750 core barrel bolts in the late 1980s, which had exhibited failure due to PWSCC (described below). Subsequent UT inspection and failure analysis confirmed that the cracking was confined to the bolt head initiating from the bolt fillet transition, but the bolt threads and shank were free from cracking. The failure mechanism of the cold-worked Type 316Ti stainless steel replacement core barrel bolts has been identified as IGSCC. To date, all known failures of core barrel bolts have been limited to the original Alloy X-750 and the replacement cold-worked Type 316Ti stainless steel in German PWRs.

4.1.2 PWSCC

Alloy X-750 has experienced numerous worldwide failures in the Westinghouse-designed reactor vessel internals involving the control rod guide tube support pins (a.k.a., split pins). As noted in IN 82-29, these failures first appeared in Japan in the late 1970s. Split pin failures prompted investigations and modifications to manufacturing practices. The original heat treatment condition AH[4] of the age-hardenable material has shown the most susceptibility to PWSCC cracking. By the early 1980s, nearly all of the original design split pins had been replaced with the improved HTH heat treatment condition.

In 1987, failures of Alloy X-750 HTH condition control rod guide tube support pins in French PWRs occurred at much shorter times and lower stresses than expected. Foucault, et al., showed that these early failures were due to the surface condition of the pins. Any heat treatment after machining degrades the performance of Alloy X-750. The greatest resistance to IGSCC was found when machining or polishing was performed after heat treatment, which removes an oxide layer from the surface of the material. Additional refinements have since been made to the manufacturing practices used to produce a newer version of Alloy X-750 HTH split pins.

Alloy X-750, in a condition similar to AH, was used for the baffle-to-former plate bolts in the German Biblis-type reactors. After about four years of service, several bolts were found either cracked or severed. The cracking occurred in the bolt head-to-shank fillet area and was attributed to IGSCC (a.k.a., PWSCC in nickel-base materials). The bolt stress levels were reportedly at the yield strength of the material.

Failures have been attributed to three factors:

1. Heat treatment condition
2. High peak stresses
3. Surface damage due to fabrication processes

Failures of Alloy X-750 clevis insert bolts were reported by one Westinghouse-designed plant in 2010. The lower clevis structure works with the radial keyways on the core barrel to provide rotational alignment for the lower internals. The Alloy X-750 bolting was used to fasten the Alloy 600 clevis inserts to the RV lugs. Although the failed clevis insert bolts were not removed for metallurgical examination, it can be surmised that the most likely cause of failure was PWSCC. The clevis insert bolting had been heat treated in a condition similar to the AH treatment that has proven to be susceptible to PWSCC in the guide tube support pins. The relatively long time to failure in the clevis insert bolting may be attributed to the lower service temperature.

4.1.3 IASCC

A considerable amount of PWR vessel internals IASCC has been observed in European PWRs since the 1980s, with emphasis on cracking of baffle-former bolting. Ultrasonic (UT) testing of baffle-former bolts in six French PWRs discovered failure rates ranging from 1.2% to 11% of the

[4] Hot rolled, “equalized” at 1625ºF (885ºC) followed by 20 hours at 1300ºF (704ºC).

960 total bolts. For this reason, the U.S. PWR owners and operators began a program to inspect the baffle-former bolting in order to determine whether similar problems might be expected in U.S. plants. One benefit of this program was the experience gained with the UT examination techniques used in the inspections. In addition, the industry began laboratory testing projects in order to gather the materials data necessary to support future inspections and evaluations.

At one U.S. domestic unit with Type 347 stainless steel baffle-former bolting, all 728 baffle-former bolts were inspected by UT in 1998, with 55 bolts (7.5%) having indications that exceeded the UT acceptance criteria. At another unit, 639 out of the 728 baffle-former bolts were examined in 1999, with 59 bolts (9.2%) having indications failing to meet the UT acceptance criteria. At the first unit, on-site underwater mechanical testing of the removed baffle-former bolts indicated that the actual number of defective bolts was lower than suggested by the UT inspection. However, these known European or domestic baffle-former bolt IASCC indications are not necessarily applicable to all PWR designs. To date, the incidents have been generally associated with cold-worked Type 316 stainless steel or Type 347 stainless steel.

Bolts fabricated from solution-annealed Type 304 stainless steel appear to be less susceptible. An inspection was performed at one B&W-designed unit in 2005 on all 864 baffle-former bolts and no UT indications were observed.

In 2010, one Westinghouse plant reported finding several broken Type 347 stainless steel baffle-former bolt heads and Type 304 stainless steel locking bars on the lower core plate during a normal refueling outage. Subsequent investigation identified a region containing approximately 40 broken or severely damaged bolts. The damage was limited to the upper half of a single baffle plate. The root cause evaluation for this event is ongoing.

Baffle-former bolt inspections have been conducted under the guidance of MRP-227 at three different Westinghouse-design U.S. domestic plants. The original bolting material at all three plants is Type 347 stainless steel; though, one of the units did replace a subset of the bolts in 1999 with Type 316 stainless steel.

The first of these inspections was a full UT examination of the baffle-former bolts conducted in 2010. The UT inspection detected one likely flaw out of 1088 bolts. Additionally, visual inspection of the baffle-former assembly detected two baffle-former bolts with missing lock bar welds. Each lock bar should have two welds to hold it in place, and these two bolts only had one weld. The missing welds were dispositioned as fabrication errors. The missing welds and flawed bolt were left in service.

The second inspection was a full UT examination of the baffle-former bolts conducted in 2011. The UT inspection detected two likely flaws out of 1088 bolts. Visual examination of the baffle-former assembly did not detect any reportable indications. The flawed bolts were left in service.

The one unit which replaced a subset of bolts in 1999 conducted a UT examination on a subset of bolts. All of the 1999 replacement bolts were inspected and approximately 100 of the original bolts were inspected. Additionally, a small number of bolts were removed and replaced. When possible, these bolts were inspected by UT after removal. Of the bolts inspected, only one defective bolt was detected. This bolt was left in service.

4.1.4 Fatigue

In the earlier PWRs, a number of incidents occurred indicating that thermal shields and their support system could be vulnerable to the high flow forces in the vessel-core barrel downcomer, as the result of flow-induced vibration. Westinghouse, CE, and B&W responded to these experiences in different ways. The Westinghouse approach was to add vibration resistance to the shields and to embark on a program to develop advanced thermal shield designs for future plants. For CE plants, thermal shields were removed from operation for all but one facility, which has maintained integrity through positioning pin replacement, tightening, and inspection. Difficulties with B&W thermal shields have been generally addressed by repair and modification.

The dominant degradation mechanisms in thermal shields are high-cycle fatigue and SCC resulting from flow-induced vibration, with mechanical wear as a potential consequence. These degradation events appeared predominantly in the earliest thermal shield designs. Typically, the degraded components were fasteners or thermal shield support structures, not the thermal shield itself.

Two CE plants reported cases where failures in the thermal shield resulted in damage to the core barrel. The thermal shields were removed from both plants and the damage to the core barrels was mitigated.

Three early Westinghouse plants identified thermal shield degradation. The thermal shield degradation in these three plants was repaired; however, they are no longer operating and no operating plant has the same thermal shield design. Two additional Westinghouse plants have reported isolated failures of core barrel bolting that may be linked to flow-induced vibration.

4.2 Loss of Material

4.2.1 Wear

Wear of the in-core instrumentation thimble tubes was observed in the top part of the Zircaloy-4 thimble tubes at three PWR CE-designed units. These tubes experienced through-wall tube degradation as a result of flow-induced vibration in the vicinity of the fuel alignment plate. This particular wear phenomenon was addressed by making modifications to the fuel alignment plate to alter the flow conditions in the vicinity of the entry point of the thimble tubes into the plate. Wear as a result of flow-induced vibration has not been observed in these components after implementing the modifications to the fuel alignment plate. Accordingly, these components are not considered susceptible to this type of wear in the future.

Problems were noted involving the original locking devices for the B&W-design vent valve jackscrews in the late 1970s and early 1980s. The jackscrew locking mechanism was vibrating and wearing through the locking cup. A new locking mechanism was designed and supplied to most B&W units. At least four of the eight vent valves were modified with the redesigned locking devices. The four vent valves next to the two outlet nozzles were replaced. In the late 1970s and early 1980s, problems were also noted involving the original jackscrew guide bushing, which was found to be improperly secured on some valves. Procedures were developed to install the modified locking device on the jackscrew and to secure the bushing when necessary.

Wear of the Westinghouse control rod guide tube assembly guide cards has been reported at several domestic and international plants. The wear enlarges the guide card holes that guide the control rods through the assembly and maintain the alignment of the rods. A program is currently in progress through the PWROG to establish guidelines for managing this wear.

The wear surfaces on the radial keyways and clevis inserts are routinely examined as part of the reactor vessel internals ASME Section XI inservice inspection programs. While reports of scratches, superficial wear, or both are common in these inspections, one European plant has reported significant wear scars at these surfaces. Efforts to establish quantitative acceptance criteria are ongoing.

In all currently operating Westinghouse and B&W plants, the incore flux detectors are directed through the reactor vessel bottom head via thimble tubes or guideways. For the bottom-mounted instrumentation design, the thimble tubes are retractable, and the insertion and retraction of these tubes are directed by long-radius guides below the bottom head and by internals guides between the bottom head and fuel assemblies. There is significant variation among plants with regard to thimble tube diameters (outer and inner), thimble tube-to guide path clearance, length of thimble tube exposed to coolant, and flow conditions.

The primary historical concerns with flux thimble degradation have been obstruction of the flux detector pathways, wear due to flow-induced vibration of the thimble tube, flow-induced vibration fatigue damage to thimble tube guideways, and damage to in-core instrumentation flange seating surfaces at refueling. The obstruction problem can often be mitigated by appropriate cleaning procedures at refueling. All Westinghouse plants are required by NRC Inspection & Enforcement (IE) Bulletin 88-09 to have an inspection program to periodically confirm incore neutron monitoring system thimble tube integrity. Reductions in wall thickness due to wear are normally monitored with an eddy-current inspection. Many plants have chosen to replace the flux thimbles with improved designs. These programs have been successful in managing thimble tube degradation.

A visual inspection in 1973 at one CE-designed plant revealed worn areas in the reactor vessel flange and head resulting from inadequate hold-down spring design and subsequent reactor vessel internals vibration. Prior to shutdown, higher than normal ex-core neutron detector readings had suggested the possibility of excessive internals vibration. Wear was found on the mating surfaces, alignment keys and slots, snubbers, and outlet nozzle faces. The worn surfaces were repaired and a new design using Belleville spring assemblies greatly increased hold-down capacity and mitigated the issue.

In the Westinghouse-design and in two CE-designed units, reactor vessel internals hold-down rings (or springs) were fabricated with Type 304 stainless steel. The subsequent CE-designed units switched to a modified Type 403 stainless steel hold-down ring, which shows less reduction in preload over the lifetime of the component. At least one international Westinghouse-designed plant has replaced their Type 304 stainless steel hold-down rings. Those that have not are managing potential degradation through physical measurement.

4.3 Change in Dimension

4.3.1 Irradiation-Induced Growth

Although irradiation-induced growth of zirconium alloys in CE-designed plants was not explicitly identified in MRP-175 as an age-related degradation mechanism to be evaluated as part of the screening process, irradiation-induced growth in the axial direction of the in-core instrumentation thimble tubes has reduced the clearance between the thimble nose and the bottom of the fuel assembly. Some plants had observed that the thimble tube support plate was raised above its normal support position when the upper internals structure was set in place after fuel reload. This indicated that for some of the thimbles the gap tolerance between the thimble tube and the bottom end fitting of the fuel assembly had been reduced until the tube contacted the bottom fitting of the fuel assembly and was being loaded in compression. Ten plants affected by this issue have taken actions. Six of these plants have already replaced the thimble tube assemblies with modified designs that are shorter in length to accommodate the expected irradiation-induced growth. Two additional plants have replacement designs in fabrication and have made preparations to install the replacement thimbles in an upcoming outage. The remaining two plants have not yet begun preparations for a full replacement of the thimble tubes, but one of these two has instead taken the intermediate step of raising the thimble support plate to accommodate additional axial growth. These plants are planning to execute a thimble assembly replacement program during a future refueling outage that is not currently encumbered with other large-scale replacements of major components. All affected plants will likely have replaced their thimble tubes prior to license extension.

4.4 Miscellaneous

4.4.1 B&W-Design Vent Valves

Vent valve jackscrew locking cup damage has also been observed at some units, which was due to an interaction with the plenum assembly during insertion and removal activities. Vent valves are replaceable items and as noted above, have been replaced as necessary.

4.4.2 Mechanism Unidentified to Date

Visual examinations at one B&W-designed unit in 2005 indicated that three or four internal baffle-to-baffle bolts were found protruding. The bolt heads extended beyond the baffle plate surface. This was an indication that the locking devices, and potentially the bolts as well, had failed. As noted above, a UT inspection of 100% of the baffle-former bolts was performed, with no detected indications of broken bolts. No UT inspection was performed on the internal baffle-to-baffle bolts, and the potentially failed baffle-to-baffle bolts have yet to be removed to confirm failure and, if failed, the mechanism. As a result of the observations, AREVA performed a unit-specific evaluation to assess operational and safety functions for continued operation. That evaluation included thermal hydraulic evaluation, structural evaluation, fuel evaluation, and loose parts evaluation.

4.5 Plant-Specific Operating Experience

This sub-section should include any plant-specific operating experience pertinent to the management of age-related degradation effects in either the plant core support structures or in the other plant internals.

4.6 Reactor Vessel Internals Component Replacements

Upper internals in Westinghouse and CE designed plants have been replaced.

Beginning in 2004, replacement of the complete internals (upper and lower internals) at three Japanese PWRs has also been performed. It has been stated that these replacements have been performed for the following reasons:

- To keep and improve operational reliability, safety, and a high load factor for the nuclear power units
- To maintain the plant against aging degradation of the reactor vessel internals
- To mitigate degradation risks that would rise with increasing operational time in the future

5
PLANT SPECIFIC ACTION ITEMS

The NRC staff final Safety Evaluation (SE) of MRP-227-A [3b] included eight plant-specific action items that are required to be addressed by a license renewal applicant who references MRP-227-A [1b], or any subsequent versions of the guidance approved by the NRC staff, in their AMP submittal. Each of these plant-specific action items is described below, with proposed language to address that item provided.

5.1 Applicant/Licensee Action Item 1 (SE Section 4.2.1)

The SE provided an assessment of the explicit assumptions documented in Section 2.4 of MRP-227-A, which stipulated base load operation (i.e., no load following) and conversion of high-leakage core loading patterns to low-leakage core loading patterns after no more than 30 years of operation. In addition, the SE referred implicitly to assumptions used in the Failure Modes, Effects, and Criticality Analyses (FMECA) reports [10, 11, 12] and the functionality analyses [13, 14] that provide technical support to MRP-227-A. In particular, the SE expressed concern about power plant uprates and the continued relevance of the guidelines to such uprated plants. This concern had been expressed by a Request for Additional Information (RAI) during the staff's review of Reference 1a, with the response to that RAI stipulating that each applicant/licensee was responsible for assessing its plant's operating history and demonstrating that the approved version of MRP-227 was applicable to its facility. The applicant/licensee action item states that

"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."

Therefore, each applicant/licensee that references [1b] or one of its successor versions is required to assess its plant design and operating history, in order to determine the degree to which the guidance in such a document is applicable to the facility. The applicant/licensee may wish to assert that the assumptions regarding plant design and operating history made in References 10, 11, or 12 are appropriate for their plant and that, therefore, no differences in component inspection categories are required at that plant. The licensee/applicant may also choose to describe the history of the core loading patterns at their plant, documenting when the transition of a high leakage core loading pattern to a low leakage core loading pattern took place. The licensee/applicant should then assert that their facility is bounded by the assumptions in the relevant documents. Finally, the licensee/applicant should affirm that their facility has always operated as a base-load plant that operates at fixed power levels without varying those levels on a calendar or load demand schedule.

5.2 Applicant/Licensee Action Item 2 (SE Section 4.2.2)

Reference 1b and its successor versions do not provide guidance for all PWR internals components within the scope of license renewal – only those PWR internals components that were screened in as being potentially susceptible to one or more of the eight age-related degradation mechanisms and their effects. The SE expressed some concern that perhaps Reference 1b had not fully considered every possible PWR internals component, and that each licensee/applicant would not be able to determine that one or more PWR internals components within the scope of license renewal was not covered by the guidance in Reference 1b. Therefore, an action item was added by the SE to require that each licensee/applicant referencing [1b] review either Reference 10 or 12, as appropriate, to assure that all PWR internals components within the scope of license renewal at their facility are listed in Tables 4-1 and 4-2 in Reference 10, or Tables 4-4 and 4-5 of Reference 12, whichever is applicable to their facility. If, for some reason, the tables in the relevant supporting document do not identify all the PWR internals components that are within the scope of license renewal for its facility, the applicant or licensee is required to:

" 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."

In order to respond to this requirement, the licensee/applicant should compare the listing of PWR internals components within the scope of license renewal at their facility with the corresponding tables in References 10 or 12, as appropriate, and assert whether the listed PWR internals components agree.

5.3 Applicant/Licensee Action Item 3 (SE Section 4.2.3)

Reference 1b identified three sets of PWR internals components -- CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (see Section 4.3.2 in Reference 1b), and Westinghouse guide tube support pins (split pins) (see Section 4.3.3 in Reference 1b) – for which plant-specific evaluation of an existing program is needed to verify the continued acceptability of that program, or to identify changes to the that program that should be implemented to manage the aging of these components for the period of extended operation. Reference 1b referred to plant-specific evaluation for these programs because, although plant-specific programs were known to be in place for the management of aging effects for these components, the level of detail needed for generic evaluation was not available.

As a result, the SE chose to call out this plant-specific actions as a licensee/applicant action that would include the following, as appropriate:

"……applicants/licensees ……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, Revision 0), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227, Revision 0)."

In addition, this licensee/applicant action item stipulates that:

"CE fuel alignment pins are susceptible to IASCC, wear, fatigue, irradiation embrittlement, and irradiation-enhanced stress relaxation. The applicants/licensees shall evaluate the adequacy of their plant-specific existing program with respect to CE fuel alignment pins and ensure that the synergistic effects of aforementioned degradation mechanisms are adequately monitored during the extended period of operation."

For Westinghouse or CE-designed plants, this licensee/applicant action item requires some level of plant-specific assessment of their existing program for the listed components. As an example, for Westinghouse plants that have replaced guide tube support pins, the following paragraph could be used to satisfy the action item:

"At Plant X, the original X750 guide tube support pins (split pins) were replaced in xxxx with an improved X750 Revision B material made from more selective material with more continuous carbide coverage grain boundaries and tighter quality controls, to provide greater resistance to stress corrosion cracking," or "At Plant Y, the original X750 guide tube support pins (split pins) were replaced in yyyy with cold-worked 316 stainless steel that is a significant improvement over the X750."

5.4 Applicant/Licensee Action Item 4 (SE Section 4.2.4)

Reference 1b identified the core support structure upper flange weld for B&W-designed plants as a PWR internals component that required No Additional Measures to supplement any existing plant inservice inspection commitments based on the ASME Code Section XI. The reason for this finding was that this particular weld was below the screening criteria for all aging degradation mechanisms, including SCC, because the applied stress on this component is low and weld residual stresses have been alleviated by a stress relief heat treatment during the original fabrication. This finding was quite different from the upper flange welds in Westinghouse and CE-design plants, which were designated in Reference 1b as Primary components requiring augmented examinations. The SE accepted the technical argument, but concluded that each B&W-designed plant applicant/licensee was responsible to confirm the accuracy of the finding for its particular facility.

Therefore, the SE requires that each B&W-designed plant applicant/licensee referencing [1b] shall:

"confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the RPV 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."

No action is required by Westinghouse and CE-designed plant applicants/licensees. However, for B&W-designed plant applicant/licensees, fabrication records and other documentation should be reviewed, in order to confirm the applicability of the findings in Reference 1b to their facilities, and a declaration to that effect should be included in the facility AMP for internals.

5.5 Applicant/Licensee Action Item 5 (SE Section 4.2.5)

Reference 1b identified periodic physical measurements as a method for managing aging effects for some PWR internals components. In particular, physical measurements were specified for monitoring the potential loss of compressibility in Westinghouse-designed plant hold down springs; and for measuring gap distortion caused by void swelling between the top and bottom core shroud segments in CE-designed plants with core barrel shrouds assembled in two vertical sections. The SE expressed a concern that no specific acceptance criteria for the periodic physical measurements were provided in Reference 1b. In addition, the SE expressed a concern that the baffle-to-baffle bolts and the core barrel-to-former bolts for B&W-designed plants were classified as "Expansion" components in Reference 1b, subject to further evaluation depending upon the examination results for their associated "Primary" components, the baffle-to-former bolts. However, because of inaccessibility issues, no examination requirements are provided in Reference 1b for the "Expansion" components, with structural integrity determined based on evaluation or replacement. The SE indicates that some form of physical measurements or some type of novel examination would be a preferred alternative.

Therefore, the SE requires that applicant/licensees

"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 as part of their submittal to apply the approved version of MRP-227."

In addition, for B&W-designed plants, the applicants/licensees are required to:

"perform a plant-specific analysis on the effect of loss of closure integrity on the functionality of the core barrel assembly and propose physical measurements or examinations, if necessary, to confirm that adequate closure integrity will be maintained over the period of extended operation."

For Westinghouse and CE-designed plants, applicants/licensees should develop plant-specific acceptance criteria for the physical measurements to be made on hold down springs and core shroud segment gap distortion prior to the first required set of physical measurements, and should explain how the proposed acceptance criteria are consistent with the plant licensing basis and the need to maintain the functionality of the component under all licensing basis conditions. This information should be submitted to the NRC as part of the submittal to apply the approved version of MRP-227.

For B&W-designed plants, should the "Primary" examination of baffle-to-former bolts indicate the need to elevate the associated "Expansion" components into the inspection sample, a plant-specific analysis to evaluate the effect of loss of preload on closure structural integrity is the firsts step in determining the need for some degree of examination or alternative physical measurement (or possible bolt replacement) to assure continuing closure integrity.

5.6 Applicant/Licensee Action Item 6 (SE Section 4.2.6)

The guidance in Reference 1b identified several PWR internals components for B&W-designed plants as "Expansion" components with no examination requirements, due to inaccessibility. These components include the core barrel cylinder and its vertical and circumferential seam welds, the former plates, the external baffle-to-baffle bolts and their locking devices, and the core barrel-to-former bolts and their locking devices. Reference 1 also identified the core support shield vent valve disc shafts or hinge pins as inaccessible "Primary" components, also with no examination requirements.

The SE requires that B&W-designed plant applicants/licensees:

"justify the acceptability of these components (B&W core barrel cylinders and their 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 support shield vent valve disc shafts or hinge pins) for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components for NRC review and approval."

While this action item does not apply to Westinghouse or CE-designed plants, B&W-designed plants will be required to submit the evaluation described in the SE, or provide a component replacement schedule.

5.7 Applicant/Licensee Action Item 7 (SE Section 4.2.7)

The guidance in Reference 1b requires plant-specific analysis in order to demonstrate continued structural integrity and functionality during the period of extended operation for a number of cast austenitic stainless steel components. For example, in-core monitoring instrumentation (IMI) guide tube assembly spiders in B&W-designed plants, which are "Primary" components; CRGT assembly spacer castings in B&W-designed plants, which are "Expansion" components; lower support columns in CE-designed plants, which are "Primary" components; and lower support column bodies in Westinghouse-designed plants, which are "Expansion" components, all require such plant-specific analysis. In the first two cases, the purpose of the analysis is to determine the number of spiders or the number of spacer castings needed to maintain operability, while the last two cases require an analytical demonstration of continued functionality. The SE has formalized the plant-specific requirements as follows:

"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. These analyses should also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement. 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."

The appropriate plant commitment would be to confirm that plant specific analyses to demonstrate that the cast austenitic stainless steel {name the components} will maintain their functionality during the period of extended operation, and will consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement. The analyses will be consistent with the plant licensing basis and the need to maintain the functionality of the {name the components} under all licensing basis conditions of operation. The plant should commit to submit this information to the NRC as part of the submittal to apply the approved version of MRP-227.

5.8 Applicant/Licensee Action Item 8 (SE Section 4.2.8)

Applicant/Licensee Action Item 8 requires the formal submittal to the NRC review and approval of the plan to implement the approved version of Reference 1b. The formal requirement is:

"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."

Section 3.5.1 of the SE identifies at least two items to be included in that submittal, as listed below:

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

2. To ensure the MRP-227, Revision 0 program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components."

Therefore, the submittal may use the information contained in this document to prepare the AMP (see Section 3 of this document) and the inspection plan that implements the guidance in Reference 1b or its successor versions, as approved by NRC. As indicated by the above, any deviations from Reference 1b are to be documented in the inspection plan.

It should be pointed out that Section 3.5.1 of the SE also covers three other items outside the scope of this document – updating the facility FSAR to incorporate the Reference 1b activities, as appropriate; Technical Specification changes needed to manage the effects of aging for PWR internals; and any time-limited aging analyses (TLAAs) that apply to PWR internals.

6
SUMMARY

This template provides the information necessary for developing either a license renewal aging management program (AMP) submittal, in conjunction with supplementary tables, figures, and text obtained directly from MRP-227-A [1b] and MRP-228 [2], or their NRC-approved successor documents. In addition to the background discussion in Section 1 and the process discussion in Section 2, the template includes in Section 3 an amplified version of Chapter XI.M16A of Reference 6, including text that can be used directly for the actual AMP itself, covering nine of the ten required AMP attributes. Then, Section 4 contains an amplified version of operating experience discussion that can be combined with plant-specific operating experience to cover the tenth required AMP attribute. Finally, Section 5 contains both a discussion of the eight Licensee/Applicant Action Items that are required to be addressed in the AMP, as outlined in the NRC staff SE for Reference 1b, as well as suggested text to respond to those requirements.

The template is considered sufficiently general to also aid licensees in the preparation of an inspection plan that can be submitted to the NRC staff prior to implementation of the guidance contained in MRP-227-A or its successor versions. While the template emphasizes the AMP attributes, the information contained therein, when combined with the appropriate tables, figures, and text from References 1b and 2, can be used to form the essential bases for the inspection plan.

7 REFERENCES

[1a]. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 0)*, Electric Power Research Institute, Palo Alto, CA: 2008. 1016596. (ADAMS Accession Number ML090160204).

[1b]. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*, Electric Power Research Institute, Palo Alto, CA: December 2011. 1022863. (ADAMS Accession Numbers ML12017A191, ML12017A193, ML12017A194, ML12017A196, and ML12017A197).

[2]. *Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228)*. Electric Power Research Institute, Palo Alto, CA: 2009. 1016609. (Non-publicly available ADAMS Accession Number ML092120574; the non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number ML092750569).

[3a]. *Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines*," U.S. Nuclear Regulatory Commission, Washington, DC, June 22, 2011 (ADAMS Accession Number ML111600498).

[3b]. *Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*," U.S. Nuclear Regulatory Commission, Washington, DC, December 16, 2011 (ADAMS Accession Number ML11308A770).

[4]. *NRC Regulatory Issue Summary (RIS) 2011-07, License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management*, U.S. Nuclear Regulatory Commission, Washington, DC, July 21, 2011 (ADAMS Accession Number ML111990086).

[5]. *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, NUREG-1800, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, December 2010 (ADAMS Accession Number ML103490036).

[6]. *Generic Aging Lessons Learned (GALL) Report*, NUREG-1801, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, December 2010 (ADAMS Accession Number ML103490041).

[7]. Draft LR-ISG-2011-04, *Updated Aging Management Criteria for PWR Reactor Vessel Internal Components*, Appendices A and B, U.S. Nuclear Regulatory Commission, Washington, DC, March 2012 (ADAMS Accession Numbers ML12004A149 and ML12004A150).

[8]. WCAP-17096, *Reactor Internals Acceptance Criteria Methodology and Data Requirements*, PWROG, July 2009.

[9]. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)*. EPRI, Palo Alto, CA: 2005. 1012081.

[10]. *Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals (MRP-189-Rev. 1)*. EPRI, Palo Alto, CA: 2009. 1018292.

[11]. *Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)*. EPRI, Palo Alto, CA: 2006. 1013233.

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

[13]. *Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals (MRP-229)*. EPRI, Palo Alto, CA: 2008. 1016598.

[14]. *Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (MRP-230)*. EPRI, Palo Alto, CA: 2008. 1016597.

[15]. *Materials Reliability Program: Aging Management Strategies for B&W PWR Internals (MRP-231)*. EPRI, Palo Alto, CA: 2008. 1016592.

[16]. *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232)*. EPRI, Palo Alto, CA: 2008. 016593.

[17]. ASME Boiler & Pressure Vessel Code, Section XI, Division 1, "*Rules for Inservice Inspection of Nuclear Power Plant Components*," The American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004 (as approved in 10 CFR 50.55a).

[18]. EPRI TR-1014986, *PWR Primary Water Chemistry Guidelines*, Volume 1, Revision 6, Electric Power Research Institute, Palo Alto, CA, December 2007. (Non-publicly available ADAMS Accession Number ML081140278). The non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number ML081230449.

[19]. ASME Boiler & Pressure Vessel Code, Section V, *Nondestructive Examination*, American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.

[20]. 10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Pla*nts, Office of the Federal Register, National Archives and Records Administration, 2009.

[21]. EPRI TR-102134, PWR Secondary Water Chemistry Guideline-Revision 3, Electric Power Research Institute, Palo Alto, CA, May 1993.

[22]. 10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

[23]. Appendix D: Materials Guidelines: Implementation Protocol, in *"Guidelines for the Management of Materials Issues," NEI 03-08, Nuclear Energy Institute, Washington, DC, Latest Edition*.

[24]. *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in Reactor Internals (MRP-134)*. EPRI, Palo Alto, CA: 2005. 1008203.

[25]. B&W Report No. BAW-2248, *Demonstration of the Management of Aging Effects for the Reactor Vessel Internals*, Framatome Technologies (now AREVA Technologies), Lynchburg VA, July 1997. (NRC Microfiche Accession Number A0076, Microfiche Pages 001 - 108).

[26]. Westinghouse Non-Proprietary Class 3 Report No. WCAP-14577, Revision 1-A, *License Renewal Evaluation: Aging Management for Reactor Internals*, Westinghouse Electric Company, Pittsburgh, PA, March 2001. Report was submitted to the NRC Document Control Desk in a letter dated April 9, 2001. (ADAMS Accession Number ML011080790).

[27]. NRC RAI No. 11 in the *NRC's Request for Additional Information* to the Mr. Christen B. Larson, EPRI MRP on Topical Report MRP-227 dated November 12, 2009.

[28]. NRC Safety Evaluation from C. I. Grimes [NRC] to R. A. Newton [Chairman, Westinghouse Owners Group], *Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled "License Renewal Evaluation: Aging Management for Reactor Internals,"* WCAP-14577, Revision 1, February 10, 2001. (ADAMS Accession Number ML010430375).

[29]. NRC Safety Evaluation from C. I. Grimes [NRC] to W. R. Gray [Framatome Technologies], *Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,"* February 10, 2001. (ADAMS Accession Number ML993490288).

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