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Aging Management Strategy and Requirements Of Pressurized Water Reactor Internal Components

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1. Introduction

Pressurized water reactor (PWR) internals components are to hold a core, fix fuel assemblies in their positions, distribute coolant flow in a reactor, support and protect control rods, decrease the neutron effects onto reactor vessel. Components of PWR internals are consisted of austenitic stainless steel materials which are known to have good strength, ductility, toughness and corrosion resistance. However, when placed in a PWR environment, these properties undergo changes due to long-term exposure to neutron irradiation, high temperature, reactor water, and loading.

The demonstration that the effects of degradation in the components of PWR internals are adequately managed is essential for maintaining a healthy fleet and ensuring the continued functionality of the reactor internals. It is also very important to determine when and where irradiation susceptibility may occur for the continued operation. This paper introduces the aging management strategies and requirements for PWR internals components and discusses effects of irradiation aging results from the functionality assessments based on the categorization of internal components.

2. Aging Management Strategy and Requirements

2.1 Status of RV internal Aging Management

All PWRs in the U.S. must have in place a program for managing aging of RVIs components in accordance with MRP-227 [1]. Each PWR in the U.S. shall implement the tables in MPR-227 [1] for inspection of (Primary and Expansion category) internals components for the applicable design if required to meet license renewal commitments. Examinations of PWR internals components must be also conducted per the required schedule in MRP-227 [1] and in accordance with the inspection standard, MRP-228 [2].

Nuclear Regulatory Commission is reviewing many of the PWR internals aging management program documents and inspection plans and questioning utilities through Request for Additional Information on the basis for their aging management program. Implementation and inspections of PWR internals are related to effective full power year and the data plants enter the license renewal period. A number of plants have already performed the first round of internals inspections and generally no major degradation found.

2.2 Process of RV internal Aging Management

Aging management process in MRP-227 [1] consists of three process steps. The first step is screening, categorization and ranking through the screening criteria and failure modes and effects analysis (FMECA). The second step is functionality assessment for components and assemblies. The third step is to establish appropriate aging management strategy which combined the results of functionality analysis with component accessibility, operating experience, existing evaluations, prior examination results, and baseline and subsequent examination timing. Fig. 1 shows the links between the categorization based on screening criteria, functionality assessment, the aging management strategy, and the inspection & examination (IE) guidelines.

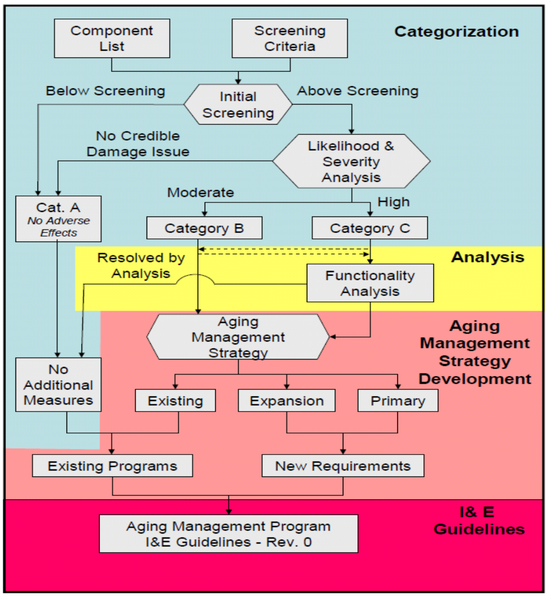

Fig. 1 Guideline Process Flow Chart [1]

Prioritization of significance of aging degradation in PWR internals can be determined by considering potential for aging based on risk factors such as fluence, temperature, stress and material and for loss of function with materials degradation. Screening criteria from MRP-175 [3] are summarily described as irradiation embrittlement, thermal aging embrittlement, stress corrosion cracking, irradiation-assisted stress corrosion cracking, wear, fatigue, void swelling and irradiation growth and thermal and irradiation-enhanced stress relaxation or irradiation enhanced creep.

After assessing each component against criteria, initial screening categorization is summarized as “A” through “C” Category. Component that met the screening criteria for all eight age-related degradation mechanisms is placed into “A” Category. Component that failed the screening criteria for at least one age-related degradation mechanism and found to be highly susceptible to at one age-related degradation mechanism, gets into “C” Category. Component that failed the screening criteria for at least one age-related degradation mechanism and found to be moderately susceptible to at one age-related degradation mechanism is placed into “B” Category.

Category B and C components are assessed further using FMECA [4,5,6]. Experts verify screening criteria results and review the probability of occurrence of degradation and assess the probability of failure and determine the consequence of failure. Some components assess further for functionality including the degradation effects [7].

The ways that screening of components based on thresholds, identifying cracking susceptibility factors and categorization of components based on prioritization and ranking of susceptibility may be useful for identifying lead components and eliminating regions from consideration for inspection. The screening and categorization will also consider results of functionality analyses which show no loss of function if the component is flaw tolerant.

All components ended up in one of four categories. Primary is highly susceptible/accessible or moderately susceptible and most accessible component. Expansion is highly or moderately susceptible component but functionality analysis shows tolerance and added when examination results warrant a larger inspection sample. Existing Programs is when current generic or plant-specific examinations are adequate to manage aging degradation effects. No Additional Measures are that effects of all eight aging mechanisms are below screening criteria. Component categorization results [7] for Westinghouse-designed internals indicate that 29 Category B and C components; 8 Primary, 7 Expansion, 8 Existing Program, and 6 No Additional Measures.

2.3 Aging Management Requirements

Aging management activities for PWR internals are based on the inspection and evaluation, which generally consist of the following;

- Selection of items for aging management
- Selection of type of examination or other methodologies appropriate for each applicable degradation effect and
- Specification of the required level of examination qualification
- Schedule of examination frequency
- Sampling, coverage, and accessibility
- Examination scope expansion, as required
- Examination acceptance criteria
- Methods for evaluating examination results not meeting the examination acceptance criteria
- Repair, replacement, and mitigation strategies
- Updating aging management program, based on industry-wide results.

Aging management approaches for PWR internals use established aging management methodologies that should be appropriate for the characterization of particular age-related degradation effects. The selected aging management methodologies emphasize existing, well-proven techniques that have been subject to widespread, relevant application. Aging management methodologies include visual examinations, surface examinations, volumetric examinations, and physical measurements. Visual (VT-3) examinations are conducted to determine the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.

The visual (VT-1) examination is used for the detection of surface discontinuities such as gaps, while the enhanced visual (EVT-1) examination is used for the detection of surface breaking flaws. Surface examination is an examination that indicates the presence of surface discontinuities, and eddy current testing (ET) is considered as surface examination alternatives. Ultrasonic examination (UT) is selected where visual or surface examination is unable to detect the effect of the age-related degradation in bolting, and UT is also permissible as an alternative or supplement to the specified visual examinations for other configurations such as plates and welds.

The aging management requirements for Primary and Expansion PWR internals are listed in tables, which contain columns describing the component; any particular applicability requirement for that component; the degradation effect to be detected; the examination method; the examination coverage; and any linkage between the Primary and Expansion components. There are no specified examinations where inadequate coverage is anticipated to be an issue. If the examination coverage is questionable with respect to meeting the intent of the guidelines, the condition should be entered in the corrective action program for disposition. For example, Table 1 and 2 show the Primary and Expansion requirements for Westinghouse and CE internals.

Table. 1 Primary Component Table for
CE Designed Plants

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Core shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 2)	Core support column bolts, Barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (see Note 3). Heads are accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figure 4-24.
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of central flange and horizontal stiffeners. See Figures 4-12 and 4-14.
Core Shroud Assembly (Welded) Shroud plates	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds, Ribs and rings	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (± three feet in height) as visible from the core side of the shroud. See Figure 4-13.

Table. 2 Primary Component Table for
Westinghouse Designed Plants

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Internals Assembly Upper core plate	All plants	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
Lower Internals Assembly Lower support forging or castings	All plants	Cracking Aging Management (TE in Casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure 4-33.
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads (Note 2). See Figure 4-23.
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant-specific justification (Note 2). See Figures 4-32 and 4-33.

3. Aging Degradations Effects

3.1 Functionality Assessments

In the process of the development of aging management and inspection strategies in PWR internals, several components have been identified as Category C, components and recommended for the highest priority functionality assessments. These assessments involve the analyses of the effects of irradiation-induced degradation of mechanical and/or physical properties of these austenitic stainless steel components.

For the Westinghouse designed reactor internals, the category C components for which the functionality assessments are to be performed includes baffle-former bolts, barrel-former bolts, upper core barrels and associated welds, lower core barrels and associated welds, welded core shroud, bolted low core plates and welded core support plate. Fig. 2 shows the category C core surrounding components.

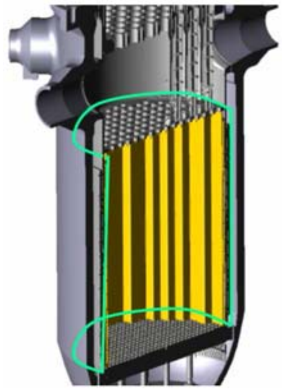

Fig. 2 Category C core surrounding components

3.2 Functionality Analysis Procedure

The functionality analysis selected components of Category C by subjecting them to representative core thermal and irradiation loading for 40 cycles or 60 years. These analyses are being used to determine when and where irradiation susceptibility may occur. Fig. 3 shows the functionality analysis procedure for the components which are judged to be susceptible to irradiation aging degradation.

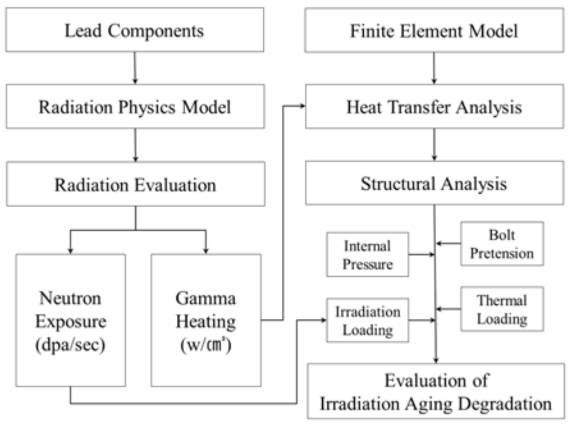

Fig. 3 Functionality Analysis Procedure

3.3 Functionality Analysis Assumptions

In the performance of the functionality analyses, conservative core loading, heat transfer, mechanical preloads, and fabrication loads such as weld residual stresses are applied. The material model [8] represents susceptibility to irradiation-induced degradation of mechanical properties using an ANSYS-based subroutine [9]. The loading patterns consisted of 20 fuel cycles of Out-In core loading followed by 20 fuel cycles of Low-Leakage core loading. A friction factor of 0.1 is applied to all standard contact interfaces in the structural analyses. Each load cycle was initiated from a cold thermal standby loading state in nuclear power plant.

3.4 Irradiation-Induced Aging Results

Core surrounding components is held together using baffle-former, barrel-former, and baffle-edge bolts. The most highly irradiated locations in the core surrounding components are occurred at the internal corners of the baffle plates. The peak value in the baffle plates is due to swelling-induced contact between the two baffle plates. The former plates show localized indications of irradiation assisted stress corrosion cracking (IASCC) due to thread-to-former coupling. The baffle-former bolts show the highest results with several indicating IASCC susceptibility. The baffle plates show localized indications of IASCC susceptibility due to contact. The bolts show greater susceptibility to IASCC at the head-shank interface than at the thread-former interface. The core barrel and core barrel weld have very low stresses and show little susceptibility to IASCC.

3. Conclusions

This paper introduces aging management strategies and requirements for PWR internals components. The aging management requirements for PWR internals are specified in four final component groups, which are Primary, Expansion, Existing Program and No Additional Measures. Among these groups, Primary groups include any restriction on general applicability, degradation mechanism, forward link to any Expansion components, examination method, initial examination and frequency, and examination coverage and accessibility. Expansion groups are backward link to the Primary component. Also, the functionality assessment is performed for the core surrounding category C components based on the results of the screening, ranking, and categorization of reactor internals components. Especially bolts at all former levels where is nearest to the core indicate the potential of the IASCC.

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