Materials Reliability Program: Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals (MRP-153)

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REPORT SUMMARY

This report, a key element in an overall strategy for managing the effects of aging in pressurized water reactor (PWR) internals, describes inspection methods and flaw tolerance evaluations that can be applied to the different categories of internals components.

Background

Management of aging effects, such as loss of material, reduction in fracture toughness, dimensional changes, or cracking, depends on a demonstrated capability to detect, evaluate, and potentially correct conditions that could affect system, structure, or component function. For PWR internals, utilities have identified the general elements of aging effects management programs, including existing in-service inspection and monitoring, with the possibility of enhancement or augmentation depending on future research and development findings. This report describes inspection methods and flaw tolerance evaluations that can be applied to PWR internals components.

Objectives

To establish inspection and flaw evaluation approaches for managing effects of aging in PWR internals.

Approach

The principal investigators first summarized ASME Section XI examination methods and definitions such as VT-3 and VT-1 visual examination and ultrasonic testing (UT) volumetric examination. Subsequently, the investigators described the enhanced VT-1 (EVT-1) examination procedure developed by the EPRI BWR Vessel and Internals Program (BWRVIP) and discussed how this inspection technique could be adapted to PWR internals. Finally, the investigators described limit load analysis, EPFM (elastic-plastic fracture mechanics) and LEFM (linear elastic fracture mechanics) approaches for flaw tolerance evaluation and provided illustrative examples to show their applications.

Results

This report describes inspection and flaw tolerance evaluation approaches that can be applied to PWR internals components having varying degrees of degradation susceptibility, with a particular emphasis on considering degradation effects during extended plant operation. The important elements of the approaches are:

- Enhanced visual examination (EVT-1) for some PWR reactor internals components, based on developments in the BWRVIP

- UT volumetric examination of other PWR reactor internals components, based on the need for assessment of aging effects in locations not accessible for visual examination
- Limit load analysis, EPFM, and LEFM for actual and hypothetical flaw evaluations.

EPRI Perspective

The EPRI MRP Reactor Internals Issue Task Group (RI-ITG) has been conducting studies to develop technical bases to support aging management of PWR internals, with a particular attention to utility license renewal commitments. This Inspection and Flaw Evaluation Strategy report is the third of a three-part document series on an overall strategy for managing the effects of aging in PWR internals. The first document in the series, *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134)* (EPRI report 1008203, June 2005), focuses on the overall framework and strategy. The second document, *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)* (EPRI report 1012081, forthcoming), will detail degradation mechanisms and screening and threshold values.

Based on the strategies developed in these studies, the RI-ITG is focusing on performing screening and functionality and safety evaluation of the effects of degradation in PWR internals components. In parallel, hot cell testing to quantify aged/irradiated materials behavior and performance is continuing. These studies and results, together with the three-part document series on aging management strategy, will provide a basis for developing Inspection and Evaluation (I&E) Guidelines for utility applications.

Keywords

PWR internals
Aging management
Inspection
Flaw evaluation
License renewal

ABSTRACT

Demonstrating that the effects of aging degradation in PWR internals are adequately managed is essential for maintaining a healthy fleet and to assure functionality of core internals components. As part of the EPRI Material Reliability Program (MRP) of the Reactor Internals Issue Task Group (RI-ITG), this report is a key element in an overall strategy for managing the effects of aging in PWR internals through the use of knowledge of internals design, materials, and material toughness properties and the application of screening methodologies for known aging mechanisms. The report describes inspection methods and flaw tolerance evaluations that can be applied to the different categories of internals components. The categorization depends on an initial screening for susceptibility and functionality of the components. Two related MRP documents are *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134)* (EPRI report 1008203), and *Reactor Vessel Internals Aging Degradation Mechanism Screening and Threshold Values* (EPRI report 1012081). The strategy described in these reports incorporates existing knowledge of design, materials, and degradation mechanisms from available research programs including the EPRI MRP RI-ITG, Owners Group programs, and the EPRI BWR Vessel and Internals Program (BWRVIP) for BWR internals.

Other key results from the EPRI MRP program will focus on aging mechanisms and screening for susceptibility and will provide more detailed functionality evaluations of the effects of aging degradation on PWR internals components. Additional data for PWR conditions are expected in a number of areas, including crack initiation, crack growth, fracture toughness, and void swelling; and this information will provide additional insight into the degradation mechanisms and will directly impact the final inspection and flaw evaluation guidelines and acceptance criteria to be developed for PWR internals aging management.

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INTRODUCTION

Management of aging effects, such as loss of material, reduction in fracture toughness, dimensional changes, or cracking, depends upon the demonstrated capability to detect, evaluate, and potentially correct conditions that could affect system, structure, or component function. For PWR internals, utilities have identified the general elements of aging effects management programs, including existing inservice inspection and monitoring, with the possibility of enhancement or augmentation depending on ongoing and future research and development findings. For example, a visual examination of removable PWR internals components is performed periodically by each utility as required by their ASME Code Section XI inservice inspection program. The objective of the examination is to detect "relevant conditions," defined in Section XI to include distortion, cracking, loose or missing parts, wear, or corrosion. If a relevant condition is discovered, an evaluation must then follow to determine the effect on functional integrity and, if significant, some form of corrective action must be taken to restore functionality.

The present surveillance techniques required for PWR internals include:

1. Visual (VT-3) examination, in accordance with Examination Category B-N-3 of the ASME Code Section XI, Subsection IWB

2. Loose parts detection monitoring system

3. Reactor coolant system (RCS) chemistry monitoring system

When relevant conditions are detected by the VT-3 examination of Examination Category B-N-3, the ASME Code (Section IWB-3142) provides options for evaluating or correcting the relevant condition, such as:

1. Supplemental examinations (e.g., surface or volumetric examinations) to characterize the indication more accurately,

2. Analytical justification for continued service of the affected component that may involve more frequent examination, or

3. Repair/replacement of the component.

Regulatory review of early license renewal applications [References 3 - 8] called into question the adequacy of these existing surveillance techniques to manage aging effects in PWR internals. In particular, Examination Category B-N-3, with its requirements for visual distance between the

examiner and the component, and for its character recognition height, was thought to be inadequate, with the staff of the U. S. Nuclear Regulatory Commission (NRC) calling for enhanced or augmented examinations for component locations with potentially significant aging effects. It is not clear whether the supplemental ASME Code examinations of relevant conditions were given proper credit in this regulatory determination, since the Code supplementary examinations – if triggered by the detection of a relevant condition – are as or more rigorous than the proposed enhanced or augmented examinations.

This report prepared for the MRP RI-ITG describes preliminary approaches on inservice inspection and flaw evaluation for future development of PWR internals components inspection and evaluation guidelines in conjunction with functionality analysis.

Chapter 2 describes the functions performed by PWR internals that must be shown to continue in the presence of aging degradation effects. Chapter 3 describes the existing inservice examination requirements for PWR internals contained in the ASME Code Section XI. Chapter 4 discusses alternative inservice examination procedures beyond those contained in the ASME Code Section XI for BWR reactor internals as developed by the BWRVIP. Chapter 5 adapts the BWRVIP inservice examination elements to PWR internals inspection options for enhanced or augmented visual examination of internals. Chapter 6 provides step by step flaw evaluation approaches suggested for PWR internals, including recommendations for flaw growth and flaw acceptance criteria based on the state of knowledge. Chapter 7 summarizes the results of this study, with references provided in Chapter 8. Appendix A provides generic standards for visual inspection of reactor internals components. Appendix B contains sample flaw tolerance evaluations for PWR internals support structures based on the 1999 state of knowledge. Appendix C provides a list of acronyms, as used in this report, and a glossary of terms for visual examination of PWR internals.

2
BACKGROUND

2.1 PWR Internals Functions

The reactor internals are designed to perform several functions:

1. Provide support and orientation of the reactor core (i.e., fuel assemblies).
2. Provide support, orientation, guidance and protection of the rod control cluster assemblies (RCCA) in Westinghouse plants. These are referred to in the Combustion Engineering and Babcock & Wilcox plants, respectively, as control element assemblies (CEA) and control rod assemblies (CRA).
3. Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
4. Provide a passageway for support, guidance, and protection for in-vessel/core instrumentation.
5. Provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.
6. Provide gamma and neutron shielding for the reactor vessel.

The fuel assemblies rest on the lower support structure of the lower assembly, which transfers the resulting load to the core barrel and then to the core barrel flange which rests on the reactor vessel flange. The upper assembly is clamped under the reactor vessel head flange and provides the upper structure interface with the fuel assemblies. During refueling operations, the upper assembly is removed from the reactor vessel to allow access to the fuel assemblies. This provides an opportunity to perform inspections of the upper internals components. The core barrel also provides a flow boundary for the reactor coolant as illustrated in Figure 2-1.

When the primary coolant enters the reactor vessel, it impinges on the side of the core barrel and is directed downward through the annulus formed by the gap between the outside diameter of the core barrel and the inside diameter of the vessel. The flow then enters the lower plenum between the bottom of the lower support plate and the vessel bottom head and is redirected upward through the core. After passing through the core, the coolant enters the upper core support region and then proceeds outward through the reactor vessel outlet nozzles. The perforations in the various components, such as the lower support, control and distribute the flow to the core. In some reactor internals designs, a small amount of bypass flow is allowed to enter the vessel closure head plenum for cooling purposes.

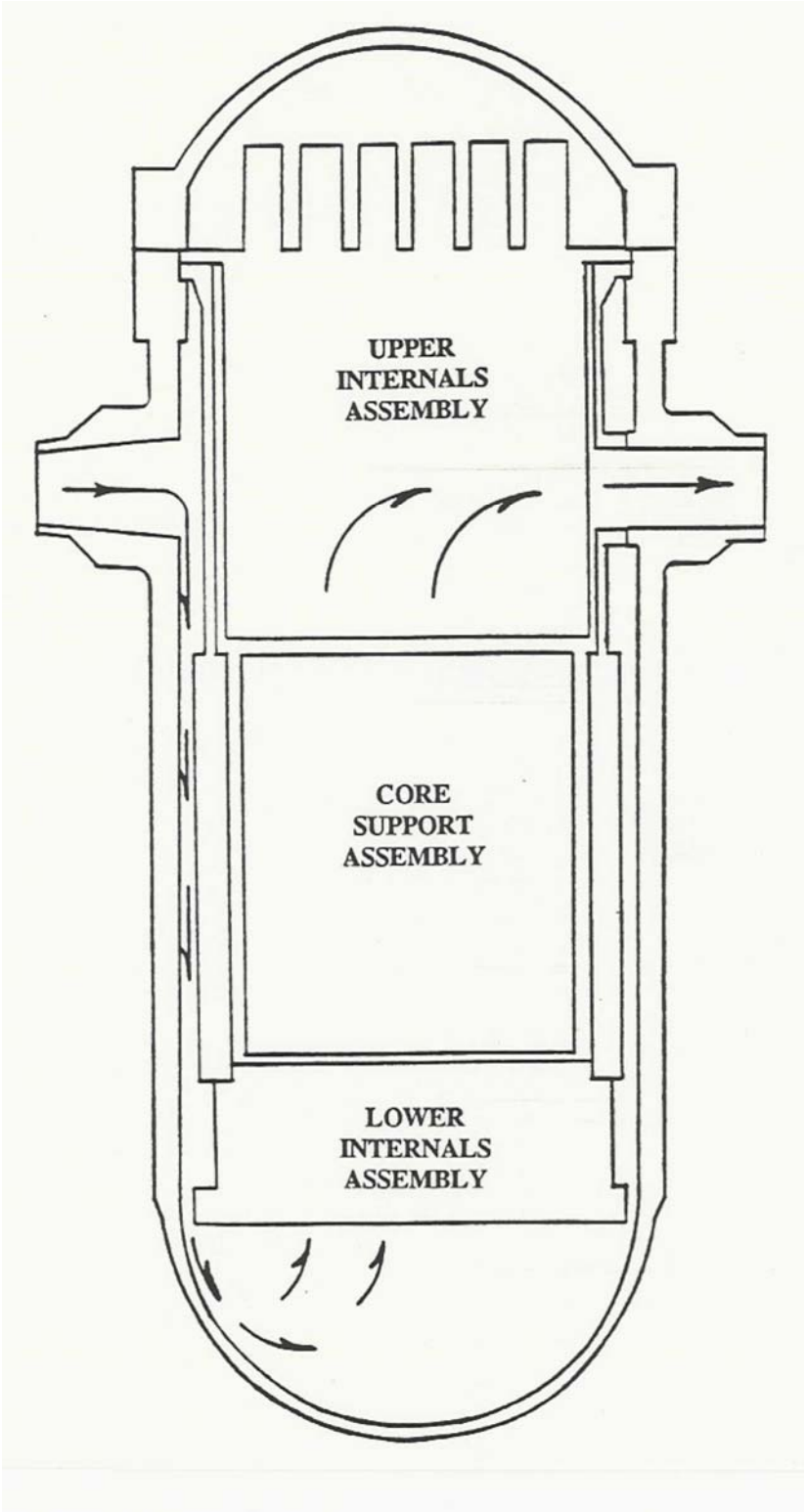

Figure 2-1
PWR RPV Internals Structural Assembly Groupings

Before the development of the ASME Code requirements specifically applicable to reactor internals, the design of reactor internals was based on criteria specific to each vendor. However, Section III of the ASME Boiler and Pressure Vessel Code was used as a guideline for the design criteria for the reactor vessel internals. PWR internals, whose contract dates followed the issuance of the 1974 Edition the ASME Code Section III, were designed to satisfy Subsection NG, Core Support Structures. Among the requirements contained in Subsection NG are rules for fatigue evaluation and categorization of internals loads. The rules for elevated temperature service of metals whose temperatures exceed the ASME Section III allowables are in Code Case N-201.

2.2 Categorization of PWR Internals Components

Four categories of components are considered for classification of the significance for susceptibility to aging effects. The categories described here are defined in MRP-134 [1], and the complete definitions of the categorization are contained in that reference. These categories are based on the significance of the aging effects and will be related to the type of inspections to be used for managing the effects:

Category A

Category A components are those for which aging effects are below the screening criteria, so that aging degradation significance is minimal. Typically, only the required ASME B&PV Code Section XI Examination Category B-N-3 ISI visual examinations (VT-3) will be performed on these components to assess potential aging effects.

Category C

Category C PWR internals components are those “lead” components for which aging effects are above screening levels, which have moderate or high susceptibility to degradation. Enhanced inspections (e.g., Enhanced VT-1, UT, etc.) and/or surveillance sampling will typically be warranted to assess aging effects and verify functionality of these components.

Category B

Category B includes those PWR internals components that are moderately susceptible to the aging effects, such that the effects on function cannot easily be dispositioned by screening and are not “lead” components. Category B components may require additional evaluations to be shown tolerant of the aging effects with no loss of functionality (i.e., damage tolerant).

Category B'

Category B' components are those "lead” components that can be shown to be tolerant of the aging effects through a functionality assessment. These components are candidates for an expanded inspection program.

Given these categories for grouping or "binning" of the PWR internals components, a process was developed to identify the aging degradation significance as a key step in developing inspection guidelines for PWR internals. The steps in this process are shown in Figure 2-2, and are described in greater detail in Reference 1.

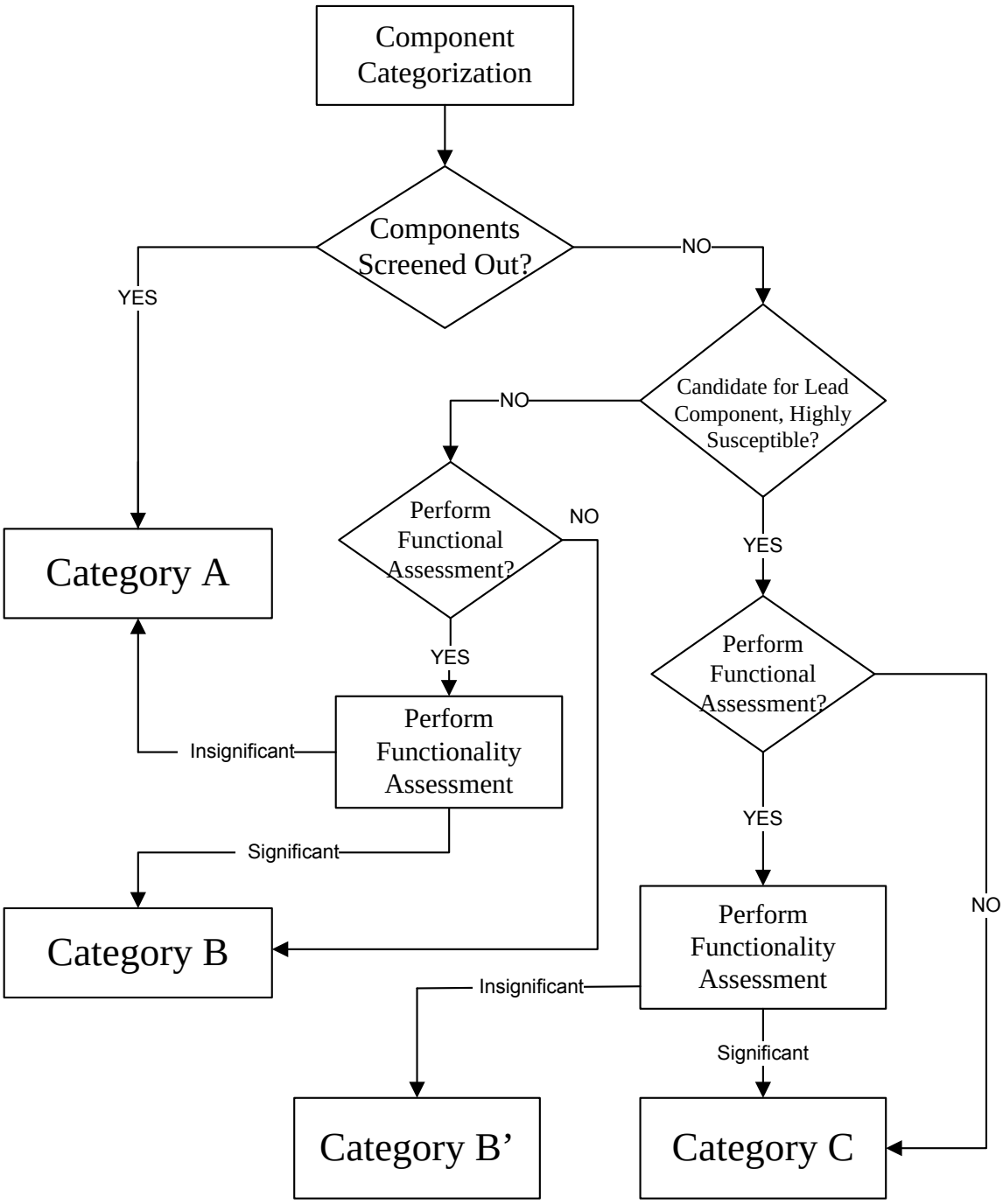

Figure 2-2
Process for Categorization of PWR Internals Components

Note: See MRP-134 [1] for detailed discussion

3
ASME SECTION XI EXAMINATIONS

Visual examinations and flaw evaluations are accepted elements of nuclear power plant component inservice examination programs conducted in accordance with Section XI of the ASME Boiler & Pressure Vessel Code [9], both for cases involving the evaluation of actual flaws detected by the inservice inspections [10] and for postulated flaws [11]. In addition, a form of flaw tolerance involving a postulated (or reference) flaw is included in a non-mandatory appendix [12] to the construction code for nuclear power plant components in the U.S., as a means to address the potential for fast fracture of pressure vessels.

3.1 Existing Section XI Visual Examinations

ASME Section XI visual examinations are relied upon for detection of mature cracks in a variety of systems, components, and structures at commercial nuclear power plants. This is particularly true for PWR internals components that may be subject to a variety of potential cracking mechanisms, whether assisted by irradiation or not. Table IWB-2500-1 of Section XI lists the inservice inspection requirements. ASME Section XI Examination Category B-N-1 calls for the visual examination (VT-3) of accessible areas of the reactor vessel interior surface during each refueling outage. ASME Section XI Examination Category B-N-2 calls for visual examination (VT-1) for accessible attachment welds within the vessel beltline region, with VT-3 visual examination for accessible attachment welds beyond the beltline region of interior attachment welds. The periodicity of these B-N-2 examinations is approximately every ten years. These visual examinations include the attachment welds themselves and one-half inch of the base metal surface adjacent to the weld.

Finally, and of most importance for this discussion, ASME Section XI Examination Category B-N-3 calls for visual (VT-3) examination of the accessible surfaces of removable PWR core support structures. It must be emphasized that this examination addresses only accessible surfaces of PWR core support structures that have been removed from the reactor vessel for the examination. The periodicity of these examinations is also on the order of ten years.

ASME Section XI IWB-3520 provides the acceptance criteria for these inservice inspections. In particular, IWB-3520.1 and IWB-3520.2 list the relevant conditions for the VT-1 and VT-3 visual examinations. A relevant condition is a condition observed during a visual examination that requires supplemental examination, some form of corrective measure (e.g., correction by repair/replacement activities), or analytical evaluation (IWA-9000). Relevant conditions for the VT-1 examination include “crack-like surface flaws on the welds joining the attachment to the vessel wall that exceed the allowable linear flaw standards of IWB-3510.” Relevant conditions

for the VT-3 examination include "loose, missing, cracked, or fractured parts, bolting, or fasteners."

ASME Section XI IWB-3142.1 stipulates that *any* relevant condition is unacceptable unless either a supplemental examination (surface or volumetric) shows that the condition meets Section XI limits, or that the relevant condition is corrected by a repair/replacement activity, or that an analytical evaluation demonstrates acceptability. Note that a supplemental examination is *not* an augmented or an enhanced visual examination. Note also that acceptance by analytical evaluation involves successive re-examinations, in accordance with IWB-2420, to assure that the relevant condition is not deteriorating. IWB-2420 (b) stipulates that, when a component is accepted for continued service based on analytical evaluation, the areas containing flaws or relevant conditions must be reexamined during the next three inspection opportunities (e.g. three subsequent refueling outages). If the reexaminations show that the flaws or relevant conditions remain essentially unchanged for those three successive inspection intervals, the component examination schedule may revert to the original (ten-year) schedule of successive inspections. Note that these reexaminations are required even when the analytical evaluation shows that the flaw remains acceptable, based on flaw growth analysis (see Chapter 6 of this report), for the complete nominal ten-year inspection interval. A potential option will be considered that would allow the reexamination interval to be determined by analysis, based on component location flaw tolerance, functional robustness, and conservative degradation rate assumptions for specific PWR reactor internals component locations.

3.2 Visual Examination Regulatory Concerns

As the result of technical evaluations related to license renewal, the need for specific augmentations of existing visual examination requirements for stainless steel internals components for PWRs has been identified. These specific augmentations would address the perceived deficiencies in the existing ASME Code Section XI requirements for PWR internals – a visual (VT-3) examination of accessible surfaces of removable core support structures. The perceived deficiencies are apparently [13, 14] based on the nominal standoff distances for the VT-3 examination and on the associated demonstration of ability to recognize characters with a prescribed nominal height. In addition, the perceived deficiencies may involve flaw tolerance, or fitness-for-service, demonstrations that rely on the relationship between the frequency/coverage of inservice examinations (and, therefore, on inservice inspection detection sensitivity); the reference flaw location, orientation and size; service loads expected to occur during the period of operation between examinations; any growth of the reference flaw during this interval, based on appropriate crack growth rates for PWR environments; and the critical flaw size that serves as a surrogate for component failure.

The NRC staff in their Safety Evaluation Reports (SERs) have not been willing to permit full credit to license renewal applicants for periodic, continuing VT-3 visual examinations of PWR internals components as the basis for managing all types of cracking during the license renewal term. Instead, in many cases, the staff have requested that the utility adopt enhanced or augmented inservice inspection programs, such as upgrading VT-3 visual examinations to VT-1 visual examinations or enhancement of VT-1 inspections, as a part of the license renewal process.

The apparent source of the staff's major concern is the capability of a visual examination to detect cracking, even for cases when the component is tolerant of all but the very largest flaws. That concern is related to the less rigorous distance, character recognition, and lighting requirements of the VT-3 visual examination, in comparison to those for a VT-1 visual examination. These differences are reflected in the less prescriptive relevant conditions for VT-3 versus VT-1, especially for the detection of surface cracking.

The visual acuity and maximum direct examination distance requirements for the VT-1 and VT-3 visual examinations are given in Table IWA-2210-1, which is duplicated in the table below.

Table 3-1
Table IWA-2210-1 Visual Examinations

Visual Examination	Minimum Illumination f_c	Maximum Direct Examination Distance, ft (mm)	Maximum Procedure Demonstration Lower Case Character Height, in. (mm)
VT-1	50	2 (610)	0.044 (1.11)
VT-3	50	N/A	0.105 (2.66)

The maximum direct examination distance for VT-1 visual examination is given as 2 feet (610 mm), with the character recognition heights for the two methods given as 0.044 and 0.105 inches (1.11 mm and 2.66 mm), respectively. There are no direct visual examination distance requirements for VT-3 visual examination, provided that the examiner is able to satisfy the character recognition requirements specified in Table IWA-2210-1. In other words, VT-1 visual examinations require the observer to recognize smaller objects, by a factor of 3. It should be pointed out that the distances listed in Table IWA-2210-1 are the ***maximum*** for direct examination, and that a closer distance can be used. Since the regulatory concern is the visual examination of ***accessible*** surfaces of ***removable*** PWR core support structures, proximity to the accessible surface should not be an issue. In fact, IWA-2210 specifies "Visual examinations shall be conducted in accordance with Section V, Article 9, Table IWA-2210-1, and the following." Section V, Article 9, T-952 says "Direct visual examination may usually be made when access is sufficient to place the eye within 24 in. (610 mm) of the surface to be examined …".

Furthermore, IWA-2210(c) states "Remote examination may be substituted for direct examination." Section V, Article 9 defines "remote visual examination" as "a visual examination technique used with visual aids for conditions where the area to be examined is inaccessible for direct visual examination." Section V, Article 9 also defines "enhanced visual examination" as "a visual examination technique using visual aids to improve the viewing capability, e.g., magnifying aids, borescopes, video probes, fiber optics, etc." It should also be pointed out that IWA-2210(c) requires the remote examination procedure to meet the character recognition tests. Therefore, remote visual examination techniques, such as those using cameras or fiber-optic devices, can be substituted for direct visual examinations, but are required to meet the same qualifications.

The visual acuity requirements of Table IWA-2210-1 are not directly related to the length or crack-opening width of a surface-breaking crack that is subject to detection. Instead, the character recognition height should be treated as a requirement that any features on the surface to be inspected be discernible to the eye. A discontinuity on the surface caused by a surface-breaking crack is more readily detectable and recognized than a letter or numerical character of the same dimension. This topic -- the difference between a crack discontinuity and character recognition -- has been discussed in Reference 15, using the results from an earlier EPRI study on ultrasonic detectability of thermal fatigue cracking in BWR feedwater piping.

An excellent example of ASME Code requirements for VT-3 versus VT-1 visual examination is Nuclear Code Case N-481, which provides alternative rules to the Section XI inservice volumetric examinations of reactor coolant pump casings. Code Case N-481 permits the substitution of visual examinations of the internal and external surfaces of the cast austenitic stainless steel pump casings, plus a flaw tolerance evaluation, in lieu of volumetric examination. The external surface inspection is required to meet VT-1 visual examination requirements, while the internal surface inspection is only required to meet VT-3 visual examination requirements, and is only required when the pump is disassembled for maintenance (i.e., only when the internal surface is accessible). The justification of VT-3 for the internal surface examination is the recognition that standoff distances for the examination will be much less than the maximum permitted for VT-3, by necessity.

In summary, the NRC staff do not accept the existing ASME Code Section XI VT-3 visual inservice examination requirements (Examination Category B-N-3) for removable PWR core support structures as the basis for managing cracking during the license renewal term. Augmented or enhanced visual examination will be required. The alternatives for such augmented or enhanced visual examinations are described in Chapter 5 of this report.

3.3 Section XI Surface and Volumetric Examinations

IWB-3200 (b) permits supplemental surface or volumetric examinations to determine the extent of relevant conditions detected by the Examination Category B-N-3 VT-3 visual examinations. This Section XI provision would permit, for example, UT examination of a PWR internals component location, in order to determine the length and depth of a surface-breaking flaw that was detected by visual examination. This provision would also permit the ultrasonic examination of that same component location, in order to size both the length and depth of that same surface-breaking flaw found by visual examination. In addition, IWB-3200 (b) would permit the ultrasonic examination of bolts for which the Examination Category B-N-3 visual VT-3 examination detected “loose, missing, cracked, or fractured” bolting, in accordance with IWB-3520.2 (b).

Another provision of ASME Section XI that permits surface and/or volumetric examinations, in lieu of the requirements of Table IWB-2500-1, is IWA-2240, which stipulates that “Alternative examination methods, a combination of methods, or newly developed methods may be substituted for the methods specified in … this Division, provided the Inspector is satisfied that the results are demonstrated to be equivalent or superior to those of the specified method.”

The provision cited in the above paragraphs offers wide latitude for substituting inservice examination methods for the prescribed methods in Table IWB-2500-1, provided that appropriate equivalence or superiority to the prescribed methods is demonstrated. The application of these provisions to potentially superior inservice examination methods are discussed in more detail in Chapter 5.

4
BWR VESSEL AND INTERNALS PROGRAM

Among the various alternatives for augmented or enhanced visual examination for removable PWR core support structures is the enhanced visual (VT-1) examination standard being implemented by the BWR Vessel and Internals Program (BWRVIP). This standard, which is referred to as EVT-1, has been the subject of extensive development work by the industry for over a decade, and was documented in the original version of BWRVIP-03 [16], and in subsequent revisions to that original document. This alternative has been recognized by the NRC staff as potentially applicable to PWR internals as well. The standard and its supporting information are summarized in the following paragraphs.

4.1 Visual Examination Demonstrations

Comprehensive demonstration of visual examination detection capability has been one of the objectives of the BWRVIP [17]. The demonstration was originally directed at the detection of potential intergranular stress corrosion cracking (IGSCC) in BWR core shrouds, and used crack-like simulations to test the visual acuity of potential examiners. Small (0.0005-inch (0.013 mm) diameter) stainless steel wires were placed 20 feet (6 meters) underwater against various backgrounds and under various lighting conditions. Cameras were used as the remote visual examination device. Note that the wire diameter is very small in comparison to both the VT-1/ VT-3 character recognition heights and to the typical surface crack opening displacement of a mature stress corrosion flaw, as defined in Chapter 6 of this report.

The BWRVIP studies found that detection of the wires was assured from a distance of 16 inches (0.4 m), provided that the lighting was adequate and reflection from the various backgrounds minimized. Contrast was not a concern. Note that the remote camera distance was somewhat less than the maximum examination distance for either VT-1 or VT-3 (e.g., 16 inches (0.4 m) versus 24 inches (0.6 m)). However, the 16-inch (0.4-meter) standoff distance is not untypical of the distance that might be used for a remote (or enhanced) visual examination.

The study found that flaws detected and sized (length) by visual examination tend to be undersized with respect to length, since extreme crack tightness affects the flaw length sizing and the length is only measured over the portion of the flaw that is visible. In addition, poor surface cleaning can also cause underestimation of the flaw length. Landmarks and other surface features assist in length sizing. Reference 17 found that, when the flaws are at least 12 inches (300 mm) long, the error in length sizing is less than 10 %. Figure 4-1 shows the statistical data from the sizing tests.

4.2 BWRVIP Flaw Evaluation Guidelines

In addition to the visual examination demonstrations and the development of an enhanced VT-1 visual examination standard, the BWRVIP has also provided a technical basis for evaluation of flaws detected and sized for length by visual examination, or detected and sized for both length and depth by ultrasonic examination. BWRVIP-76 [19] contains the documentation for both the BWR core shroud inspection and flaw evaluation guidelines. Major elements of the flaw evaluation process are listed below:

5
INSPECTION AND SURVEILLANCE APPROACHES

ASME Section XI Subsection IWB Examination Category B-N-3 provides inservice inspection requirements for accessible surfaces of removable PWR reactor internals classified as core support structures (Class CS). The NRC staff have determined that Examination Category B-N-3 is inadequate, in part, as a program -- or an element of a program -- for managing some of the effects of aging during the license renewal term. In particular, deficiencies in the capability of Examination Category B-N-3 to manage the effects of cracking have been cited in the Generic Aging Lessons Learned (GALL) report [13]. The apparent reason for this inadequacy is due to the relevant conditions cited in IWB-3520.2, compared with those of IWB-3520.1 for VT-1 visual examination, together with the difference between the required character recognition height for VT-1 and VT-3 in Table IWA-2210-1. The character recognition height is three times as large for VT-3 as for VT-1. The implication is that detection and length sizing of crack-like indications using VT-3 visual examination is subject to uncertainty and potentially significant error.

In order to address these concerns, in part, the industry is considering whether or not the basic requirements are to be supplemented, where appropriate, by enhanced visual examination, ultrasonic examination, or other means to monitor and manage aging degradation of reactor internals components.

Similarly, the EPRI MRP RI-ITG is considering whether, for PWR internals component locations that are classified as lead locations (Category C) for potentially significant age-related degradation, enhanced visual examination (EVT-1), supplemented, as necessary, by ultrasonic volumetric examination (UT) – should be used for aging management. For other component locations, either the existing Examination Category B-N-3 visual examination requirements, perhaps supplemented by standard VT-1 visual examination, continue to be adequate. The guidelines for these two aging management alternatives are described in more detail in 5.1 (Enhanced VT-1 Visual Examination) and 5.2 (VT-1 Visual Examination).

These considerations are deemed to be responsive to NRC concerns about the effectiveness of existing ASME Section XI inservice examination programs for PWR reactor internals.

In addition, 5.3 describes recommended inspection and surveillance program elements for managing the effects of cracking and stress relaxation for PWR internals. Chapter 5.4 recommends methods for determining the frequency of inspection and surveillance.

5.1 Enhanced VT-1 Visual Examination

For the accessible surfaces of removable PWR internals components subject to relatively high service or residual stresses, including relatively high preload stresses, enhanced visual examination (EVT-1) is capable of surface-breaking crack detection and sizing without excessive uncertainty. The definition of this type of enhanced visual examination is not that of the ASME Code Section V, Article 9, but instead follows the definition given in BWRVIP-03 [16]. Enhanced visual examination (EVT-1), as defined in BWRVIP-03, is a visual examination method where the equipment and the environmental conditions are such that the detection of a 1/2 mil (0.0005 inches or 0.0127 mm) target can be demonstrated (see Chapter 4).

That detection resolution is demonstrated through the application of the Sensitivity, Resolution and Contrast Standard (SRCS) prepared by the BWRVIP and published in BWRVIP-03 (see Appendix A of this report for more detail). The critical elements of that standard are repeated here for emphasis.

5.2 VT-1 Visual Examination

For the accessible surfaces of removable PWR internals components subject to relatively low service or residual stresses, VT-1 visual examination is capable of detecting and assessing the general mechanical condition of exposed surfaces, including surface-breaking crack detection and sizing of mature fatigue cracks. The EPRI MRP RI-ITG is considering whether the following ASME Code Section XI provisions should be followed when carrying out these VT-1 examinations.

1. As required by ASME Code Section XI, the maximum examination distance for the VT-1 visual examination shall be 16 inches (0.4 m).

2. The character recognition height for direct VT-3 visual examination is 0.105 inches (2.66 mm). In view of the selection of 16 inches (0.4 m) as the maximum VT-1 visual examination distance, the character recognition height shall be reduced to 0.044 inches, or 1.11 mm, from 24 inches, or 0.6 m.

5.3 Augmented Inspection/Surveillance

Some component locations for PWR internals are subject to potentially significant age-related degradation effects during the license renewal term. These lead component locations include those that are subject to the potentially significant effects of stress relaxation (loss of preload), void swelling (excessive dimensional change), and cracking (e.g., IASCC). Depending upon the particular lead component and the location within the component, visual examination – whether EVT-1 or VT-1 – may be unable to detect the effect of the age-related degradation. For example, IASCC in baffle/former bolts may occur under the bolt head – in the shank or threaded region – and will be undetectable by visual examination unless the bolt is removed and subject to visual examination over its entire length. Loss of preload in baffle/former bolts may be undetectable by visual examination unless the loss is total (e.g., a loose or broken bolt) and the capturing mechanism is absent. Again, unless the bolt is removed, and the residual preload is estimated or measured during the removal process, the amount of degradation is not known. Finally, the locations within the core baffle structure where void swelling is potentially maximum are found in the so-called re-entrant corners with three immediate neighboring fuel elements and where the neutron dose and temperature due to gamma heating are greatest. The zones that are potentially susceptible to void swelling are quite localized. Swelling is expected to occur first in the solution annealed (Type 304) stainless steel baffle plates followed at significantly higher doses by any adjacent cold worked stainless steel baffle bolts. Thus visual examination for dimensional changes and distortion should be focused particularly on the baffle plates at the most susceptible high dose, high temperature locations. Localized volume increases of the order of 5 %, or greater, are of potential concern. Bolts removed for other reasons may also be examined microscopically for any evidence of the early stages of swelling.

Based upon the above reasoning, aging management of the full range of age-related degradation effects can require more than EVT-1 or VT-1 visual examination.

The ASME Code Section XI permits supplemental and alternative examinations under such circumstances, as discussed in Chapter 3 of this report. For example, IWB-3200 (b) permits supplemental surface or volumetric examinations to determine the extent of relevant conditions detected by the Examination Category B-N-3 VT-3 visual examinations, such as in-situ ultrasonic examination of baffle bolts for which the Examination Category B-N-3 visual VT-3 examination detected "loose, missing, cracked, or fractured" bolting, in accordance with IWB-3520.2 (b). Even in the absence of relevant conditions from a VT-1 or EVT-1 visual examination, IWA-2240 permits alternative surface and/or volumetric examinations, provided that the alternative methods are shown to be "equivalent or superior" to the required visual examinations.

In-situ ultrasonic (UT) methods are potentially capable of detection of substantial manifestations of all three age-related degradation effects (mature cracking under the bolt head, significant dimensional change along the bolt length, and complete loss of preload). However, the demonstration of this capability is subject to two factors:

- The capability to couple transducers to the relatively complex geometries of the bolt heads; and
- Performance demonstration results on baffle/former bolt mockups.

In addition, functional analysis results may show that certain bolts can – in the aggregate – tolerate mature cracking, measurable dimensional change along the bolt length, and complete loss of preload, thereby reducing or potentially eliminating any need for UT examination.

Therefore, the options for managing aging effects in baffle bolts would appear to be: (1) UT examination of a sufficient number of bolts to assure that an acceptable pattern of functional bolts can be demonstrated to exist, possibly coupled with some very limited bolt replacement to achieve an acceptable pattern; or (2) pre-emptive bolt replacement of some prescribed number and pattern of baffle bolts.

With at least some baffle bolt replacement likely, an opportunity could be to establish an integrated surveillance program for the baffle/former bolts. The objective would be to create a comprehensive material testing and evaluation database for the industry through measurements on selected baffle/former bolts removed periodically from operating plants, and to use this information for determining the extent of aging effects on other PWR internals components of the same material composition. Related testing and evaluation has already been performed on bolts removed from three US operating plants. The results from this surveillance program could be integrated with existing and future research data (e.g., IASCC, void swelling, and stress relaxation) being generated through industry sponsored programs.

The integrated surveillance program would entail the removal and laboratory testing of baffle/former (and possible barrel/former) bolts from a cross section of PWR plants (Westinghouse, B&W and perhaps CE plants). The results of the laboratory testing would be used to generate an aging degradation database for a range of environmental conditions. Data being generated through industry sponsored programs would also be added to the aging

degradation database. The database would be used to develop improved aging correlations for assessment of results from functionality and safety analyses.

The surveillance program should consider the full range of range of baffle/former bolting materials and heat treatments used in US PWR plants, with an emphasis on bolting at the upper end of the neutron irradiation and operating temperature exposure range. The elevations and locations from which such bolting samples could optimally be extracted are known generally from plant-specific operating conditions. Hot cell examinations and tests can augment the existing database in terms of engineering material properties and microscopic damage. Both types of measurements are needed for further development and benchmarking of aging degradation models.

Further details of such an integrated baffle/former bolting surveillance program depends upon future activities within the EPRI MRP RI-ITG research and development effort, which will lead to decisions on the need for such a program and its capability to supplement EVT-1 and volumetric examination requirements.

5.4 Frequency of Inspection/Surveillance

The ASME Code Section XI defines the nominal inspection and surveillance intervals for nuclear power plant components in IWA-2430. Regardless of whether Inspection Program A (IWA-2431) or Inspection Program B (IWA-2432) is selected by the utility, the nominal inspection interval is ten years. However, under the provisions of Inspection Program A, the inspections are spread throughout the ten-year period, with the first inspection interval three years after the start of initial plant commercial service, the second inspection interval seven years after initial commercial startup, the third inspection interval after thirteen years, the fourth after seventeen years, and so on. Under the provisions of Inspection Program B, the inspections are performed at ten-year intervals throughout the life of the plant. However, even under the provisions of Inspection Program B, Table IWB-2412-1 lists a schedule of inspection completions by the three-year and seven-year period within an inspection interval. The inspection periods for both Inspection Program A and Inspection Program B are sufficiently flexible to permit coincidence with plant maintenance and refueling outages.

However, these inspection intervals and inspection periods within intervals represent the nominal frequency only. As discussed in Chapter 6 of this report, the frequency of inspections is increased when flaws and relevant conditions are detected and engineering evaluations are required to justify continued operation of the affected components. IWB-2420 (b) requires, when a component is accepted for continued service based on analytical evaluation, the areas containing flaws or relevant conditions must be reexamined during the next three inspection periods. If the reexaminations show that the flaws or relevant conditions remain essentially unchanged for those three successive inspection periods, the component examination schedule may revert to the original (ten-year) schedule of successive inspections.

As discussed in Chapter 6 of this report, the analytical evaluation of flaws and relevant conditions may show that the inspection frequency cannot be sustained at a ten-year interval.

Some PWR internals component locations may contain flaws and be subject to postulated crack growth rates that cause an existing flaw to grow to an unacceptable size in a time period much less than either the next inspection interval, or even – in some extreme cases – less than the next inspection period. The decision in such cases is the choice between permitting operation of the component until the end of the next inspection period, or prematurely shutting the plant down for a repair/replacement activity.

Because of such considerations, the EPRI MRP RI-ITG has not yet developed a firm technical position on any changes in the frequency of inservice examinations relative to existing ASME Code requirements, but will decide on that position prior to the publication of the I&E Guidelines.

5.5 Expansion Criteria for Additional Examinations

Criteria for acceptability of flaws detected during examination will be based on flaw tolerance evaluations. If flaws exceeding the acceptable limits are detected, consideration of expansion criteria for additional examinations beyond the original scope of original examinations may be necessary. No specific recommendations for sample expansion criteria have been developed in this report. The EPRI MRP RI-ITG is taking the issue of sample expansion under consideration, and will decide on sample expansion criteria prior to publication of the I&E Guidelines.

6
PWR INTERNALS FLAW EVALUATION APPROACHES

Reference 16 provides the BWRVIP flaw evaluation guidelines for BWR core shrouds with flaws detected and sized by visual examination, possibly augmented by supplementary volumetric examinations. Chapter 4 of this report provides a summary of those guidelines.

The purpose of the supplementary volumetric examinations recommended by the BWRVIP is to characterize the depth of the flaw. Based upon visual examination alone, only the flaw length can be determined, so that, in the absence of any flaw depth information from supplementary examination, the flaw must be assumed to extend completely through the thickness of the internals component. A supplementary volumetric examination permits actual measured flaw depth to be used in the evaluations.

A similar set of preliminary flaw evaluation approaches is proposed here for application to PWR vessel internals. The evaluation is carried out in seven steps, as described below. Any thresholds, crack growth rates, evaluation criteria or other technical details are based on preliminary available data and provided here for illustration only. These may and will change when all new and existing data are compiled and with further evaluation.

1) The flaw characteristics could be determined, based on the length of any flaws detected and sized during inservice examination, either by VT-3 visual examination (see Appendix C of this report) or by an enhanced visual examination (EVT-1) (see Chapter 5.1 of this report). Because of the potential for flaw length undersizing, the flaw length could be sized for analytical purposes at 110 % of the length determined by VT-3 or enhanced visual examination (EVT-1). The undersizing margin could be increased relative to that for EVT-1, if deemed necessary. The flaw depth could be assumed to be through the thickness of the component, unless supplementary volumetric examination is used to characterize the flaw depth. An alternative approach is to assume a maximum flaw depth, based on fracture mechanics principles.

When a postulated, versus an actual, flaw is to be evaluated, such as for a flaw tolerance evaluation, the length may be assumed, consistent with the length of flaws that have been shown to be detectable by visual examination. In this case, the standard practice for a flaw depth is an aspect ratio of 6 to 1.

2) The neutron irradiation fluence needs to be determined for the component location at or near where the flaw was detected or assumed. Extensive neutron irradiation reduces the ductility, increases the yield and ultimate strengths, and reduces the fracture toughness of austenitic stainless steel material. Therefore, the fluence level needs to be estimated in order to select the method and the mechanical properties to be used in the flaw evaluation.

For accumulated neutron exposure of less than 10^{20} n/cm^2 (E > 1 MeV) (0.14 dpa), the changes in mechanical properties are negligible. When the accumulated exposure reaches $7x10^{20}$ n/cm^2 (E > 1 MeV) (1 dpa), the changes start to become noticeable. Figures 6-1 and 6-2 in MRP 129 [21] show the significant yield strength and uniform elongation reduction at 3 to 5 dpa (2.1 to 3.5 $x10^{21}$ n/cm^2 (E > 1 MeV)).

For accumulated neutron fluence of the order of 3 to 5 dpa (2.1 to 3.5 $x10^{21}$ n/cm^2 (E > 1 MeV)), the ductility remains relatively high, Figure 6-3 in MRP 129 [21]. However, fracture toughness resistance is reduced to very low levels. Figure 6-4 in MRP 160 [22] shows fracture toughness vs fluence data for some PWR irradiated 300 series SS. Figure 6-5 shows the crack growth resistance curve for stainless steel material exposed in service to estimated fluence levels of $8x10^{20}$ n/cm^2 (E > 1 MeV) (1.14 dpa) [23]. Figures 6-6 to 6-8 shows crack growth resistance (J-R) curves of 304 PWR irradiated specimens with various neutron irradiation exposures [22] (MRP 160). The sensitivity of fracture toughness properties with respect to neutron irradiation fluence implies that the determination of component locations within the toughness ranges is a critical step in the evaluation process.

3) The flaw evaluation methodology needs to be selected**.** The general recommendations adopted by BWRVIP are also recommended for PWR internals. For all neutron fluence levels, the flaw needs to satisfy limit load requirements, following procedures similar to those given in the ASME Code Section XI, Appendix C [10]. For neutron fluence levels exceeding $3x10^{20}$ n/cm^2 (E > 1 MeV), either an elastic-plastic fracture mechanics (EPFM) evaluation or a linear elastic fracture mechanics (LEFM) evaluation should be performed, in order to assure continued structural integrity. However, for neutron fluence above $3x10^{20}$ n/cm^2 (E > 1 MeV) but below $3x10^{21}$ n/cm^2 (E > 1 MeV), EPFM would normally be preferred. For neutron fluence above $3x10^{21}$ n/cm^2 (E > 1 MeV), LEFM is recommended.

4) The applied stresses need to be calculated**.** The stresses for the flawed component need to be known or calculated for the design-basis service loadings. These loadings may include expected (ASME Code Service Level A and B) loadings and unexpected (ASME Code Service Level C and D) loadings.

5) The flaw growth during the next inspection interval needs to be calculated. Prior to the limit load and fracture mechanics calculations, the cyclic and time-dependent flaw growth from the current time to the next inservice inspection needs to be calculated. For example, if the inservice inspection interval is ten years, the flaw growth needs to be calculated for a ten-year period. If the end-of-period flaw exceeds limits, the inservice inspection interval may be less than ten years.

6) Limit load requirements need to be satisfied for the flawed component at the end of the current inservice inspection interval for all levels of neutron irradiation exposure. The limit load calculation is carried out to find the critical flaw parameters (location of the cross section neutral axis and the effective flaw length) that cause the cross section to reach its limit load. This proposed criterion will be adopted as a strategy in the interim for PWR applications. No fracture toughness requirements need to be met for neutron fluence exposures less than $3x10^{20}$ n/cm^2 (E > 1 MeV).

7) Fracture toughness requirements need to be satisfied for the flawed component at the end of the current inservice inspection interval. For $3x10^{20}$ n/cm^2 (E > 1 MeV) < neutron fluence exposure < $3x10^{21}$ n/cm^2 (E > 1 MeV), the preferred methodology is that of elastic-plastic fracture mechanics (EPFM). Reference 23 has demonstrated generically, using the crack growth resistance curve of Figure 6-5, that stainless steel components represented by typical geometries satisfy these requirements for fluence levels that are less than or equal to $8x10^{20}$ n/cm^2 (E > 1 MeV) and for applied stress levels that range from 10 ksi to 30 ksi. The geometries covered include columnar supports and toroidal shells with edge cracks and corner cracks. If the component under consideration does not fall within the geometries evaluated in Reference 23, a plant-specific analysis to show that the flaw being evaluated continues to have stable crack growth during the subsequent operating interval is needed.

7
SUMMARY

This report is the third report in the series for recommended supplementary elements to be considered for programs to manage the effects of aging in PWR reactor internals. Figure 7-1 represents a chronological sequence of actions by the EPRI MRP RI-ITG in this regard. The top block describes the overall framework and strategy report, Report 1 in the series, for managing these aging effects [1]. The information blocks on the second tier and, to some extent, on the third tier, refer to other reports that are contemporary with this report. For example, the development of constitutive equations needed for functionality analyses of irradiated internals components and assemblies [24]. Report 2 in the series focuses on degradation mechanisms screening criteria for categorization [2]. All of these steps and strategies are intended to culminate in developing PWR Internals Inspection & Evaluations Guidelines.

The strategy -- as shown in the decision block between the mechanisms and the components -- uses knowledge of internals design, materials and material toughness properties to select (screen) the components that are most affected by aging. The most-affected components are defined in this report to be “lead” components. As shown in the bottom portion of Figure 7-2, the strategy and the preliminary guidance in this report calls for augmented inspections, surveillance, and flaw tolerance evaluations for these lead components. It is recognized that functional assessments and other generic evaluations will most likely reduce the lead component populations, and that this population would also be subject to reduction or enlargement, depending on the findings from the augmented inspection, surveillance, and flaw tolerance evaluation program elements.

Chapter 2 describes the functions performed by PWR internals that must be shown to continue in the presence of aging degradation effects. Chapter 3 describes the existing inservice examination requirements for PWR internals contained in the ASME Code Section XI. Chapter 4 discusses alternative inservice examination procedures beyond those contained in the ASME Code Section XI for BWR reactor internals as developed by the BWRVIP. Chapter 5 provides the preliminary inspection and surveillance program approaches for the lead components.

The preliminary inspection and surveillance studies introduce three new program elements:

1. Enhanced visual examination (EVT-1) for some PWR reactor internals components, based on developments in the BWRVIP;

2. Ultrasonic volumetric examination (UT) of other PWR reactor internals components, based on the need for assessment of aging effects in locations not accessible for visual examination; and

3. An integrated surveillance program for baffle/former bolting, in order to accumulate material degradation data that might be beneficial in the assessment of other PWR internals components fabricated from the same material.

Three open issues that require further consideration by the EPRI MRP RI-ITG were identified in Chapter 5: (1) specific selection of aging management program elements, and the extent to which these program elements should be supplemented by an integrated surveillance program activity; (2) criteria for the expansion of the population of internals locations requiring enhanced examination and surveillance; and (3) justification for relief from ASME Code Section XI requirements for successive examination of known flaw locations, based on flaw tolerance and functional robustness of particular internals component locations. These three open issues will be subject to further study prior to the development of the I&E Guidelines.

Chapter 6 provides preliminary approaches for flaw evaluations. This approach considers a combination of BWRVIP procedures and data specific to PWR reactor water environments. All data specific to PWR reactor water environments including crack growth rates will be selected based on available published data, ongoing testing and planned testing, and their technical bases documented prior to the development of the I&E Guidelines.

The analytical evaluation of flaws and relevant conditions per the approach outlined in Chapter 6 may show that the inspection frequency cannot be sustained at a ten-year interval. Some PWR internals component locations may contain flaws and be subject to postulated crack growth rates that cause an existing flaw to grow to an unacceptable size in a time period much less than either the next inspection interval, or even – in some extreme cases – less than the next inspection period. In such cases, the decision is the choice between permitting operation of the component until the end of the next inspection period, or prematurely shutting the plant down for a repair/replacement activity. Because of such considerations, the EPRI MRP RI-ITG has not yet developed a firm technical position on any changes in the frequency of inservice examinations relative to existing ASME Code requirements, but will decide on that position prior to the publication of the I&E Guidelines.

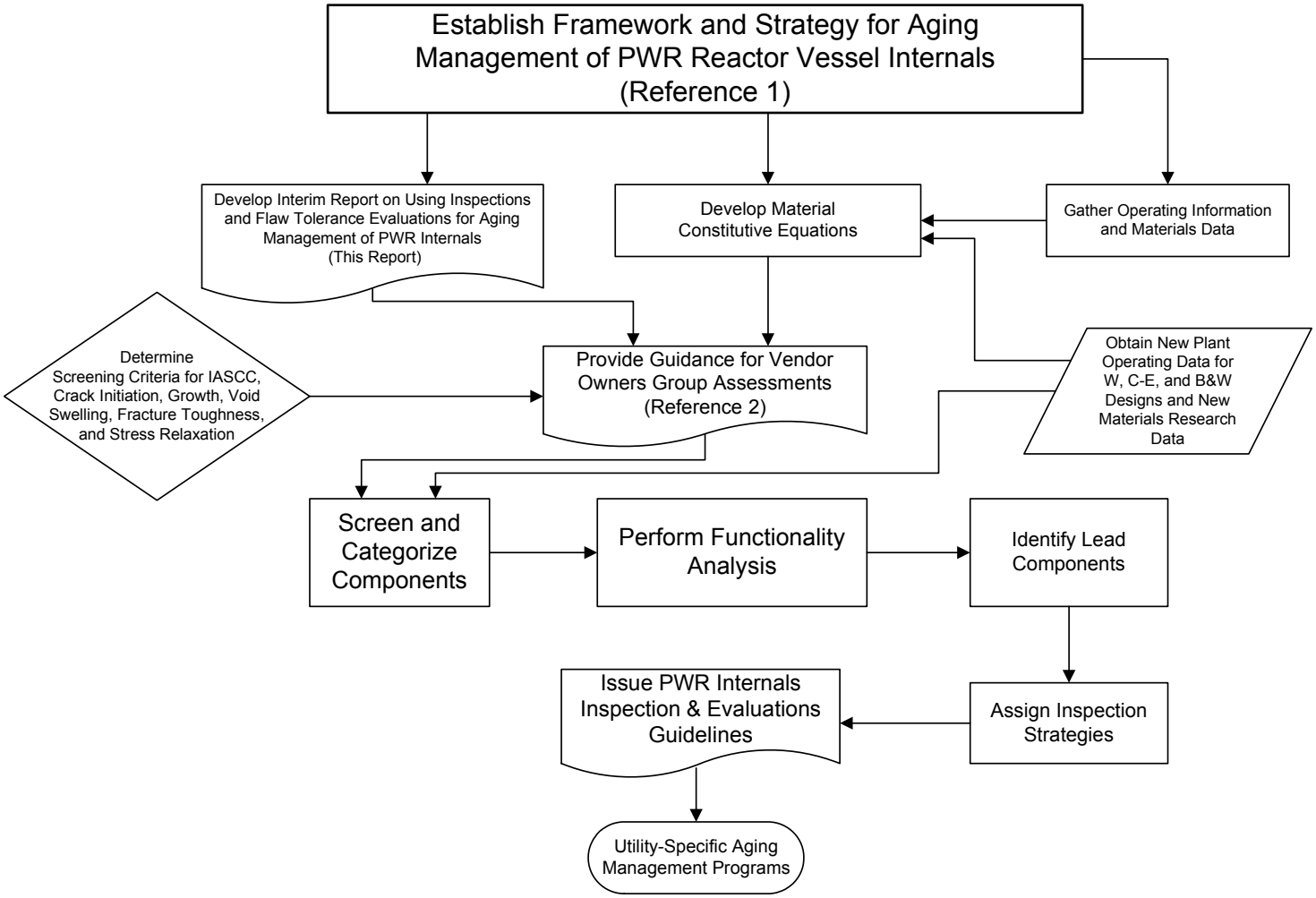

Figure 7-1
Framework for Implementation of Aging Management Using Screening, Functionality Evaluations, and Inspections [1]

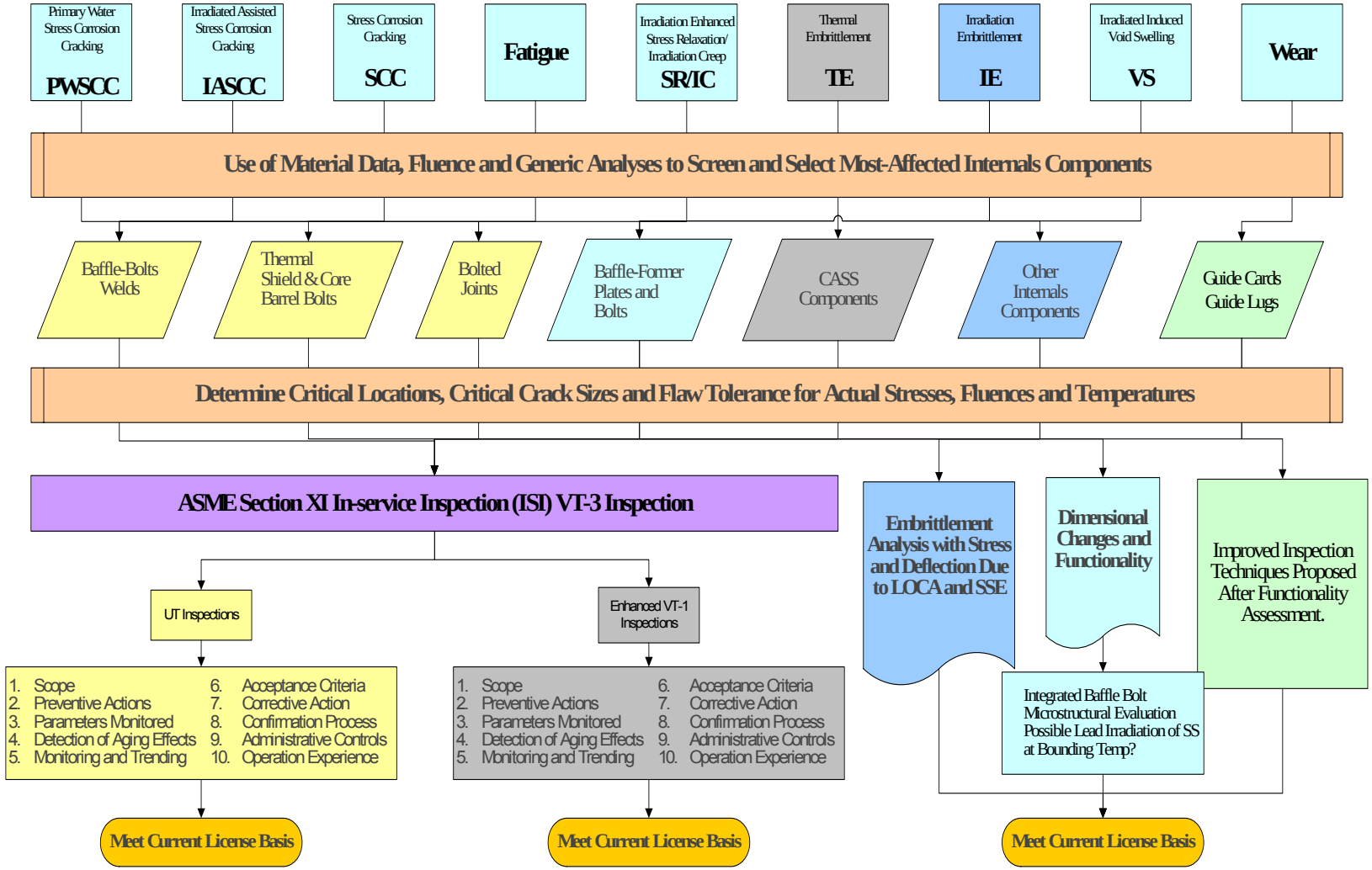

Figure 7-2
Example of Reactor Internals Aging Management Strategy Using Inspections

8
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A

APPENDIX A: GENERIC STANDARDS FOR VISUAL INSPECTION OF REACTOR INTERNALS COMPONENTS

B

APPENDIX B: SAMPLE FLAW TOLERANCE EVALUATIONS FOR PWR INTERNAL SUPPORT STRUCTURES

C
APPENDIX C: ACRONYMS AND GLOSSARY OF TERMS FOR EXAMINATION OF PWR INTERNALS

CLB	Current Licensing Basis
NEI	Nuclear Energy Institute
MRP	Material Reliability Program
RI-ITG	Reactor Internals Issues Task Group
MTAG	Materials Technical Advisory Group
DM	Degradation Matrix
IMT	Issue Management Table
SCC	Stress Corrosion Cracking
PWSCC	Primary Water Stress Corrosion Cracking
IASCC	Irradiation Assisted Stress Corrosion Cracking
CASS	Cast Austenitic Stainless Steel
DH	Dissolved Hydrogen
ASME	American Society of Mechanical Engineers
BWRVIP	Boiling Water Reactor Vessel and Internals Project
GALL	Generic Aging Lessons Learned Report
JOBB	Joint Owners Baffle Bolt Program
EFPY	Effective Full Power Years
FSAR	Final Safety Analysis Report
RDC	Resolution Demonstration Check
SRCS	Sensitivity, Resolution and Contrast Standard
AMP	Aging Management Program
AMR	Aging Management Review
IE	Irradiation Embrittlement
TE	Thermal Embrittlement
VS	Void Swelling

Inservice Inspection	Methods and actions for assuring the structural and pressure-retaining integrity of safety-related nuclear power plant components in accordance with the rules of the ASME Code Section XI (adapted from IWA-9000).
Inservice Examination	The process of visual, surface, or volumetric examination performed in accordance with the rules and requirements of Division 1 of the ASME Code Section XI (adapted from IWA-9000).
Nondestructive Examination	An examination by the visual, surface, or volumetric method (IWA-9000). The development and application of technical methods to examine materials and/or components in ways that do not impair future usefulness and serviceability in order to detect, locate, measure, interpret, and evaluate flaws (Article 1, I-130, ASME Code Section V).
Visual Examination	A nondestructive examination method used to evaluate an item by observation, such as: the correct assembly, surface conditions, or cleanliness of materials, parts, and components used in the fabrication and construction of ASME Code vessels and hardware (Article 9, ASME Code Section V).
Direct Visual Examination	A visual examination technique performed by eye and without any visual aids (excluding light source, mirrors, and/or corrective lenses (Article 9, ASME Code Section V).
Remote Visual Examination	A visual examination technique used with visual aids for conditions where the area to be examined is inaccessible for direct visual examination (Article 9, ASME Code Section V).
Enhanced Visual Examination	A visual examination technique using visual aids to improve the viewing capability, e.g., magnifying aids, borescopes, video probes, fiber optics, etc. (Article 9, ASME Code Section V).
VT-1 Visual Examination	A visual examination technique conducted to detect discontinuities and imperfections on the surfaces of components, including such conditions as cracks, wear, corrosion, or erosion, in accordance with the requirements of Table IWA-2210-1 (adapted from IWA-2211).

VT-3 Visual Examination	A visual examination technique conducted to determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical displacements; and to detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion, in accordance with the requirements of Table IWA-2210-1 (adapted from IWA-2213, ASME Code Section XI).
Character Recognition Demonstration	The demonstration that VT-1 and VT-3 examination techniques are capable of representative lower case characters of dimensions, at distances from, and under illumination conditions specified in Table IWA-2210-1. For VT-1 examination, the specified character height is 0.044 in. (1.1 mm) and the maximum standoff distance is 24 in. (610 mm). For VT-3 examination, the specified character height is 0.105 in. (2.7 mm) and the maximum standoff distance is 72 in. (1219 mm) (adapted from IWA-2210, ASME Code Section XI).
Relevant Condition	A condition observed during a visual examination that requires supplemental examination, corrective measure, correction by repair/replacement activities, or analytical evaluation (IWA-9000, ASME Code Section XI).
Supplemental Examination	A surface or volumetric examination to determine the extent of the unacceptable conditions and the need for corrective measures, analytical evaluation or repair/replacement activities, based on the detection of relevant conditions by visual examination (adapted from IWB-3200, ASME Code Section XI).
Indication	The response or evidence from the application of a nondestructive examination (IWA-9000, ASME Code Section XI).
Relevant Indication	An indication detected by nondestructive testing that is caused by a condition or type of discontinuity that requires evaluation (Adapted from Article 30, ASME Code Section V).
Flaw	An imperfection or discontinuity that may be detectable by nondestructive testing and is not necessarily rejectable (Article 30, ASME Code Section V).
Defect	A flaw (imperfection or unintentional discontinuity) of such size, shape, orientation, location, or properties as to be rejectable (IWA-9000). One or more flaws whose aggregate size, shape, orientation, location, or properties do not meet specified acceptance criteria and are rejectable (Article 30, ASME Code Section V).

Discontinuity	A lack of continuity or cohesion; an interruption in the normal physical structure of material or a product (IWA-9000).
Linear Elastic Fracture Mechanics	The analytical procedure that relates the stress-field magnitude and distribution in the vicinity of a crack tip, resulting from the nominal stress applied to the structure, to the size of a crack that would cause non-ductile failure (Appendix A, ASME Code Section XI).
Crack Initiation	The onset of flaw extension due to an increase in component loading (Appendix A, ASME Code Section XI).
Crack Growth in Austenitic Components	The stable flaw extension caused by cyclic fatigue flaw growth, stress corrosion cracking under sustained load, or a combination of both (adapted from C-3200, Appendix C, ASME Code Section XI).
Mature Crack	A surface-breaking crack propagated to a depth under applied load such that the crack opening surface displacement is of the same order of magnitude as the character recognition height demonstration requirements of Table IWA-2210-1 of the ASME Code Section XI (new definition).
Crack Tightness	The characteristic magnitude of the crack opening surface displacement of a surface-breaking crack following removal of the applied load causing crack propagation (new definition).
Liquid Penetrant Examination	A nondestructive test that uses suitable liquids that penetrate discontinuities open to the surface of solid materials and, after appropriate treatment, indicate the presence of discontinuities (Article 30, ASME Code Section V).
Magnetic Particle Examination	A nondestructive test method utilizing magnetic leakage fields and suitable indicating materials to disclose surface and near-surface discontinuity indications (Article 30, ASME Code Section V).
Eddy Current Testing	A nondestructive test method in which eddy current flow is induced in the test object. Changes in the flow caused by variations in the specimen are reflected into a nearby coil, coils, or Hall effect device for subsequent analysis by suitable instrumentation and techniques (Article 30, ASME Code Section V).
Radiographic Inspection	The use of X-rays or nuclear radiation, or both, to detect discontinuities in material, and to present their images on a recording medium (Article 30, ASME Code Section V).

Ultrasonic Testing	A nondestructive method of examining materials by introducing ultrasonic waves into, through, or onto the surface of the article being examined and determining various attributes of the material from effects on the ultrasonic waves (Adapted from Article 30, ASME Code Section V).
Core Support Structures	Those structures or parts of structures that are designed to provide direct support or restraint of the core (fuel and blanket assemblies) within the reactor pressure vessel (IWA-9000).
Examination Category	A grouping of items to be examined or tested (IWA-9000).
Examination Category B-N-1	The examination category that includes accessible areas of the reactor vessel interior using VT-3 visual examination techniques (adapted from Table IWB-2500-1).
Examination Category B-N-2	The examination category that includes accessible welds for interior attachments within the reactor vessel beltline using VT-1 visual examination techniques, and accessible welds for interior attachments outside the reactor vessel beltline using VT-3 visual examination techniques (adapted from Table IWB-2500-1).
Examination Category B-N-3	The examination category that includes accessible (or made accessible by removal) surfaces of core support structures using VT-3 visual examination techniques (adapted from Table IWB-2500-1).
Acceptance By Visual Examination	A component whose visual examination confirms the absence of the relevant conditions described in the standards of Table IWB-3410-1 shall be acceptable for service (IWB-3122.1(a)). A component whose visual examination detects the relevant conditions described in the standards of Table IWB-3410-1 shall be unacceptable for service, unless such components meet the requirements of IWB-3122.2 or IWB-3122.3 prior to placement of the component in service (IWB-3122.1(b)).

VT-3 Visual Examination Standards	The following relevant conditions shall require correction in meeting the requirements of IWB-3122 prior to service or IWB-3142 prior to continued service: (a) structural distortion or displacement of parts to the extent that component function may be impaired; (b) loose, missing, cracked, or fractured parts, bolting, or fasteners; (c) foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel; (d) corrosion or erosion that reduces the nominal section thickness by more than 5 %; (e) wear of mating surfaces that may lead to loss of function; or (f) structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 % (IWB-3520.2).