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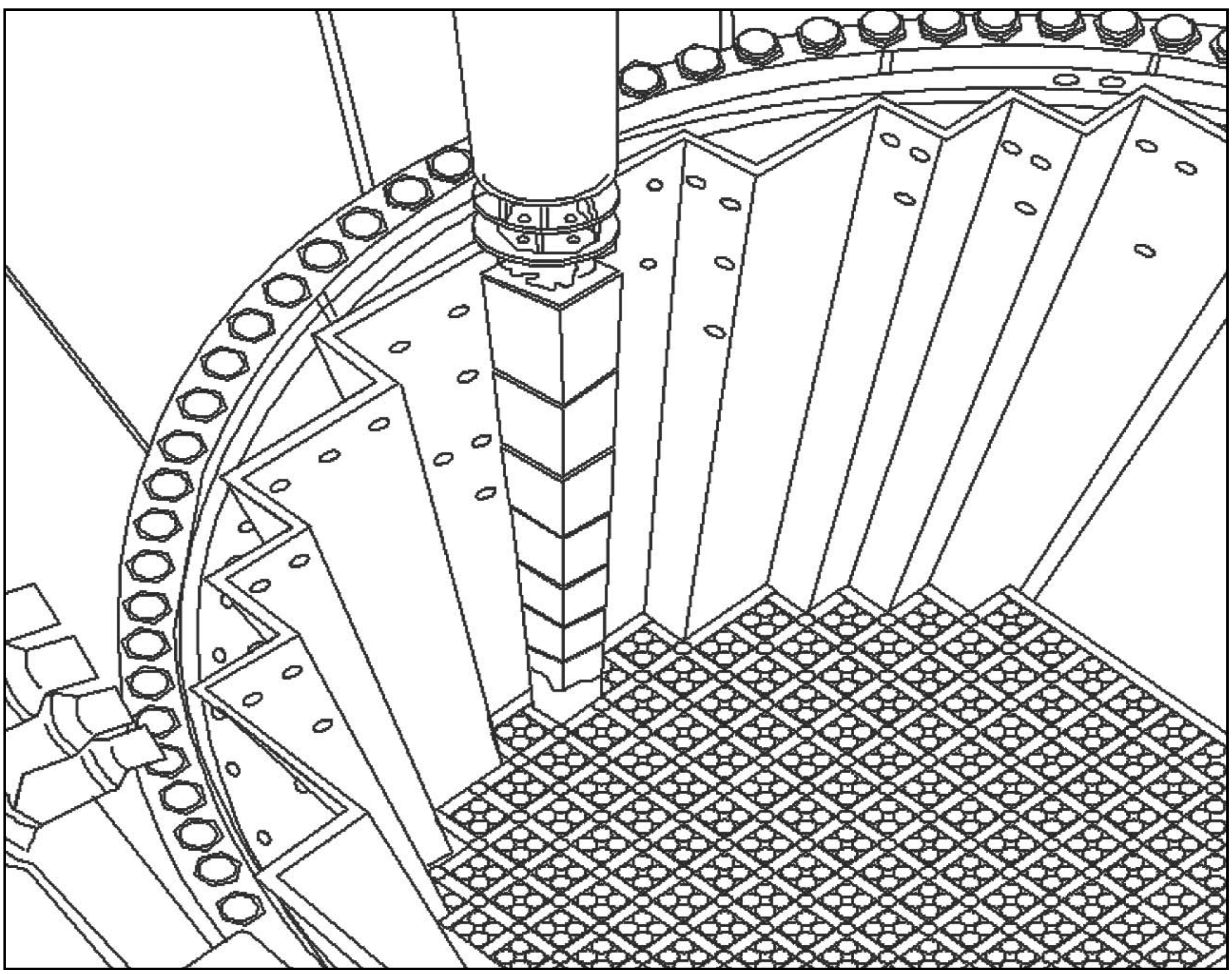
Materials Reliability Program: Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals (MRP-153)

Technical Report



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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

Demonstration that the effects of aging degradation in PWR internals are adequately managed is essential for maintaining a healthy fleet and to assure functionality of the core internals components. As part of the EPRI Material Reliability Program (MRP) 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 using knowledge of internals design, materials and material toughness properties, and applying 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. Related MRP documents will be published on a Framework and Strategy for Managing Aging Effects in PWR Internals [1], and Reactor Vessel Internals Aging Degradation Mechanism Screening and Threshold Values [2]. 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 to perform their function. Additional data for PWR conditions are expected in a number of areas, including crack initiation, crack growth, fracture toughness, void swelling, etc., 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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1 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.

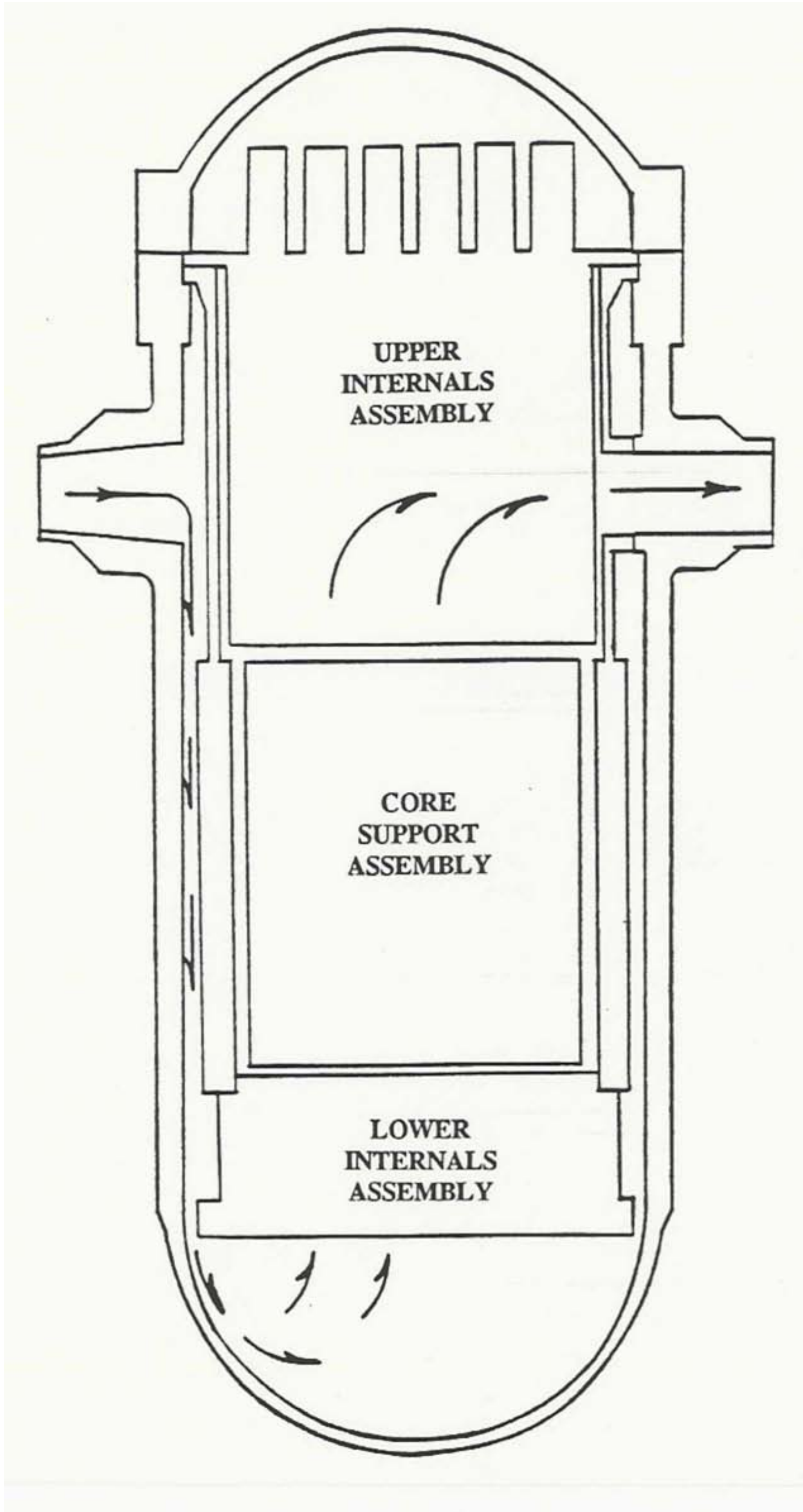

Figure 2-1
PWR RPV Internals Structural Assembly Groupings

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.

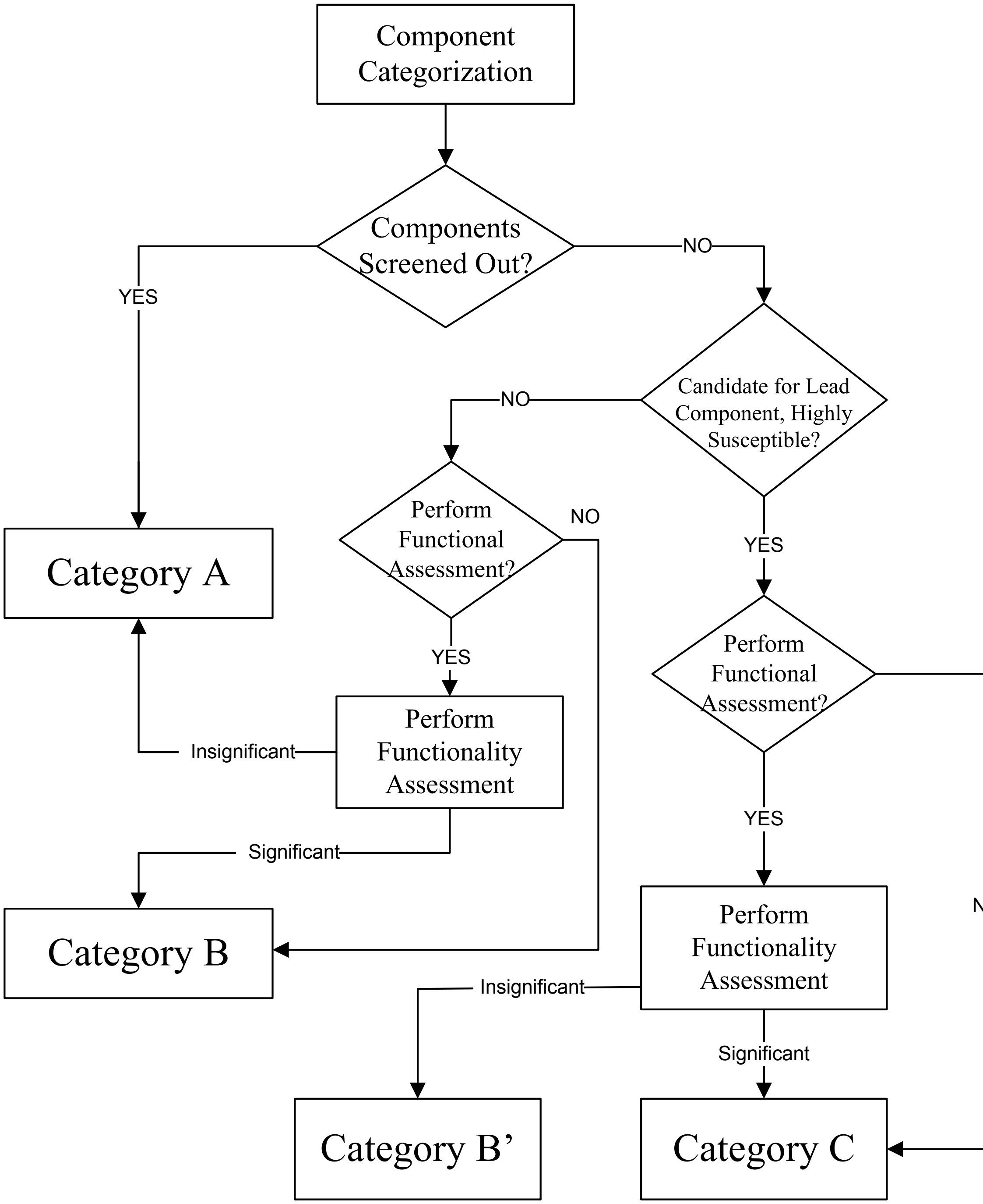

Figure 2-2
Process for Categorization of PWR Internals Components

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. The average undersize was 1.91 inches, with a maximum undersize of 5.81 inches out of a total length of 71.06 inches. The maximum oversize was 14.75 inches out of a total length of 47.5 inches.

The findings from Reference 17 are summarized below:

- Crack length can only be determined over the visible portion of the crack
- Poor surface cleaning can cause underestimation of crack length
- Cameras and lighting require some form of landmark feature for sizing estimation
- Extreme crack tightness can cause severe undersizing error
- In general, crack lengths are undersized, with the average undersize of 1.91 inches (48.5 mm)
- When cracks are at least 12 inches (300 mm) long, the error is less than 10 %.

The current BWRVIP position on inclusion of NDE uncertainty is documented in a BWRVIP Response to an NRC Request for Additional Information* that states:

For the purposes of flaw evaluation, no adjustment to the measured flaw size is required if the following criteria are met:

1. *For UT depth sizing, the RMS value of the flaw depth measurement errors experienced during performance demonstration is less than 0.125 in.*

2. *For UT length sizing, the RMS value of the flaw length errors experienced during performance demonstration is less than 0.75 in.*

3. *For VT or ET length sizing, the RMS value of the flaw length measurement errors experienced during performance demonstration is less than 0.75 in.*

4. *For UT or VT/ET length sizing of cylindrical piping components, the RMS value of the flaw length errors experienced during performance demonstration is less than 2% of the circumference (1% at each crack tip) or 0.75 in., whichever is smaller.*

If the inspection techniques do not meet the above criteria, the Evaluation Factors identified in the current version of BWRVIP-03 [16] must be added to each flaw.

In addition to the small-diameter wire simulation of cracking discontinuities, the BWRVIP also developed core shroud weld mockups for ultrasonic inspection demonstration, using fatigue pre-cracking to generate real flaws. The fatigue cracks were found to be sufficiently tight to have crack widths less than the widths of typical IGSCC cracks detected and sized in BWR stainless steel piping. The minimum surface crack opening displacements for any of the fatigue cracks was 0.00008 inches (0.002 mm) and the maximum surface crack opening displacement was 0.004 inches (1 mm). The average surface crack opening displacement was 0.0002 inches (0.005 mm). Note that the average crack opening displacement is smaller than the diameter of the calibration wire used for visual examination demonstration. The observations on relative crack tightness between fatigue and SCC cracks are not generally true although if the residual stress field relieved by SCC is large, as could be the case in BWR shrouds, then the SCC cracks would

* BWRVIP letter 2004-192 from Robin Dyle/Tom Mulford to All BWRVIP Committee Members, "BWRVIP Response to NRC RAI on NDE Flaw Sizing Uncertainty," dated May 25, 2004

gape. The general application of the principle may need to be verified for certain geometry specific situations.

Based on this study, the BWRVIP has recommended the implementation of an improved visual examination procedure, called EVT-1 (enhanced VT-1) that requires demonstration of capability to detect a 0.0005-inch target. As pointed out earlier, this definition of enhanced visual examination differs from that given in Article 9 (Visual Examination), Appendix I (Glossary of Terms for Visual Examination) of the ASME Code Section V [18].

The ASME Code definition for ***enhanced visual examination*** is "***a visual examination technique using visual aids to improve the viewing capability, e.g., magnifying aids, borescopes, video probes, fiber optics, etc***." Reference 16 states that "***Enhanced VT-1 (EVT-1) as used in this document is a visual inspection method where the equipment and environmental conditions are such that they can achieve a ½ mil resolution***."

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:

- A plant specific neutron transport analysis should be performed to determine the azimuthal and axial flux/fluence variation for the internals components (e.g., core shroud welds) of interest. The neutron transport calculations should be consistent with vessel surveillance fluence methods. Only the calculated best estimate fluence is required for comparison to threshold values, such as the $3x10^{20}$ n/cm^2 (E > 1 MeV) threshold for fracture mechanics analysis of flaws in irradiated stainless steel.
- When a visual examination technique is used to determine crack length, the cracking should be assumed to be completely through the component thickness; if volumetric examination techniques are used to detect and size the crack, the measured crack depth may be used. For visual examination and determination of flaw length, uncertainty with respect to flaw length sizing should be accounted for. ASME Code proximity rules should be used for closely spaced flaws.
- To account for the effects of irradiation changes to mechanical properties, three different analytical techniques should be used: (1) limit load techniques; (2) linear elastic fracture mechanics (LEFM) methods; and (3) elastic-plastic fracture mechanics (EPFM) methods.
- Limit load techniques can be used as the sole flaw evaluation method for component neutron irradiation exposures less than or equal to $3x10^{20}$ n/cm^2 (E > 1 MeV).
- Because data have shown that substantial ductility remains up to and including neutron irradiation exposures of $8x10^{20}$ n/cm^2 , E > 1 MeV, limit load techniques should continue to be used up to exposure levels of $1x10^{21}$ n/cm^2 (E > 1 MeV) supplemented by either LEFM or

EPFM calculations. Above $1x10^{21}$ n/cm^2 (E > 1 MeV), limit load assessment must be supplemented by LEFM calculations.

- For the limit load evaluation, any volume of material with fluence above the $3x10^{20}$ n/cm^2 (E > 1 MeV) threshold shall be removed from the analytical model of the weld being analyzed, including any region that is cracked. The limit load calculation should be applied to the remainder of the volume of material.
- The acceptance criterion for the limit load evaluation is that the limit moment divided by the section modulus of the uncracked cross section plus any applied primary membrane stress shall be greater than or equal to the appropriate safety factor multiplied by the applied primary bending plus applied primary membrane stress.
- The acceptance criterion for any LEFM calculations is that the applied stress intensity factor is less than the material fracture toughness divided by the appropriate safety factor.
- The acceptance criterion for any EPFM calculations is that the flaw is shown to be stable under the applied loads, with appropriate safety factors taken into account. The stability can be demonstrated by J-R curve or tearing modulus approaches, as described in Appendix K of Section XI of the ASME Code.
- Safety factors of 2.77 for normal and upset loading conditions (Level A/B) and 1.39 for emergency and faulted loading conditions (Level C/D) were set.

A set of preliminary flaw evaluation approaches similar to that of BWRVIP is proposed for application to PWR vessel internals as discussed in Chapter 6 of this report.

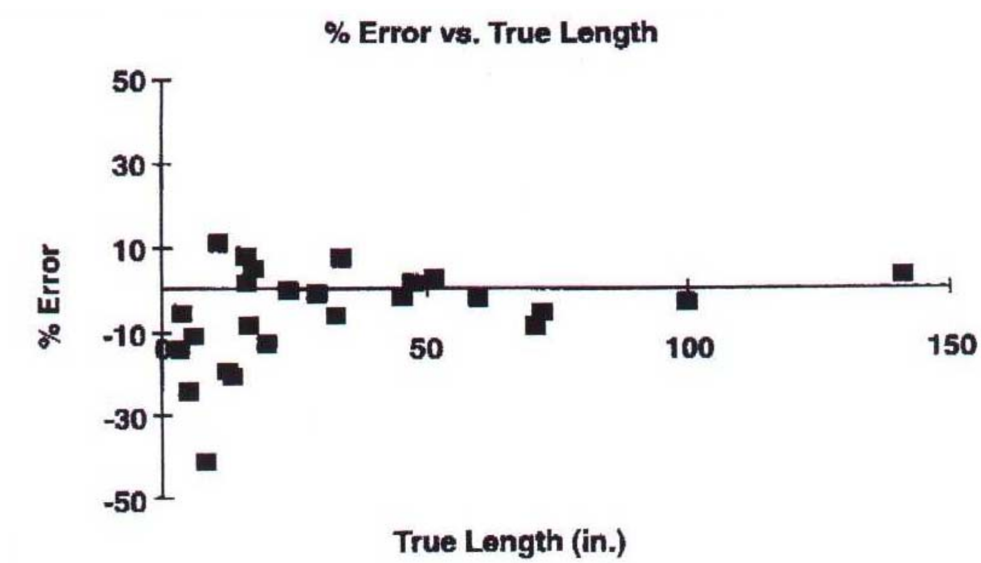

Figure 4-1
VT Length Measurement Performance on Simulated Cracks (Pieces of Tape) in NDE Center's 20-foot-deep Water Tank

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.

1. The SRCS demonstration test article shall be fabricated with a surface texture (reflectivity, color, and finish) that is representative of the actual items to be examined. The SRCS may consist of the component to be examined with a target superimposed over the area of interest.

2. Targets approximately 1/2 mil wide and of sufficient length to demonstrate the required detection resolution across the entire field of view of the camera system shall be affixed or embedded into the SRCS. At least one such target shall be oriented in the horizontal direction and at least one such target shall be oriented in the vertical direction. Targets may be wire, electro-discharge machined notches, laser cut notches, etc. An alternative SRCS may be used provided it is shown to produce equal or higher sensitivity.

3. The detection resolution capabilities of the inspection equipment and technique shall be demonstrated using the SRCS, in an environment representative of the area in which inspections will be performed, prior to performing inspections and at any time a key element of the procedure is changed. A detection resolution demonstration check (RDC) shall also be performed at the beginning and end of each inspection.

4. The maximum distance the camera should be from the inspection surface is to be determined based on the level of detail that can be seen on the inspection surface. Industry experience has shown that a quality inspection can be performed provided that surface anomalies such as grinding and machining marks, weld beads and ripples, undercuts, arc strikes, etc. can readily be seen in the field of view. As a minimum, inspections shall be conducted within the parameters established during the RDC. The screen image should include a small amount of weld or other surface feature within the area of interest as a landmark. (The EPRI MRP RI-ITG is considering whether, as a starting point, the maximum examination distance for the remote visual examination should be chosen to be 16 inches (0.4 m).)

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, as reported in MRP-129 [21] remains relatively high, Figure 6-3. However, fracture toughness resistance is reduced to very low levels. Figure 6-4 shows fracture toughness vs fluence data for some PWR irradiated 300 series SS [22] (MRP-160). 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. As an illustration of the threshold fluence level for application of either LEFM or EPFM evaluation methods, the data reported in MRP 160 [22] is applied. The trend lines shown in Figures 6-6 to 6-8 are fitted by the power law

$$J_{mat} = C\,(\Delta a)^n.$$

where the expression for C in KJ/m^2 is given by

$$C = 262^* F^{-0.382}$$

and the expression for n is given by

$$n = 0.288^* F^{-0.163}$$

F is in dpa for both C and n.

When the fluence value of 10^{21} n/cm^2 (E > 1 MeV) (1.43 dpa) is substituted into these expressions, C = 228.6 and n = 0.272. The value of J_{mat} can then be determined at a crack extension of 1.5 mm to be approximately 255 KJ/m^2. When this value is converted to English units, the 1453 in-lb/in^2 represents elastic-plastic material toughness essentially identical to the 1450 in-lb/in^2 toughness used in the elastic-plastic flaw evaluations described in Reference 23. Elastic-plastic fracture mechanics methods would be acceptable at such fluence levels. However, when a fluence of 3x10^{21} n/cm^2 (E > 1 MeV) (4.3 dpa) is substituted into the bounding expressions, C = 150.3 and n = 0.227. The corresponding J_{mat} at 1.5 mm of crack extension is equal to 165 KJ/m^2 (939 in-lb/in^2) and J_{mat} at 2.5 mm of crack extension is equal to 185 KJ/m^2 (1054 in-lb/in^2). These toughness levels are about 2/3 of the toughness level used in Reference 23. Therefore, at fluence levels greater than $3x10^{21}$ n/cm2 (E > 1 MeV), LEFM analyses can be used. Note that Figures 6-6 to 6-8, which provided the basis for the equations used to interpolate the parameters C and n, shows elastic-plastic fracture toughness values in the range of 150 to 160 KJ/m^2 (850 to 900 in-lb/in^2) at 5.9 dpa for crack extensions of 1.5 to 2.5 mm. Such values imply that elastic-plastic fracture mechanics methods may continue to have application at fluence levels greater than $3x10^{21}$ n/cm^2 (E > 1 MeV) (4.3 dpa).

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.

Based on currently available PWR test data, a time-dependent flaw growth rate of $1x10^{-7}$ mm/sec ($1.4x10^{-5}$ inches/hour) for the length direction of the flaw is proposed, Figure 6-9 [19], and half of the length crack growth rate, $0.5x10^{-7}$ mm/sec ($.7x10^{-5}$ inches/hour), is proposed for the depth direction following the approach of BWRVIP [20].

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. The BWRVIP has proposed that a safety factor of 2.77 for expected loadings and 1.39 for unexpected loadings be maintained for the applied membrane and bending stresses, while insuring against the formation of a hinge. 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.

If the fluence is greater than $3x10^{21}$ n/cm^2 (E > 1 MeV), the preferred demonstration methodology is that of linear elastic fracture mechanics (LEFM). Reference 26 provides a generic treatment for BWR core shroud flaws, assuming a very low applied stress level of 6 ksi. For such low stress levels, even the reduced fracture toughness at fluence much greater than 10^{21} n/cm^2 (E > 1 MeV) does not compromise the continued function of the component.

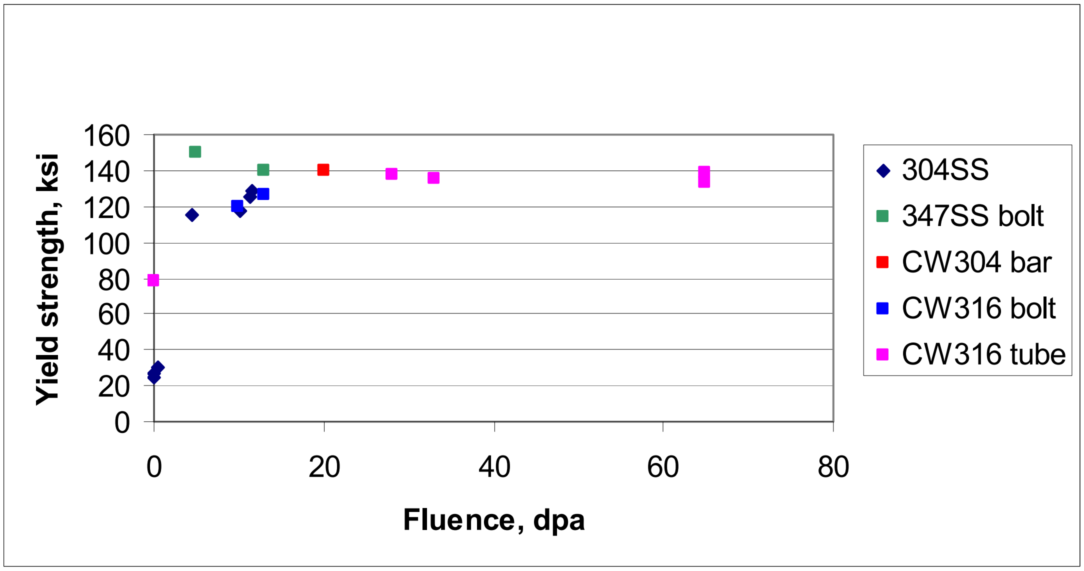

Figure 6-1
608°C Yield Strength Increase as a Function of Fluence [21]

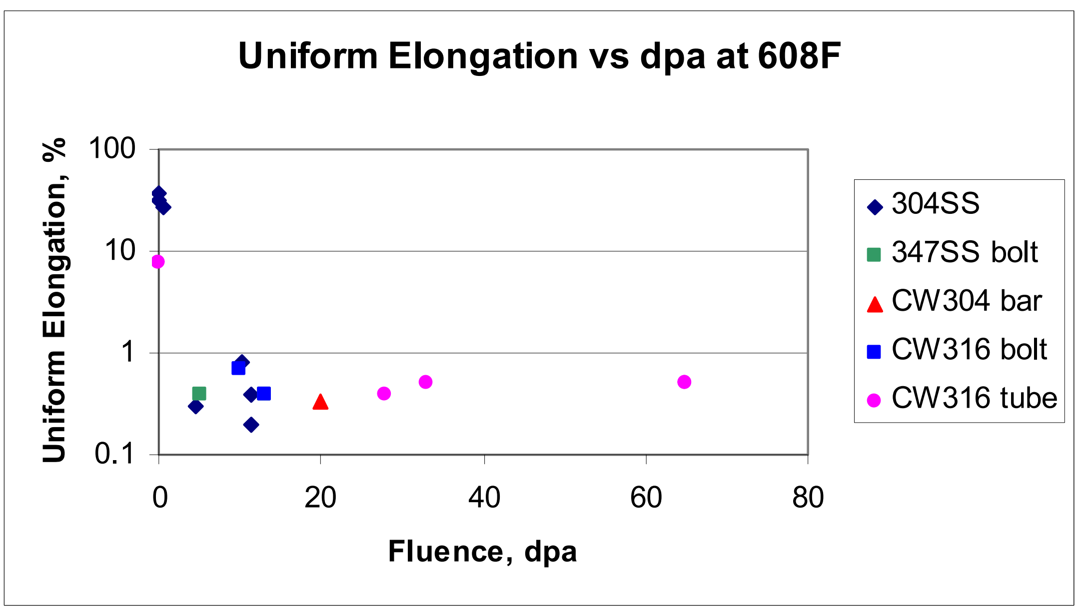

Figure 6-2
Uniform Elongation as a Function of Fluence [21]

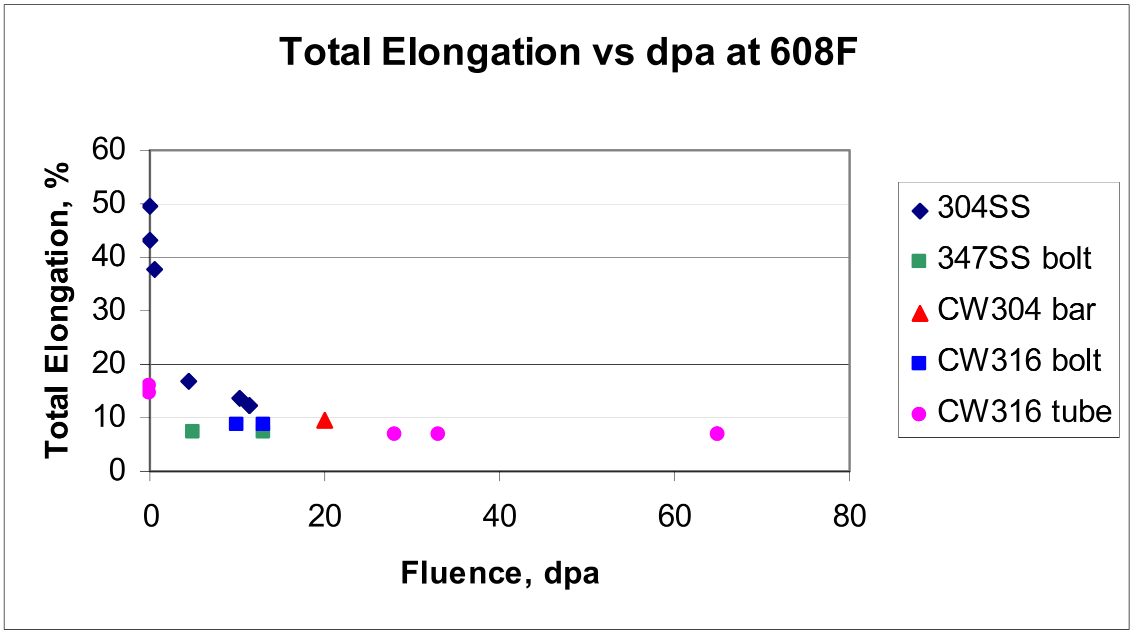

Figure 6-3
Total Elongation as a Function of Fluence [21]

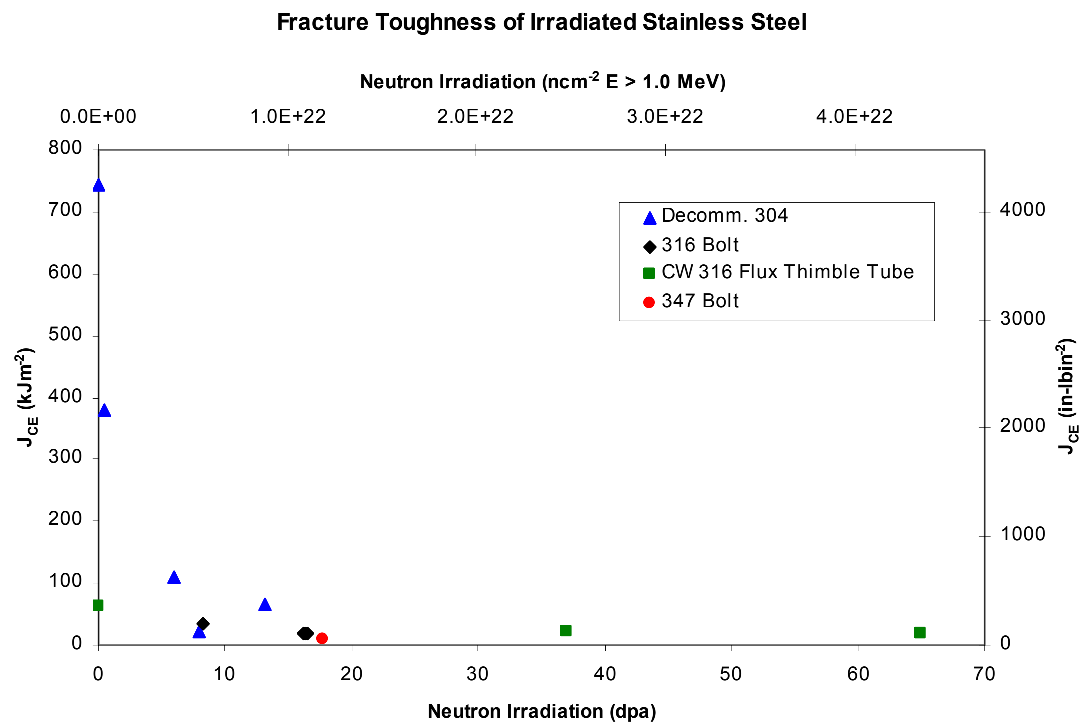

Figure 6-4
Fracture Toughness as a Function of Fluence [22]

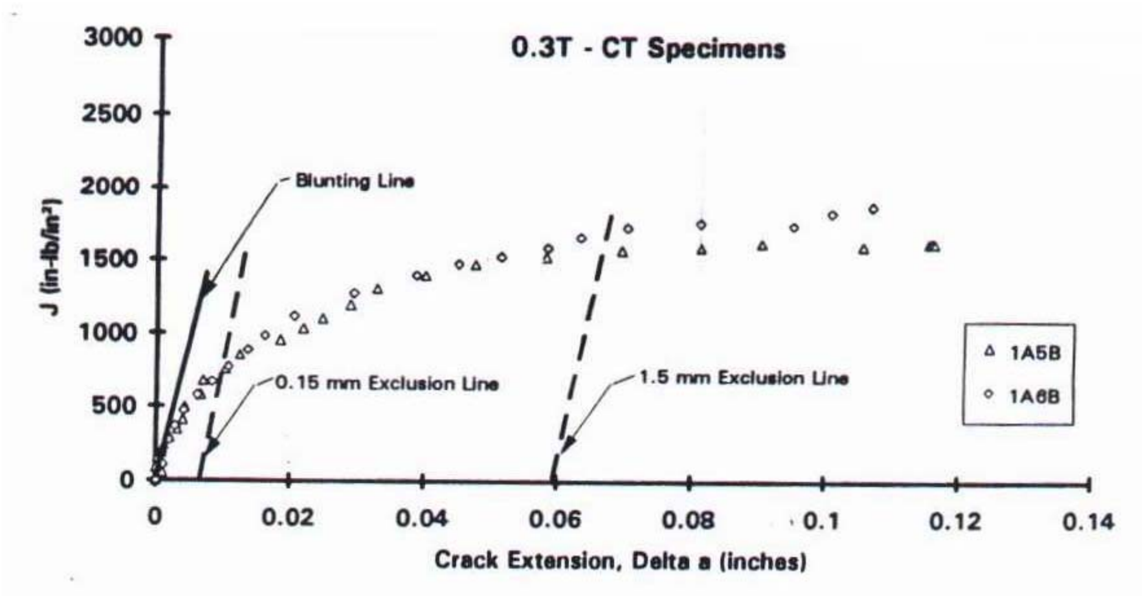

Figure 6-5
J-R Curves for Two Irradiated Stainless Steel Specimens at Fluence of $8x10^{20}$ n/cm^2 (E > 1 MeV) [23]

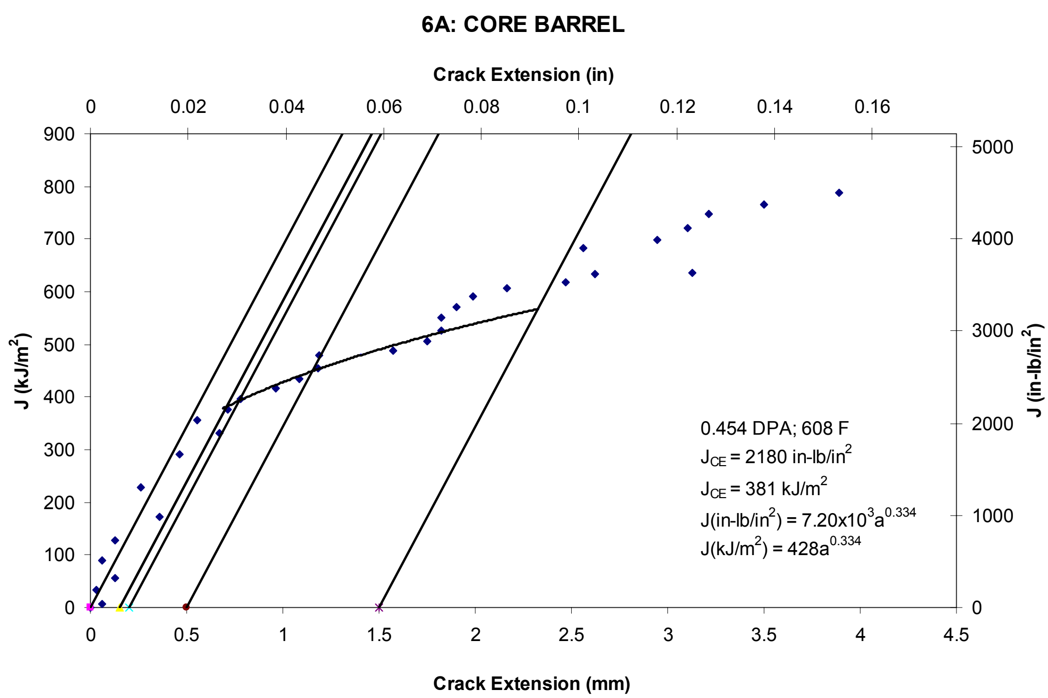

Figure 6-6
$J_{material}$ vs Crack Growth at 0.454 dpa Fluence for 304 SS [22]

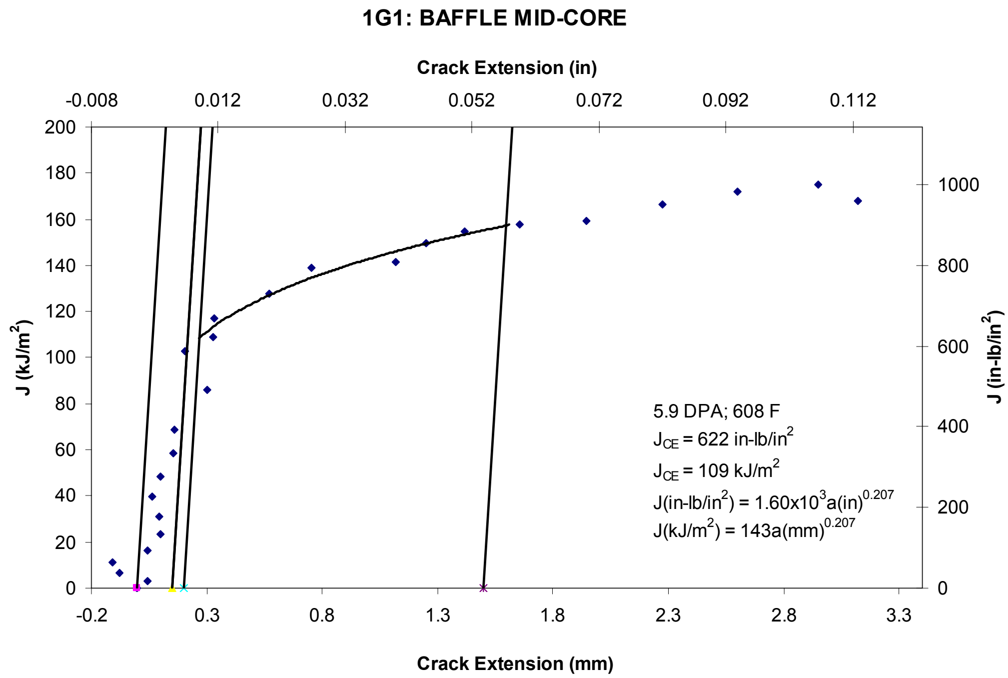

Figure 6-7
$J_{material}$ vs Crack Growth at 5.9 dpa Fluence for 304 SS [22]

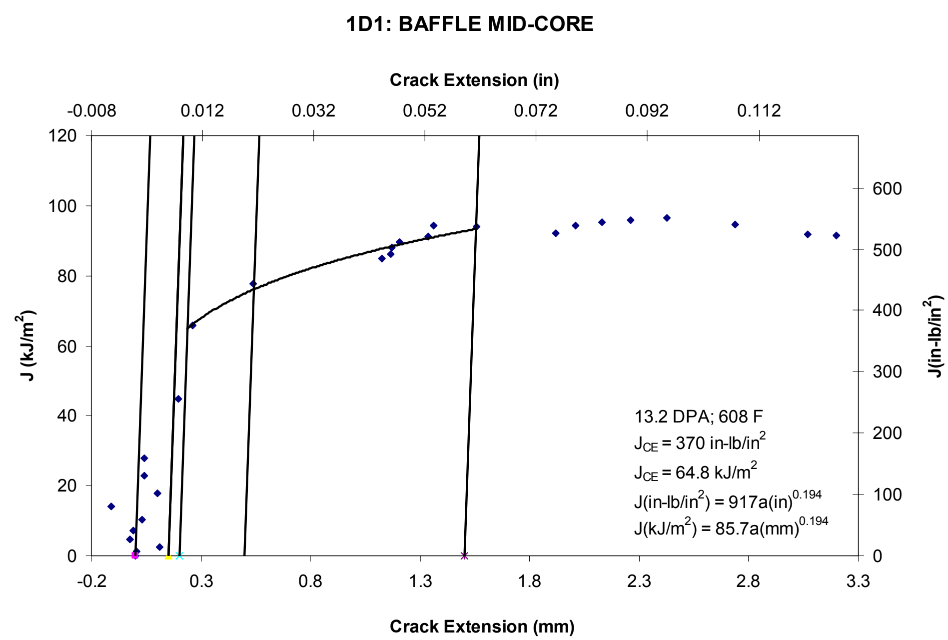

Figure 6-8
$J_{material}$ vs Crack Growth at 13.2 dpa Fluence for 304 SS [22]

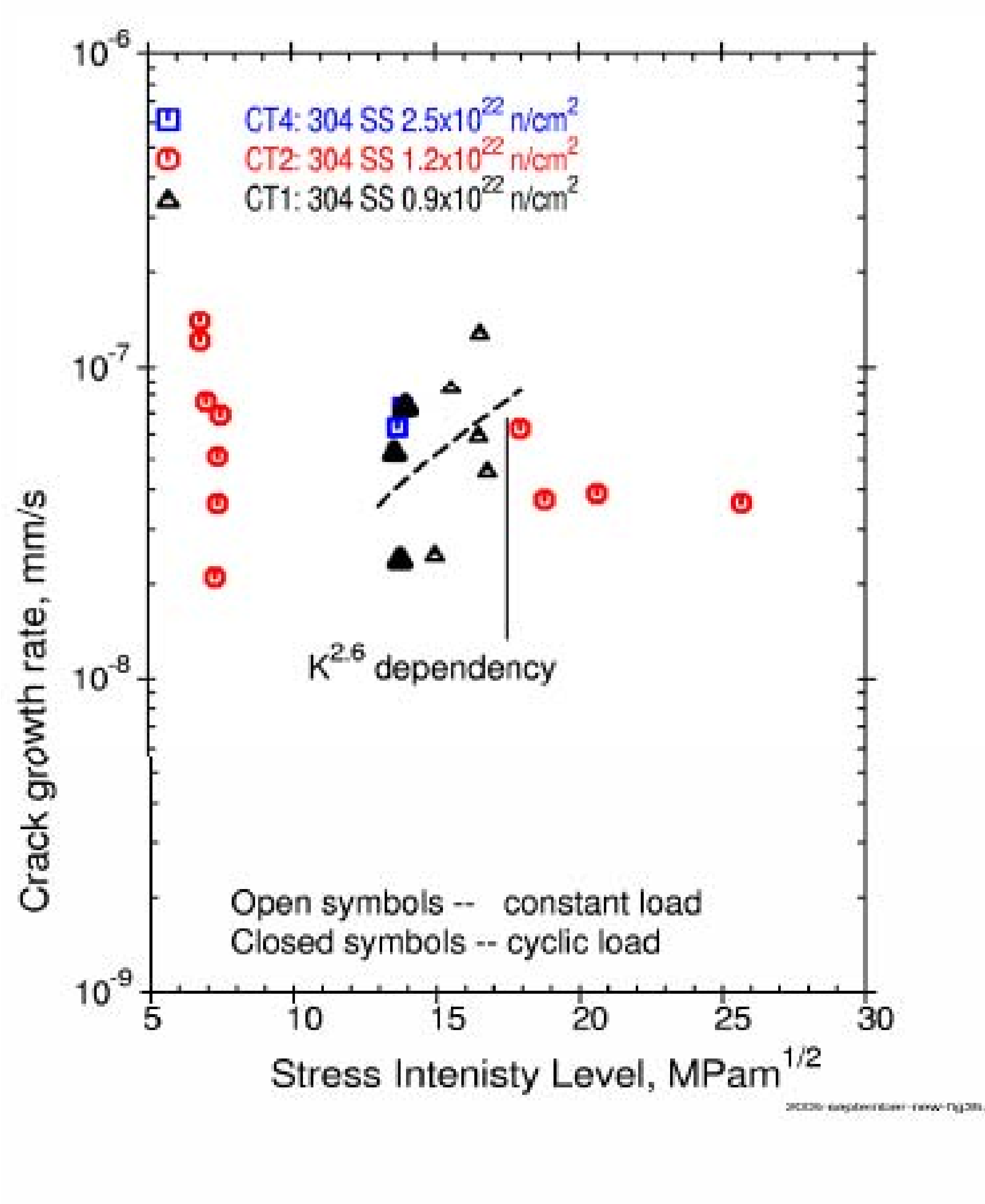

Figure 6-9
Crack Growth Data of LWR Irradiated 304 Tested in Pile in PWR Environment [19]

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.

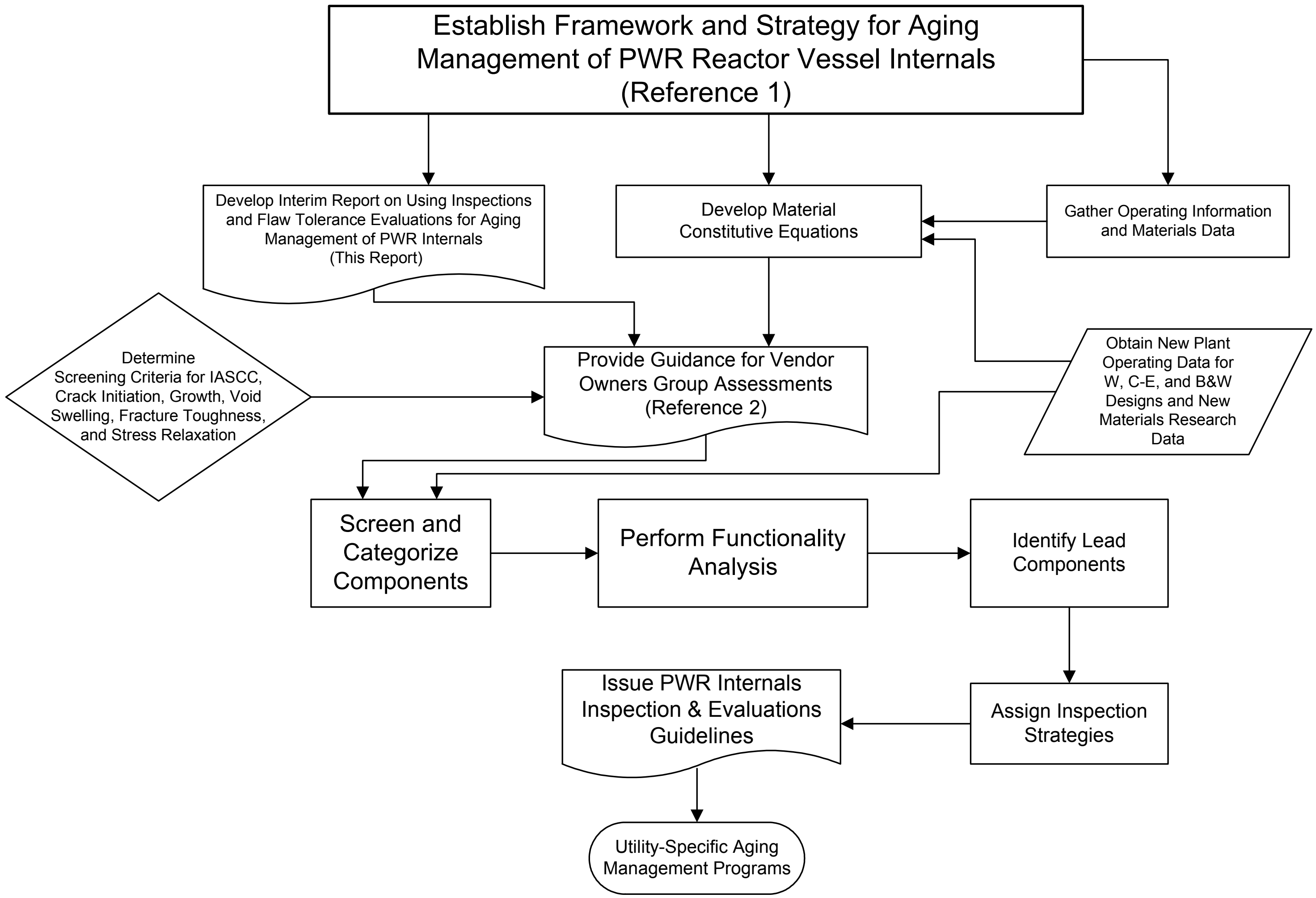

Figure 7-1
Framework for Implementation of Aging Management Using Screening, Functionality Evaluations, and Inspections [1]

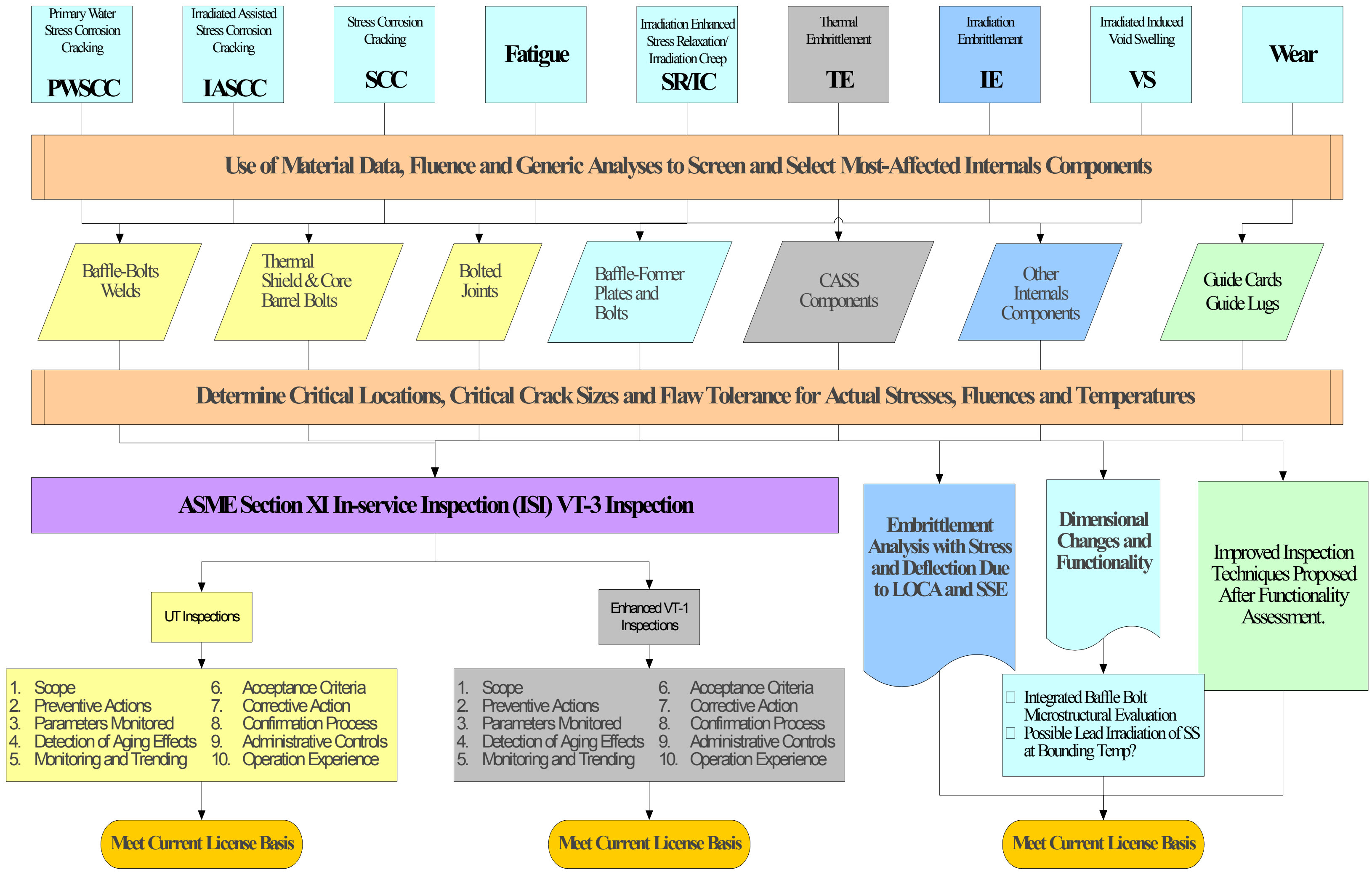

Figure 7-2
Example of Reactor Internals Aging Management Strategy Using Inspections

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

The following is from BWRVIP-03 [16].

1.0 Purpose

This document delineates the minimum requirements and recommendations for the performance of underwater in-vessel visual inspections (IVVI) of reactor pressure vessel (RPV) internals, components, and associated repairs.

2.0 Scope

2.1 This document is to be used by boiling water reactor utilities when performing Enhanced VT-1 (EVT-1) on RPV internals, components, and associated repairs to meet the recommendations set forth in Boiling Water Reactor Vessel and Internals (BWRVIP) documents.

2.2 The inspection requirements and recommendations contained herein are generic and the user should review the specific inspection requirements contained in the applicable BWRVIP document.

2.3 This document is not intended to replace a utility's specific American Society for Mechanical Engineer's (ASME) Boiler and Pressure Vessel Code Section XI visual inspection program. When a VT-1 or VT-3 inspection is specified in a BWRVIP document, it is intended that those inspections be performed in accordance with the utilities ASME Section XI requirements for those types of inspections.

3.0 Definitions

3.1 Indication:

Evidence of an apparent interruption in the normal structure of a material of product. Indications may be relevant or non-relevant.

3.2 Relevant Indication:

3.2.1 Welds - Cracks, or indications that exhibit characteristics of cracking are considered relevant.

3.2.2 Components - Cracking or other significant degradation that could impair the ability of the component to perform its design function is considered relevant.

3.3 Non-relevant Indication:

An indication that is evaluated as not being relevant per 3.2 above.

3.4 Sensitivity, Resolution and Contrast Standard (SRCS):

The standard used to qualify visual inspection techniques and equipment.

3.5 Key Element:

Any element, component, or combination of the inspection equipment that if changed, could affect the ability of the inspection equipment to detect indications or an evaluator's ability to evaluate indications. This includes, but is not limited to the camera, camera tube or board, camera lens; video processor, monitor, recording device or recording medium (if evaluations will be performed from recordings); light intensity and source; and inspection conditions, such as water clarity, lens-to-subject distance, etc.

3.6 Enhanced VT-1:

Enhanced VT-1 (EVT-1) as used in this document is a visual inspection method where the equipment and environmental conditions are such that they can achieve a ½ mil resolution.

3.7 Resolution Demonstration:

The process of demonstrating the ability of the IVVI equipment, equipment setup, inspection area environment, and inspection technique to resolve the appropriate target on an SRCS.

4.0 Personnel Training/Experience

4.1 Personnel evaluating inspection data shall be certified as a Level II or a Level III examiner in the VT-1 and/or VT-3 method, as appropriate, to a written practice meeting the requirements of ASME, Section XI.

4.2 In addition, personnel evaluating inspection data shall receive a minimum of four (4) hours of indoctrination training on inspection methods and experience specific to the component(s) being examined. **The employer is responsible for documenting this training.** This training shall include the following, as a minimum:

4.2.1 A review of actual inspection tapes, which show the types of flaws typically encountered in the inspection(s) being performed.

4.2.2 Various types of non-relevant indications that may be encountered such as grinding, machining, crud build-ups, mechanical marks from fabrication/erection, etc.

4.2.3 Identification of areas prone to cracking, including details and characteristics of cracks that might be found in these areas.

4.2.4 The effects of surface conditions on detecting and evaluating indications, including the masking effects of crud build-ups and other extraneous materials.

4.2.5 Inspection techniques which maximize the ability to detect and evaluate indications, including:

4.2.5.1 Techniques which aid in determining relevancy or non-relevancy of indications.

4.2.5.2 Camera and lighting source angle to the inspection surface to eliminate shadows and glare.

4.2.5.3 Camera lens effects, including: field (angle) of view, magnification, depth of field, distortion, size of area observed and speed of camera movement during inspections.

4.2.6 Radiation effects on camera equipment.

4.2.7 Effects of changes in key elements of the inspection system.

4.2.8 Measurement techniques.

4.3 Utilities shall also conduct site specific training for all personnel evaluating inspection data. The training shall also include utility-specific procedural requirements, configuration details, previous inspection results, operation of inspection equipment, specific outage inspection scope, and any other pertinent information related to inspection, evaluation and/or reporting, as applicable. The training is to be conducted prior to inspections for each refueling outage. **The length of training, although at the discretion of the utility, shall be based on the outage inspection scope, the inspection history, and the familiarity of data evaluators with the site. This training shall be documented by the utility.**

4.4 Personnel evaluating inspection data shall have a minimum of ten (10) hours of work time experience performing VT-1 and/or VT-3 IVVI, as appropriate, under the direction of a Level II or Level III IVVI examiner qualified in the method being performed. **The employer is responsible for documenting this work time experience.**

4.5 Utilities shall ensure that all personnel involved in the evaluation of inspection data meet the requirements of this document.

4.6 Personnel who have had previously documented in-vessel visual inspection evaluation experience (which included crack detection and characterization), are considered to have met the indoctrination training and work experience requirements contained herein and are exempt from the requirements of 4.2 and 4.4, except as noted below. Note: For personnel exempted from the requirements of 4.2 and 4.4 based upon previous inspection experience, the items outlined in 4.2.1 through 4.2.3 must be included in the site-specific training specified in 4.3. The length of this training is at the discretion of the utility.

5.0 Equipment Requirements

5.1 Sensitivity, Resolution, and Contrast Standard

5.1.1 An SRCS shall be fabricated which is representative of the surface texture (reflectivity, color, and finish) of the item to be examined. The SRCS may consist of the component being examined with the target superimposed over the area of interest.

5.1.2 Targets, approximately ½ mil wide, of sufficient length to demonstrate the required resolution across the entire field of view of the camera system shall be affixed or embedded into the SRCS. At least one such target shall be oriented in the horizontal direction and at least one such target shall be oriented in the vertical direction. Targets may be wire, electro-discharge machined notches, laser cut notches, etc. Alternative SRCS may be used provided they are demonstrated to produce equal or higher sensitivity.

5.2 Underwater Cameras

5.2.1 Black and white or color cameras may be used provided the entire system meets the requirements of 6.3. Generally, a camera having a minimum resolution of 450 horizontal lines along with the ability to operate in macro will help ensure the system's ability to pass the equipment resolution demonstration.

5.2.2 Camera resolution and auxiliary features have significant impact on the ability to perform quality inspections and evaluate inspection results. Tube cameras should be adjusted to achieve the highest resolution possible using both the optical (pot adjustment on the scan board) and electrical (face panel adjustment) focus. Failure to electronically align the camera as described above following tube or board replacement may result in foggy or yellowed images, which could result in loss of resolution.

5.2.3 Resistance to radiation fields (both magnitude and duration) should be considered in the selection of camera and related equipment. Generally, nuvicon tubes are less tolerant to radiation but provide higher sensitivity with a given intensity of external lighting. Vidicon tubes are more tolerant to high radiation fields but require significantly more external lighting to obtain the same sensitivity.

5.3 Camera Lenses

5.3.1 Lens selection and use are at the discretion of the utility, provided the entire system meets the requirements of 6.3.

5.3.2 Industry experience has shown that narrow angle of view lenses (e.g., 25 mm) produce higher levels of magnification than wide angle-of-view lenses (e.g., 9 mm) at a given camera-to-subject distance. However, narrow angle-of-view lenses reduce the depth of field and the size of the target area, thus increasing the time and scanning required to obtain complete coverage.

5.3.3 Lenses with zoom features, capable of variable magnification and depth of field, may be used provided the requirements of 6.3 are met for the entire range of the lens.

5.4 Lighting

5.4.1 Care must be taken to control the angle of lighting during inspections. Lighting can cast shadows into the area of interest and prevent the detection of indications. Care must also be taken to avoid saturating the area of interest with light since this can create a glare, which can mask the indications.

5.4.2 Lighting sources should be equipped with a control that permits varying the intensity. This is a valuable tool for evaluating the relevancy of indications.

5.5 Viewing Monitors

5.5.1 Monitors used for viewing can have a direct impact on the ability to discern indications. Monitors should have, as a minimum, an equal number of horizontal line resolution as the camera to be used for the inspection.

5.5.2 Although adjusting the brightness and contrast controls on the monitor can better enable inspection personnel to discern indications, adjustments to the monitor do not alter the images being recorded. Therefore, when recording devices are being used, these controls should be set to the neutral position. The contrast and brightness adjustments should first be made by adjusting the lighting and camera iris, and then the contrast and brightness controls on the hard copy video processor (if used). Adjusting the picture as described will help to ensure that the recorded image will closely match that being viewed live by inspection personnel.

5.6 Video Data Recording

Note: Although it is not a requirement of this standard, industry experience has shown that video recording of the inspection images can be very beneficial to the utility. It provides a means for independent review of inspection results off of critical path, allows the comparison of current inspections with those performed in the past, and provides a useful tool to communicate inspection information to analysis personnel, repair vendors, regulators, etc.

5.6.1 When inspections are to be recorded, high-resolution (i.e., super VHS) equipment has been shown to accurately record the images received from the camera.

5.6.2 Recording on standard play has been shown to produce better results than recording on extended play.

5.6.3 Alternative recording media (i.e., CD-ROM, digital discs, etc.) are also available and may be used at the discretion of the utility.

5.7 Hard Copy Processors

5.7.1 The use of hard copy processors is at the discretion of the utility. These devices take a snap shot image of a video signal and are useful in providing immediate feedback of inspection results. Also, when set up per 5.8.1, hard copy processors with real time contrast and brightness controls provide a means to adjust the contrast and brightness of the image being viewed before that image is recorded. This helps to ensure that the recorded image closely matches that being viewed live.

5.8 Equipment Setup

5.8.1 Inspection system setup can have a significant impact on the accuracy of the recorded image. If inspections are to be recorded, Figure A-1 should be referred to for equipment setup recommendations that will optimize camera system performance and recording accuracy, regardless of the brand or equipment used.

6.0 Inspection Requirements

6.1 Surface Conditions

6.1.1 Surfaces to be inspected shall be sufficiently free from deleterious materials such as crud deposits and other conditions that could prevent detection of the smallest expected indication.

6.1.2 Industry experience has shown that a cleaning assessment must be performed prior to inspection. This applies to surfaces "as-found" condition and in the post-cleaned condition. The following industry best practices can be used to assess the suitability of surfaces for inspection.

6.1.2.1 Surfaces shall be considered suitable for inspection when surface texture identifiers such as grinding and machining marks, weld beads and ripples, undercut, arc strikes, etc. are readily visible.

6.1.2.2 If it is indeterminate whether a surface should be cleaned prior to inspection, then the following technique may be used.

a) Perform a pre-cleaning inspection of a worst case area that includes the identifiers listed in 6.1.2.1 above, then clean the area and perform a post-cleaning inspection. If the identifiers seen in the post-cleaning inspection are significantly clearer or sharper than in the pre-cleaning inspection, or previously undetected landmarks are identified, then the existing surface conditions have the potential for masking crack indications, and cleaning shall be performed prior to inspecting.

6.1.3 Cleaning methods shall not smear surface sediment, which could mask or hinder the detection of indications. Cleaning methods also shall not produce a polished surface finish that could cause excessive glare and prevent the detection of indications.

6.1.4 Industry experience has shown that abrasive type pads can smear sediments and hinder the detection and evaluation of indications. Nylon bristle brushes and hydrolazing have been shown to be effective in removing these materials <u>without</u> producing a shiny, reflective surface.

6.1.5 A cleaning assessment may determine that pre-inspection cleaning is needed, but that it cannot be performed due to physical restraints that prevent access for cleaning equipment. When this is the case, a utility should perform a "best-effort" inspection of the area and document the reason(s) that no cleaning was performed.

6.2 Minimum Water Clarity

6.2.1 Water clarity, throughout the inspection, should remain equal to or better than that used during the resolution demonstration. If water clarity becomes suspect, inspections should be discontinued and a resolution demonstration performed per 6.3.

6.3 Equipment Resolution Demonstration Requirements

6.3.1 The resolution capabilities of the inspection equipment and technique shall be demonstrated using an SCRS, in an environment representative of the area inspections will be performed in, prior to performing inspections and at any time a key element is changed. A resolution demonstration check (RDC) shall be performed at the beginning and end of each inspection or series of inspections.

Note: Although not a requirement of this standard, if inspections are to be recorded, it is recommended that the RDC be recorded. Whenever an RDC is recorded, a review of the videotape should be performed prior to continuing inspections to ensure that the required target is visible on the recorded image.

6.3.2 The RDC shall be considered adequate provided the system is capable of discerning the required target. The lens-to-object distance required to discern the target on the SRCS becomes the maximum distance inspections can be performed from the inspection surface.

6.3.3 If an RDC fails to meet the requirements of 6.3.2, all inspections performed since the last valid RDC shall be considered void and the areas reexamined.

Note: In cases where a camera experiences sudden, complete failure (e.g., electronic failure, mechanical breakage, etc.), and the inspection was video recorded, an evaluation may be performed to determine the need for reinspection. Such evaluations shall include, as a minimum, a comparison of the recorded images made immediately after the last acceptable RDC and those made just prior to the camera failure by a Level II (or Level III) IVVI examiner qualified to this standard. This review shall focus on inspection landmarks to determine whether resolution just prior to the failure was equivalent to that experienced immediately after the last acceptable RDC. All such evaluations shall be independently reviewed by a Level III IVVI examiner qualified to this standard, and shall be approved by the utility.

6.3.4 In order to avoid the potential for extensive re-inspection, consideration should be given to performing a RDC more frequently when inspections are performed in excessively high radiation fields.

CAUTION High radiation fields may cause cameras to "burn out." This is evidenced by foggy or hazy areas or spots on the monitor for black and white cameras and "snowy" images for color cameras. This condition can be temporary or permanent. If it occurs, the camera should be moved to a lower radiation field and a RDC performed. If the targets on the SRCS can be resolved, continued use of the camera is warranted, however, consideration should be given to using a more-radiation-tolerant camera in the area that produced the condition. If, however, the SRCS targets cannot be resolved, the camera shall be repaired or replaced and all areas inspected since the last acceptable RDC shall be repeated.

6.4 Area(s) of Interest

6.4.1 The area of interest for components shall consist of all accessible surfaces of the components. For welds, the area of interest is dependent upon the type and thickness of the base material.

6.4.1.1 For welds in low alloy or stainless steel materials one-half inch or less in thickness the inspection area shall include, as a minimum, the toe of the weld and three quarters of an inch of the adjacent base material on each side of the weld.

6.4.1.2 For welds in low alloy or stainless steel materials greater than one half inch in thickness the inspection area shall include, as a minimum, the toe of the weld and one-half inch of the adjacent base material on each side of the weld.

6.4.1.3 For welds in nickel based materials one-half inch and less in thickness the inspection area shall include, as a minimum, the entire width of the weld and three quarters of an inch of the adjacent base material on each side of the weld.

6.4.1.4 For welds in nickel based materials greater than one-half inch in thickness the inspection area shall include, as a minimum, the entire width of the weld and one-half inch of the adjacent base material on each side of the weld.

6.5 Inspection Technique

6.5.1 The distance the camera lens is to the inspection surface is highly dependent on several factors. Among these are the environment, lighting, and type of equipment, water clarity, and surface condition. Since all of these variables can change throughout the course of an inspection, it is important to ensure a proper inspection is being performed at all times during the inspection.

6.5.2 The maximum distance the camera should be from the inspection surface is to be determined based on the level of detail that can be seen on the inspection surface. Industry experience has shown that a quality inspection can be performed provided surface anomalies such as grinding and machining marks, weld beads and ripples, undercut, arc strikes, etc. can be conducted within the parameters established during the RDC. The screen image should include a small amount of the weld, as well as the area of interest, for use as a landmark.

6.5.3 Where inspections are to be recorded, commentary and evaluations by examiners, dubbed onto the recording medium during inspection, is a valuable tool. It provides reviewers first-hand information about the inspection. Character generators also provide the useful information for evaluating and identifying data. Information that has proven useful when dubbed or generated onto the recording medium includes component and/or weld identification, azimuthal references, explanation of indications, etc.

6.5.4 When performing inspections, great care should be taken to eliminate shadows in the area of interest. If shadows are present, re-inspection of those areas should be performed with the shadows removed.

6.5.5 Inspections should be overlapped in a manner that allows documentation of complete coverage of the specified area of interest.

6.5.6 When access and component geometry permits, the angle at which the camera views the inspection surface should not be less than 30 degrees (not greater than 60 degrees from surface normal) or that demonstrated during the resolution demonstration.

7.0 Evaluation of Indications

7.1 Classification of Indications

Note: Indications should be interpreted and evaluated as the inspections are being performed (real time). Industry experience has shown that real time evaluation is the most accurate and efficient way of evaluating indication relevancy. It also affords examiners an opportune time for additional cleaning, re-looks from different angles, etc. However, if recording mediums produce the required resolution sensitivity, they may also be used for evaluation.

7.1.1 Indications shall be classified as either "relevant" or "non-relevant." If an indication cannot be classified during initial inspections, then additional cleaning, re-looks, etc. shall be performed. If the indication still cannot be classified through visual inspection, it shall be considered as "relevant."

7.1.2 Process enhancements such as electronic digital image enhancement devices may be used for evaluation provided they produce the required resolution sensitivity. However, caution should be taken when using these devices since their features simply provide a means to highlight and enhance visual images of indications and furnish photographic quality hard copy prints. They do not magnify the image so that an enhanced evaluation can be performed.

7.1.3 Industry experience has shown that an independent review of the inspection, performed by a second qualified individual, significantly increases the level of confidence that indications have been properly identified and evaluated. Recording of the inspection provides a means for this review to be accomplished off of critical path.

7.2 Measurement of Relevant Indications

7.2.1 Determining indication length can be challenging due to remote operation of measuring tools. However, rulers and specially made devices have been used to measure indication lengths with acceptable accuracy and repeatability. Azimuthal locators (shroud, vessel, seam weld location, etc.) have also been used to measure lengths with acceptable accuracy.

7.2.2 Measurements may be taken by any means that can be demonstrated under actual conditions as having a repeatable accuracy. Training of inspection personnel may be needed for the specific methods.

8.0 Documentation of Results

8.1 Documentation should, as a minimum, include the following:

8.1.1 The inspection procedure, date(s) of inspection(s) and evaluation(s), and the inspection and evaluation personnel performing the inspections, including data reviewers.

8.1.2 The location and extent of the areas examined, including an estimate of the percentage of the examination area that was examined with EVT-1 quality.

8.1.3 Relevant indications found, including: location, length, method used to measure length (estimation or direct measurement using a ruler), orientation, unique identification, and disposition.

8.1.4 Any resolution evaluations conducted following camera failures per 6.3.3.

8.1.5 Equipment used for the inspection.

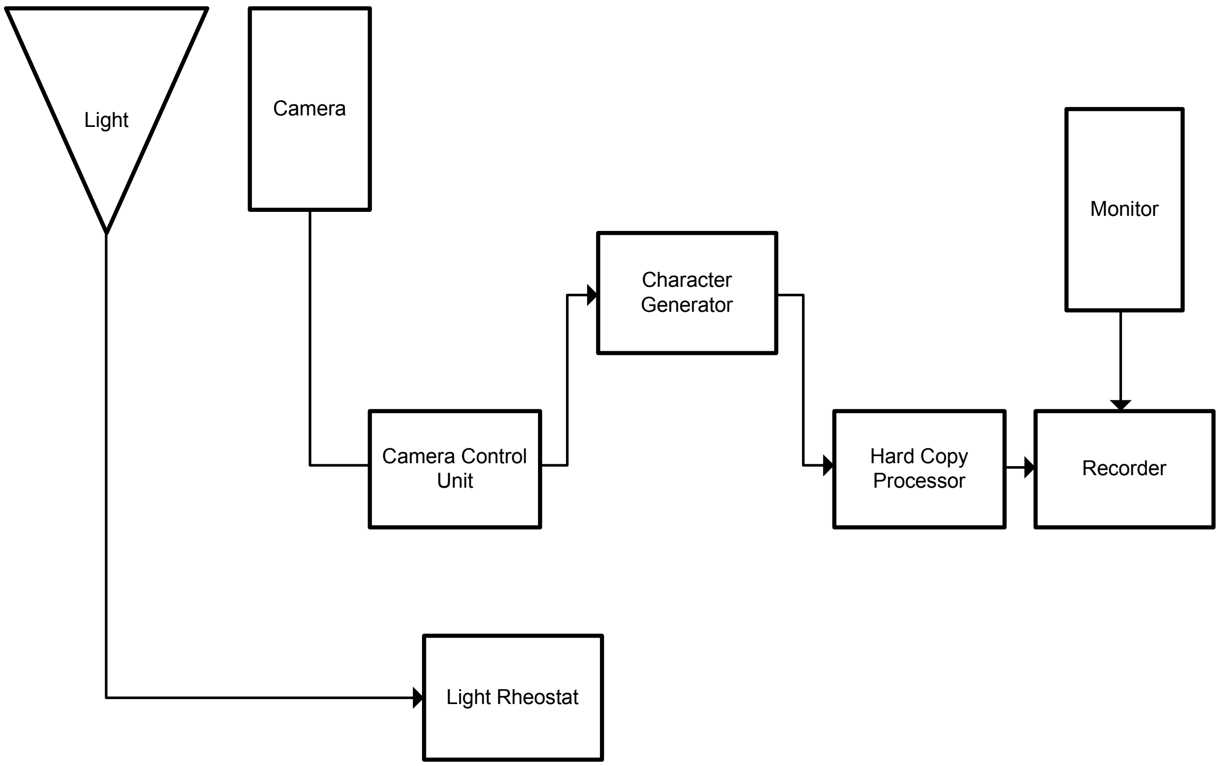

Figure A-1
System Setup

B

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

Reference 23 evaluated crack growth resistance and flaw stability for a set of representative PWR austenitic stainless steel core support structures that were assumed to be embrittled from neutron irradiation exposure. These components are subject to VT-3 visual examination in accordance with Examination Category B-N-3. Five different core support structures were evaluated, ranging from a rectangular parallelepiped representing a columnar support (Figure B-1) to a hollow circular cylinder representing a core barrel assembly (Figure B-2).

Flaws of various sizes were postulated at worst-case tensile stress locations and in worst-case orientations. Then, applied-J integrals were calculated for the postulated flaws subject to tensile stresses from nominal, design-basis, and bounding load conditions. The J-integrals were determined from linear elastic fracture mechanics (LEFM) solutions, with a conversion to elastic-plastic crack driving force valid for localized plasticity at the crack tip. As a check on the validity of the approach, approximate limit loads were calculated for the uncracked sections using mechanical properties for irradiated stainless steel. Finally, the evaluation procedures specified in Appendix K of the ASME Code Section XI were used to demonstrate flaw stability. Results for two of the core support structures – a columnar support and a toroidal ring support – are discussed below.

B.1 Columnar Core Support Structure

For the columnar support geometry, the crack driving forces for a variety of non-dimensional flaw lengths, depths and aspect ratios were evaluated. The non-dimensional flaw length was defined to be 2c/W, where 2c is the flaw length and W is the characteristic width dimension of the component (W is the width of the columnar support or the circumference of the toroidal support). The non-dimensional flaw depth was defined to be a/t, where a is the flaw depth and t is the thickness of the component. 2c/a is the aspect ratio of the flaw. Flaw lengths were selected to be 2.4 inches, 3.6 inches, 4.8 inches, and 6 inches. For each flaw length, three flaw depths were examined: (1) aspect ratio = 0.2 (flaw depths = 0.24 inches, 0.36 inches, 0.48 inches, and 0.6 inches); (2) aspect ratio = 0.4 (flaw depths = 0.48 inches, 0.72 inches, 0.96 inches, and 1.2 inches); and aspect ratio = 1.0 (flaw depths = 1.2 inches, 1.8 inches, 2.4 inches, and 3 inches). For each of the flaw depths, three different uniform remote tensile stress states were examined, 10 ksi, 20 ksi, and 30 ksi. The first of these represents the expected level of remote tensile stress under LOCA or seismic loads, the second represents a value of the remote tensile stress equivalent to the maximum design membrane stress for the material, and the third represents 150 % of that maximum design membrane stress.

For each postulated flaw length and depth, the LEFM stress intensity factor in ksi√in for each remote tensile stress level was calculated and is listed in the appropriate column of Table B-1. The column to the right of these calculated LEFM stress intensity factors shows the applied J-integral, in units of in-lb/in^2. The threshold value for crack growth resistance was selected in Reference 23 to be about 1500 in-lb/in^2. With the exception of the case where the flaw is 3 inches deep (a/t = 0.5), and the flaw length extends completely along the width of the component (2c/W = 1.0), with a remote stress is 30 ksi, all of the calculations satisfy the crack growth resistance criterion.

This is considered to be a bounding case. Note that the flaw area is 14.14 in^2, about 40 % of the cross-sectional area of the component. As a check, the applied load, P, is equal to the remote stress (30 ksi) multiplied by the cross-sectional area (36 square inches), or 1,080,000 lbs. The value of P_o depends upon the choice of the yield strength in the irradiated condition, which could vary from its initial value (e.g., 35 ksi) all the way up to 100 ksi after prolonged exposure. For purposes of comparison, a value of σ_o equal to 65 ksi was chosen. In this case, when 65 ksi is multiplied by the area of the uncracked cross section, the approximate limit load is 1,420,000 lb, well above the applied load. Therefore, as the result of increases in the yield strength caused by neutron irradiation, elastic behavior is expected to control this set of fracture mechanics analyses.

The crack depths for the various cases evaluated range from 0.24 inches to 3 inches, with the core support structure tolerant of flaws as deep as 2.4 inches at the highest tensile loads. For an applied tensile stress of 20 ksi, the core support structure is tolerant of flaws at least 3 inches deep. These are definitely mature cracks that would have large crack surface displacements in the unloaded condition.

B.2 Toroidal Core Support Structure

Results for the toroidal core support structure were obtained for three flaw lengths – 8 inches, 12 inches, and 20 inches. For each flaw length, three flaw depths were examined. These flaw depths ranged from 1.6 inches to 8 inches deep. All of these flaws can be considered mature. The remote circumferential stresses ranged from 10 ksi to 20 ksi to 30 ksi.

The complete set of results, including the LEFM stress intensity factors converted to elastic-plastic crack driving force, is given in Table B-2. Note that only two of the calculations have elastic-plastic crack driving forces that are relatively close to, or exceed, the 1500 in-lb/in^2, crack growth resistance value at a crack depth of 0.1 inches. One of these calculations corresponds to a postulated through-wall flaw 10 inches deep and 20 inches in length, subjected to 10 ksi remote circumferential tensile stresses. The applied J-integral in this case is over 1700 in-lb/in^2.

The other calculation corresponds to a postulated flaw that is 6 inches deep and 12 inches long, subjected to 30 ksi remote circumferential tensile stresses, with an applied J-integral that is well below the stability threshold. (As a point of reference, the average circumferential stress for a cylindrical shell with a diameter of 80 inches and a wall thickness of 10 inches subjected to an internal pressure of 2250 psi is about 9 ksi.) The cross-sectional area of the flaw in the latter case is 56.5 in^2, which comprises about 28 % of the 200 in^2 cross-sectional area of the torus. A somewhat larger flaw would be needed to cause unstable crack growth, even at the very high stress levels. As a check, the limit load calculation again shows that elastic behavior controls because of the increase in yield strength from neutron irradiation.

Again, the results for this case show that these core support structures are extremely tolerant of mature flaws, except in the most limiting cases, and that the crack surface displacements after unloading will be of the same order as the character recognition heights for both VT-1 and VT-3 visual examinations.

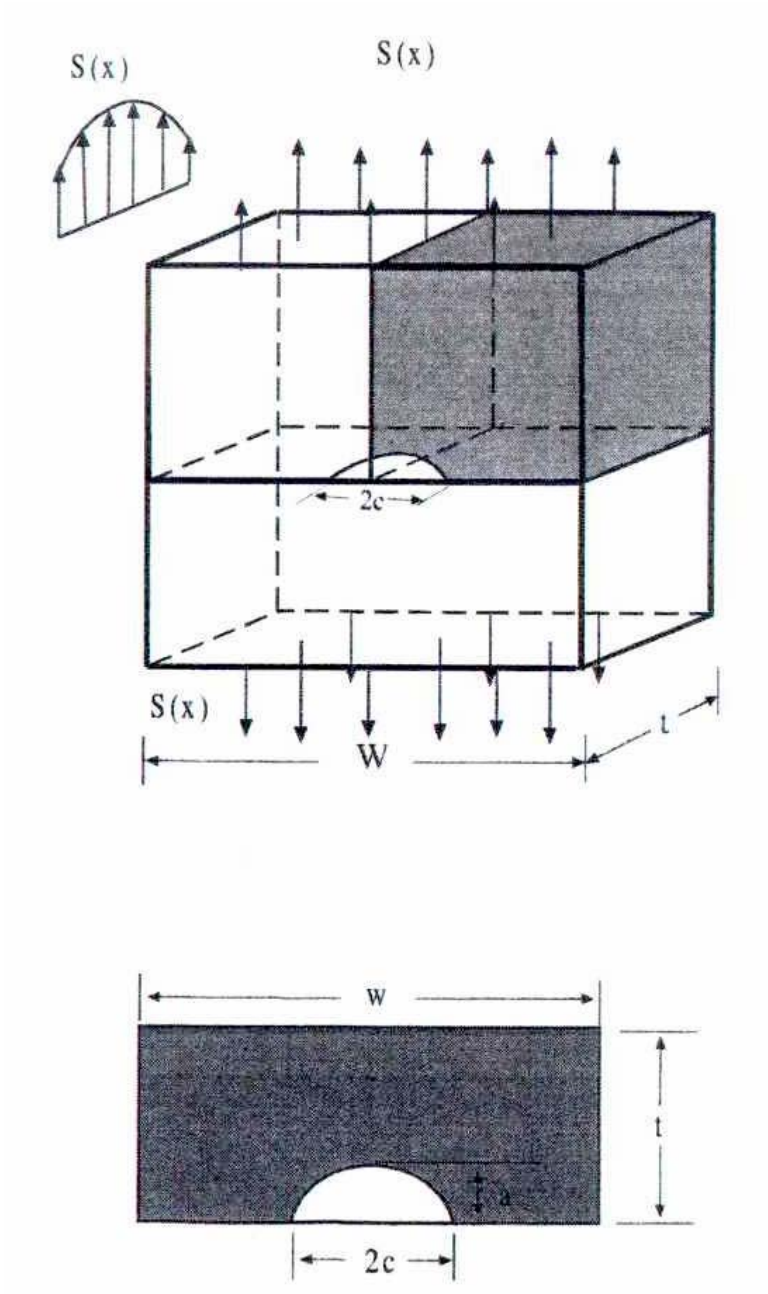

Figure B-1
Geometry and Coordinate System for Surface Edge Crack in Rectangular Parallelepiped Representing Columnar Core Support Structure

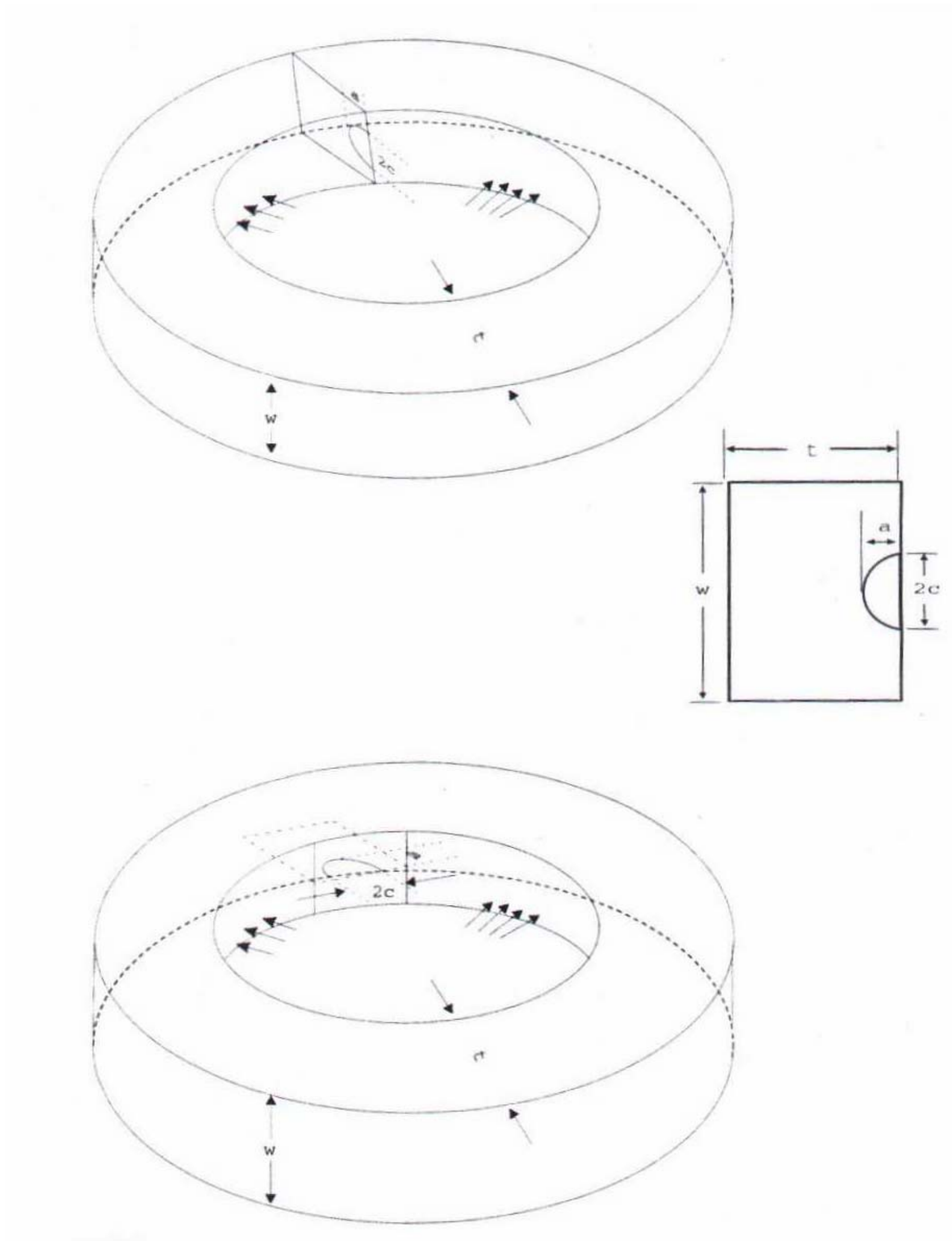

Figure B-2
Top: Geometry and Coordinate System for Axial Surface Crack in Toroid Representing Cylindrical Core Support Structure
Bottom: Geometry and Coordinate System for Radial Surface Crack in Toroid Representing Cylindrical Core Support Structure

Table B-1
Results for Rectangular Parallelepiped, W=t=6 inches

2c/W	a/t	K_I (10 ksi)	J_{appl}	K_I (20 ksi)	J_{appl}	K_I (30 ksi)	J_{appl}
0.4	0.04	9.57	3	19.14	14	28.71	31
0.4	0.08	12.87	6	25.73	25	38.60	56
0.4	0.20	19.21	14	38.41	56	57.62	125
0.6	0.06	11.72	5	23.44	21	35.16	47
0.6	0.12	16.04	10	32.09	39	48.13	87
0.6	0.30	26.81	27	53.61	108	80.42	244
0.8	0.08	14.02	7	28.04	30	42.05	67
0.8	0.16	19.52	14	39.04	58	58.56	129
0.8	0.40	40.49	62	80.98	247	121.47	557
1.0	0.10	15.94	10	31.89	38	47.83	86
1.0	0.20	23.12	20	46.23	81	69.35	181
1.0	0.50	66.71	168	133.42	672	200.13	1511

Table B-2
Results for Torus with Rectangular Cross Section, w = 30 inches, t = 10 inches

2c/W	a/t	K_I (10 ksi)	J_{appl}	K_I (20 ksi)	J_{appl}	K_I (30 ksi)	J_{appl}
0.4	0.08	17.6	12	35.3	47	52.9	106
0.4	0.16	23.7	21	47.4	85	71.1	191
0.4	0.40	38.1	55	76.2	219	114.3	493
0.6	0.12	22.0	18	43.9	73	65.9	164
0.6	0.24	30.6	35	61.2	141	91.8	318
0.6	0.60	58.1	127	116.3	510	174.4	1148
1.0	0.20	30.7	36	61.3	142	92.0	319
1.0	0.40	50.6	97	101.3	390	151.9	871
1.0	1.00	212.5	1704	-----	------	------	------

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).

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