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Screening in Reactor Vessel Internals for Aging Degradation Mechanisms

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1. Introduction

In order to establish aging management program for Reactor Vessel Internals(RVIs) in domestic Pressurized Water Reactor(PWR), it is necessary to precede characterization and identification of components, materials, and operating conditions. Based on the results, potential degradation mechanisms should be analyzed for each component.

In this study, development process of aging management program is introduced for RVIs based on MRP-191[1] and MRP-227-A[2], and lists of susceptible aging degradation mechanisms are provided by applying screening criteria for RVIs of domestic plants.

2. Characterization of Domestic RVIs Components

The 6 Westinghouse(WH)-type, 8 Combustion Engineering(CE)-type, and 2 Framatome-type plants are being operated in Korea. In keeping with the overall objective of this study to screen age-related degradation mechanisms for RVIs components, it is necessary to identify the components and materials for each type plant. Table 1 shows a list of WH-type plant RVIs components and the materials based upon Final Safety Analysis Report(FSAR) and Periodic Safety Review Report(PSAR). The RVIs components in the CE-type and Framatome-type plants are also characterized based upon FSAR and PSAR.

Table 1. Components and materials for Kori 1,2,3,4 and Hanbit 1,2 units.

Assembly	Component	Material
Upper Internals Assembly	Upper support plate	304 SS
	Upper core plate	304 SS
	Fuel alignment pin	316SS(CW)
	Upper support column	304 SS
	Upper support column bolt	316 SS(CW)
	Control rod guide tube	304 SS
Lower Internals Assembly	Lower core plate	304 SS
	Lower support column	304 SS
	Core barrel flange	304 SS
	Core barrel	304 SS
	Radial support key	304 SS
	Baffle plate	304 SS
	Former plate	304 SS
	Baffle-former bolt	316 SS(CW)
	BMI column	304 SS
	BMI flux thimble	316 SS(CW)
Interfacing Component	Clevis insert(Kori 1)	304 SS
	Clevis insert (Kori 2,3,4 Hanbit 1,2)	Alloy 600
	Clevis insert bolt	X-750
	Hold down spring	304 SS

3. Operating Conditions of RVIs Components

The screening process requires information on the operating conditions for each RVIs component. EPRI had been performed to characterize the conditions from available documents and expert interviews under consideration of six basic concepts as follows [3, 4];

1. Could the operating stress be over 207 MPa(30 ksi)?
2. Locations of components relative to the core.
3. Wear potential.
4. Could the cumulative fatigue usage factor be over 0.1 at 40 operating years?
5. Existence of the structural welds.
6. Is the component preloaded?

Neutron fluence estimates require detailed results in connection with second item mentioned above. The expert panels of WH selected two typical representative plants (WH-type and CE-type), and detailed fluence maps were generated for the RVIs to 60 years of reactor operation. The maps define six distinct fluence regions.

The estimated operating conditions are used in the screening evaluation of susceptibility to the materials aging degradation mechanisms for the RVIs components.

4. Aging Degradation Mechanisms and Screening Criteria

The screening criteria set thresholds for evaluating 8 degradation mechanisms [5].

1. Stress Corrosion Cracking (SCC): Stainless steel welds and parts with > 20% cold work are potentially susceptible to SCC. For cast austenitic stainless steels and stainless steel welds, < 5% ferrite contents is identified as contributing to SCC. SCC could accelerate at higher stress than about 600 MPa for precipitation-hardened alloys, and martensitic stainless steels.

2. Irradiation Assisted Stress Corrosion Cracking (IASCC): The criteria for IASCC apply to all materials in terms of both stress and fluence. The threshold fluence for IASCC is at approximately 3 dpa. Above this fluence, the required stress drops extremely.

3. Wear: The components should be screened in for wear when two surfaces are in contact and have relative motion. Locations with clamping force and pre-loaded bolts or springs should be also screened in for wear.

4. Fatigue: A cumulative usage factor of 0.1 at the end of the design life (40 years) is suggested as the screening criterion.

5. Thermal aging Embrittlement (TE): The screening criteria screen in all martensitic stainless steel, cast austenitic stainless steel, welds with high ferrite contents.

6. Irradiation Embrittlement (IE): For austenitic stainless steel, the fluence of 1.5 dpa is suggested as threshold value for IE. The criterion for cast austenitic stainless steel is set more conservatively as 1 dpa.

7. Void swelling: The screening criteria established for void swelling are a minimum fluence of 20 dpa, and a minimum temperature of 320 ^{o}C, which is well above the core inlet temperature.

8. Stress relaxation or irradiation creep: The stress relaxation screening criteria are deemed to apply to all bolts and springs that require pre-stress with irradiation to fluence greater than 0.2 dpa.

5. Screening Results for Domestic RVIs

In the application of the screening criteria, certain simplifying assumptions are made to provide clarity as follows [1];

· SCC: All austenitic stainless steel components with heavy deformation or welds are initially screened in regardless of the effective stress. All cast stainless steel having stress levels above 207 MPa are screened in for SCC, without a specific appraisal of the ferrite content. The components having failure cases with SCC degradation in service are screened for SCC without consideration of the screening criteria.

· IASCC: All components below 1.5 dpa are screened out for IASCC, and all components above 15 dpa are screened in for IASCC. Between 1.5 and 15 dpa, all components having effective stress above 207 MPa are screened in for IASCC.

· TE: All cast austenitic stainless steel are screened in for TE without consideration of ferrite content.

· Void swelling: In the case of void swelling, A slightly more conservative screening value, 15 dpa, is used in place of screening value of 20 dpa determined in MRP-175 [5].

The parts of the screening results for domestic WH-type RVIs components are presented in Table 2. Based on the described process above, the domestic CE and Framatome-type RVIs components are also screened in for the aging degradation mechanisms. Additional revise of the screening results is necessary for the Framatome-type RVIs, because those are screened in by applying the analysis results for WH and CE-type RVIs without own analysis.

Table 2. Screening results for Kori 1,2,3,4 Hanbit 1,2 units.

Assembly	Component	Aging mechanism	Note
Upper Internals Assembly	Upper core plate	Fatigue	-
		Wear	-
	Fuel alignment pin	Wear	-
	Upper support column bolt	Fatigue	Synergy effect with stress-relaxation
		Wear	Synergy effect with stress-relaxation
		Stress-relaxation	-
	Control rod guide tube	Wear	-
Lower Internals Assembly	Lower core plate	IASCC	$\Phi > 15$ dpa
		Fatigue	-
		Wear	-
		IE	-
	Core barrel flange	SCC	Welds
		IASCC	-
		Fatigue	-
	Core barrel	SCC	Welds
		IASCC	-
		Fatigue	-
		IE	-
	Baffle plate/ Former plate	IASCC	$\Phi > 15$ dpa
		Void swelling	-
	Baffle-former bolt	IASCC	Failure case in service
		Fatigue	Synergy effect with stress-relaxation
		IE	-
		Void swelling	-
		Stress relaxation	Synergy effect with void swelling
Interfacing Component	Clevis insert	Wear	-
	Clevis insert bolt	SCC	Failure case in service
		IASCC	-
		Fatigue	-
		Wear	-
		Stress relaxation	-

REFERENCE

[1] EPRI, 2006, Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components, MRP-191.
[2] EPRI, 2011, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227-A.
[3] WCAP-14577, 2001, Revision 1A, License Renewal Evaluation: Aging Management for Reactor Internals.
[4] USNRC, 2001, Generic Aging Management Review Report for the Reactor Internals, CE NPSD-1216, Rev.0.
[5] EPRI, 2006, Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values, MRP-175.